CHAPTER 4 – REACTOR COOLANT SYSTEM

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4.0 REACTOR COOLANT SYSTEM

4.1 DESIGN BASES

4.1.1 PERFORMANCE OBJECTIVES

4.1.1.1 Steam Output

The reactor is licensed to operate at a power level of 2568 MWt. The Reactor Coolant System (RCS) is designed to operate at a core power level of 2568 MWt and transfer a total of 2584 MWt (including 16 MWt input from the reactor coolant pumps) to the steam generators. The system will produce a total steam flow of 11.0 million lb/hr.

The design information presented in this section is for a reference design reactor power level of 2568 MWt.

4.1.1.2 Transient Performance

The RCS will follow step or ramp load changes under automatic control without relief valve or turbine bypass valve action as follows:

a. Step load changes - increasing load steps of 10 percent of full power in the range between 20 percent and 90 percent full power and decreasing load steps of 10 percent of full power between 100 percent and 20 percent full power.

b. Ramp load changes - increasing load ramps of 10 percent per minute in the range between 20 percent and 90 percent full power are acceptable, or decreasing load ramps of 10 percent to 15 percent full power. From 15 percent to 20 percent and from 90 percent to 100 percent full power, increasing ramp load changes of 5 percent per minute are acceptable.

4.1.1.3 Partial Loop Operation

The RCS will permit operation with less than four reactor coolant pumps in operation. The steady-state operating power levels for combinations of reactor coolant pumps operating are as follows:

Reactor Coolant Pumps Operating*	Rated Power (%)
4	100
3	75
1 pump each loop	49

* Section 7.1.2 describes the protective system that allows partial loop operation.

4.1.2 DESIGN CONDITIONS

4.1.2.1 Pressure

The RCS components are designed for an internal pressure of 2500 psig.

4.1.2.2 Temperature

The RCS components are designed for a temperature of 650 °F with the exception of the pressurizer, and surge line, both of which are designed for 670 °F.

4.1.2.3 Reactor Loads

All components in the RCS are supported and interconnected so that piping reaction forces result in combined mechanical and thermal stresses in equipment nozzles and structural walls within established code limits. Equipment, pipe supports, and restraints are designed to absorb design basis piping rupture reaction loads for elimination of secondary accident effects such as pipe motion and equipment foundation shifting. Protection of RCS is described in Section 4.2.1.2.

4.1.2.4 Cyclic Loads

All RCS components are designed to withstand the effects of cyclic loads due to system temperature and pressure changes. Design transient cycles are shown in Table 4.1.1.

Design transients including cycle limits are described in AREVA document 18-1173549, "Function Specification Reactor Coolant System for Three Mile Island Unit One." These are the analyzed limits for transients.

It is identified in the NRC SER for License Renewal, for certain components the projected 60-year Environmentally Adjusted Fatigue (EAF)-adjusted Cumulative Usage Factor (CUF) values exceed the fatigue limit, the applicant performed additional fatigue evaluations for these components to establish a set of new transient cycle administrative limits which would result in acceptable EAF-adjusted CUF values during the period of the extended operation. These limits will be implemented by site-specific T&RM ER-TM-470-1000.

Flow-induced vibration analyses have been performed for the fuel assembly, including fuel rods, and for the reactor internal components. The analyses and design criteria for the thermal shield, flow distributor assembly, and shroud tubes, and the U-baffles are given in Reference 14. For more information on these components, refer to Chapter 3.

Components subjected to cross flow are checked for response during design, so that the fundamental frequencies are those associated with cross flow, and these are the vortex shedding frequencies. It has also been conservatively determined that the flow-induced pressure fluctuations acting on the disc of the vent valve are such that for normal operation there is always a positive net closing force acting on the disc. Emergency operational modes are covered in References 15 and 16.

4.1.2.5 Seismic Loads and Loss of Coolant Loads

RCS components are designated as seismic Class I equipment and are designed to maintain their functional integrity during earthquake. Design is in accordance with the seismic design bases shown below. The loading combination and corresponding design stress criteria for

internals, vessel, integral support attachments, and piping are given in this section. The design criteria (loading combinations and corresponding stress limits) for the reactor vessels and internals of the Oconee and TMI-1 units are identical. Refer to Section 5.4.4 and 5.4.5 for a further discussion of piping system analyses. A discussion of each of the cases of loading combinations follows:

a. Seismic Loads

Case I - Design Loads Plus Design Earthquake Loads - For this combination, the reactor must be capable of continued operation; therefore, all components of the RCS are designed to Section III of the ASME Boiler and Pressure Vessel Code for Reactor Vessels. The primary piping is designed according to the requirements of USAS B31.7. The seismic design of the hot leg piping portion between the flow meter inlet and steam generator inlet nozzles is in accordance with the ASME B&PV Code, Section III, Division 1, NB-3600, 1992 Edition through 1993 Addenda. The Sm values for all components, excluding bolting, are those specified in Table N-421 of the ASME Code. Sm values for bolts are those specified in Table N-422 of the ASME Code Section Section III. RCS component codes are listed in Table 4.1-2.

Case II - Designed Loads Plus Maximum Hypothetical Earthquake Loads - In establishing stress levels for this case, a no-loss-of-function criterion applies, and higher stress values than in Case I can be allowed. The multiplying factor of (1.2) has been selected in order to increase the code-based stress limits and still ensure that for the primary structural materials, i.e., 304 SS, 316 SS, SA302B, SA212B, and SA106C, an acceptable margin of safety will always exist. A more detailed discussion of the adequacy of these margins of safety is given in Reference 15. The Sm values for all components are those specified in Table N-421 of the ASME Code, Section III.

b. Loss of Coolant Loads

A loss of coolant accident (LOCA) coincident with a seismic disturbance has been analyzed to assure the ability to initiate and maintain reactor shutdown and emergency core cooling. Two additional cases are considered as follows:

Case III - Design Loads Plus Pipe Rupture Loads - For this combination of loads, the stress limits for Case I are imposed for those components, systems, and equipment necessary for reactor shutdown and emergency core cooling.

Case IV - Design Loads Plus Maximum Hypothetical Earthquake Loads Plus Pipe Rupture Loads - Two-thirds of the ultimate strength has been selected as the stress limit for the simultaneous occurrence of maximum hypothetical earthquake and reactor coolant pipe rupture. As in Case III, the primary concern is to maintain the ability to shut the reactor down and to cool the reactor core. This limit assures that a material strength margin of safety of 50 percent of the design load will always exist.

The design allowable stress of Case IV loads is given in Figure C-1 of Reference 15 for 304 stainless steel. This curve is used for all reactor vessel internals including bolts. It is based on adjusting the ultimate strength curves published by U.S. Steel in ADUSS 43-1089 to minimum ultimate strength values by using the ratio of ultimate strength

given by Table N-421 of Section III of the ASME Boiler and Pressure Vessel Code at room temperature to the room temperature strength given by U.S. Steel.

In Cases II, III, and IV, secondary stresses were neglected, since they are self-limiting. Design stress limits in most cases are in the plastic region, and local yielding would occur. Thus the conditions that caused the stress are assumed to have been satisfied. See Reference 15 for a more extensive discussion of the stresses, the deflections, the margin of safety, the effects of using elastic equations, and the use of limit design curves for reactor internals. Effects of asymetric LOCA loadings are described in Reference 17.

Stress Limits for
Seismic, Pipe Rupture, and Combined Loads

Case	Loading Combination	Stress Limits
I	Design loads + design earthquake	$P_m \le 1.0\ S_m$ $P_L + P_b \le (1.5\ S_m)$
II	Design loads + maximum hypothetical earthquake loads	$P_m \le 1.2\ S_m$ $P_L + P_b \le 1.2\ (1.5\ S_m)$
III	Design loads + pipe rupture loads	$P_m \le 1.2\ S_m$ $P_L + P_b \le 1.2\ (1.5\ S_m)$
IV	Design loads + maximum hypothetical earthquake loads + pipe rupture	$P_m \le 2/3\ S_u$ $P_L + P_b \le 2/3\ S_u$

Where*:

P_L = Primary local membrane stress intensity

P_m = Primary general membrane stress intensity

P_b = Primary bending stress intensity

S_m = Allowable membrane stress intensity

S_u = Ultimate stress for unirradiated material at operating temperature

* All symbols have the same definition or connotation as those in ASME B&PV Code, Section III, Nuclear Power Plant Components.

All components are designed to insure against structural instabilities regardless of stress levels. See Section 4.3.3 for Reactor Vessel Irradiation.

4.1.2.5.1 Seismic Loads and Loss of Coolant Loads for the Steam Generators

The steam generators are designed as Seismic Class I equipment and are designed to maintain their functional integrity during earthquakes. Design is in accordance with the design bases shown below. The loading combinations and corresponding design stress criteria for internals, vessel and integral support attachments are given in this section. The loading conditions and transients outlined below are described in the applicable equipment and functional specifications (References 67 and 68).

a. Design Loading Conditions

Design - Design Loads Plus Operating Basis Earthquake (OBE) Loads - For this combination, OBE loads are combined with the other design and normal loads and compared to the stress limits given by the ASME Code for design loading conditions. The steam generators must be capable of continued operation and sustain no damage, internal or external, when subjected to design loading conditions and the limits as outlined in paragraph NB-3221 of Section III of the ASME Code.

b. Normal Loading Conditions

Normal - Normal Loads Plus Normal Transients - Internal normal mechanical loads, deadweight and thermal loads are combined with loads due to normal condition transients and compared to the stress limits given by the ASME Code for stress limit level "A". The steam generator must be capable of continued operation and sustain no damage, internal or external, when subjected to normal loading conditions and the limits as outlined in paragraph NB-3222 of Section III of the ASME Code.

c. Upset Loading Conditions

Upset - OBE Loads Plus Upset Loads and Transients - Operating Basis Earthquake (OBE) loads are combined with internal upset mechanical loads, upset condition transients, deadweight and thermal loads and compared to the stress limits given by the ASME Code for stress limit level "B". The steam generators must be capable of sustaining damage without requiring repair when subjected to upset loading conditions and the limits as outlined in paragraph NB-3223 of Section III of the ASME Code.

d. Emergency Loading Conditions

Emergency - Emergency Loads Plus Emergency Transients - Internal emergency mechanical loads, deadweight and thermal loads are combined with loads due to emergency condition transients and compared to the stress limits given by the ASME Code for stress limit level "C." The steam generators may experience large deformations in areas of structural discontinuity, when subjected to emergency loading conditions and the limits as outlined in paragraph NB-3224 of Section III of the ASME Code, which may necessitate the removal of the component from service for inspection or repair of damage.

e. Faulted Loading Conditions

Faulted - Faulted Loads and Transients Plus Safe Shutdown Earthquake (SSE) Loads - SSE loads are combined with, pipe rupture loads, internal faulted mechanical loads,

normal loads and loads due to faulted condition transients and compared to the stress limits given by the ASME Code for stress limit level "D." The steam generators may experience gross general deformations with some loss of dimensional stability and damage requiring repair, when subjected to faulted loading conditions and the limits as outlined in paragraph NB-3225 of Section III of the ASME code, which may necessitate the removal of the component from service for inspection or repair of damage.

Loading Conditions and ASME Stress Levels
for the Steam Generators
(Source: Reference 64)

Loading Conditions	Service Loads/Combinations (Note 1)	ASME Service Stress Limit Level	Stress Limit
Design	Deadweight Operating Basis Earthquake (OBE) Design Pressure Design Temperature Design Thermal Flow Internal Normal Mechanical Loads	Design	$P_m \leq S_m$ $P_L + P_b \leq 1.5\ S_m$
Normal	Deadweight Thermal Internal Normal Mechanical Loads Normal Condition Transients (Note 2)	A	$P_L + P_b + Q \leq 3\ S_m$ $P_L + P_b + Q + F \leq S_a$
Upset	Deadweight Thermal OBE (Note 4) Internal Upset Mechanical Loads Upset Condition Transients (Note 2)	B	$P_m \leq 1.1\ S_m$ $P_L + P_b \leq 1.65\ S_m$ $P_L + P_b + Q \leq 3\ S_m$ $P_L + P_b + Q + F \leq S_a$
Emergency	Deadweight Thermal Internal Emergency Mechanical Loads Emergency Condition Transients (Note 2)	C	$P_m \leq 1.2\ S_m$ or S_y (Note 5) $P_L + P_b \leq 1.8\ S_m$ or $1.5\ S_y$ (Note 5)
Faulted	Deadweight Thermal Safe Shutdown Earthquake (SSE) (Note 3) Internal Faulted Mechanical Loads Pipe Rupture Loads (Note 3) Faulted Condition Transients (Note 2)	C	$P_m \leq 2.4\ S_m$ or $0.7\ S_u$ (Note 6) $P_L + P_b \leq 3.6\ S_m$ or $1.05\ S_a$ (Note 6)

Where:

P_L = Primary local membrane stress intensity
P_m = Primary general membrane stress intensity
P_b = Primary bending stress intensity
S_m = Allowable membrane stress intensity
S_u = Ultimate strength for unirradiated material at operating temperature
S_a = Allowable amplitude of the alternating stress intensity

Notes:

1. Loading responses are combined using the absolute sum method with the exception of those addressed in Notes 3 and 5.
2. Transients are applied one at a time unless otherwise noted.
3. For faulted condition evaluations, the effects of SSE and pipe rupture are combined using the square root of the sum of the squares (SRSS) method. (Reference NUREG-0484)
4. For fatigue analysis, stresses due to OBE are combined with the Normal (Level A) and Upset (Level B) transients which produce the maximum positive and negative stresses.
5. The greater value was used to establish the stress limit.
6. The lesser value was used to establish the stress limit.

4.1.2.6 Service Lifetime

The original design service lifetime for the major RCS components was 40 years. The allowed number of cyclic system temperature and pressure changes is described in UFSAR Section 4.1.2.4.

4.1.2.7 Water Chemistry

The water chemistry is maintained to provide the necessary boron content for reactivity control and to minimize corrosion of the RCS surfaces. The design water quality is listed in Table 9.2-3. The reactor coolant chemistry is discussed in further detail in Section 9.2.

4.1.2.8 Vessel Radiation Exposure

The reactor vessel is the only RCS component exposed to a significant level of neutron irradiation and is therefore the only retaining component of concern subject to material radiation damage. The maximum exposure from fast neutron (E > 1.O MeV) has been computed to be less than 3.0×10^{19} n/cm^2 over a 40 year life with an 80 percent load factor. Reactor vessel irradiation calculations are described in Section 4.3.3.

4.1.3 CODES AND CLASSIFICATIONS

The ASME codes listed in this Subsection include the code addenda and case interpretations issued through Summer 1967 (June 30, 1967) unless noted otherwise. Quality control and quality assurance programs relating to the fabrication and erection of system components are summarized in Section 4.3.11. RCS component codes are listed in Table 4.1-2.

4.1.3.1 Vessels

The design, fabrication, inspection, and testing of the reactor vessel and closure head and pressurizer is in accordance with the ASME Boiler and Pressure Vessel Code, Section III, for Class A vessels as indicated in Table 4.1-2. Replacement Once Through Steam Generator design, fabrication, inspection, and testing is in accordance with ASME Boiler and Pressure Vessel Code, Section III, for Class 1 Vessels.

4.1.3.2 Piping

The design, fabrication, inspection, and testing of the reactor coolant piping including the pressurizer surge line and spray line is in accordance with USAS B31.7, Code for Pressure

Piping, Nuclear Power Piping, February 1968 Draft including June 1968 Errata. The feedwater header and the auxiliary feedwater header for the steam generator meet the requirements of the Code for Pressure Piping USAS B31.1-1967. A portion of the RCS hot leg piping between the flow meter inlet and steam generator inlet for both RCS loops A and B was replaced during T1R18. The material, fabrication, inspection, and testing of the replacement RCS hot leg piping is in accordance with ASME B&PV Code, Section III, Division 1, 2001 Edition through 2003 Addenda. Design analyses for the replacement piping were performed in accordance with ASME B&PV Code, Section III, Division 1, 1992 Edition through 1993 Addenda; however, the lesser of the material allowable stresses were selected from the 1993 Addenda and 2003 Addenda.

4.1.3.3 Reactor Coolant Pumps

The reactor coolant pumps are designed, fabricated, inspected, and tested to meet the intent of the ASME Boiler and Pressure Vessel Code, Section III for Class A vessels, but are not code stamped.

4.1.3.4 Relief Valves

The pressurizer code safety valves and the pilot operated relief valve comply with Article 9, Section III, of the ASME Boiler and Pressure Vessel Code.

4.1.3.5 Attachments to Loop

Nozzles on reactor coolant piping replaced in T1R18 comply with ASME Section III, Subsection NB (2001 Edition through 2003 Addenda). Remaining nozzles on the reactor coolant piping comply with USAS B31.7 February 1968 Draft June 1968 Errata. Nozzles on the vessels comply with Subsection 4.1.3.1 above.

4.1.3.6 Welding

Welding qualifications are in accordance with the ASME Boiler and Pressure Vessels Code, Section III and Section IX, as applicable.

TABLE 4.1-1
(Sheet 1 of 1)

TRANSIENT CYCLES

Design transients including cycle limits are described in AREVA document 18-1173549, "Function Specification Reactor Coolant System for Three Mile Island Unit One." These are the analyzed limits for transients.

TABLE 4.1-2
(Sheet 1 of 2)

REACTOR COOLANT SYSTEM COMPONENT CODES

Component	Codes	Addendum
Reactor Vessel	ASME III Class A	Summer 1967 [1]
Pressurizer	ASME III Class A	Summer 1967 [7]
Pressurizer Vent Nozzle	ASME III Class 1	1998 Edition through 2000
Abandoned and Relocated Pressurizer Thermowell	ASME III Class 1	1998 Edition through 2000 [3]
Pressurizer Upper Level Sensing Nozzles	ASME III Class 1	1998 Edition through 2000
Pressurizer Lower Level Sensing Nozzles	ASME III Class 1	1998 Edition through 2000
Pressurizer Sampling Nozzle	ASME III Class 1	1998 Edition through 2000
Reactor Coolant System Piping [4]	USAS B31.7 February 1968 Draft	Errata Through June 1968
Feedwater Header	USAS B31.1.0	1967
Safety and Relief Valves	ASME III Art. 9	Winter 1968
Welding Qualifications	ASME III and IX	Summer 1967
Steam Generator (primary and secondary sides)	ASME III Class 1	2001 Edition through 2003
Control Rod Drive Mechanisms	ASME III Class 1 Appurtenance	1971 Edition through 1972 / 1974 Edition [2]
Hot Leg Piping [5]	ASME III Class 1	2001 Edition through 2003 [6]
Cold Leg Letdown Nozzle [8] (Full Structural Weld Overlay)	ASME III Class 1	2004 Edition, No Addenda

[1] Welded joints tested in accordance with requirements of Article 7, Winter 1965 Addenda

[2] 1971 Edition with Addenda through Summer 1972, including Code Case 1337-4; 1974 Edition with no Addenda including Code Cases 1337-9 and 1337-10.

TABLE 4.1-2
(Sheet 2 of 2)

[3] Modification designed to ASME III Class 1, 1998 Edition through 2000 Addenda; however, replacement thermowell designed to ASME III, Class 1, 1989 Edition no Addenda.

[4] Not including the hot leg piping portion between the flow meter inlet and SG inlet nozzles

[5] Portion between the flow meter inlet and SG inlet

[6] Material, fabrication, inspection, and testing is in accordance with 2001 Edition through 2003 Addenda. Design analyses are in accordance with 1992 Edition through 1993 Addenda except that the lesser of the material stresses were selected from the 1993 Addenda and 2003 Addenda

[7] Pressurizer Upper Heater Bundle (ECR 04-00375):

- N-405-1 applicable to heater element end plug to sheath weld.
- Code case 2142-2 for F-Number Grouping of Ni-Cr-Fe Filler Metals.
- Pressurizer upper bundle is designed to ASME Section III, 1986 Edition, No Addenda.
- Pressurizer middle and lower bundles are designed to ASME Section III, 1998 Edition with Addenda through 2000.

[8] ASME Section III, Nuclear Power Plant Components, Division 1, 2004 Edition, No Addenda and Code Case N-720-2, Full Structural Dissimilar Metal Weld Overlay for Repair or Mitigation of Class 1, 2, and 3 items, Section XI, Division 1, as modified by 10 CFR 50.55a and reconciled to the original USAS B31.7 February 1968 Draft Errata through June 1968 Code.

4.2 SYSTEM DESCRIPTION AND OPERATION

4.2.1 GENERAL

4.2.1.1 System

The RCS consists of the reactor vessel, two vertical once through steam generators, four shaft-sealed reactor coolant pumps, an electrically heated pressurizer, and interconnecting piping. The system is arranged in two heat transport loops, each with two reactor coolant pumps and one steam generator. The reactor coolant is transported through piping connecting the reactor vessel to the steam generators and flows downward through the steam generator tubes, transferring heat to steam and water on the shell side of the steam generator. In each loop, the reactor coolant is returned to the reactor through two lines, each containing a reactor coolant pump. In addition to serving as a heat transport medium, the coolant also serves as a neutron moderator and reflector and as a solvent for the soluble poison (boron in the form of boric acid). The RCS schematic is shown on Drawing C-302-650.

4.2.1.2 System Protection

a. Interacting Forces

BAW-1999, "TMI-1 Nuclear Power Plant Leak-Before-Break Evaluation of Margins Against Full Break for RCS Primary Piping," was submitted to the NRC in April 1987 (Reference 65). This topical report and its subsequent NRC safety evaluation (Reference 66) establish that LOCA dynamic effects may be excluded as a design basis using Leak-Before-Break (LBB). As part of performing the replacement of the TMI-1 steam generators and portion of the hot leg riser and the hot leg riser elbow, evaluations were performed which document that material usage and installation welding activities meet the acceptance criteria of the BAW-1999 topical report. Accordingly, UFSAR Section 4.2.1.2(a) text that describes support functions pertaining to LOCA load effects should be viewed historically in light of the NRC approval of the LBB design basis for TMI-1.

Several precautions have been taken in the design and the arrangement of piping inside the Reactor Building to minimize the possibility of interacting pipe failures (e.g., secondary accident). These precautionary measures assure adequate core cooling and Reactor Building integrity under assumed accidents. Reference 17 has been submitted to the NRC on the consequences of asymetric loading to the reactor, reactor support, etc. due to a LOCA. All main reactor coolant piping and components are arranged and/or restrained to minimize the possibility and consequences of an interacting failure involving the secondary system.

The secondary side of the steam generator is protected from jet reaction forces due to RCS failures in the area of the steam generator feedwater header.

Engineered safety features and associated systems are protected from missiles and jet forces which might result from a LOCA. Protection is provided by concrete shielding and/or segregation of redundant components.

The reactor vessel is surrounded by a concrete primary shield wall and the heat transport loops are surrounded by a concrete secondary shield wall. These shielding walls provide missile protection for the Reactor Building liner plate and equipment located outside the secondary shielding. Reactor coolant piping which passes through the primary shield has been restrained adequately to preclude pressure from exceeding the shield cavity design pressure due to a pipe rupture.

Removable shields over the reactor vessel also provide shielding and missile protection.

All steam and feedwater piping has been routed and/or restrained to minimize the possibility of an interacting rupture of RCS. All steam feedwater piping has been routed and/or restrained to minimize the possibility of an interacting rupture of other steam or feedwater piping in the same loop and to preclude an interacting rupture of the undamaged loop.

b. Seismic

The RCS is analyzed for maximum hypothetical earthquake to determine that resultant stresses do not jeopardize the safe shutdown of the reactor and removal of decay heat.

4.2.1.3 System Arrangement

The arrangement of the reactor coolant system is shown on Figures 4.2-2 and 4.2-3. Drawings IE-153-02-002 and IE-153-02-006 depict the system arrangement in relation to shielding walls, the Reactor Building, and other equipment in the building.

4.2.2 MAJOR COMPONENTS

4.2.2.1 Reactor Vessel

The reactor vessel consists of a cylindrical shell, cylindrical support skirt, spherically dished bottom head, and a ring flange to which a removable reactor closure head is bolted. The reactor closure head is a single piece, forged, spherically dished head and ring flange that is bolted to the reactor vessel flange. The reactor vessel general arrangement is shown on Figure 4.2-4. The general arrangement of the reactor vessel with internals is shown on Figures 3.2-44 and 3.2-45. Reactor vessel design data is listed in Table 4.2-1. The number and size of reactor vessel nozzles are also shown in Table 4.2-1. All coolant inlet, coolant outlet, core flooding, and control rod drive nozzles are located above the level of the top of the core. The reactor vessel is vented through the control rod drives.

The core flood nozzle contains a venturi type flow restrictor shown on Figure 3.2-46. The restrictor and attachment weld are designed in accordance with ASME Section III. The significant transients which affect the restrictor and weld are RCS heatup and cooldown including the core flooding system periodic test transient and decay heat removal initiation. All transients are considered as normal operating conditions and considered in determining thermal stresses and the fatigue usage factor. The fatigue analysis includes a strength reduction factor of two on the weld per ASME Section III. The weld has been designed to withstand a differential pressure of up to 2250 psi which may occur because of a core flood line LOCA. A dynamic magnification factor of two was applied to the pressure to account for instantaneous application. Based on these criteria, the average shear stress in the weld yields

a safety margin of 1.4. These assumptions and safety margin are sufficient to ensure the structural integrity of the nozzle, restrictor, and weld for all operating and faulted conditions. During core flooding transient, the maximum delta-P across the nozzle is expected to be approximately 200 psi. This is a factor of greater than 20 times less than the design loading assumptions. During operation of the decay heat system, the delta-P loads on the restrictor are insignificant.

The reactor closure head flange and the reactor vessel flange are joined by sixty 6 1/2 inch diameter studs. Two metallic o-rings seal the reactor vessel when the reactor closure head is bolted in place. Leakoff taps are provided in the annulus between the two o-rings to dispose of leakage. To insure uniform loading of the closure seal, the studs are hydraulically tensioned.

The reactor vessel contains the core support assembly, upper plenum assembly, fuel assemblies, control rod assemblies, axial power shaping rod assemblies, and incore instrumentation. Guide lugs welded to the inside of the reactor vessel wall limit the reactor internals and core to vertical drop of 1/2 inch or less and prevent rotation of the core internals about the vertical axis in the unlikely event of a major core barrel or core support shield failure. The reactor vessel internals are designed to direct the coolant flow, support the reactor core, and guide the control rods throughout their full stroke. The internals and the core are supported from the reactor vessel flange. The control rod drives mechanisms are supported by the nozzles in the reactor vessel head.

Surveillance specimens, made from appropriately selected specimens of reactor vessel steel, had been maintained in the surveillance capsule holder tubes located between the reactor vessel wall and thermal shield. In the original core, these specimens were located so as to afford the desired fast neutron exposure lead time with respect to the vessel wall, and were examined at appropriate intervals to evaluate reactor vessel material NDTT changes**.** Due to damage to the holder tubes the specimens were subsequently removed for further exposure of the capsules performed in the Crystal River reactor.

External reactor cavity dosimetry has been installed to measure vessel neutron fluence. Solid state track recorder cavity dosimetry hardware is installed in the source calibration tube located in the out-of-core instrumentation well which also houses NI-4 and NI-8 between the W and Z axis. The dosimetry equipment requires no power supply and has no interconnections with any plant systems. The hardware is classified Not-Important-To-Safety (other)/Non-Seismic. Section 4.4.5 provides further description of this surveillance program.

4.2.2.2 Steam Generator

The steam generator general arrangement is shown on Figure 4.2-5. Principal design data are tabulated in Table 4.2-2.

The original steam generators were replaced in 1R18 with OTSGs that are similar in performance and function to the original OTSGs. The original OTSGs were designed and licensed to operate with 20% of the tubes plugged. Similarly, the replacement OTSGs are designed to operate with 20% plugged tubes. The licensing basis for the replacement OTSGs, however, credits only 5% tube plugging. As such, a new safety analysis will be required if the number of tubes plugged in the replacement OTSGs exceed 5%.

The steam generator is a vertical, straight-tube and shell heat exchanger which produces superheated steam at constant pressure over the power range. Reactor coolant flows downward through the tubes and transfers heat to generate steam on the shell side. The high pressure (reactor coolant pressure) parts of the unit are the hemispherical heads, the tubes sheets, and the tubes between the tube sheets. The tubes are spaced and supported laterally by 15 stainless steel support plates. Tube support plates maintain the tubes in a uniform pattern along their length. The unit is supported by a skirt attached to the bottom head and lateral restraints at the top of the steam generator.

The shell, the outside of the tubes, and the tube sheet form the boundaries of the steam-producing section of the vessel. Within the shell, the tube bundle is surrounded by a cylindrical baffle. There are openings in the baffle at the feedwater inlet nozzle elevation to provide a path for steam to afford contact feedwater heating. The upper part of the annulus formed by the baffle plate and the shell is the superheat steam outlet, while the lower part is the feedwater inlet heating zone.

Vent, drain, instrumentation nozzles, inspection manways and handholes are provided on the shell side of the unit. The reactor coolant side has manway openings and handholes in both the top and bottom heads. Originally, to drain the steam generator reactor coolant from the bottom head, each OTSG included a drain nozzle and servicing piping. Because each Enhanced OTSG (EOTSG) has a flat bottom bowl configuration that requires no bottom bowl drain connection, the drain lines were removed from service with the replacement of the OTSGs with the EOTSGs. Venting of the reactor coolant side of the unit is accomplished by vent connection on the reactor coolant inlet pipe to each unit.

A secondary side blowdown system is provided for the OTSG to assist in controlling secondary side water chemistry. This system is described in Section 9.2.2.

The two reactor coolant outlet nozzles can be fitted with inflatable seal nozzle dams.
The purpose of the nozzle dams, which may be used only during an outage/refueling, is to prevent water from entering the steam generator lower head when the primary piping and refueling canal are full of water. Before installing the nozzle dams, the RCS cold legs must be drained to permit steam generator entry. Once in place, the RCS cold legs can be re-flooded and the RCS level increased to any point up to the highest allowed level for the Fuel Transfer Canal. The cold legs must be re-drained to permit removal of the nozzle dams. The nozzle dams are designed with a "wet" and a "dry" seal to provide redundancy and a passive seal to ensure that a loss of seal air supply does not result in a catastrophic failure of the nozzle dam. If air deflation to both wet and dry seals should occur, the nozzle dam will remain in place and limit the resulting leak rate to $\leq$ 2 gpm.

Emergency feed (condensate or river water) is supplied to the steam generator through an auxiliary feedwater ring located at the top of the steam generator to assure natural circulation of the reactor coolant following the unlikely event of the loss of all reactor coolant circulating pumps.

Four heat transfer regions exist in the steam generator as feedwater is converted to superheated steam. Starting with the feedwater inlet, these are as follows:

a. Feedwater Heating Region

Feedwater is heated to saturation temperature by direct contact heat exchange. The feedwater entering the unit is sprayed into the downcomer annulus formed by the shell and the cylindrical baffle around the tube bundle. Steam is drawn by aspiration through holes in the baffle into the downcomer and heats the feedwater to saturation temperature. The steam-water mixture flows down the downcomer annulus and through an adjustable orifice and enters the tube bundle.

The saturated water level in the downcomer provides a static head to balance the static head in the nucleate boiling section, the required head to overcome pressure drop in the circuit formed by the downcomer, the boiling section, and the bypass steam flow to the feedwater heating region. The downcomer water level varies with steam flow from approximately 15 to 100% unit load. A constant minimum level is held below approximately 15% unit load.

b. Nucleate Boiling Region

The saturated water enters the tube bundle just above the lower tube sheet and the steam-water mixture flows upward on the outside of the tubes countercurrent to the reactor coolant flow. The vapor content of the mixture increases almost uniformly until DNB is reached, and then film boiling and superheating occurs.

c. Film Boiling Region

Dry saturated steam is produced in the film boiling region of the tube bundle.

d. Superheated Steam Region

Saturated steam is raised to final temperature in the superheated region. Changes in temperature, pressure, and load conditions cause an adjustment in the length of the individual heat transfer regions and result in a change in the inventory requirements. The amount of surface available for superheat varies inversely with load. As load decreases, the superheat section gains surface from the nucleate and film boiling regions. Mass inventory in the steam generator increases with load as the length of the heat transfer region vary. If the inventory is greater than that required, the pressure increases. Inventory is controlled automatically as a function of load by the feedwater controls in the Integrated Control System.

Steam generator feedwater quality requirements are shown in Table 9.2-2.

4.2.2.3 Pressurizer

The pressurizer general arrangement is shown on Figure 4.2-7 and principal design data are tabulated in Table 4.2-3.

The electrically heated pressurizer establishes and maintains the RCS pressure within prescribed limits and provides a steam surge chamber and a water reserve to accommodate reactor coolant density changes during operation.

The pressurizer is a vertical cylindrical vessel with a bottom surge line penetration connected to the reactor coolant piping at the reactor outlet. The pressurizer contains removable electric

heaters in its lower section and a water spray nozzle in its upper section to maintain RCS pressure within desired limits. The pressurizer vessel is protected from thermal effects by a thermal sleeve in the surge line and by an internal diffuser located above the surge pipe entrance to the vessel.

During outsurges, as RCS pressure decreases, some of the pressurizer water flashes to steam, thus assisting in maintaining the existing pressure. Heaters are then actuated to restore the normal operating pressure. During insurges, as system pressure increases, water from the reactor vessel inlet piping is sprayed into the steam space to condense steam and reduce pressure. Spray flow and heaters are controlled by an RCS pressure controller.

Since all sources of heat in the system-core, pressurizer heaters, and reactor coolant pumps interconnected by reactor coolant piping with no intervening isolation valves, relief protection is provided on the pressurizer. Overpressure protection consists of two code safety valves and one pilot (electromatic) operated relief valve (PORV) (See Section 4.2.4).

To eliminate abnormal buildup or dilution of boric acid within the pressurizer, and to minimize the cooldown of the coolant in the spray and surge lines, a bypass flow is provided around the pressurizer spray control valve. This continuously circulates approximately 1 gpm of reactor coolant from the heat transport loop. A sampling connection to the liquid volume of the pressurizer is provided for auditing boric acid concentration. A steam space sampling line provides capability for monitoring of or venting accumulated gases. During cooldown and after the decay heat system is placed in service, the pressurizer can be cooled by circulating water through a connection from the discharge of the low pressure injection pump to the pressurizer auxiliary spray line. Refer to Section 9.5 for more information.

Electroslag welding was utilized in the fabrication of the pressurizer, only in the longitudinal seams of the shell courses. A total of three individual electroslag welds were made in fabrication of each pressurizer. The electroslag welding process and quality control was the same as described in Subsection 4.2.2.2.

A potential high energy line break (HELB) of the ten (10) inch pressurizer surge line has been evaluated for potential impact on NSCCW piping inside the D-ring. The only portion of the NSCCW piping that could be impacted is the piping leading to reactor coolant pumps RC-P-1A and RC-P-1B. The results of the evaluation concluded:

a. The Surge Line has a whip restraint installed near its attachment to the hot leg, therefore, from this end of the Surge Line, only jet impingement is a concern. A detailed review of the jet indicates that the jet cone would not reach the NSCCW piping, assuming a jet cone half angle of 15º.

b. At the Pressurizer end of the Surge Line there are no whip restraints, therefore, if a HELB occurred, the pipe would whip around the "D" Ring with its initial movement downward, applying a jet directly to the Pressurizer and nowhere else. Therefore, pipe whip and jet impingement from this terminal end would not effect the NSCCW piping leading to Reactor Coolant Pump RC-P-1A.

c. For intermediate longitudinal pipe breaks in the Surge Line, the only break of this type which would affect the NSCCW piping would be a longitudinal break of 1 X pipe area, on top of the pipe, close to the pressurizer terminal end. A jet from

this break would impact the NSCCW piping leading to pump RC-P1-A. Therefore, a jet impingement analysis of the piping was performed. The results indicate that the NSCCW piping will not rupture during the HELB event described above. This analysis was performed in accordance with the criteria described in FSAR Section 5.2.5.3. The JET FORCE (ENTHALPY) was assumed to remain constant over the linear distance to the NSCCW line. The area of jet impingement was conservatively determined by assuming a 15º half angle for jet cone expansion.

4.2.2.4 Reactor Coolant Piping

The general arrangement of the reactor coolant piping is shown on Figures 4.2-2 and 4.2-3. Principal design data are tabulated in Table 4.2-4.

The major piping components in this system are the 28 inch inside diameter cold leg piping from the steam generator to the reactor vessel and the 36 inch inside diameter hot leg piping from the reactor vessel to the steam generator. (The straight sections of the 28 inch inside diameter and 36 inch inside diameter piping are hollow-forged.) Also included in this system are the 10 inch surge line and the 2 1/2 inch spray line to the pressurizer. The system piping incorporates the auxiliary system connections necessary for operation. In addition to drains, vents, pressure taps, injection, and temperature element connections, there is a flow meter section in 36 inch line to the steam generator to provide a means for determining the flow in each loop.

The 28 inch and 36 inch piping is carbon steel clad with austenitic stainless steel. Short sections of stainless steel transition piping are provided between the pump casing and the 28 inch carbon steel lines. Stainless steel or Inconel safe-ends are provided for field welding the nozzle connections to smaller piping. The piping safe- ends are designed so that there will not be any furnace sensitized stainless steel in the pressure boundary material. This is accomplished either by installing stainless steel safe-ends after stress relief or by using Inconel. Smaller piping, including the pressurizer surge and spray lines, is austenitic stainless steel. All piping connections in the RCS were butt-welded except for the flanged connections on the pressurizer for the relief valves.

Thermal sleeves are installed where required to limit the thermal stresses developed because of rapid changes in fluid temperatures. They are provided in the following nozzles: the four high pressure injection nozzles on the reactor inlet pipes and spray line nozzle on the pressurizer.

4.2.2.5 Reactor Coolant Pumps

Each reactor coolant loop contains two vertical single stage centrifugal type pumps which employ controlled leakage seal assembly. A cutaway view of the pump is shown on Figure 4.2-8 and the principal design parameters for the pumps are listed in Table 4.2-5. The estimated reactor coolant pump performance characteristic is shown on Figure 4.2-9.

Reactor coolant is pumped by the impeller attached to the bottom of the rotor shaft. The coolant is drawn up through the bottom of the impeller, discharged through passages in the guide vanes and out through a discharge in the side of the casing. The pump impeller can be removed from the casing for maintenance or inspection without removing the casing from the

piping. All parts of the pumps in contact with the reactor coolant are constructed of austenitic stainless steel or equivalent corrosion resistant materials. A list of pressure containing materials is given in Table 4.2-6.

The pump employs a controlled leakage seal assembly to restrict leakage along the pump shaft, as well as provide a path for water to flow past the pump radial bearing to maintain radial bearing cooling. The seal assembly has three (3) redundant stages that can each handle full RCS pressure. The seal assembly is also equipped with an abeyance seal that will actuate and limit leakage from the seal assembly in a scenario where the mechanical seals have failed.

A portion of the high pressure water flow from the makeup pumps is injected into the reactor coolant pump between the impeller and the controlled leakage seal. Part of the flow enters the RCS through a labyrinth seal on the lower pump shaft to serve as a buffer to keep reactor coolant from entering the upper portion of the pump. The remainder of the injection water flows along the drive shaft, through the controlled leakage seal, and finally out of the pump via the seal leak off lines. The majority of the seal leakage is returned to the makeup tank through the Controlled Bleed Off (CBO) line, while a small amount of seal leakage goes to the RCDT.

Intermediate cooling water is supplied to the thermal barrier cooling coil.

4.2.2.6 Reactor Coolant Pump Motor Flywheel

4.2.2.6.1 Reactor Coolant Pump Motor Flywheel RC-P-1A/B/D

The reactor coolant pump motors are large, vertical, squirrel cage, induction machines. Each is equipped with an antireverse device to prevent backward rotation and reduce motor starting time. The motors have flywheels to increase the rotational inertia, thus prolonging pump coastdown and assuring a more gradual loss of main coolant flow to the core in the event pump power is lost. Two flywheel assemblies are provided: a large assembly at the upper end of the rotor and a smaller flywheel at the lower end of the motor. The upper flywheel is a three-piece assembly consisting of two segments approximately the same diameter as the rotor and a third segment with a diameter of 72 inches. The flywheels are made from ASTM Specification A-516-66 Grade 65 steel plate which is a grade used in pressure vessels. The plate was normalized to produce grain refinement and to improve the toughness. Chemical analysis and physical tests to determine the physical properties including yield strength, tensile strength, and elongation were performed and certified by the supplier.

a. Preservice Tests/Inspection of the Reactor Coolant Pump Motor Flywheels

   Each of the flywheels of the reactor coolant pump motors was ultrasonically inspected, after final machining but before final assembly. Both straight and angle beam techniques were used to detect any material flaws (radial flaws were of primary concern). The acceptance standard was based on a flaw size considerably smaller than the critical flaw size.

   The finished bores were subjected to magnetic particle inspection to detect surface or slightly subsurface discontinuities. The acceptance standards imposed in the magnetic particle inspection were the same criteria imposed by the ASME Boiler & Pressure Vessel Code.

Upon assembly with the motor rotor, the flywheels were reinspected using a procedure specially developed and qualified for the limited accessibility of the flywheels for inspection. Inspection ports in the motor casing and cooling baffles provide access to the upper face and rim of the upper flywheel and to the lower face of the lower flywheel. Special calibration holes were drilled into two flywheels to facilitate test equipment calibration and qualify the procedure.

The assembled motor flywheel baseline data were taken using this procedure. The preservice baseline inspection consisted of a full inspection of the upper and lower flywheels to the extent possible in both the axial and radial direction through the access ports.

b. Inservice Inspection of the Reactor Coolant Pump Motor Flywheels

An Inservice Inspection Program has been developed to satisfy the technical specification requirements for inservice inspection and the frequency of the inspection of the reactor coolant pump motor flywheels. It is based on suspected flaw orientation and locations.

c. Critical Flaw Size for Reactor Coolant Pump Motor Flywheels

A through-wall flaw size of 6 inches originating at the bore has been calculated to be the critical size necessary to cause failure of the flywheel. Based on flaw growth analysis, there must be present a through-wall flaw of 5.4 inches at beginning- of-life in order to propagate to the critical size of 6 inches after 500 starts. This calculation is based on the method provided in Reference 10 and material physical properties taken from Reference 11.

4.2.2.6.2 Reactor Coolant Pump Motor Flywheel RC-P-1C

The reactor coolant pump motors are large, vertical, squirrel cage, induction machines. RC-P-1C is equipped with an antireverse device to prevent backward rotation and reduce motor starting time. This motor has one flywheel to increase the rotational inertia, thus prolonging pump coastdown and assuring a more gradual loss of main coolant flow to the core in the event pump power is lost. One flywheel assembly is provided: a large assembly at the upper end of the rotor. The flywheel is a single one-piece disc made from ASTM A533 Type B Class 1 steel plate. (Reference AREVA document # 32-9124760).
Chemical analysis and physical tests (Ultrasonic inspection and Dye Penetrant Examination) to determine the physical properties including yield strength, tensile strength and elongation were performed and certified by the supplier.

a. Preservice Tests/Inspection of the Reactor Coolant Pump Motor Flywheel

The flywheel of the reactor coolant pump motor was ultrasonically inspected after final machining but before final assembly. The acceptance standard was based on a flaw size considerably smaller than the critical flaw size.

Upon assembly with the motor rotor a baseline test was performed using a procedure specially developed and qualified for the limited accessibility of the flywheel for inspection.

The assembled motor flywheel baseline data were taken using this procedure. The preservice baseline inspection consisted of a full inspection of the flywheel to the extent possible in both the axial and radial direction through the access ports.

b. Inservice Inspection of the Reactor Coolant Pump Motor Flywheels

An Inservice Inspection Program has been developed to satisfy the technical specification requirements for inservice inspection and the frequency of the inspection of the reactor coolant pump motor flywheel. It is based on suspected flaw orientation and locations.

c. Critical Flaw Size for Reactor Coolant Pump Motor Flywheel

The AREVA RCP Motor flywheel critical flaw size has been calculated for three operating speeds, normal operation, design overspeed (1.25 x nominal speed), and critical speed (2 x nominal speed).

In all cases, the critical flaw size of a radially oriented crack in the flywheel which could cause catastrophic fracture is far bigger than the minimal defect that can be detected (Refer AREVA Document 32-9156246)

4.2.2.7 Reactor Coolant Equipment Insulation

The reactor coolant system components are insulated with stainless steel metal reflective type insulation (MRI), with the exception of the pressurizer which has both metal reflective insulation and NUKON™ flexible fiberglass blanket insulation. The insulation attached to the steam generators, their affected connecting piping, and the pressurizer is supported by the use of friction rings, and seismically qualified in order that the insulation remains intact during a seismic event to prevent damage to adjacent Nuclear Safety Related Systems, Structures, or Components (SSCs).

The insulation units are removable and are designed for ease of removal and installation in such areas as field welds, nozzles, and bolted closures. The insulation units permit free drainage of any condensate or moisture from within the insulation unit.

(Historical): The original reactor coolant system components were insulated entirely with stainless steel MRI. The entire pressurizer and steam generator upper heads were reinsulated during 6R (1986-87) and 7R (1986), respectively, with NUKON ™ flexible fiberglass blanket insulation. New replacement insulation consisting entirely of MRI was installed to the Enhanced OTSGs (EOTSGs) installed in the Fall of 2009 and to the upper cylindrical section of the pressurizer, above the support lugs and below the upper head, in the Fall of 2011.

4.2.2.8 Valves

Valves in the RCS are fabricated from either castings or forgings. The types of non-destructive testing to which the major pressure containing parts of a valve are subjected depend on whether it is fabricated from a casting or a forging and are shown in Table 4.2-7. In addition, the PORV main body assembly weld has been radiographed.

4.2.3 SYSTEM PARAMETERS

4.2.3.1 Flow

The RCS was originally designed on the basis of 176,000 gpm flow rate in each heat transport loop. Since actual RCS flow was 108% of the system design flow in cycle one, a minimum required flow rate of 106.5% of original system design flow has been used in all subsequent core thermal hydraulic analysis. RC system minimum required operating flow rate is provided on Tech Spec Figure 2.1-3.

4.2.3.2 Temperatures

RCS temperatures as a function of power are shown on Figure 4.2-10. The system is controlled to a constant average temperature throughout the power range for 15 percent to 100 percent full power. The average system temperature is decreased between 15 percent and 0 percent of full power to the saturation temperature (532ºF) at 900 psia. Limited temperature reductions near end of cycle life are sometimes performed to extend 100 percent full power operation.

4.2.3.3 Heatup

All RCS components are structurally designed for a continuous heatup rate of 100ºF/hr during transient cycles described in Table 4.1-1.

4.2.3.4 Cooldown

All RCS components are structurally designed for a continuous cooldown rate of 100ºF/hr. System cooldown to 250ºF is accomplished by use of the steam generators and by bypassing steam to the condenser with the turbine bypass system. The decay heat removal system provides the heat removal for system cooldown below 250ºF.

4.2.3.5 Volume Control

a. Letdown

The only coolant removed from the RCS is that which is letdown to the makeup and purification system. The letdown flow rate is set at the desired rate by the operator positioning the letdown control valve and/or opening the stop valve for the letdown orifice.

b. Makeup

To maintain a constant pressurizer water level, total makeup to the RCS must equal that which is letdown from the system. Total makeup consists of the seal injection water through the reactor coolant pump shaft seals and makeup returned to the system through the reactor coolant volume control valve. The pressurizer level controller provides automatic control of the valve to maintain the desired pressurizer water level. Reactor coolant volume changes during plant load changes may exceed the capability of the reactor coolant volume control valve, and thus result in variations in pressurizer level. The level is returned to normal as the system returns to steady-state conditions.

The primary method of monitoring the inventory of the RCS is the pressurizer water level measurements. Any excessive makeup or deficient makeup would result in a change in the pressurizer water level. During normal operation, the pressurizer level is automatically maintained at approximately 18 ft (216 inches). One of two independent pressurizer level signals is used for operator indication and automatic makeup control. If the normal level control system fails, there are four pressurizer level alarms (two high level and two low level) which would be actuated by any one of the two independent sensors.

The pressurizer normally contains 800 ft^3 (5,984 gallons) of water and 702 ft^3 (5,246 gallons) of steam for a total volume of 1502 ft^3 (11,230 gallons). The addition of 970 gallons of water to the RCS (pressurizer) would produce the first pressurizer high-level alarm and the addition of 2280 gallons of water to the RCS (pressurizer) would produce another high-level alarm. The first low-level alarm would be produced by the removal of 460 gallons of water at 200 inches, and the second low-level alarm would be produced by the removal of 3360 gallons of water, corresponding to 80 inches.

The redundancy of pressurizer level transmitters in combination with independent computer analog alarms and indication provide adequate information concerning pressurizer water level (RCS inventory), and therefore excessive makeup or deficient makeup would not go undetected.

4.2.3.6 Chemical Control

Control of reactor coolant chemistry is a function of the chemical addition and sampling system. Sampling lines from the letdown line of the makeup and purification system provide samples of the reactor coolant for chemical analysis. All chemical addition is made from the chemical addition and sampling system to makeup and purification system. See Chapter 9 for detailed information concerning the chemical addition and sampling system and makeup and purification system. RC quality is in Table 9.2-3.

a. Boron

Boron in the form of boric acid is used as a soluble poison in the reactor coolant. Concentrated boric acid is stored in the boric acid mix tank and is transported to the RCS in the same manner as described above for chemical addition. A second source of stored concentrated boric acid is that which is reclaimed from the coolant and stored in the reclaimed boric acid tanks. This reclaimed boric acid is pumped to the makeup and purification system which transports it to the RCS. All bleed and feed operations for changing boric acid concentration of the reactor coolant are made between the makeup and purification system and the liquid waste system. Chapters 9 and 11 contain detailed information concerning these two systems.

b. pH

The pH of the reactor coolant is controlled to minimize corrosion of the RCS surfaces which minimizes coolant activity and radiation levels of the components (see Table 9.2-3).

c. Water Quality

The reactor coolant water chemistry specifications, listed in Table 9.2-3, have been selected to provide the necessary boron content for reactivity control and to minimize corrosion of RCS surfaces. The solids content of the reactor coolant is maintained below the design level by minimizing corrosion through chemistry control and by continuous purification of the letdown stream of reactor coolant in the letdown filter and purification demineralizer of the makeup and purification system. Excess hydrogen is maintained in the reactor coolant to chemically combine with the oxygen produced by radiolysis of the water.

4.2.3.7 Flow Measurement

Reactor coolant flow rate is measured for each heat transport loop by a flow tube welded into the reactor outlet pipe. The power/flow monitor of the reactor protective system utilizes this flow measurement to prevent reactor power from exceeding a permissible level for the measured flow. This is discussed in further detail in Section 7.3.2.

4.2.3.8 Leak Detection

The entire RCS is located within the secondary shielding and is inaccessible during reactor operation. Any normally maintained leakage drains to the Reactor Building sump. Any coolant leakage to the atmosphere will be in the form of fluid and vapor. The fluid will drain to the sump and the vapor will be condensed in the Reactor Building coolers and also reach the sump via a cooler drain line.

Most RCS power operated valves have their leakoff connections capped. However, some power-operated valves containing reactor coolant, located in the Reactor Building, have two sets of stem packing, with a leakoff connection between the packing sets. Leakoff from valve packing is piped to the RC drain tank.

For the reactor coolant pumps, leakage past the first and second stages of the mechanical seal is returned to the makeup tank. Leakage past the third stage of the mechanical seal is piped to the reactor coolant drain tank. An additional flow path is available for the third stage seal leakoff to over flow through a loop seal to the Reactor Building sump.

Locating the actual point of RCS leakage can most readily be accomplished when the reactor is shut down, thereby allowing personnel access inside the secondary shielding. Leaks can be located by visual observation of escaping steam or water, or of the presence of boric acid crystals deposited near the leak site by evaporation of the leaking coolant.

If reactor coolant leakage is to the containment, it may be identified by one or more of the following methods:

a) The containment radiation monitor is a three channel monitor consisting of a particulate channel, an iodine channel, and a gaseous channel. All three channels read out in the Control Room and alarm to indicate an increase in containment activity.

   The containment particulate channel is sensitive to the presence of Rb-88, a daughter product of Kr-88, in the containment air sample. Since this activity originates predominantly in the Reactor Coolant System, an increase in monitor readings could be

indicative of increasing RCS leakage. The sensitivity of the particular monitor is such that a leakrate of less than 1 GPM will be detected within one hour under normal plant operating conditions.

b) Deleted.

c) The mass balance technique is a method of determining leakage by stabilizing the Reactor Coolant System and observing the change in water inventory over a given time period. Level decreases in the makeup tank may also serve as an early indication of abnormal leakage.

4.2.3.9 Reactor Coolant System Venting

The function of the reactor coolant venting system is to permit venting of gases trapped in the reactor vessel head and in both hot legs and also to provide pressurizer degassing capability from a remote location when postaccident radiation and contamination would not permit access to systems inside the containment. Drawing C-302-650 schematically shows the system layout. The pressurizer vent, the hot legs vent, and the reactor vessel head vent are installed.

Vent piping and valves are designed to the same conditions as the RCS. The lines are sized such that the failure to completely close off any one vent path will not allow a loss of reactor coolant at a rate in excess of the normal capability of the makeup system at full RCS design pressure.

Small break loss of coolant accidents (LOCAs) can lead to RCS depressurization in which steam and/or non-condensable gases may accumulate in the reactor vessel head, the upper portion of the hot legs, and in the pressurizer. Following repressurization of the RCS by high pressure injection (HPI), the steam bubbles collapse and remotely controlled vents on the upper hot legs and pressurizer can be used to vent non-condensable gases to promote water solid natural circulation for core cooling.

The function of a reactor coolant venting system is to permit venting, from a remote location, of gases trapped at high points in the reactor coolant system when postaccident radiation and contamination levels will not permit access to systems inside the containment.

The hydrogen generation design basis for the system will be loss of coolant accident (LOCA) with hydrogen generation rates calculated in accordance with NRC Regulatory Guide 1.7.

The system is capable of venting a volume of non-condensable gas equivalent to one half of the reactor coolant system volume in one hour.* The system in performing its design function will not degrade nor defeat any features of the existing reactor coolant system. The remote operated vent valves are solenoid operated (except for the normal "degas" valve from the pressurizer to the RCDT which is motor operated). The power supplies and any instrumentation for the vent valve operators are Class 1E and from on-site power sources.

This vent rate does not apply to the pressurizer since the pressurizer is vented to the RCDT and a rapid vent rate poses the risk of blowing the RCDT ruptured disc and subsequent hydrogen burns flash explosions. The vent from the pressurizer will be operated at a slow

'degassing' vent. Subsequently, the RCDT may be vented to the reactor building atmosphere through valves WDG-V134 and/or WDG-V135 (See Drawing 302694).

Based on the pipe size selected and/or suitable restrictions, the vent piping is designed to preclude challenges to the high-pressure injection function of the Makeup and Purification system. The vent lines are sized so that an inadvertent opening of a pair of vent valves in a single vent line will not result in out-flow greater than the makeup capacity of a Makeup Pump. On this basis a LOCA analysis was not required.

Two remotely actuated valves in series in each vent line provide RCS integrity. Each valve is supplied with a control switch and a position indicating light. FSAR Section 7.3.2.2.C provides a detailed discussion of reactor coolant system vent controls and instrumentation.

The power-operated vent from the pressurizer to the reactor coolant drain tank provides remote venting capability for the pressurizer during normal operation and following loss of coolant accidents. Use of the pressurizer vent during a postaccident condition provides degassing capability for the RCS, maintains the pressurizer level as an accurate indication of RCS inventory, and maintains system pressure control.

The new solenoid operated valve in the vent line from the pressurizer to the reactor coolant drain tank has a keylock operating switch. When the switch is turned off or when power to the solenoid valve is lost, the valve closes.

The pressurizer vent valve (RC-V-28) is not assumed to operate to mitigate any design basis accident analyses. The pressurizer vent is referenced in plant emergency procedures and is preferred over the PORV for venting because of the smaller capacity and more gradual effect on RCS pressure. The pressurizer vent is used to minimize subcooling margin during an OTSG tube rupture event and following an excessive primary-to-secondary heat transfer event. This use of the pressurizer vent only subjects the equipment inside containment to higher temperature and pressure for a short duration without creating a harsh radiation environment. The pressurizer vent is also used during RCS superheat conditions (cladding temperature of 1400°F or higher) to remove non-condensible gases.

The pressurizer vent valve is environmentally qualified in accordance with the requirements of NUREG 0737, Item II.B.1 and the May 23, 1980 Commission Order and Memorandum. The pressurizer vent valve and other RCS high point vents are not within the scope of 10CFR50.49.

Inservice inspection is as per the requirements of the ASME B&PV Code, Section XI, for system design and inspection. Technical Specifications provide periodic test and surveillance requirements to ensure operability of the reactor coolant system vents.

4.2.3.10 Reactor Coolant System Drains

Drain lines are shown on the system diagram, Drawing C-302-650. They are located at the low points of the RCS and provide the means for draining the heat transport loops and pressurizer. The reactor vessel cannot be drained below the reactor outlet nozzle. Each line contains two manual valves in series. Drain lines are routed to a header connected to the suction of the reactor drain pump.

4.2.4 PRESSURE CONTROL AND PROTECTION

Normal RCS pressure control is by the pressurizer steam cushion in conjunction with the pressurizer spray, PORV, and heaters. The system is protected against overpressure by reactor protection system circuits such as the high pressure trip and by pressurizer relief valves located at the top head of the pressurizer. The schematic arrangement of the relief valves is shown on Drawing C-302-650. Since all sources of heat in the system, i.e., core, reactor coolant pumps, and pressurizer heaters, are interconnected by the reactor coolant piping with no intervening isolation valves, all relief valves are located on the pressurizer. RCS pressure settings and relief valve capacities are listed in Table 4.2-8.

4.2.4.1 Pressurizer Code Safety Valves

Two pressurizer code safety valves are mounted on individual nozzles on the top head of the pressurizer. The valves have a closed bonnet with bellows and supplementary balancing piston. The valve inlet and outlet is flanged to facilitate removal for maintenance or set point testing. Refer to Section 4.3.7 and Table 4.2-8.

4.2.4.2 Pilot (Electromatic) Operated Relief Valve (PORV)

The PORV is mounted on a separate nozzle on the top head of the pressurizer. The main valve operation is controlled by the opening or closing of a pilot valve which causes unbalanced forces to exist on the main valve disc. The pilot valve is opened or closed by a solenoid in response to the opening and closing pressure set points. Flanged inlet and outlet connections provide ease of removal for maintenance purposes.

The PORV (RC-RV2) normally remains in closed position except when pressure in the RCS exceeds set point, as sensed by a selected pressure transmitter. The operator may manually select one of two pressure transmitters to control RCS pressure. The console switch allows the operator to have the option of manually opening the PORV to vent RCS pressure or to allow automatic venting of pressure.

Manual venting RCS pressure would permit anticipatory reduction of pressure following an event which would normally result in high RC pressure and thus avoid lifting the code relief valves. In the event of the failure of the PORV to close, the block valve can be remote manually operated from the Control Room to close this path.

The Control Room operator is provided with the status of the PORV and the pressurizer code safety valves (RC-RV1A and RC-RV1B). Discharge flow is measured by differential pressure transmitters connected across elbow taps downstream of each of the valves. In addition, the pilot operated relief valve is monitored by accelerometers mounted on the valve. These detect flow if the valve opens. Alarms and indications are provided in the Control Room to inform the operator if any of these valves are open.

A reliable and unambiguous control grade indication is provided to the Control Room operator if the pressurizer PORV or code safety valves open. The monitoring system will remain functional in containment conditions associated with any transient for which valve status is required by the operator. Redundant and diverse means are provided for monitoring the PORV. The monitoring systems will remain functional during a loss of offsite power. All equipment inside containment is seismically mounted.

In response to NUREG 0737, Item II.K.3.1 - Automatic PORV Isolation, it was concluded that the requirements of II.K.3.1 and II.K.3.2 are met with the existing PORV safety valve and reactor high-pressure trip setpoints and that an automatic PORV isolation system to protect against a small break LOCA caused by a stuck open PORV is not required for TMI-1 (Reference 35).

4.2.4.3 Pressurizer Spray

The pressurizer spray line originates at the discharge of a reactor coolant pump in the same heat transport loop that contains the pressurizer. Pressurizer spray flow is controlled by an electric motor operated valve using on/off control in response to opening and closing pressure set points. An electric motor operated valve in series with the spray valve provides a backup means of securing flow in the event the spray valve should stick open.

4.2.4.4 Pressurizer Heaters

The pressurizer heaters replace heat lost during normal steady state operation, raise the pressure to normal operating pressures during RCS heatup from a cooled down condition, and restore system pressure following transients. The heaters are arranged in thirteen (13) groups and are controlled by the pressure controller. The first six (6) operational groups utilize modulating control and will normally operate at partial capacity to replace heat lost, thus maintaining pressure at setpoint within a reasonable margin of difference. A basic "on" or "off" control is used for the remaining seven (7) groups. A low-level interlock prevents energizing of the heaters while they are uncovered.

Redundant emergency power for 107KW of pressurizer heaters is required to maintain natural circulation conditions in the event of a loss of offsite power. Refer to Section 8.2.3.5 for a detailed description of the power system.

4.2.4.5 Relief Valve Effluent

Effluent from the pressurizer PORV and code safety valves discharges to and is collected by the reactor coolant drain tank. This is shown schematically on Drawing C-302-650. After the reactor coolant drain tank receives relief or safety valve effluent, the tank contents are cooled by circulating the fluid from the tank through the reactor coolant drain tank heat exchanger and returned to the tank by spraying into the tank vapor space. The head created by the reactor coolant drain tank pump causes flow through the heat exchanger and the spray. The reactor coolant drain tank is protected against overpressure by a rupture disc sized for the total combined relief capacity of the two pressurizer code safety valves and the pressurizer PORV. The reactor coolant drain tank is vented to the gaseous waste disposal system (Chapter 11 see Table 11.2-1).

4.2.4.6 Cooldown

Pressure reduction during RCS cooldown is accomplished by the pressurizer spray supplied from the reactor vessel inlet piping. Below a system temperature of approximately 250ºF, the decay heat removal system is used for system heat removal, the steam generators and reactor coolant pumps are removed from service. During this period, spray flow is provided by a branch line from decay heat removal line to the pressurizer spray line for further pressure reduction or complete depressurization of the RCS.

4.2.4.7 Sampling

A sample line from the pressurizer steam space to chemical addition and sampling systems permits detection of non-condensable gases in steam space. This sample line also permits bleeding operation from the vapor space to the letdown line of the makeup and purification system to transport accumulated non-condensable gases in the pressurizer to the makeup tank.

4.2.5 INTERCONNECTED SYSTEMS

4.2.5.1 Decay Heat Removal

The decay heat removal system provides the capability for cooling the RCS below about 250F during plant cooldown. During this mode of operation, coolant is drawn from the RCS through a nozzle on the reactor outlet pipe, circulated through the decay heat removal coolers by the decay heat removal pumps, and then injected into the RCS through two core flood nozzles on the reactor vessel into inlet side of the core. The heat received by this system is rejected to decay heat cooling water system. Components in these two systems are redundant for reliability purposes. These two systems are described in Chapter 9.

The decay heat removal system also performs an emergency injection function for a LOCA and provides long term emergency core cooling; this is described in Chapter 6.

4.2.5.2 Makeup and Purification

The makeup and purification system controls RCS coolant inventory changes resulting from temperature-caused water density changes during normal reactor operation, recirculates RCS inventory through letdown for water quality maintenance and reactor coolant boric acid concentration control, provides seal water to the reactor coolant pumps, supplies borated makeup water to Core Flood Tanks, and emergency high pressure injection into the RCS in the event of a LOCA. Installed components are redundant for reliability purposes (Section 9.1).

Makeup and emergency high pressure injection flow from the makeup pump(s), enters the RCS through a nozzle in each reactor inlet pipe (coldleg) located downstream of the reactor coolant pumps. Water from the makeup tank and/or Borated Water Storage Tank (BWST) is used for makeup and high pressure injection. The high pressure injection feature which operates after a loss of coolant accident (LOCA) is described in Section 6.1.2.1.

Letdown from the RCS is through a nozzle on the outlet pipe from one steam generator. The letdown reactor coolant temperature is reduced in a cooler. Impurities other than Boron are removed in the purification demineralizer. The process letdown fluid is then returned to the makeup tank.

4.2.5.3 Core Flooding System

The core flooding system recovers the core at intermediate-to-low RCS pressures so as to maintain core integrity in the event of a LOCA. Connection to the reactor vessel is through the two nozzles described above for low-pressure injection. The decay heat removal and core

flooding lines tie together and connect to the same nozzle on the reactor vessel. The core flooding system is described in item c of Section 6.1.2.1.

4.2.5.4 Secondary System

The principal decay heat removal system interconnected with RCS is the steam and power conversion system. The RCS is dependent upon the steam and power conversion system for decay heat removal at normal operating conditions and for all reactor coolant operating temperatures above 250F. The system is discussed in detail in Section 10.0.

The turbine bypass system routes steam to the condensers when the turbine has tripped or is shutdown and also during large plant load reduction transients when steam generation exceeds the demand. Overpressure protection for the secondary side of the steam generators is provided by the Main Steam Safety Valves (100 percent of total), the Turbine Bypass Valves (approximately 33% of steam flow at 100% percent reactor power), and Atmospheric Dump Values (approximately 8.8% of steam flow at 100% reactor power). The emergency feedwater system will supply water to the steam generators in the event that the main feedwater system is inoperative. The physical layout of the RCS provides natural circulation of the reactor coolant to ensure adequate core cooling following a loss of all reactor coolant pumps.

4.2.6 COMPONENT FOUNDATIONS AND SUPPORTS

4.2.6.1 Reactor Vessel

The reactor vessel is bolted to reinforced concrete foundation designed to support and position the vessel and withstand the forces imposed on it by a combination of loads, including the weight of vessel and internals, thermal expansion of the piping, design basis earthquake, and dynamic load following reactor trip.

The foundation, in addition, restrains the vessel during the combined forces imposed by circumferential rupture of a 36 inch reactor outlet line and a simultaneous maximum hypothetical earthquake. The vessel foundation is also designed to provide accessibility for the installation and later inspection of incore instrumentation, piping and nozzles, to contain ductwork and vent space for cooling air, to remove heat losses from the vessel insulation, and, to provide a drainage line for leak detection.

4.2.6.2 Pressurizer

The pressurizer is supported on a structural steel foundation by eight (8) lugs welded to the side of the vessel.

The foundation and supports are designed to withstand loads imposed by thermal expansion of the pressurizer, the weight of the pressurizer, including its contents and attached piping, relief valve reaction forces, and forces imposed by the design basis earthquake. In addition, foundation and supports will restrain the vessel during the combined forces imposed by the circumferential rupture of the 10 inch surge line coupled with the maximum hypothetical earthquake.

The foundation is designed to permit accessibility to pressurizer surfaces for inspection.

4.2.6.3 Steam Generator

The steam generator foundation is designed to support and position the generator. The foundation is designed to accept the loads imposed by the generators the attached reactor coolant and feedwater piping filled with water, and steam lines under normal operation conditions.

Forces imposed on the generator by a design basis high energy line break, by the design basis earthquake, or by the maximum hypothetical earthquake are transferred to the shielding walls by restraints located near the top of the generator.

4.2.6.4 Piping

The reactor coolant piping, inlet and outlet lines, are supported by the reactor vessel and steam generator nozzles. The piping will withstand the forces imposed on it by the design basis earthquake and the maximum hypothetical earthquake.

4.2.6.5 Pump and Motor

The reactor coolant pump casing, internals, and motor weight are supported by the 28 inch coolant lines and constant load hangers attached to the motor. In the cold condition, the coolant piping will support the coolant pump and motor without the hangers.

4.2.6.6 LOCA Load Effect

BAW-1999, "TMI-1 Nuclear Power Plant Leak-Before-Break Evaluation of Margins Against Full Break for RCS Primary Piping," was submitted to the NRC in April 1987 (Reference 65). This topical report and its subsequent NRC safety evaluation (Reference 66) establish that LOCA dynamic effects may be excluded as a design basis using Leak-Before-Break (LBB). As part of performing the replacement of the TMI-1 steam generators and portion of the hot leg riser and the hot leg riser elbow, evaluations were performed which document that material usage and installation welding activities meet the acceptance criteria of the BAW-1999 topical report. Accordingly, UFSAR Section 4.2.6.6 text that describes support functions pertaining to LOCA load effects should be viewed historically in light of the NRC approval of the LBB design basis for TMI-1.

Each steam generator has restraints located at the elevation of the upper tube sheet which transfer forces from the generator into shield walls in the event of a design basis high energy line break.

Each reactor coolant pump has a restraint system to limit motion of the pump in the event of a rupture of the cold leg piping.

A restraint around the surge line (with integral portions around the loop A hot leg) is provided to maintain the integrity of the secondary system. The restraint was designed to withstand the loads that would result from combining the seismic loads due to maximum hypothetical earthquake and pipe loads due to the design basis accident.

Primary coolant pump support systems are provided and were originally designed for guillotine breaks or longitudinal splits of pump suction or discharge piping at any location and direction between the steam generator and reactor vessel. Pipe rupture loads were combined with the maximum seismic loads due to a hypothetical earthquake. Stresses in the support

system were limited to yield stress. The displacement is small and will not result in the rupture of the main feedwater header, or rupture of other lines in the primary system.

Structural analyses performed in support of OTSG replacement take advantage of leak-before-break methodologies to remove ruptures in the reactor coolant loop piping from the structural design basis. In these analyses, the largest design basis pipe ruptures remaining after consideration of leak-before-break, including guillotine ruptures of the surge line, decay heat line, core flood line, main steam lines and main feedwater lines, combined with maximum hypothetical seismic loads are considered. Stresses in the supports are maintained within the original design basis allowables.

Lateral restraints at the top of each steam generator are provided to maintain the steam generators essentially in rigid condition so that only minimal loads from the design basis accident are transmitted via the steam generator to the steam and feedwater piping.

The dynamic response of the original OTSG lateral support was determined due to rupture of the hot leg. The lateral loads were combined with the maximum seismic loads of a hypothetical earthquake. Stresses in the lateral support were limited to yield stress. The displacement was small and did not jeopardize the integrity of the steam generator.

Structural analysis performed in support of OTSG replacement take advantage of leak-before-break methodologies to remove ruptures in the reactor coolant loop piping from the structural design basis. In these analyses, the largest design basis pipe ruptures remain after consideration of leak-before-break, including guillotine ruptures of the surge line, decay heat line, core flood line, main steam lines and main feedwater lines, combined with dead load, thermal, and maximum hypothetical seismic loads are considered. Stresses in the supports are limited to allowable stresses according to ASME Section III Code.

In addition, secondary shield walls are arranged around each steam generator and its pair of reactor coolant pumps to preclude failure in one area from damaging equipment or piping in the other. Safety related piping has been routed to obtain adequate separability within the secondary shield area or restrained as required.

4.2.7 MISSILE PROTECTION

The major components, including reactor vessel, reactor coolant piping, reactor coolant pumps, steam generators, and the pressurizer, are located within three shielded cubicles. Each of two cubicles contains one steam generator, two coolant pumps, and associated piping. One of the cubicles also contains the pressurizer. The reactor vessel is located within the third cubicle or primary shield. The reactor vessel head and control rod drives extend into the fuel transfer canal.

Penetrations in the cubicles are located such that missiles which may be generated, such as valves, valve bonnets, valve stems, or reactor coolant temperature sensors, will not escape the cubicles or possess sufficient energy to damage the Reactor Building liner plate. Special missile shields are provided as required to supplement the major shield structures in protecting the liner plate or other essential components within the major shield structures.

Pipe lines carrying makeup and purification (high pressure injection) water are routed within the secondary shield wall. Missiles which may be generated in one cubicle cannot rupture makeup

and purification (high pressure injection) lines for the other loop. Decay heat removal (low pressure injection) lines and core flooding lines connect outside the secondary shield walls. The decay heat and core flooding lines penetrate the secondary shield walls at opposite sides of primary shield and enter north and south nozzles of the reactor vessel through penetrations in the primary shield. Thus, the decay heat removal/core flooding lines are protected from potential missiles which might jeopardize their integrity.

Missile shields are located above the control rod drives to stop a control rod drive should it become a missile. The shields are removed during refueling.

Table 4.2-1
(Sheet 1 of 2)

REACTOR VESSEL DESIGN DATA

Design Pressure, psig	2,500
Design Temperature, °F	650
Coolant Operating Temperature, Inlet/Outlet, °F	554/604
Hydrotest Pressure, psig	3,125
Coolant Volume (Hot, Core and Internals in Place), cubic feet	4,058
Reactor Coolant Flow, lb/hr	131.32×10^6
Number of Reactor Closure Head Studs	60
Diameter of Reactor Closure Head Studs, inches	6-1/2
Vessel Dimensions	
Overall Height of Vessel and Closure Head, ft-inches[1]	40 - 8-3/4
Shell ID, inches	171
Flange ID, inches	165
Straight Shell Minimum Thickness, inches	8-7/16
Shell Cladding Minimum Thickness, inches	1/8
Shell Cladding Nominal Thickness, inches	3/16
Insulation Thickness, inches	3
Closure Head Minimum Thickness, inches	6-5/8
Lower Head Minimum Thickness, inches	5

[1] Instrument nozzle to CRD flange.

Table 4.2-1
(Sheet 2 of 2)

REACTOR VESSEL DESIGN DATA

Vessel Nozzles

Function	No.	ID (inches)	Material
Coolant Inlet	4	28	Carbon Steel, SS Clad
Coolant Outlet	2	36	Carbon Steel, SS Clad
Core Flooding - LP Injection[4]	2	14 Sch 140	Carbon Steel,[2] SS Clad
Control Rod Drive	61	2.76	Inconel[3]
Axial Power Shaping Rod Drive	8	2.76	Inconel[3]
Instrumentation	8	3/4 Sch 160	Inconel[3]
In core Instrumentation	52	3/4 Sch 160	Inconel

Dry Weight, lb

Vessel	646,000
Closure Head	158,300
Studs, Nuts, and Washers	39,500

[1] Instrument nozzle to CRD flange
[2] With stainless steel safe end added after stress relief
[3] With stainless steel flanges
[4] With venturi type flow restrictor

TABLE 4.2-2
(Sheet 1 of 3)

STEAM GENERATOR DESIGN DATA
(Data per Steam Generator)

Steam Conditions at Full Load, Outlet Nozzles	
Steam Flow, lb/hr	5.5×10^6
Steam Temperature, °F	570
Steam Pressure, psig	910
Feedwater Temperature, °F	460
Reactor Coolant Flow, lb/hr	65.66×10^6
Reactor Coolant Side	
Design Pressure, psig	2,500
Design Temperature, °F	650
Hydrotest Pressure, psig	3,125
Coolant Volume (Hot), ft^3	2,021
Full Load Temperature Inlet/Outlet, °F	604/554
Secondary Side	
Design Pressure, psig	1,150
Design Temperature, °F	605
Hydrotest Pressure, psig	1,438
Net Volume, ft^3	3,663
Dimensions	
Tubes, OD/min Wall, inches	0.625/0.034
Overall Height (Including Skirt), inches	881
Shell OD, inches	150.62

Table 4.2-2
(Sheet 2 of 3)

STEAM GENERATOR DESIGN DATA
(Data per Steam Generator)

Shell Minimum Thickness, inches	3.31
Shell Minimum Thickness, (at Tube Sheets and Feedwater Connect), inches	5.31
Tube Sheet Thicknesses, inches	24
Dry Weight, lb	991,019
Tube Length, ft-inches	52 - 1-3/8

Nozzles - Reactor Coolant Side

Function	No.	ID (inches)	Material
Inlet	1	36	Carbon Steel - SS Clad
Outlet	2	28	Carbon Steel - SS Clad
Manways	2	16	Carbon Steel - SS Clad
Handholes	1	5	Carbon Steel - SS Clad

Table 4.2-2
(Sheet 3 of 3)

STEAM GENERATOR DESIGN DATA
(Data per Steam Generator)

Nozzles - Secondary Side

Function	No.	ID (inches)	Material
Steam	2	24*	Carbon Steel
Vent	1	1-1/2 Sch 80	Carbon Steel
Drains	2	1-1/2 Sch 80	Carbon Steel
	4	2 Sch 80	
Drain	2	1 Sch 80	Carbon Steel
Level Sensing	8	1 Sch 80	Carbon Steel
Temperature Sensing	3	1-1/2	Carbon Steel
Manways	1	16	Carbon Steel
Feedwater Connect	32	3 Sch 80	Carbon Steel
Feedwater Nozzles	32	4 Sch 40	Inconel 690
Emergency Feedwater Connect	7	3 Sch 80	Carbon Steel
Handholes	6	8	Carbon Steel
Inspection Handholes	15	3	Carbon Steel
Orifice Plate Adjustment Ports	2	8	Carbon Steel
Orifice Plate Locking Ports	2	8	Carbon Steel

* Venturi throat diameter of 7.82 inches. Three venturi per nozzle for total venturi flow area of 1.0 ft^2

TABLE 4.2-3
(Sheet 1 of 2)

PRESSURIZER DESIGN DATA

Design/Operating Pressure, psig	2,500/2155
Design/Operating Temperature, °F	670/648
Steam Volume, ft^3	702
Water Volume, ft^3	800
Hydrotest Pressure, psig	3,125
Electric Heater Capacity, kW	1,638 *
Dimensions	
Overall Height, ft-inches	44 - 11-3/4
Shell OD, inches	96-3/8
Shell Minimum Thickness, inches	6.188
Dry Weight, lb	301,800

* Actual heater capacity may be less than this amount due to heater failures. Total capacity shall not exceed this amount. This vale is the heater capacity at initial installation that consisted of three heater bundles with each heater bundle having 39 single phase heater elements rated for 14kW at 460VAC.

TABLE 4.2-3
(Sheet 2 of 2)

PRESSURIZER DESIGN DATA

Nozzles

Function	No.	ID (inches)	Material
Surge Line	1	10 Sch 140	Carbon Steel, SS Clad[1]
Spray Line Nozzle	1	4 Sch 120	Carbon Steel, SS Clad[2]
Relief Valve	3	2-1/2	Carbon Steel, SS Clad[1]
Vent Line	1	1 Sch 160	SA-479, Type 316 SS
Sample Line	1	1 Sch 160	Carbon Steel, SS Clad
Temperature Well Remnant	1	3/8	Inconel
Level Sensing	6	1 Sch 160	Carbon Steel, SS Clad
Heater Bundle	3	19-1/8	Carbon Steel, SS Clad, SS
Manway	1	16	Carbon Steel, SS Clad
New Relocated Temperature Well	1	3/8	Alloy 690

[1] With stainless steel safe end added after stress relief.
[2] With Inconel safe end.

TABLE 4.2-4
(Sheet 1 of 3)

REACTOR COOLANT PIPING DESIGN DATA

Reactor Inlet Piping	
Pipe ID, inches	28
Design Pressure/Temperature, psig/°F	2,500/650
Hydrotest Pressure, psig	3,125
Minimum Thickness, inches	2-1/4
Coolant Volume (Hot - System Total), ft^3	1,102
Dry Weight, System Total, lb	214,000
Reactor Outlet Piping	
Pipe ID, inches	36
Design Pressure/Temperature, psig	2,500/650
Hydrotest Pressure, psig	3,125
Minimum Thickness, inches	2-7/8
Coolant Volume (Hot - System Total), ft^3	979
Dry Weight, System Total, lb	200,000
Pressurizer Surge Piping	
Pipe Size, inches	10, Sch 140
Design Pressure/Temperature, psig/°F	2,500/670
Hydrotest Pressure, psig	3,125
Coolant Volume, Hot, ft^3	20
Dry Weight, lb	5,000

TABLE 4.2-4
(Sheet 2 of 3)

REACTOR COOLANT PIPING DESIGN DATA

Pressurizer Spray Piping

Pipe Size, inches	2-1/2, Sch 160
Design Pressure/Temperature, psig/°F	2,500/650
Hydrotest Pressure, psig	3,125
Coolant Volume, Hot, ft^3	2
Dry Weight, lb	650

Nozzles

Function	No.	ID (inches)	Material
On Reactor Inlet Piping			
High Pressure Injection	4	2-1/2 Sch 160	Carbon Steel, SS Clad[3]
Pressurizer Spray	1	2-1/2 Sch 160	Stainless Steel
Drain/Letdown	1	2-1/2 Sch 160	Carbon Steel, SS Clad
Drain	3	1-1/2 Sch 160	Carbon Steel, SS Clad[1]
Pressure Sensing	4	1 Sch 160	Carbon Steel, SS Clad
Temperature Well	4	0.375	Inconel
Temperature Sensing	4	0.613	Inconel
On Reactor Outlet Piping			
Decay Heat	1	2 Sch 140	Carbon Steel, SS Clad[2]
Vent	2	Sch 160	Carbon Steel, SS Clad[2]
Conn. on Flow Meters	4	1 Sch 160	Carbon Steel, SS Clad[2]

[1] With Inconel safe end
[2] With stainless steel safe end added after stress relief
[3] With austenitic stainless steel safe end

TABLE 4.2-4
(Sheet 3 of 3)

REACTOR COOLANT PIPING DESIGN DATA

Function	No.	ID (inches)	Material
Pressure Sensing	4	1 Sch 160	Carbon Steel, SS Clad[2]
Temperature Sensing	6	0.686	Alloy 690
Surge Line	1	10 Sch 140	Carbon Steel, SS Clad[1]
On Pressurizer Surge Piping			
Drain	1	1 Sch 160	Stainless Steel
On Pressurizer Surge Piping			
Drain	1	1 Sch 160	Stainless Steel
On Pressurizer Spray Piping			
Auxiliary Spray	1	1 1/2 Sch 160	Stainless Steel
Spray Valve Bypass	2	1/2 Sch 160	Stainless Steel

[1] With Inconel safe end
[2] With stainless steel safe end added after stress relief

TABLE 4.2-5
(Sheet 1 of 2)

REACTOR COOLANT PUMPS - DESIGN DATA
(Data per Pump)

Design Pressure/Temperature, psig/°F	2,500/650
Hydrotest Pressure, psig	3,125
RPM at Nameplate Rating	1,190
Developed Head, ft	350
Capacity, gpm	88,000
Seal Water Injection, gpm	8
Seal Water Leakoff, gpm	3
Injection Water Temperature, min, normal, max, °F	60, 130, 150
Cooling Water Temperature, min, normal, max, °F	60, 80, 105
Cooling Water Flow, gpm	40
Cooling Water Pressure, psig	150
Pump Discharge Nozzle ID, inches	28
Pump Suction Nozzle ID, inches	31
Overall Height (Pump-Motor for RC-P-1A/B/D), ft-inches	31 - 10.5
Overall Height (Pump-Motor for RC-P-1C), ft-inches	28 – 8.45
Dry Weight Without Motor, lb	97,200
Coolant Volume, ft^3	56
Pump-Motor Moment of Inertia (RC-P-1A/B/D), $lb\text{-}ft^2$	70,000
Pump-Motor Moment of Inertia (RC-P-1C), $lb\text{-}ft^2$	70,598

TABLE 4.2-5
(Sheet 2 of 2)

REACTOR COOLANT PUMPS - DESIGN DATA
(Data per Pump)

Motor Data

Type	Squirrel Cage Induction Single-Speed, Water-Cooled
Voltage	6,600
Phase	3
Frequency, H_z	60
Insulation Class	F
Starting Current, amperes (RC-P-1A/B/D)	4,350
Starting Current, amperes (RC-P-1C)	4,000
Power, HP (Nameplate)	9,000

TABLE 4.2-6
(Sheet 1 of 2)
MATERIALS OF CONSTRUCTION

Component	Section	Materials
Reactor Vessel	Pressure Plate	SA-302, Grade B, as modified by Code Case 1339. *See Note.
	Pressure Forgings	A-508-64, Class 2 (Code Case 1332-3)
	Cladding	18-8 Stainless Steel or Ni-Cr-Fe
	Studs, Nuts, and Washers	A-540, Grade B23 (Code Case 1335-2)
	Thermal Shield and Internals	Sa-240, Type 304
	Guide Lugs	Ni-Cr-Fe, SB-168 (Code Case 1336)
Steam Generator	Pressure Plate	SA-508 Gr. 3 Cl. 2
	Pressure Forgings	SA-508 Gr. 3 Cl. 2
	Cladding for Heads	308L/309L Stainless Steel
	Cladding for Tube sheets	Ni-Cr-Fe
	Tubes	Ni-Cr-Fe, SB-163 Alloy 690
	Studs - Reactor Coolant Side	SA-193 Grade B7
	Nuts - Reactor Coolant Side	SA-194 Grade 7
	Studs - Secondary Side	SA-193 Grade B7
	Nuts - Secondary Side	SA-194 Grade 7

Note: The material is metallurgically identical to SA-533, Grade B, Class 1.

TABLE 4.2-6
(Sheet 2 of 2)
MATERIALS OF CONSTRUCTION

Component	Section	Materials
Pressurizer	Shell, Heads, and External Plate	SA-212, Grade B
	Forgings	A-508-64, Class 1 (Code Case 1332-3)
	Cladding	18-8 Stainless Steel
	Studs and Nuts	SA-320, Grade L43
	Internal Plate	SA-240, Type 304
	Internal Piping	SA-312, Type 304
Reactor Coolant Piping	28 inch and 36 inch	SA-516, Grade 70 or SA-508 G3 C12 (Elbows) A-106, Grade C (Straight)
	Flow Meter Piping	SA-350 GR LF-2, Class 1 with Alloy 690 Clad
	Cladding	18-8 Stainless Steel
	10 inch Surge Line and 2-1/2 inch Spray Line	A-403, Grade WP 316 (Elbows A-376, Type 316 (Straight)
	Piping Safe Ends	A-376, Type 316, SA-182 Type 316/316L and Ni-C SB-166
Reactor Coolant	Forging	Stainless Steel, SA182,
Pumps	Static Casting	Stainless Steel, SA351, Gr. CF8
	Tubing and Pipe	Stainless Steel, SA-213, Type 316 or 304 and SA-3 or 312 (Seamless) Type 3 or 316
	Bolting Material	SA-193
	Main Flange Bolt/Stud/ Nut Material	SA-540 Grade B24, Class 4 SA-540 Grade B23/B24, Class 3 SA-540 Grade B23, Class 4
	Welding Filler Metals	SA-298 or SA-371
	Plate, Sheet and Strip	SA-240
Valves	Pressure-Containing Parts	A-351, Grade CF8M; A-182, F316 and F347

TABLE 4.2-7
(Sheet 1 of 7)

FABRICATION INSPECTIONS

Component	RT	UT	PT	MT	ET
1. Reactor Vessel					
1.1 Forgings					
1.1.1 Flanges		X[1]	X		
1.1.2 Studs, Bar		X			
1.1.3 Studs After Final Machining				X	
1.1.4 Skirt Adaptor		X[1]			
1.1.5 Nozzle Shell Forgings		X		X	
1.1.6 Main Nozzle Forgings		X		X	
1.1.7 Dutchman Forging		X[1]	X		
1.1.8 CRD Mechanism Adaptor		X	X		
1.1.9 CRD Mechanism Housing		X	X		
1.2 Plates					
1.2.1 Head and Shell Plate		X[1]	X*		
1.2.2 Support Skirt		X[1]	X*		
1.3 Instrumentation Tubes		X	X		
1.4 Closure O-Rings		X	X		
1.5 Weldments					
1.5.1 Longitudinal and Circumferential Main Seams	X			X	
1.5.2 CRD Mechanism Adaptor to Shell		X			

TABLE 4.2-7
(Sheet 2 of 7)

FABRICATION INSPECTIONS

Component	RT	UT	PT	MT	ET
1.5.3 CRD Mechanism Adaptor to Flange	X		X		
1.5.4 Main Nozzles	X			X	
1.5.5 Instrumentation Nozzle Connection			X		
1.5.6 Nozzle Safe-Ends, Weld Deposit		X	X		
1.5.7 Temporary Attachment After Removal				X	
1.5.8 All Accessible Welds After Hydrotest			X or	X	
1.5.9 O-Ring Closure Weld	X		X		
1.5.10 Cladding, Sealing Surfaces		X[2*]	X		
1.5.11 Cladding, All Other		X[3]	X		
1.5.12 Insulation Support Lugs				X	
2. Steam Generator					
2.1 Tube Sheet					
2.1.1 Forging		X[1]			
2.1.2 Cladding		X[2*]	X		
2.2 Heads					
2.2.1 Plate		X[1]		X[*]	
2.2.2 Cladding		X[3*]	X		

TABLE 4.2-7
(Sheet 3 of 7)

FABRICATION INSPECTIONS

Component	RT	UT	PT	MT	ET
2.3 Shell					
2.3.1 Plates		X[1]	X		
2.4 Tubes		X	X[5*]		
2.5 Nozzles (Forgings)		X		X	
2.6 Studs, Bar		X		X	
2.7 Studs After Final Machining				X	
2.8 Weldments					
2.8.1 Deleted					
2.8.2 Deleted					
2.8.3 Shell, Circumferential	X			X	
2.8.4 Cladding, Sealing Surfaces		X[2*]	X		
2.8.5 Cladding, All Other		X[3*]	X		
2.8.6 Nozzle to Shell	X	X[3]	X		
2.8.7 Level Sensing and Drain Connections	X			X	
2.8.8 Instrument Connections			X		
2.8.9 Support Skirt	X			X	
2.8.10 Tube-to-Tube Sheet (4)			X		

TABLE 4.2-7
(Sheet 4 of 7)

FABRICATION INSPECTIONS

Component	RT	UT	PT	MT	ET
2.8.11 Temporary Attachment After Removal				X	
2.8.12 After Hydrostatic Test (All Accessible Welds)				X	
2.8.13 Lifting Lugs				X	
2.8.14 Deleted				X	
3. Pressurizer					
3.1 Heads					
3.1.1 Plate		X[1]		X	
3.1.2 Cladding		X[3*]	X		
3.2 Shell					
3.2.1 Forging		X[1]		X	
3.2.2 Plate		X[1]		X	
3.2.3 Cladding		X[3*]	X		
3.3 Heater Bundles					
3.3.1 Cover Plate		X		X	
3.3.2 Diaphragm and Spacer Plate		X	X		
3.3.3 Studs, Bar		X			
3.3.4 Studs and Nuts After Final Machining				X	

TABLE 4.2-7
(Sheet 5 of 7)

FABRICATION INSPECTIONS

Component	RT	UT	PT	MT	ET
3.3.5 Heaters					
3.3.5.1 Tubing		X	X*		
3.3.5.2 Positioning of Heater Element in Tube	X				
3.4 Nozzle (Forgings)		X		X	
3.5 Weldments					
3.5.1 Shell, Longitudinal as Deposited By Submerged Arc	X			X	
3.5.2 Shell, Longitudinal as Deposited by Electroslag	X	X		X	
3.5.3 Shell, Circumferential	X			X	
3.5.4 Cladding, Sealing Surfaces		X2*	X		
3.5.5 Cladding, All Other		X3*	X		
3.5.6 Nozzle to Shell	X			X	
3.5.7 Nozzle Safe-Ends (If Weld Deposit)		X	X		
3.5.8 Nozzle Safe-End (If Forging or Bar)	X		X		
3.5.9 Instrumentation and Vent Connections			X		
3.5.10 Support Brackets				X	

TABLE 4.2-7
(Sheet 6 of 7)

FABRICATION INSPECTIONS

Component	RT	UT	PT	MT	ET
3.5.11 Heater Guide Tube Pad		X	X		
3.5.12 Temporary Attachment After Removal				X	
3.5.13 All Accessible Welds After Hydrotest				X	
3.5.14 Insulation Support Pads				X	
4. Piping					
4.1 Pipe					
4.1.1 Forgings		X[1]		X	
4.1.2 Claddings		X[3*]	X		
4.2 Bends					
4.2.1 Plate		X[1]		X[*]	
4.2.2 Claddings		X[3*]	X		
4.3 Nozzle Forgings		X		X	
4.4 Weldments					
4.4.1 Longitudinal	X			X	
4.4.2 Circumferential	X			X	
4.4.3 Cladding, Elbows		X[3*]	X		
4.4.4 Cladding, Straight		X[3*]	X		
4.4.5 Nozzles to Run Pipe	X			X	
4.4.6 Thermowell Connections		X			
4.4.7 Insulation Support Lug Pads				X	

TABLE 4.2-7
(Sheet 7 of 7)

FABRICATION INSPECTIONS

Component	RT	UT	PT	MT	ET
5. Reactor Coolant Pumps					
5.1 Castings	X		X		
5.2 Forgings		X	X		
5.3 Weldments					
5.3.1 Circumferential	X		X		
6. Valves					
6.1 Castings	X		X		
6.2 Forgings		X	X		

[1] 100% scanning for longitudinal wave technique and 100% shear wave technique.
[2] UT of cladding defects and bond to base metal.
[3] UT of cladding bond to base metal (spot check).
[4] Also gas leak test -- B&W requirement.
[5] Over 12-inch length on each end.

[*] Additional B&W requirement.

Notes:

RT: Radiographic
UT: Ultrasonic
PT: Dye Penetrant
MT: Magnetic Particle
ET: Eddy Current

TABLE 4.2-8
(Sheet 1 of 1)
REACTOR COOLANT SYSTEM PRESSURE SETTINGS

	Pressure (psig)	TOTAL CAPACITY (lb/hr) Steam
Design Pressure	2,500	
Pressurizer Code Safety Valves	2,500	595,692[3]
High Pressure Reactor Trip[1]	2,355[2]	
Pressurizer Pilot Operated Relief Valve[1]		
Open	2,450	100,000[4]
Close	2,400	
High Pressure Alarm[1]	2,255[1]	
Pressurizer Spray Valve[1]		
Open	2,205	
Close	2,155	
Operating Pressure[1]	2,155	
Low Pressure Alarm[1]	2,055	
Low Low Pressure Alarm[1]	1,700	
Low Pressure Reactor Trip[1]	1,900	
Hydrotest Pressure	3,125	

1 At sensing nozzle on reactor outlet pipe
2 Refer to FSAR Sections 7.1.2.2.c. 6), 7.1.2.2.c. 8), and Table 7.1.1.
3 Rated flow at 3 percent accumulation or 2575 psig.
4 Rated flow at 2300 psig; the PORV setpoint was raised to be above the reactor trip setpoint by NRC request, IE Bulletin 79-05B, Nuclear Incident at Three Mile Island, April 21, 1979.

4.3 RCS STRUCTURAL DESIGN EVALUATION

4.3.1 FUNCTIONAL DESIGN BASES

The RCS is designed structurally for 2500 psig and 650F. The system will normally operate at 2240 psig at the primary pump outlet and 604F at the reactor vessel outlet.

The number of design transient cycles specified in Table 4.1-1 for fatigue analysis is conservative. Six heatup and cooldown cycles per year are specified, where the system may not be required to complete more than one or two cycles per year in actual operation. A heatup rate of 100°F/hr was used in the fatigue analysis of Transients 1 and 2 in Table 4.1-1.

4.3.2 MATERIAL SELECTION

Each of the materials used in the RCS has been selected for the expected environment and service conditions. The major component materials are listed in Table 4.2-6. RCS materials normally exposed to coolant are corrosion resistant materials, consisting of 304 or 316 stainless steel, Inconel, 17-4PH(H1100), Zircaloy, or weld deposits with corrosion resistant properties equivalent to or better than those of 304 SS. These materials were chosen for specific uses at various locations within a system because of their compatibility with the reactor coolant. There are no novel material applications in the RCS. Nitrogen was not added to any stainless steel type 304 or type 316 to enhance the material strength nor was it added for any other purpose.

To assure long steam generator tube lifetime, feedwater quality entering the steam generator is maintained within the specifications given in Table 9.2-2 in order to prevent deposits and corrosion inside the steam generator. These required feedwater specifications have been successfully used in comparable once-through nuclear steam generators.

The selection of materials and manufacturing sequence for RCS components is arranged to ensure that no RCS pressure boundary material is furnace-sensitized stainless steel. There are no stainless steel pressure boundary parts or load-bearing stainless steel members which become furnace-sensitized during the fabrication sequence.

Safe ends were provided on those carbon steel nozzles of the system vessels and piping which connect the stainless steel piping. Dissimilar metal welds (DMW) were made using Alloy 82/182 weld material (a.k.a Alloy 600 material) to connect stainless steel pipe and safe ends to carbon steel vessel and piping nozzles.

This weld material (Alloy 600) has shown a vulnerability to primary water stress corrosion cracking (PWSCC), especially in components subjected to higher operating temperatures, such as the pressurizer. The RCS locations where Alloy 600 weld material was used are being programmatically addressed. Table 4.3-0 shows all locations where Alloy 600 material was used as pressure retaining component of the RCS boundary and how the vulnerability to PWSCC was addressed.

4.3.3 REACTOR VESSEL

a. Stress Evaluation

A stress evaluation of the reactor vessel has been performed in accordance with Section III of the ASME Code. The evaluation shows that stress levels are within the ASME Code limits. Table 4.3-1 lists the reactor vessel steady-state stresses at various load points. The results of the transient analysis and the determination of the fatigue usage factor at the same load points are listed in Table 4.3-2. As specified in the ASME Code, Section III, Paragraph 415.2(d) (6), the cumulative fatigue usage factor is less than 1.0 for the design cycles tested in Table 4.1-1. Table 4.3-3 presents a summary of the major reactor vessel material physical properties, including the results of fabrication material sample impact tests. Table 4.3-4 lists the chemical analysis results for the same materials.

The stress level of the material in a reactor vessel, or any other component of the coolant system, is a combination of stresses caused by internal pressures and temperature gradients. The maximum steady-state stress resulting from gamma heating in the vessel is a relatively low value, and no problems are anticipated from thermal stresses in the reactor vessel wall.

b. Nil-Ductility Transition Temperature (NDTT)

The reactor vessel plate material opposite core was purchased to mill practices which improve material toughness and result in lower range of NDTT values for heavy sections. The raw material was purchased to be capable of meeting Charpy V-notch values of 30 ft/lb or greater at a temperature of 10F. The material was tested during vessel fabrication after forming to show conformity to specified requirements or to determine the actual temperature at which the specified 30 ft/lb Charpy V-notch value was met.

The unirradiated or initial NDTT of pressure vessel base plate material was measured by the Charpy V-notch impact test (Type A) given in ASTM E23. Using the Charpy V-notch test, the NDTT was defined as the temperature at which the energy required to break the specimen is a certain "fixed" value. For SA-302B or SA-335B steel, the ASME Code, Section III, Table N-332 specified an energy value of 30 ft/lb. A curve of the temperature versus the energy absorbed in breaking the specimen was plotted. To obtain this curve, at least two specimens were tested at a minimum of five different temperatures. Available data indicate NDTT differences as great as 40F between curves plotted through the minimum and average values, respectively. The intersection of the minimum energy versus temperature curve with the 30 ft/lb ordinate was designated as the NDTT. The determination of NDTT from the average curve was considered representative of the material and was consistent with procedures specified in ASTM E23. The material for these tests was treated by the methods outlined in ASME Code Section III, Paragraph N-313. The test coupons were taken at a distance of T/4 (1/4 of the plate thickness) from the quenched surfaces and at a distance of T from the quenched edges. These tests were performed by the material supplier or B&W, in accordance with ASME Code, Section III, Paragraphs N-313 and N-330.

The basic determination of vessel operation from cold startup and shutdown to full pressure and temperature operation was originally performed in accordance with a "Fracture Analysis Diagram" per Reference 2.

At temperatures below the design transition temperature (DTT), which is equal to NDTT + 60F, the pressure vessels are operated so that the stress levels are restricted to a value that will prevent brittle failure. These levels are:

1) Below the temperature of DTT minus 200F, a maximum stress of 10 percent yield strength.

2) From the temperature of DTT minus 200F, a maximum stress which will increase from 10 to 20 percent yield strength.

3) At the temperature of DTT, a maximum stress of 20 percent yield strength.

With the stresses held within the above limits (Items 1 through 3 above), brittle fracture will not occur. This statement is based on data reported in References 3 and 4. These stress values are interpreted in terms of operating temperatures and pressures, and it can be shown that stress limits can be controlled by imposing operating procedure limits through control of pressure and temperature during heatup and cooldown pursuant to References 1 and 3. This procedure assures that the stress levels do not exceed those specified in Items 1 through 3 above.

Subsequent to the original design, fabrication and material testing, new regulatory requirements (10CFR50 Appendices G and H and 10CFR50.61) imposed revised rules for addressing the reactor vessel embrittlement. These requirements and resulting operating limits are discussed in Section 4.3.3.h.

c. Flux and NVT at Reactor Vessel Wall

The design value for the fast neutron flux greater than 1.0 MeV at the inner surface of the reactor vessel is 3.0×10^{10} n/cm^2-s at the reactor reference design power of 2568 MWt. The corresponding calculated maximum fast neutron flux at the vessel wall is 2.2×10^{10} n/cm^2-s. This calculated value includes a lifetime average axial peaking factor of 1.3 and an azimuthal peaking factor of 1.29. For 40 years at 80 percent load this corresponds to an nvt of 2.0×10^{19} n/cm^2-s (maximum) for the vessel wall.

The attenuation of the neutron flux from the core to the reactor vessel was computed using the NRN program (Reference 5). This is a one dimensional multigroup removal-diffusion program in slab, cylindrical, or spherical geometry. This code uses the method in which the uncollided and strongly forward-scattered neutrons (removal groups) are computed by integration of an energy- dependent attenuation kernel over the source volume. Scattering of neutrons out of the removal groups forms a source term for the multigroup diffusion calculations. Neutron slowing-down is handled by elastic and nonelastic scattering matrices for both the removal and diffusion groups.

Neutron fluxes at the vessel wall were computed with the core represented as a slab source equal in thickness to the equivalent core diameter. A lifetime average power distribution through the thickness of the core was determined from calculated power profiles over several core cycles and at various times during each cycle. The neutron energy spectrum was represented by 26 energy groups with 14 of these groups covering the range above 1.0 MeV.

Local flux peaking on the vessel wall due to fuel assemblies extending beyond the equivalent core diameter (azimuthal peaking) was determined with PDQ-5 (Reference 6), a two-dimensional diffusion program. The lifetime-average axial flux peak at the vessel is the same as that in the two outer rows of fuel assemblies.

Calculations with the NRN code were compared with data from various experiments including measurements on the R2-0 reactor at Studsvik Research Center (Reference 7), on the LIDO pool reactor at Harwell (Reference 8), and on a shielding mockup of the reactor vessel and internals at the B&W Critical Experiment Laboratory (CEL) (Reference 9). At the R2-0 reactor, measurements were made through about 3 ft of water with thresholds detectors which included ^{115}In (n,n') ^{115}In (1.5 MeV), ^{32}S (n,p), ^{32}P (3 MeV), and ^{27}Al (n, α),^{24}Na (6 MeV). Energies shown are threshold energies for the reaction.

In the LIDO pool, thermal flux measurements were made through laminations of iron and water over a penetration distance of about 4 ft. In the experiment at B&W's CEL, sulphur foil data was taken at points covering the distance between the core and the reactor vessel.

In all cases, and over the entire penetration distances, the calculations either were in agreement with the data or predicted higher flux and activation levels. It is thus concluded that the NRN code provides a conservative method for the calculation of vessel nvt.

d. Expected NDTT Shift

As a result of fast neutron bombardment of vessel metal in the region surrounding the core, the reactor vessel material ductility will change. The effect is an increase in the NDTT. For the 40 year exposure (Section 4.1.2.8), the predicted NDTT shift, shown on Figure 4.3-1, is 250F. This based on the "Maximum Curve for 550F Data," as shown on Figure 4.3-2. The "Trend Curve for 550F Data," as shown on Figure 4.3-2, represents irradiated material test results and was compiled from the reference documents listed in Table 4.3-5.

The reactor vessel design provides for vessel material surveillance specimens which will permit an evaluation of the actual neutron exposure-induced shift for material NDTT.

Test coupons of welds, heat-affected zones, and base material for the material used in the reactor vessel, are incorporated in the reactor vessel surveillance program, as described in Section 4.4.5.

The NDTT shift is factored into the plant startup and shutdown procedures so that full operating pressure is not attained until the reactor vessel temperature is about DTT. The total stress in the vessel wall due to both pressure and the associated heatup and cooldown transient is restricted to 5000 to 10,000 psi, which is below the threshold of concern for safe operation. An adjusted 100F/hr heatup rate can be maintained throughout life. An adjusted rate is one in which the pressure is held constant to maintain stresses at the desired low level while temperatures are at a level below DTT. A 100F/hr temperature increase is maintained until DTT is passed and pressure can be raised to a new higher level.

e. Fracture Mode Evaluation

Analyses have been made to demonstrate that the reactor vessel can accommodate without failure the rapid temperature change associated with the postulated operation of the emergency core cooling system (ECCS) at end of vessel design life.

The state of stress in the reactor vessel during the LOCA was evaluated for an initial vessel temperature of 603F. The inside of the vessel wall was rapidly subjected to 90°F injection water of the maximum flow rate obtainable. The results of these analyses show that the integrity of the vessel is not violated. Reference 18 provides the details of the analyses.

f. Closure

The reactor closure head was bolted to a ring flange on the reactor vessel. The vessel closure seal was formed by two concentric metal O-ring seals with provisions for leakoff between the O-rings. Reactor closure head leakage is negligible from the annulus between the metallic O-ring seals during vessel steady-state and virtually all transient operating conditions. Only in the event of a rapid transient operation, such as an emergency cooldown, will there be some leakage past the innermost O-ring seal. A stress analysis on a similar vessel design indicates this leak rate would be approximately 10 cc/min and no leakage would occur past the outer O-ring seal.

The reactor closure head was attached to the reactor vessel with sixty 6-1/2 inch diameter studs. The stud material is A-540, Grade B23 (ASME Code Section III, Case 1335), which has minimum yield strength of 130,000 psi. The studs, when tightened for operating conditions, have a tensile stress of approximately 46,000 psi.

The studs are an intimate part of closure geometry which are evaluated for temperature and pressure effects considering transient conditions identified in RCS functional specifications. All normal, upset, emergency, and faulted conditions were considered in the analysis. The stress analysis of the studs included a fatigue evaluation. The stress analysis of the closure verified the structural integrity of the studs in accordance with ASME Section III criteria. The analysis also evaluates the sealing ability of gaskets, thereby providing the assurance that a leak will not occur during design and normal operating conditions.

The fatigue usage factors for the reactor vessel components are included in Table 4.3-2.

g. Control Rod Drive Service Structure

The control rod drive service structure was designed to support control rod drives to assure no loss of function in the event of combined LOCA and maximum hypothetical earthquake. Requirements for rigidity, imposed on the structure to avoid adversely affecting the natural frequency of vibration of the vessel and internals, as well as space requirements for service routing, resulted in stress levels considerably lower than design limits. The structure is more than adequate to perform its required function.

h. Current Reactor Vessel Embrittlement Evaluations

The following are the most recent evaluations of the reactor vessel integrity, in accordance the present license regulatory requirements. The results of these evaluations form the basis for the present operating requirements and procedures. See Reference 59 for the complete basis for the 29 EFPY revision.

1) Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock (PTS Rule)

The projected reactor vessel beltline material RT_{PTS} values for operation through 50.2 EFPY were developed in accordance with 10 CFR 50.61 "Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock" (PTS Rule). The bases for the RT_{PTS} values determined are provided in References 73 & 74. (Historical): In response to GPU Nuclear's submittals to GL 92-01, Revision 1, Supplement 1, for TMI-1, the NRC issued a closeout letter (Reference 50). In this letter, as well as Reference 51, the NRC requested that an assessment of the application of the ratio procedure, as described in Position 2.1 of Regulatory Guide (RG) 1.99 Revision 2, to the TMI-1 pressure-temperature (PT) limit curves and low temperature overpressure (LTOP) limits be provided. GPUN chose to revert back to the use of the RG Position 1 and the chemistry factor (CF) from the RG table as the basis for the revised assessment and operating limits.

The beltline material most susceptible to PTS is the WF-70 lower nozzle belt to upper shell circumferential weld. This RT_{PTS} value considers fluence levels for plant operation to 50.2 EFPY. This RT_{PTS} falls below the 10 CFR 50.61 maximum RT_{PTS} screening value of 300°F for this material; therefore, the TMI-1reactor vessel meets the toughness requirements of 10 CFR5 0.61 to 50.2 effective full power years.

The PTS Rule 10CFR50.61 requires that the projected assessment of the RT_{PTS} must be updated whenever changes in core loadings, surveillance measurements or other information (including changes in capacity factor) indicate a significant change in the projected values. Significant being defined as those which would result in the projected RT_{PTS} value for the limiting material to equal or exceed the PTS Rule screening value.

2) Operational Pressure/Temperature (P/T) Limits

The operating P/T limit curves for heatup and cooldown are periodically revised based on the results of the surveillance capsule testing program described in Section 4.4.5, projections of vessel fluence, and changes in regulatory requirements to which the licensee is committed to comply.

The present TMI-1 operating P-T limits are based upon the projected fluence at 50.2 EFPY at RV beltline axial weld SA-1526 (per Reference 73 & 74).

3) Charpy Upper Shelf Evaluation

In order to demonstrate adequate margin due to ductile failure of the vessel, 10CFR50, App. G requires that the Charpy upper shelf energy (Cv-USE) remain above 50 ft-lbs at the EOL. If the Cv-USE is projected to fall below this value, then a demonstration of adequate margin for such a postulated failure must be submitted and approved by the NRC. Since the Cv-USE of the vessel weld metals had been projected to fall below this value, an equivalent margins analysis (Reference 75) was produced, which demonstrates that the RV beltline welds still provide margins of safety against fracture equivalent to those required by the ASME B&PV Code, Section XI, Appendix G.

As a result of fast neutron irradiation in the beltline region of the core, there will be an increase in the RT_{NDT} with accumulated nuclear operations. The adjusted reference temperatures have been calculated by adding the predicted radiation-induced RT_{NDT} and the unirradiated RT_{NDT} for each of the reactor coolant beltline materials.

Analyses of the activation detectors contained in the TMI-1 surveillance capsules have provided estimates of reactor vessel wall fast neutron fluxes for Cycles 1 through 4 per Reference 24. Extrapolation of reactor vessel fluxes, and corresponding fluence accumulations based on predicted fuel cycle design conditions during the past and future years of operation, are described in Reference 52 and Reference 63.

The inside surface fluence value at each weld location has been calculated on the basis of the peak flux values for each of the operating cycles, the total EFPY per cycle, and azmuthal and axial adjustment factors contained in BAW-2108 (Ref. 42). The fluence attenuation through the vessel wall has been calculated by using the R.G. 1.99 R-2 fluence factor correlation.

The P-T limits developed for operation to 50.2 EFPY have been established in accordance with the requirements of 10 CFR 50, Appendix G, utilizing the analytical methods and flaw acceptance criteria of BAW-10046A, Rev. 2 and ASME B&PV Code, Section XI, Appendix G as modified by ASME B&PV Code Cases N-640 and N-588. These P-T limits, as provided in TMI-1 Technical Specification 3.1.2, protect the RV against non-ductile failure during normal heat-up, normal cooldown, and in-service leak hydrotest conditions.

The pressure limit lines on T.S. 3.1.2 Figure Nos. 3.1-1 and 3.1-2 have been established considering the following:

a. A 25 psi error in measured pressure.

b. A 12°F error in measured temperature.

c. System pressure is measured in RCS “A” loop hot leg. RCS “A” is most conservative and bounds use of “B”.

d. Maximum differential pressure between the point of system pressure measurement and the limiting reactor vessel region for the allowable operating pump combinations.

The heatup and cooldown rate limits shown in T.S. 3.1.2, Figure Nos. 3.1-1 and 3.1-2 are based on linear heatup and cooldown ramp rates which by analysis have been extended to accommodate 15°F step changes at any time with the appropriate soak (hold).

4) Potential Low Temperature Over-pressurization Events Evaluation

At lower operating temperatures, such as during cold shutdown, heatup and cooldown, the reactor vessel materials are less tough (i.e., nonbrittle) than at hot shutdown or power operation temperatures. The operating P/T curves reflect this in terms of lower allowable pressures at lower temperatures. The potential for inadvertent rapid pressurization of the RCS above the operating P/T limits exists and has occurred at other PWR plants at a sufficient frequency to prompt a concern and subsequent issuance of regulatory guidelines regarding low temperature overpressurization (LTOP) protection.

The issue was originally addressed by Reference 13. Subsequently, as part of the TMI-1 Restart activities, the issue was readdressed and resulted in a change to the PORV lower setpoint by reduction from 550 psig to 485 psig. More recently, P-T limits developed for plant operation to 50.2 EFPY determined that a PORV setpoint of 592 psig is adequate for LTOP, along with a LTOP enable temperature of 313°F. Inadvertent actuation of HPI is prevented below LTOP enable temperatures by plant operating procedures. Erroneous opening of the CFT discharge valves is precluded by closing the motor-operated block valves, and locking out their associated breakers, before RCS pressure decreases below the CFT pressure.

A further discussion of the potential events considered is presented in Section 4.3.7.

i. R.V. Head Thermal Stress During Natural Circulation Cooldown

Natural Circulation occurs in the Reactor Coolant System (RCS) when the Reactor Coolant Pumps are tripped and heat transfer is induced in the steam generators by raising secondary side water level. Sufficient circulation exists to remove core decay heat and cooldown the RCS. During natural circulation coolant flow through the reactor vessel upper head is small with little or no mixing. If a cooldown is commenced the reactor vessel upper head remains at a higher temperature than the cylindrical portions of the vessel. Additionally, a steam bubble may develop in the upper head as the RCS is depressurized, further limiting the cooldown of the reactor vessel head.

In response to Generic Letter 81-21 B&W notified the NRC of the potential that thermal stresses in the reactor vessel and the reactor vessel closure studs could be outside the design basis. This concern was evaluated and determined to be a generic safety issue and designated Generic Issue 79 (GI-79). B&W submitted a detailed analysis of a natural circulation cooldown (NCC) transient performed with a 100°F/hr cooldown rate (Reference 36) to the NRC which was independently confirmed by a Brookhaven National Laboratory (BNL) analysis (Reference 37). The NRC concluded the following:

1) The B&W 177 is considered analyzed for NCC events that are bounded by the NCC transient profile shown in Figure 3 of NUREG-1374.

2) It is extremely unlikely that a single NCC event will cause the failure of any U.S. PWR RV, even if a cooldown rate of 100°F per hour is exceeded.

3) An NCC event that does not exceed a total cooldown of 100°F, independent of rate, would not be expected to compromise the safety of any U.S. PWR RV. However, it may result in the RV being outside its documented design basis.

4) Exposure of U.S. PWR RVs to certain NCC transients, particularly transients complicated by other factors such as a stuck-open atmospheric dump valve, amy result in a condition that is outside the documented basis of the RV."

The NRC concluded that NCC events of the type analyzed occur infrequently and that the actual severity of the event will determine the need for and the extent of actions by the licensee following the event. If it can be shown that a NCC event at TMI-1 is bounded by the above referenced analyses then no further actions would be required. Cooldown rate and pressure-temperature limit guidelines are included in the TMI-1 plant procedures. If plant conditions can be maintained with these guidelines the plant response would be well within the bounds of these analyses.

4.3.4 STEAM GENERATORS

a. Research and Development

A steam generator research and development program was completed. The program included the following elements of testing:

1) The steady-state and transient operation tests confirmed the analytically-predicted performance characteristics of the steam generator and provided the data for the control system.

2) Feedwater spray nozzle tests demonstrated that the design will satisfactorily heat the feedwater.

3) Tube leak simulation tests demonstrated that a leak in one tube will not propagate by causing failure in adjacent tubes.

4) Mechanical tests demonstrated that the tubes can withstand, without failure, the mechanical loads they may experience either during normal operation or accident conditions.

5) The analytical and experimental results demonstrated that a steam generator tube wall thickness of .0116 inches met the minimum requirements of draft Reg. Guide 1.121. This thickness corresponds to a 69 percent through-wall defect in a tube of 0.0375” nominal wall thickness.

6) Vibration testing demonstrated that the steam generators contained no undesirable resonance characteristics.

7) Tests to simulate a steam line failure or RCS failure demonstrated the integrity of the steam generator under conditions of rapid depressurization and large temperature differentials between the tubes and the shell of the steam generator.

The work is reported in References 19 and 20.

Additional testing to demonstrate the repair of the OTSG tubes in the upper tubesheet is described in GPUN Topical Report TR-008, Rev. 3 (Reference 28).

b. Stress Evaluation

Because the steam generators are a straight tube-straight shell design with a minor difference in the coefficients of thermal expansion of Inconel and carbon steel, there exists a structural limitation on the allowable mean temperature difference between tubes and shell.The plant's emergency procedures provide maximum and minimum allowable tube-to-shell temperature differences to prevent damage to the steam generator tubing.

The effect of loss of reactor coolant will impose tensile stresses on tubes and cause slight yielding across the tubes. Such a condition will introduce a small permanent deformation in tubes but will in no way violate boundary integrity. A loss of feedwater to the steam generators will cause the tubes to become warmer than the shell and may cause tube deformation. Blowdown tests simulating secondary side blowdown on a 37-tube model boiler show that, although slight buckling in the tubes occurred, there was no loss of reactor coolant.

Calculations confirm that the steam generator tube sheets will withstand loading resulting from a loss of coolant accident. The basis for this analysis was a hypothetical rupture of a reactor coolant pipe resulting in a maximum design pressure differential from the secondary side of 1050 psid. Under these conditions there was no rupture of the primary-to-secondary boundary (tubes and tube sheet).

The maximum primary membrane stress in the tube sheet with these conditions is 23,300 psi which is below the ASME Section II allowable limit of 30,000 psi (Reference 67). Under the condition postulated, stresses in the primary head show only the effect of its role as structural restraint on the tube sheet. The maximum membrane stress at the juncture of the spherical head with the tube sheet is 12,500 psi which is also below the allowable limit of 30,000 psi. It can therefore be concluded that no damage will occur to the tube sheet or primary head as a result of this postulated accident.

In regard to tube integrity under loss of reactor coolant, actual pressure tests of 5/8 inch outside diameter/0.034 inch wall Inconel tubing show collapse under an external pressure of 4950 psig. This is a factor of safety of 4.7 against collapse under the 1050 psig accidental application of external pressure to the tubes.

The rupture of a secondary pipe has been assumed to impose a maximum design pressure differential of 2500 psid across the tubes and tube sheet from the primary side. The criterion for this accident permits no violation of the reactor coolant boundary

(primary head, tube sheet, and tubes). To meet this criterion, the stress limits delineated in ASME Boiler & Pressure Vessel Code, Section III, Paragraph N-714.2 for hydrotest limitations are applicable for the aforementioned abnormal operating circumstance. The referenced section states that the primary membrane stresses in the tube sheet ligaments, averaged across the ligament and through tube sheet thickness, may not exceed 90 percent of the material yield stress at the operating temperature; in addition, the primary membrane plus primary bending stress in the tube sheet ligaments, averaged across the ligament width at the tube sheet surface location giving a maximum stress, may not exceed 135 percent of the material yield stress at the operating temperature.

An examination of stresses under these conditions showed that, for the case of a 2500 psid design pressure differential, the stresses were within acceptable limits. These stresses together with the corresponding stress limits are given in Table 4.3-6.

The basic design criterion for the tubes assumed a pressure differential of 2500 psid in accordance with ASME Code, Section III. Therefore, the secondary pressure loss accident condition imposes no extraordinary stress on the tubes beyond normally expected and considered in ASME Code, Section III requirements.

The superimposed effect of secondary side pressure loss and maximum hypothetical earthquake has been considered. For this condition, the criterion is that there be no violation of the primary to secondary boundary (tube and tube sheet). For the case of the tube sheet, the maximum hypothetical earthquake loading will contribute an equivalent static pressure loading over the tube sheet of less than 5 psig (for vertical shock).

The effect of fluid dynamic forces on the steam generator internals under secondary steam break accident conditions has been simulated in a 37-tube laboratory boiler. Results of the tests show that reactor coolant boundary integrity is maintained under the most severe mode of secondary blowdown.

The ratio of allowable stresses (based on an allowable membrane stress of 0.9 of the nominal yield stress of the material) to the computed stresses for a design pressure differential of 2500 psid is summarized in Table 4.3-7.

4.3.5 RELIANCE ON INTERCONNECTED SYSTEMS

The principal heat removal system interconnected with RCS is steam and power conversion system. This system provides capability to remove reactor decay heat in the hypothetical case where all unit power is lost. Under these conditions, decay heat removal from the reactor core is provided by natural circulation characteristics of the RCS. The emergency feedwater pumps supply feedwater to steam generators from condensate storage tanks and/or the hot well. Steam is vented to atmosphere through the atmospheric vent valves. The analysis for total loss of station electric power (Station Blackout) is presented in Chapter 14.

4.3.6 SYSTEM INTEGRITY

The reactor protective system (Chapter 7) monitors parameters related to safe operation and trips the reactor to protect against RCS damage caused by high system pressure. The

pressurizer code safety valves prevent RCS overpressure after a reactor trip as a result of reactor decay heat and/or any power mismatch between the RCS and the secondary system.

The integrity of the RCS is ensured by proper materials selection, fabrication quality control, design, and operation. A summary of fabrication inspections for components is given in Table 4.2-7. Components in the system are fabricated from materials initially having low NDTT to eliminate possibility of propagating type failures. Materials surveillance specimens inside the reactor vessel had provided a check on the predicted shift in NDTT. Information on NDTT is presently obtained from specimens located in other reactors. A complete stress analysis has been prepared for all design loadings specified in the design specification.

The analysis shows that the reactor vessel, steam generators, pressurizer and pump casing comply with the allowable stress limits of Section III of the ASME Code and the requirements of the design specification. A similar analysis of the piping shows that it complies with the allowable stress limits of USAS B31.7 February 1968 Draft, including June 1968 Errata.

As a further assurance of system integrity, the completed RCS was hydrotested at 3125 psig before initial operation. Hydrostatic testing of the replacement OTSG and hot legs was performed as part of the fabrication for these components.

4.3.7 OVERPRESSURE PROTECTION

a. High Pressure Protection

The RCS is protected against overpressure by the pressurizer code safety valves mounted on top of the pressurizer. The capacity of these valves is determined from considerations of: (1) the reactor protective system; (2) pressure drop (static and dynamic) between the point of highest pressure in the RCS and the pressurizer; and (3) accident or transient overpressure conditions. RCS pressure settings are in Table 4.2-8.

A B&W analysis completed in September of 1985 concluded that the high reactor coolant system pressure trip setpoint could be raised from 2300 psig to 2355 psig with negligible impact on the frequency of opening of the PORV during anticipated overpressurization transients. The high pressure trip setpoint was subsequently raised to 2355 psig. The potential safety benefit of this action is a reduction in the frequency of reactor trips. For more detail see "Justification for Raising Setpoint for Reactor Trip on High Pressure", BAW-1890 Rev. 0, Babcock and Wilcox, September 1985.

b. Low Temperature Over Pressurization [LTOP] Protection

For events which cause the RCS pressure to increase, the pressure will increase significantly faster in a solid water system than it will in a system with steam or gas space. The RCS always operates with steam or gas space in pressurizer; no anticipated operations involve a solid water condition, other than system hydrotest.

Considering the modest rate of pressure rise (because of non-solid pressurizer) from the events and the high level alarms in the pressurizer that would normally alert the operator, it is reasonable to expect the operator to terminate the event prior to reaching an overpressurization condition. However, without operator action, the PORV located on the pressurizer will terminate any pressure increase. A dual set point is utilized for this valve to provide overpressure

protection during startup and shutdown conditions. The lower set point is enabled by actuation of a switch in the Control Room. Characteristics of this valve at the lower set point are:

Open set point	592	psig
Close set point	542	psig
Steam Capacity at 550 psig	25,985	lb/hr
Equivalent liquid insurge volume rate into the pressurizer	2,650	gpm
Liquid Capacity at 550 psig	500	gpm
Nitrogen capacity at 550 psig	32,420	lb/hr
Equivalent liquid insurge volume rate into the pressurizer	2,350	gpm

The following events have been analyzed (References 13 and 27) and shown not to be credible (Note that the analyzed events continue to be demonstrated as not credible for the 50.2 EFPY P-T limits):

a. Erroneous Actuation of the HPI system

b. Erroneous opening of the core flood tank discharge valve

c. Erroneous addition of nitrogen to the pressurizer

This same analyses (References 13 and 27) has demonstrated the adequacy of the RCS over-pressure protection system for the following postulated events (Note that RCS over-pressure protection continues to be adequate for the 50.2 EFPY P-T limits):

a. RCS makeup control valve fails, full open

b. All pressurizer heaters erroneously energized

c. Temporary loss of decay heat removal systems capability to remove decay heat from the RCS

d. Thermal expansion of RCS after starting an RCS pump due to stored thermal energy in the steam generator

4.3.8 SYSTEM INCIDENT POTENTIAL

Potential accidents and their effects and consequences as a result of component or control failures are analyzed and discussed in Chapter 14.

The pressurizer spray line contains an electric, motor-operated backup valve which can be closed should the pressurizer spray valve malfunction and fail to close; this will prevent

depressurization of the system to the saturation pressure of the reactor coolant. An electric motor- operated valve located between the pressurizer and the pressurizer PORV can be closed to prevent pressurizer steam blowdown in the unlikely event the PORV fails to reclose after being actuated. Because of the other protective features in the plant, it is unlikely that the code valves will ever lift during operation. In addition, it is extremely unlikely these valves would stick open, since there is adequate experience to indicate the reliability of code safety valves. The analysis in Section 14.2 indicates that one high pressure injection pump is sufficient to protect the core for an opening in the system considerably larger than one pressurizer code safety valve in the open position.

4.3.9 REDUNDANCY

Each heat transport loop of the RCS contains one steam generator and two reactor coolant pumps. Operation at reduced reactor power is possible with one or more pumps out of service. For added reliability, power to each pump is normally supplied by one of two electrically separated buses (Chapter 8). The two pumps in each loop are fed from separate buses.

Two core flooding nozzles are located on opposite sides of the reactor vessel to ensure core reflooding water in the event of a single nozzle failure. Reflooding water is available from either the core flooding tanks or the decay heat removal system. The makeup and purification lines are connected to the RCS on each of the four reactor coolant inlet pipes.

4.3.10 SAFETY LIMITS AND CONDITIONS

4.3.10.1 Maximum Pressure

The RCS serves as a barrier which prevents release of radionuclides contained in the reactor coolant to the Reactor Building atmosphere. In the event of fuel cladding failure, RCS is the primary barrier against the release of fission products to the Reactor Building. The safety limit of 2750 psig (110 percent of design pressure) has been established. This represents the maximum transient pressure allowable in the RCS under the ASME Code, Section III.

4.3.10.2 Maximum Reactor Coolant Activity

Release of activity into the reactor coolant in itself does not constitute a hazard. Activity in the reactor coolant constitutes a hazard only if the amount of activity is excessive and it is released to the environment. The plant systems are designed for operation with activity in the RCS resulting from 1 percent defective fuel. Activity would be released to the environment if the reactor coolant containing gaseous activity were to leak to the steam side of the steam generator. Gaseous activity could then be released to environment by the vacuum pump on the main condenser. In 10CFR20, maximum permissible concentrations (MPCs) for continuous exposure to gaseous activity have been established. These MPCs will be used as the basis for maximum release of activity to the environment which has unrestricted access.

4.3.10.3 Leakage

RCS leakage rate is determined by comparing instrument indications of reactor coolant average temperature, pressurizer water level, and makeup tank water level over a time interval. All these indications are recorded.

4.3.10.4 System Minimum Operational Components

One pressurizer code safety valve is capable of preventing overpressurization when the reactor is not critical and the RCS temperature remains high since its relieving capacity is greater than that required by the sum of the available heat sources; i.e., pump energy, pressurizer heaters, and reactor decay heat. Both pressurizer code safety valves are required to be in service, prior to criticality, to conform to system design relief capabilities. Both steam generators are required to be operable, prior to criticality, as they are the means for normal decay heat removal at temperatures above 250°F.

A reactor coolant pump or decay heat removal pump is required to be in operation prior to reducing boron concentration by dilution with makeup water. Either pump will provide mixing which will prevent sudden positive reactivity changes caused by dilute coolant reaching the reactor.

4.3.11 QUALITY ASSURANCE

Assurance that the RCS meets its design bases, insofar as the integrity of the pressure boundary is concerned, is obtained by analysis, inspection, and testing.

4.3.11.1 Stress Analyses

Detailed stress analyses of the individual RCS components including the vessel, piping, pumps, steam generators, and pressurizer have been performed for the design bases.

The primary coolant system has been dynamically analyzed considering a model consisting of one loop and support structures for interaction effects to determine piping and nozzle stress. One loop includes a steam generator, two coolant pumps, associated piping, and reactor vessel. This analysis used a lumped mass model with the assumption that the total system remains elastic and has proportional damping so that normal mode approach can be used. After obtaining the dynamic properties such as natural frequencies and mode shapes for the system with the use of a flexibility matrix and a mass matrix, the probable maximum response was then computed from the square root of the sums of the modal squares of the maxima by using the response spectra technique. This approach permitted calculation of accelerations displacements. The reverse effective forces were calculated on mode-by-mode basis. Applying these forces on the system; internal forces, moments, displacements, rotations, and stresses, together with the support reactions were then determined. Other Class 1 piping systems were analyzed in a similar manner.

Independent thermal and dynamic analyses have been performed to ensure that piping connecting to the RCS is of the proper schedule and that it does not impose forces on the nozzles greater than allowable. Small nozzles are conservatively designed and utilize ASA Schedule 160.

The reactor coolant pump casing has been completely analyzed, including a dynamic analysis separate from the loop, to ensure that the stresses throughout the casing are below the allowable stresses for all design conditions.

Stress analysis reports required by codes for the several components have been prepared by the manufacturer and reviewed for adequacy by a separate organization prior to shipment of the components.

4.3.11.2 Shop Inspection

Inspection and non-destructive testing of the materials prior to and during manufacturing in accordance with applicable codes and additional requirements imposed by the manufacturer have been carried out for all of the RCS components and piping. The extent of these inspections and testing is listed in Table 4.2-7 for each of the components in the system. The shop testing of major components culminates with a hydrostatic test of each component was followed by magnetic particle or liquid penetrant inspection of the components external surface. (Piping has been hydrostatically tested in the field and underwent the final inspection described in Subsection 4.3.11.3.)

Components were cleaned, packaged to prevent contamination, and shipped over a preselected route to the site. For materials purchased or manufactured outside of B&W, the results of the material inspection and testing program were observed or audited by B&W. In addition, there was an independent audit by B&W's Nuclear Power Generation Department Quality Assurance Section.

4.3.11.3 Field Inspection

Field welding of the reactor coolant piping and the piping connecting to the nozzles was performed using procedures which resulted in weld quality equal to that obtained in shop welding. Non-destructive testing of the welds was identical to that performed on similar welds in the shop and is shown in Table 4.2-7. Accessible shop and field welds and weld repairs in the reactor coolant piping were inspected by ultrasonic test following the system hydrostatic test.

4.3.11.4 Testing

The RCS, including the reactor coolant pump internals, the reactor closure head, the control rod drives, and associated piping out to the first stop valve, underwent a hydrostatic test following completion of assembly. The hydrostatic test was conducted at a temperature at least 60°F greater than the highest NDT. During the hydrostatic test, a careful examination was made of all pressure boundary surfaces including gasketed joints.

The vendor of the new steam generators performed a primary side hydrostatic test in accordance with ASME Section III, 2001 Edition through 2003 Addenda, prior to delivery of the new steam generators to the site. Nondestructive Examination (NDE) was performed of the completed RCS installation welds in accordance with Section III of the ASME Code, which was then followed by an RCS pressure leak test in accordance with Section XI of the ASME Code.

TABLE 4.3-0
(Sheet 1 of 2)

MITIGATION OF ALLOY 600 MATERIALS

LOCATION (where Alloy 600 material was used on RCS pressure boundary)	Repair / Mitigation	Design Change Document
Reactor Vessel - Incore Nozzles	Inspection per ISI Program.	NA
CRDM Motor Tube weld (69 welds)	Inspect per ISI Program. New Type C CRDMs with stainless steel motor tubes replaced the Type A CRDMs with nickel-lined carbon/low alloy steel motor tubes.	ECR 12-00217
Reactor Vessel Closure Head Nozzles for CRDM (69) and thermocouples (8)	A new RV Closure head was installed. All new nozzles use SB-167 Alloy 690 material.	ECR 02-01410
Reactor Vessel Core Flood "A" Nozzle to Safe End Weld	A protective layer of Alloy 52M weld metal was installed on the pipe ID to prevent PWSCC.	ECR 08-00961
Reactor Vessel Core Flood "B" Nozzle to Safe End Weld	The weld configuration was modified to facilitate periodic inspection (ISI).	ECR 09-00902
OTSG tube material, tubesheet clad, tube to tubesheet weld and lower endbell drain line safe end and welds	The OTSGs were replaced using Inconel Alloy 690 tube material. The end bell drain lines were eliminated.	ECR 06-00935
Pressurizer Level Sensing and Sample Nozzle(s) weld to safe end and safe end (7 nozzles)	The safe ends were removed and the nozzles were attached to stainless steel pipe using ER309/309L weld material.	ECR 08-00293
Pressurizer upper, middle, lower heater bundle diaphragm plate and seal weld	The upper heater bundle diaphragm plate was replaced with SA 240 Type 304 and installed using Alloy 152/52 weld material. The middle and lower heater bundle diaphragm plates were replaced with SA-182, Grade F304 and installed using Alooy 52/52M weld material.	ECR 04-00375 ECR 12-00278
Pressurizer Relief Valve Nozzle to Flange welds (3 welds)	The welds were replaced using ER309/309L weld material.	ECR 05-00286
Pressurizer Spray Nozzle to safe end and safe end to pipe welds	A full SWOL using Alloy 52M was installed.	ECR 10-00309
Pressurizer Surge Line Nozzle to safe end weld	A full SWOL using Alloy 52M was installed.	ECR 07-00369

TABLE 4.3-0
(Sheet 2 of 2)

MITIGATION OF ALLOY 600 MATERIALS

LOCATION (where Alloy 600 material was used on RCS pressure boundary)	**Repair / Mitigation**	**Design Change Document**
Pressurizer Thermowell and weld to the Pressurizer shell	The abandoned Pressurizer thermowell was covered with a Alloy 52/52M weld pad. The replacement thermowell is made of Alloy 690 and was installed using Alloy 52/52M weld material.	ECR 08-00294
Pressurizer vent nozzle and safe end	The nozzle was replaced with one made from SA-479 Type 316 material.	ECR 05-00288
Cold Leg – DMW at Reactor Coolant Pump suction and discharge (8 welds)	Mechanical stress improvement process was implemented in the RCP discharge piping to introduce compressive stresses in the Alloy 182 attachment welds to eliminate tensile stresses necessary for the development of PWSCC.	ECR 14-00131
Cold Leg – RTE (one per leg) mounting boss and weld	Inspect per ISI Program.	NA
Cold Leg – Thermowells (one per leg) and associated welds	Inspect per ISI Program.	NA
Cold Leg C – Letdown Line safe end & welds	Inspect per ISI Program pending SWOL.	ECR 12-00277
Cold Leg A & B RCP suction & discharge pressure tap - safe end & welds	Inspect per ISI Program.	NA
Cold Leg – (A, B &D) Drain Nozzle safe end & welds	Inspect per ISI Program.	NA
Cold Leg HPI Nozzle to safe end welds, safe ends and safe end to check valve welds (4 locations)	The safe ends was replaced with SA182 Type 316L material. The weld material was ER309/309L and 316/316L.	ECR 08-00962
Hot Leg - Vent and Pressure sensing nozzle safe ends and welds - Flow Measurement Nozzles, safe ends and welds - Thermowells	The upper section (from & including flow venturi to the OTSG) of each Hot Leg was replaced. All nozzles were replaced with inconel Alloy 690 or SA182 Type F316 stainless steel.	ECR 07-00576
Hot Leg A Nozzle to surge line safe end weld	A full SWOL using Alloy 52M was installed.	ECR 03-00881
Hot Leg B Decay Heat Drop Line Nozzle to safe end weld	A full SWOL using Alloy 52M was installed.	ECR 07-00369

TABLE 4.3-1
(Sheet 1 of 1)

SUMMARY OF PRIMARY PLUS SECONDARY STRESS
FOR COMPONENTS OF THE REACTOR VESSEL

Area	Stress Intensity, psi	Allowable Stress 3 Sm, psi (Operating Temperature)
Control Rod Housing	24,800	49,800
Head Flange	58,000	80,000
Vessel Flange	43,000	80,000
Closure Studs	89,400	107,400
Primary Nozzles		
Inlet/Outlet	24,000	80,000
Bottom Head to Shell	23,300	80,000
Bottom Instrumentation	10,100	69,900
Nozzle Belt to Shell	32,300	80,000
Core Flooding Nozzle	23,660	80,000
Support Skirt	88,000	93,700

TABLE 4.3-2
(Sheet 1 of 1)

SUMMARY OF CUMULATIVE FATIGUE USAGE FACTORS FOR
COMPONENTS OF THE REACTOR VESSEL

	Usage Factor[1]
Control Rod Housing	0.0
Head Flange	0.10
Vessel Flange	0.05
Stud Bolts	0.38
Primary Nozzles – Inlet	0.06
Outlet	0.06
Bottom Head to Shell	0.0
Bottom Instrumentation	0.0
Nozzle Belt to Shell	0.0
Core Flooding Nozzle	0.02
Support Skirt	0.14

[1] As defined in Section III of the ASME Boiler and Pressure Vessel Code, Nuclear Vessels.

TABLE 4.3-3
(Sheet 1 of 2)

REACTOR VESSEL - PHYSICAL PROPERTIES

Item	Heat No.	Ultimate Strength (10^3psi)	Yield Strength (10^3psi)	Elongation Range(%)	Impact Test Temp (f)	Impact Values
Upper Shell Course	C2789-1	94.0	69.5	25.8	+10	50-36-33
Upper Shell Course	C2789-2	92.0	76.0	26.6	+20	42-37-35
Bottom Shell Course	C3307-1	86.0	64.25	31.3	+30	42-41-29
Bottom Shell Course	C3251-1	87.5	66.75	28.0	+10	43-40-29
Bottom Head	B5864-2	86.25	63.0	30.0	+10	36-26-42
Upper Shell Flange	3570-VI	86.98	67.81	25.0	+10	98-103-110
Core Flood Nozzle No.1	94894	90.5	68.0	24.0	+10	45-27-43
Core Flood Nozzle No.2	94894	96.25	72.0	23.0	+10	30-25-35
Inlet Nozzle	125T338VA-1 125T338VA-2	90.0	67.5	26.6	+10	100-80-108
Inlet Nozzle	122T111VA-2 122T212VA-2	92.0	70.5	23.4	+10	90-97-68
Inlet Nozzle	125T300VA-1	82.5	62.5	28.1	+10	40-81-107
Inlet Nozzle	125T300VA-2	87.0	65.0	27.0	+10	81-98-75
Outlet Nozzle		92.0	70.5	25.0	+10	81-98-75
Outlet Nozzle		90.5	68.0	25.0	+10	82-81-79

TABLE 4.3-3
(Sheet 2 of 2)

REACTOR VESSEL - PHYSICAL PROPERTIES

Item	Heat No.	Ultimate Strength (10^3psi)	Yield Strength (10^3psi)	Elongation Range(%)	Impact Test Temp (f)	Impact Values
Closure Head Flange	123T292VA-1	94.5	73.5	24.0	+10	70-91-102
		94.5	72.5	24.0	+10	106-107-121
Closure Head Center Disc	C2331-1	88.0	66.5	29.0	+10	28-30-36
Closure Studs	6780366	166.4	152.0	13.0	+10	36 to 48
Lower Head Trans Piece	122T229VA-1	96.5	75.5	24.5	+10	45-84-100
Upper Nozzle Shell Course	123S450	88.61	72.64	24	+10	66-76-90
Lower Nozzle Shell Course	123S454	89.6	69.35	26	+10	100-117-109

TABLE 4.3-4
(Sheet 1 of 2)

REACTOR VESSEL - CHEMICAL PROPERTIES

Heat Number	Element C	Mn	P	S	Si
C2789-1	0.24	1.36	0.010	0.017	0.23
C2789-2	0.24	1.36	0.010	0.017	0.23
C3307	0.21	1.24	0.010	0.017	0.27
C3251-1	0.23	1.41	0.010	0.016	0.21
B5864-2	0.21	1.29	0.005	0.016	0.17
3570-VI	0.91	0.64	0.012	0.015	0.25
94894	0.22	0.62	0.006	0.009	0.22
C233-1	0.22	1.35	0.009	0.018	0.22
125T338VA-2	0.22	0.65	0.010	0.009	0.28
125T338VA-1	0.23	0.63	0.010	0.009	0.30
122T111VA-2	0.18	0.59	0.010	0.009	0.16
122T212VA-2	0.22	0.67	0.010	0.007	0.24
125T300VA-1	0.20	0.66	0.010	0.013	0.32
125T300VA-2	0.23	0.68	0.010	0.014	0.23
123T292VA-1	0.21	0.63	0.010	0.009	0.28
	0.22	0.63	0.010	0.011	0.26
6780366	0.41	0.75	0.009	0.010	0.26
122T229VA-1	0.22	0.63	0.010	0.008	0.28
	0.22	0.63	0.010	0.008	0.28
123S454	0.26	0.63	0.006	0.008	0.28
123S450	0.20	0.60	0.004	0.009	0.21

TABLE 4.3-4
(Sheet 2 of 2)

REACTOR VESSEL - CHEMICAL PROPERTIES

Heat	Element					
Number	Ni	Mo	Co	V	Cr	Cu
C2789-1	0.57	0.51	0.015	--	--	--
C2789-2	0.57	0.51	0.015	--	--	--
C3307	0.55	0.47	0.015	--	--	--
C3251-1	0.50	0.47	0.07	--	--	--
B5864-2	0.44	0.49	0.015	--	--	--
3570-VI	0.73	0.54	0.011	0.03	0.33	--
94894	0.87	0.60	0.016	0.00	0.33	--
C233-1	0.61	0.49	0.005	--	--	--
125T338VA-2	0.74	0.57	0.004	0.01	0.33	--
125T338VA-1	0.75	0.57	0.003	0.04	0.37	--
122T111VA-2	0.68	0.55	0.017	0.04	0.36	--
122T212VA-2	0.80	0.61	0.02	0.01	0.39	--
125T300VA-1	0.66	0.58	0.017	0.02	0.37	--
125T300VA-2	0.70	0.57	0.017	0.02	0.36	--
123T292VA-1	0.74	0.58	0.010	<.02	0.38	--
	0.72	0.58	0.010	<.02	0.39	--
6780366	1.81	0.27	--	--	0.82	--
122T229VA-1	0.74	0.60	0.013	0.02	0.38	--
	0.72	0.60	0.013	0.02	0.36	--
123S454	0.72	0.64	0.01	0.04	0.34	0.08
123S450	0.70	0.58	0.01	0.04	0.34	--

TABLE 4.3-5
(Sheet 1 of 3)

REFERENCES FOR FIGURE 4.3-1 -- INCREASE IN TRANSITION TEMPERATURE DUE TO IRRADIATION EFFECTS FOR A302B STEEL

Ref. No.	Reference	Material	Type	Temp, °F	Neutron Exposure, n/cm^2 (MeV)	NDTT, °F
1	ASME Paper No. 63-WA-100 (Figure 1)	All	Steel	(Max Curve for 550 Data)		
2	ASTM-STP 380, p 295	A302B	Plate	(Trend Curve for 550 Data)		
3	NRL Report 6160, p 12	A302B	Plate	550	5×10^{18}	65
4	ASTM-STP 341, p 226	A302B	Plate	550	8×10^{18}	85[1]
5	ASTM-STP 341, p 226	A302B	Plate	550	8×10^{18}	100
6	ASTM-STP 341, p 226	A302B	Plate	550	1.5×10^{19}	130[1]
7	ASTM-STP 341, p 226	A302B	Plate	550	1.5×10^{19}	140
8	Quarterly Report of Progress, "Irradiation Effects on Reactor Structural Materials," 11-1-64/ 1-31-65	A302B	Plate	550	3×10^{19}	120

[1] Transverse Specimens

TABLE 4.3-5
(Sheet 2 of 3)

REFERENCES FOR FIGURE 4.3-1 -- INCREASE IN TRANSITION TEMPERATURE DUE TO IRRADIATION EFFECTS FOR A302B STEEL

Ref. No.	Reference	Material	Type	Temp, °F	Neutron Exposure, n/cm^2 (MeV)	NDTT, °F
9	Quarterly Report of Progress, "Irradiation Effects on Reactor Structural Materials," 11-1-64/ 1-31-65	A302B	Plate	550	3×10^{19}	135
10	Quarterly Report of Progress, "Irradiation Effects on Reactor Structural Materials," 11-1-64/ 1-31-65	A302B	Plate	550	3×10^{19}	140
11	Quarterly report of Progress, "Irradiation Effects on Reactor Structural Materials," 11-1-64/ 1-31-65	A-302B	Plate	550	3×10^{19}	170
12	Quarterly report of Progress, "Irradiation Effects on Reactor Structural Materials," 11-1-64/ 1-31-65	A302B	Plate	550	3×10^{19}	205
13	Welding Research Supplement, Vol 27, No. 10, Oct. 1962, p 465-S	A302B	Weld	550 to 575	5×10^{18}	70

TABLE 4.3-5
(Sheet 3 of 3)

REFERENCES FOR FIGURE 4.3-1 -- INCREASE IN TRANSITION TEMPERATURE DUE TO IRRADIATION EFFECTS FOR A302B STEEL

Ref. No.	Reference	Material	Type	Temp, °F	Neutron Exposure, n/cm^2 (MeV)	NDTT, °F
14	Welding Research Supplement, Vol 27, No. 10, Oct. 1962, P 465-S	A302B	Weld	550 to 575	5×10^{18}	50
15	Welding Research Supplement, Vol 27, No. 10, Oct. 1962, p 465-S	A302B	Weld	500 to 575	5×10^{18}	37
16	Welding Research Supplement, Vol. 27, No. 10, Oct. 1962, p465-S	A302B	Weld	500 to 575	5×10^{18}	25

TABLE 4.3-6
(Sheet 1 of 1)

TUBESHEET STRESSES DUE TO A MAXIMUM DESIGN STEAM GENERATOR TUBE SHEET PRESSURE DIFFERENTIAL OF 2500 psig AT 650F

Design Conditions

Stress	Computed Value	Allowable Value
Primary Membrane	23,300 psi	30,000 psi (S_m)
Primary Membrane Plus Primary Bending	40,100 psi	45,000 psi (1.5 S_m)

Shop Hydrostatic Test Conditions

Stress	Computed Value	Allowable Value
Primary Membrane	29,100 psi	58,500 psi (0.9 S_y)
Primary Membrane Plus Primary Bending	50,100 psi	87,800 psi (1.35 S_y)

TABLE 4.3-7
(Sheet 1 of 1)

RATIO OF ALLOWABLE STRESSES TO COMPUTED STRESSES FOR A STEAM GENERATOR TUBE SHEET PRESSURE DIFFERENTIAL OF 2500 psig

Component Part	Stress Ratio
Primary Head	4.02
Primary Head Tube Sheet Joint	4.02
Tubes	1.07
Tube Sheet	(See Table 4.3-6)
Maximum Average Ligament	1.02
Effective Ligament	1.70

TABLE 4.3-8

DELETED

TABLE 4.3-9

DELETED

4.4 TESTS AND INSPECTIONS

4.4.1 INSERVICE INSPECTION OF THE REACTOR COOLANT SYSTEM

The Reactor Coolant System is identified as Class 1 in the TMI Unit 1 Inservice Inspection Program required by Section 4.2 of the Technical Specifications. A preservice inspection program was prepared and implemented prior to initial operation utilizing inspection techniques available at that time. Subsequent inservice inspections should utilize the methods used in the preservice inspections to the maximum extent possible.

The surveillance program for the Reactor Coolant System conforms to the requirements of Section XI of the ASME Boiler and Pressure Vessel Code, as incorporated by reference into 10 CFR 50.55a.

4.4.1.1 Reactor Vessel

a. Longitudinal and circumferential shell welds, including the flange to shell weld, and vessel interior are monitored using volumetric and visual techniques. A repair was performed on the RV shell plate during construction and requires periodic inservice inspection similar to the other shell welds. Location of the repair is described in a letter from E.G. Ward, B&W, to R.W. Heward, GPUSC, dated March 19, 1974.

   A 42 inch annulus was originally present between the reactor vessel and the primary shield to accommodate ultrasonic and visual inspection equipment. Access to this cavity is gained by disassembly of the shielding between the reactor vessel and primary shield at the top of the annulus, which is designed to be removable. However, since the period when the preservice exam was performed a Permanent Canal Seal Plate (PCSP) has been welded at the top of the annulus between the reactor vessel and the primary shield limiting access to this cavity for inspections. Limited access to this cavity for personnel and equipment is provided by bolted access ports around the circumference of the PCSP.

   Ultrasonic technology developed since the preservice inspection allows volumetric examination of all shell welds from the inner surface provided the core is offloaded and core support assembly is removed. This equipment results in less radiation exposure than equivalent exams conducted from the outer surface.

   Visual examination of the internal surfaces and structures is also conducted when the core is removed for the ultrasonic inspection.

b. Nozzle welds and inner radii sections were volumetrically inspected from the external surface during the preservice exam with access as discussed in a. above. However, inservice volumetric inspection (ultrasonics) and visual inspections can be done from the vessel I.D. when the core is removed. Only partial ultrasonic examinations on the two outlet nozzles can be done from the vessel I.D. without offloading the core.

c. The reactor vessel closure head was replaced during the October, 2003 refueling outage. This new head is an integrally forged head that no longer employs closure head welds. The interior and exterior of the head can be visually examined by remote means.

The reactor vessel head currently has one partial penetration "J-Groove" weld in each of the 69 control rod drive locations. The reactor vessel closure head currently has two "Thermal Couple Penetrations" that use a partial penetration "J-Groove" weld. These welds are currently examined in accordance with 10 CFR 50.55a in addition to required ASME Section XI examination.

The control rod drive housings contain four welds in each of the 69 control rod drive locations. One weld is located on the reactor vessel closure head and only the 24 peripheral drive location welds are accessible for examination from the outside surface. Three of the control rod drive housing welds are located in the removable segment of the control rod drives and are accessible for examination only while the drive is removed for inspection or maintenance.

d. Flange ligaments, closure studs, spherical washers and nuts can be directly inspected using NDE methods.

e. The vessel support weld is no longer required to be inspected as part of the inservice inspection program. This position was submitted to the NRC in 1987.

4.4.1.2 Pressurizer

a. Circumferential and longitudinal welds are accessible from the external surface, however, configuration of the pressurizer limits the volume of weld that can be ultrasonically inspected.

b. Internal surfaces are able to be visually inspected by remote means.

c. Nozzles, bolting and welded attachments are directly accessible for inspection.

4.4.1.3 Steam Generator

The external surfaces of the steam generators are accessible for non-destructive inspection.

Steam Generator Components included in the inservice inspection program include:

a. Shell to tubesheet welds
b. Head to tubesheet welds
c. Nozzle welds/inner radii
d. Manway and handhole bolting
e. Head welds
d. Support Skirt Welds

The primary side of the steam generator can be visually inspected internally by remote visual means by removing the manway covers in the heads.

Portions of the internal surfaces of the shell, feedwater nozzles and secondary side of the bottom tubesheet can be inspected by removal of the handhole covers and manway covers. Inspection of these areas is controlled by the steam generator program and is not part of the ISI program.

4.4.1.4 Reactor Coolant Pumps

a. The external surfaces of the pump casings are accessible for inspection. Internal surfaces of the pump are accessible by removal of the pump internals.

b. Main flange bolting and lower seal housing bolting are accessible for direct visual or volumetric examination or can be removed for surface examination.

c. The RC pump flywheels are inspected as required by Section 4.2 of the Technical Specification. The ultrasonic examination is conducted through access ports in the motor cover.

4.4.1.5 Piping

Reactor coolant piping, fittings and attachments to the piping are accessible for external surface and volumetric examination. No reactor coolant piping welds are located within the primary shielding. Examination of some of the piping welds is limited by the presence of LOCA supports.

On November 7, 2003, the NRC approved use of a Risk-Informed Inservice Inspection Program (RISI) for Class 1 and 2 piping welds (Reference 72).

4.4.1.6 Dissimilar Metal Welds

All pressure retaining dissimilar metal welds are accessible for inspection. Dissimilar welds on the reactor vessel include the core flooding line, incore instrumentation nozzles, and control rod drive housings. Dissimilar metal welds in the piping include the reactor coolant pump inlets and outlets. Dissimilar metal welds in the pressurizer include the surge line, the relief valve nozzles, heater bundles, level nozzles, vent nozzle, sample nozzle, the thermowell remnant, the new relocated thermowell, and the spray line connections. Other dissimilar metal welds occur at the small drain lines and instrument attachments.

4.4.2 CONSTRUCTION INSPECTION

The coolant piping for each loop was shipped to the field in eight subassemblies. The loops were then assembled in the field. In order to accommodate the small fabricating and field installation tolerances, a number of the subassemblies were fabricated with excess length. Thus, the final fitting of the coolant piping was accomplished in the field. The ends with excess length were field machined. All carbon steel-to-carbon steel field welds were back- clad with stainless steel following removal of the backing rings. Consumable inserts were used in some stainless-to-stainless welds, such as those in the surge line.

Welding of the auxiliary piping to RCS nozzles was done to the same standards as the main coolant piping. Consumable inserts were used in all cases.

Cleaning of reactor coolant piping and equipment was accomplished both before and after erection of various equipment. Piping and equipment nozzles required cleaning in the area of the connecting weldments. Piping and equipment are large enough for personnel entry and

were cleaned by locally applying solvents and demineralized water and by wire brush to remove trapped foreign particles.

4.4.3 INSTALLATION TESTING

The RCS was hydrostatically tested in accordance with USAS B31.7 February 1968 Draft, including June 1968 Errata, Nuclear Power Piping Code. The test pressure was applied to all parts of the RCS up to and including means of isolation from auxiliary systems, such as valves and blank flanges. The hydrostatic test was performed at a temperature above the design transition temperature (DTT).

The RCS relief valves were inspected and shop tested in accordance with Section III of the ASME Code. The pressure relief setting was made during the shop test.

In changing out the TMI-1 steam generators, the vendor (AREVA) of the new steam generators performed a primary side hydrostatic test in accordance with ASME Section III, 2001 Edition through 2003 Addenda, prior to delivery of the new steam generators to the site. Nondestructive Examination (NDE) was performed of the completed RCS installation welds in accordance with Section III of the ASME Code, which was then followed by RCS pressure testing in accordance with Section XI of the ASME Code during plant heatup.

4.4.4 FUNCTIONAL TESTING

Prior to initial fuel loading, the functional capabilities of the RCS components were demonstrated at operating pressures and temperatures. Measurement of pressures, flow, and temperatures were recorded for various system conditions. Operation of reactor coolant pumps, pressurizer heaters, pressurizer spray system control rod drive mechanism, and other RCS equipment were demonstrated. For descriptions of the various functional tests, refer to Chapter 13.

4.4.5 MATERIAL IRRADIATION SURVEILLANCE

Initially, surveillance specimens of the reactor vessel beltline shell course materials were installed in the reactor vessel in accordance with ASTM Specification E 185-66, Recommended Practice for Surveillance Tests on Structural Materials in Nuclear Reactors. The types of specimens in the surveillance capsules are Charpy V-notch (Type A), and tensile specimens. Materials used in fabrication of the surveillance specimens are two beltline shell course base metals, heat-affected zones from the base metals, and the beltline mid-plane circumferential weld. Also included in the capsules are dosimeters, temperature monitors, and correlation material.

Reference 21 describes the eight surveillance capsules and their original locations in the TMI-1 reactor vessel. However, after it was discovered that surveillance capsule holder tubes in the reactor were damaged, the integrated reactor vessel material surveillance program was designed, as described in Reference 22. The program is based on surveillance capsules from a reactor with damaged holders being irradiated in other reactors with similar design and operating characteristics.

The historical status of the six TMI-1 plant specific capsules is as follows:

Capsule	Historical Status
TMI1-A	Standby capsule removed from TMI-2 in 1986. Capsule opened and examined to help revalidate Owner Group Capsule for reinsertion to continue their irradiation program. (Ref. 33) The Charpy specimens from this capsule are held in storage. Contained WF-25 weld metal and had no compacts.
TMI1-B	Capsule was irradiated in Crystal River (CR-3) and removed at the end of CR-3 cycle 7 and is stored and not analyzed since it contains no weld metal or compacts and is a backup capsule.
TMI1-C	Second capsule removed and tested. Irradiated at CR-3 and removed in April 1985. Reported on in BAW-1901, March 1986. (Ref. 24) Contained WF-25 weld metal and had no compacts. The broken Charpy specimens are being held in storage.
TMI1-D	Capsule was inserted in CR-3 at the end of cycle 8 and removed at the end of cycle 11 (Fall 1999). It is not being tested since it is a backup capsule, held in storage.
TMI1-E	First TMI-1 capsule removed and tested. Irradiated for one cycle of operation. Reported on in BAW-1439, January 1977. (Ref. 23) Contained WF-25 weld metal and had no compacts. The broken Charpy specimens are being held in storage.
TMI1-F	Capsule has been irradiated in CR-3, removed at the end of CR-3 cycle 8 and is stored since it is a backup capsule.

Program Description

The capsule insertion, removal and testing program is described in BAW-1543, "Master Integrated Reactor Vessel Surveillance Program." (Ref. 22) Review and approval of Reference 22 by the NRC is performed whenever changes are made.

Dosimetry including solid state track recorders has been located external to the TMI-1 reactor vessel to enable more accurate reactor vessel fluence data measurement. The dosimetry hardware is installed in the source calibration tube located in the out-of-core instrumentation well which also houses NI-4 and NI-8. Fluence levels in the pressure vessel will be extrapolated from the fluence levels measured in the cavity. This data should provide more accurate fluence measurements for which the RCS operating pressure and temperature limits can be established under 10CFR50, Appendix G and H.

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