5 REACTOR COOLANT SYSTEM AND CONNECTED SYSTEMS

5.1 SUMMARY DESCRIPTION

5.1.1 GENERAL

The reactor coolant system, shown in Drawings 33013-1258 and 33013-1260, consists of two identical heat transfer loops connected in parallel to the reactor vessel. Each loop contains a circulating pump and a steam generator. The system also includes a pressurizer, pressurizer relief tank, connecting piping, and instrumentation necessary for operational control. The pressurizer is connected to the B loop. Auxiliary system piping connections into the reactor coolant piping are provided as necessary.

Pressure in the system is controlled by the pressurizer, where water and steam pressure is maintained through the use of electrical heaters and sprays. Steam can either be formed by the heaters or condensed by a pressurizer spray to minimize pressure variations due to contraction and expansion of the coolant. Spring-loaded steam safety valves and Pressurizer Power Operated Relief Valves (PORV) are connected to the pressurizer and the discharge to the pressurizer relief tank, where discharged steam is condensed and cooled by mixing with water.

Major components which are located inside the containment are indicated in Drawings 33013-1258 and 33013-1260 by the containment boundary. The intersection of a process line with this boundary indicates a functional penetration.

Reactor coolant system design data are listed in Tables 5.1-1 through 5.1-3.

5.1.2 PERFORMANCE OBJECTIVES

The reactor coolant system transfers the heat generated in the core to the steam generators, where steam is generated to drive the turbine generator. Demineralized water is circulated at the flow rate and temperature that are consistent with achieving the reactor core thermal-hydraulic performance presented in Chapter 4. The water also acts as a neutron moderator and reflector, and as a solvent for the neutron absorber used in chemical shim control.

The reactor coolant system provides a boundary for containing the coolant under operating temperature and pressure conditions. It serves to confine radioactive material and limits to acceptable values its uncontrolled release to the secondary system and other parts of the plant. During transient operation, the heat capacity of the system attenuates thermal transients that are generated by the core or extracted by the steam generators. The reactor coolant system accommodates coolant volume changes within the protection system criteria.

By appropriate selection of the inertia of the reactor coolant pumps, the thermal-hydraulic effects are reduced to a safe level during the pump coastdown which would result from a loss-of-flow situation. The layout of the system ensures the natural circulation capability following a loss of flow to permit plant cooldown without overheating the core.

5.1.3 DESIGN CRITERIA

The design criteria discussed in Sections 5.1.3.1 through 5.1.3.9 were used during the licensing of Ginna Station. They represent the Atomic Industrial Forum (AIF) version of proposed criteria issued by the AEC for comment on July 10, 1967. Conformance with the General Design Criteria (GDC) of 10 CFR 50, Appendix A, is discussed in Section 5.1.3.10.

The following design criteria apply to the reactor coolant system.

5.1.3.1 Quality Standards

CRITERION: Those systems and components of reactor facilities which are essential to the prevention, or the mitigation of the consequences, of nuclear accidents which could cause undue risk to the health and safety of the public shall be identified and then designed, fabricated, and erected to quality standards that reflect the importance of the safety function to be performed. Where generally recognized codes and standards pertaining to design, materials, fabrication, and inspection are used, they shall be identified. Where adherence to such codes or standards does not suffice to assure a quality product in keeping with the safety function, they shall be supplemented or modified as necessary. Quality assurance programs, test procedures, and inspection acceptance criteria to be used shall be identified. An indication of the applicability of codes, standards, quality assurance programs, test procedures, and inspection acceptance criteria used is required. Where such items are not covered by applicable codes and standards, a showing of adequacy is required (AIF-GDC 1).

The reactor coolant system is of primary importance with respect to its safety function in protecting the health and safety of the public.

Quality standards of material selection, design, fabrication, and inspection conform to the applicable provisions of recognized codes and good nuclear practice (Section 5.2.1.2). Details of the quality assurance programs, test procedures, and inspection acceptance levels are given in Section 5.2.3. Particular emphasis is placed on quality assurance in the selection of reactor vessel materials that have properties which are uniformly within tolerances appropriate to the application of the design methods of the code.

5.1.3.2 Performance Standards

CRITERION: Those systems and components of reactor facilities which are essential to the prevention or to the mitigation of the consequences of nuclear accidents which could cause undue risk to the health and safety of the public shall be designed, fabricated, and erected to performance standards that will enable such systems and components to withstand, without undue risk to the health and safety of the public the forces that might reasonably be imposed by the occurrence of an extraordinary natural phenomenon such as earthquake, tornado, flooding condition, high wind or heavy ice. The design bases so established shall reflect: (A) appropriate consideration of the most severe of these natural phenomena that have been officially recorded for the site and the surrounding area and (B) an appropriate margin for withstanding forces greater than those recorded to reflect uncertainties about the historical data and their suitability as a basis for design (AIF-GDC 2).

All piping, components, and supporting structures of the reactor coolant system are designed as Seismic Category I equipment, i.e., they are capable of withstanding the following stresses with no loss of function:

A. Code-allowable working stresses for the design seismic ground acceleration.

B. The maximum potential seismic ground acceleration acting in the horizontal and vertical direction simultaneously.

Details are given in Section 5.4.11.

The reactor coolant system is located in the containment, the design of which, in addition to being a Seismic Category I structure, also considers accidents or other applicable natural phenomena. Details of the containment design are given in Sections 3.8 and 6.2.

5.1.3.3 Records Requirements

CRITERION: The reactor licensee shall be responsible for assuring the maintenance throughout the life of the reactor of records of the design, fabrication, and construction of major components of the plant essential to avoid undue risk to the health and safety of the public (AIF-GDC 5).

Records of the design, fabrication, and construction of the major reactor coolant system components are to be maintained throughout the life of the plant.

5.1.3.4 Missile Protection

CRITERION: Adequate protection for those engineered safety features, the failures of which could cause an undue risk to the health and safety of the public, shall be provided against dynamic effects and missiles that might result from plant equipment failures (AIF-GDC 40).

The dynamic effects during blowdown following a loss-of-coolant accident are evaluated in the detailed layout and design of the high-pressure equipment and barriers which afford missile protection. Fluid and mechanical driving forces are calculated and consideration is given to possible damage due to fluid jets and secondary missiles which might be produced.

The steam generators are supported, guided, and restrained in a manner which prevents rupture of the steam side of a generator, the steam lines, and the feedwater piping as a result of forces created by a reactor coolant system pipe rupture. These supports, guides, and restraints also prevent rupture of the primary side of a steam generator as a result of forces created by a steam or feedwater line rupture.

The mechanical consequences of a pipe rupture are restricted by design such that the functional capability of the engineered safety features is not impaired.

5.1.3.5 Reactor Coolant Pressure Boundary

CRITERION: The reactor coolant pressure boundary shall be designed, fabricated and constructed so as to have an exceedingly low probability of gross rupture or significant uncontrolled leakage throughout its design lifetime (AIF-GDC 9).

The reactor coolant system, in conjunction with its control and protective provisions, is designed to accommodate the system pressures and temperatures attained under all expected modes of plant operation or anticipated system interactions, and maintain the stresses within applicable code stress limits.

Fabrication of the components which constitute the pressure-retaining boundary of the reactor coolant system is carried out in strict accordance with the applicable codes. In addition, there are areas where equipment specifications for reactor coolant system components go beyond the applicable codes. Materials of construction were chosen to lessen the probability of gross leakage or failure. Details are given in Section 5.2.3.

The materials of construction of the pressure-retaining boundary of the reactor coolant system are protected by control of coolant chemistry from corrosion phenomena which might otherwise reduce the system structural integrity during its service lifetime.

System conditions resulting from anticipated transients or malfunctions are monitored and appropriate action is automatically initiated to maintain the required cooling capability and to limit system conditions so that continued safe operation is possible.

The system is protected from overpressure by means of pressure-relieving devices, as required by Section III of the ASME Boiler and Pressure Vessel Code. The Low Temperature Overpressure Protection (LTOP) System is also provided, together with operating precautions to minimize operation under undesirable conditions. (See Section 5.2.2.)

Isolable sections of the system are provided with overpressure-relieving devices discharging to closed systems such that the system code-allowable relief pressure within the protected section is not exceeded.

5.1.3.6 Monitoring Reactor Coolant Leakage

CRITERION: Means shall be provided to detect significant uncontrolled leakage from the reactor coolant pressure boundary (AIF-GDC 16).

Positive indications in the control room of leakage of coolant from the reactor coolant system to the containment are provided by equipment which permits continuous monitoring of containment air activity (R-11 and R-12) and humidity, containment sump A level (LT-2039 and LT-2044), and of runoff from the condensate collection system under the cooling coils of the containment air recirculation (CRFC) units. This equipment provides indication of normal background which is indicative of a basic level of leakage from primary systems and components. Any increase in the observed parameters is an indication of change within the containment and the equipment provided is capable of monitoring this change. The basic design criterion is the detection of deviations from normal containment environmental conditions including air particulate activity, radiogas activity, humidity, condensate runoff, and the liquid inventory in the process systems and containment sump A.

Further details are supplied in Section 5.2.5.

5.1.3.7 Reactor Coolant Pressure Boundary Capability

CRITERION: The reactor coolant pressure boundary shall be capable of accommodating without rupture the static and dynamic loads imposed on any boundary component as a result of an inadvertent and sudden release of energy to the coolant. As a design reference, this sudden release shall be taken as that which would result from a sudden reactivity insertion such as rod ejection (unless prevented by positive mechanical means), rod dropout, or cold water addition (AIF-GDC 33).

The reactor coolant boundary is shown to be capable of accommodating, without further rupture, the static and dynamic loads imposed as a result of a sudden reactivity insertion such as a rod ejection. The rod ejection accident is described in Section 15.4.5.

The operation of the reactor is such that the severity of an ejection accident is inherently limited. Since control rod clusters are used to control load variations only and core depletion is followed with boron dilution, only the rod cluster control assemblies in the controlling groups are inserted in the core at power; at full power these rods are only partially inserted. A rod insertion limit monitor is provided as an administrative aid to the operator to ensure that this condition is met.

By using the flexibility in the selection of control rod groupings, radial locations, and position as a function of load, the design limits the maximum fuel energy for the highest worth ejected rod to a value which precludes any resultant damage to the primary system pressure boundary, i.e., gross fuel dispersion in the coolant and possible excessive pressure surges.

The failure of a rod mechanism housing causing a control rod to be rapidly ejected from the core is evaluated as a theoretical, though not a credible accident. While limited fuel damage could result from this hypothetical event, the fission products are confined to the reactor coolant system and the reactor containment. The environmental consequences of rod ejection are less severe than from the postulated loss-of-coolant accident, for which public health and safety is shown to be adequately protected.

5.1.3.8 Reactor Coolant Pressure Boundary Rapid Propagation Failure Prevention

CRITERION: The reactor coolant pressure boundary shall be designed and operated to reduce to an acceptable level the probability of a rapidly propagating type failure. Consideration is given (A) to the provisions for control over service temperature and irradiation effects which may require operational restrictions, (B) to the design and construction of the reactor pressure vessel in accordance with applicable codes, including those which establish requirements for absorption of energy within the elastic strain energy range and for absorption of energy by plastic deformation and (C) to the design and construction of reactor coolant pressure boundary piping and equipment in accordance with applicable codes (AIF-GDC 34).

The reactor coolant pressure boundary is designed to reduce to an acceptable level the probability of a rapidly propagating type failure.

In the core region of the reactor vessel it is expected that the notch toughness of the material will change as a result of fast neutron exposure. This change is evidenced as a shift in the nil ductility transition temperature which is factored into the operating procedures in such a manner that full operating pressure is not obtained until the affected vessel material is above the now higher design transition temperature and in the ductile material region. The pressure during startup and shutdown at the temperature below nil ductility transition temperature is maintained below the threshold of concern for safe operation.

The design transition temperature is a minimum of nil ductility temperature plus 60°F and dictates the procedures to be followed in the hydrostatic test and in station operations to avoid excessive cold stress. The value of the design transition temperature is increased during the life of the plant as required by the expected shift in the nil ductility transition temperature and as confirmed by the experimental data obtained from irradiated specimens of reactor vessel materials during the plant lifetime. Further details are given in Sections 5.2 and 5.3.

All pressure-containing components of the reactor coolant system are designed, fabricated, inspected, and tested in conformance with the applicable codes. Further details are given in Section 5.2.1.2.

5.1.3.9 Reactor Coolant Pressure Boundary Surveillance

CRITERION: Reactor coolant pressure boundary components shall have provisions for inspection, testing, and surveillance of critical areas by appropriate means to assess the structural and leaktight integrity of the boundary components during their service lifetime. For the reactor vessel, a material surveillance program conforming with current applicable codes shall be provided (AIF-GDC 36).

The design of the reactor vessel and its arrangement in the system provides the capability for accessibility during service life to the entire internal surfaces of the vessel and certain external zones of the vessel including the nozzle to reactor coolant piping welds and the top and bottom heads. The reactor arrangement within the containment provides sufficient space for inspection of the external surfaces of the reactor coolant piping, except for the area of pipe within the primary shielding concrete.

Monitoring of the nil ductility transition temperature properties of the core region plates, forgings, weldments, and associated heat-treated zones are performed in accordance with ASTM E185 (Recommended Practice for Surveillance Tests on Structural Materials in Nuclear Reactors). Samples of reactor vessel forging materials are retained and cataloged in case future engineering development shows the need for further testing.

The material properties surveillance program includes not only the conventional tensile and impact tests but also fracture mechanics specimens. The fracture mechanics specimens are the wedge-opening loading-type specimens. The observed shifts in nil ductility transition temperature of the core region materials with irradiation will be used to confirm the calculated limits of startup and shutdown transients.

To define permissible operating conditions below design transient temperature, a pressure range is established which is bounded by a lower limit for pump operation and an upper limit that satisfies reactor vessel stress criteria. To allow for thermal stresses during heatup or cooldown of the reactor vessel, an equivalent pressure limit is defined to compensate for thermal stress as a function of the rate of coolant temperature change. The reactor coolant temperature and pressure and system heatup and cooldown rates (with the exception of the pressurizer) are limited in accordance with the Pressure and Temperature Limits Report (PTLR). The allowable pressure-temperature relationships for the heatup and cooldown rates were developed using Regulatory Guide 1.99, Revision 2, and Appendix G of Section III of the ASME Boiler and Pressure Vessel Code and are discussed in the Technical Specifications and *Reference 1*.

For the pressurizer, the heatup and cooldown rates do not exceed 100°F per hr and 200°F per hr, respectively. An additional limitation is that spray cannot be used if the temperature difference between the pressurizer and the spray fluid is greater than 320°F.

Since the normal operating temperature of the reactor vessel is well above the maximum expected design transient temperature, brittle fracture during MODES 1 and 2 is not considered to be a credible mode of failure. A discussion of reactor vessel integrity under transient conditions is included in Sections 5.3.3.4 and 5.3.3.5.

5.1.3.10 Adequacy of Reactor Coolant System Design Relative to 1972 10 CFR 50, Appendix A, Criteria

The adequacy of the Ginna Station reactor coolant system design relative to the following General Design Criteria (GDC) is discussed in Section 3.1.2:

- GDC 14, Reactor Coolant Pressure Boundary.
- GDC 15, Reactor Coolant System Design.
- GDC 30, Quality of Reactor Coolant Pressure Boundary.
- GDC 31, Fracture Prevention of Reactor Coolant Pressure Boundary.
- GDC 32, Inspection of Reactor Coolant Pressure Boundary.
- GDC 34, Residual Heat Removal.

The use of the following Safety Guides is discussed in Section 1.8:

- Safety Guide 2, Thermal Shock to Reactor Pressure Vessels.
- Safety Guide 14, Reactor Coolant Pump Flywheel Integrity.

5.1.4 DESIGN CHARACTERISTICS

5.1.4.1 Design Pressure

The reactor coolant system design and operating pressure, together with the safety, power relief, and spray valves setpoints and the protection system setpoint pressures, are listed in Table 5.1-1. The design pressure allows for operating transient pressure changes. The selected design margin considers core thermal lag, coolant transport times and pressure drops, instrumentation and control response characteristics, and system relief valve characteristics.

Additional reactor coolant system piping and pressure drop data are listed in Tables 5.1-1 through 5.1-3.

5.1.4.2 Design Temperature

For each component, the design temperature is selected to be above the maximum coolant temperature under all normal and anticipated transient load conditions. The design and operating temperatures of the respective system components are discussed in Sections 5.3.2 and 5.4.

5.1.5 CYCLIC LOADS

All components in the reactor coolant system are designed to withstand the effects of cyclic loads due to reactor system temperature and pressure changes. These cyclic loads are introduced by normal unit load transients, reactor trips, and startup and shutdown operation. The number of thermal and loading cycles used for design purposes is shown in Table 5.1-4. During unit startup and shutdown, the rates of temperature and pressure changes are limited. The number of cycles for plant heatup and cooldown at 100°F/hr was selected as a conservative estimate based on an evaluation of the expected requirements. The resulting number, which averages five heatup and cooldown cycles per year, could be increased significantly; however, it is the intent to represent a conservative realistic number rather than the maximum allowed by the design.

Although loss of flow and loss of load transients are not included in the tabulation since the tabulation is only intended to represent normal design transients, the effects of these transients have been analytically evaluated and are included in the fatigue analysis for primary system components.

The reactor coolant system and its components are designed to accommodate 10% of full power step changes in plant load and 5% of full power per minute ramp changes over the range from 12.8% full power up to and including but not exceeding 100% of full power without reactor trip. The reactor coolant system will accept a complete loss of load from full power with reactor trip. In addition, the turbine bypass and steam dump system make it possible to accept a 50% rapid load reduction (200% per minute runback) from full power without a reactor trip for RCS full power T_{avg} values > 564.6°F. For RCS full power T_{avg} values greater than 570°F, a 50% step load reduction can be accommodated without a reactor trip or turbine trip. Additionally, a turbine trip below 50% power can be accepted without a reactor trip. The ability of the plant to withstand these plant transients at 1775 MWt was determined in Reference 2.

5.1.6 SERVICE LIFE

The service life of reactor coolant system pressure components depends upon the end-of-life material radiation damage, unit operational thermal cycles, quality manufacturing standards, environmental protection, and adherence to established operating procedures.

5.1.7 RELIANCE ON INTERCONNECTED SYSTEMS

The principal heat removal systems which are interconnected with the reactor coolant system are the steam and feedwater systems and the safety injection and residual heat removal systems. The reactor coolant system is dependent upon the steam generators and the steam, feedwater, and condensate systems for decay heat removal from normal operating conditions to a reactor coolant temperature of approximately 350°F. The layout of the system ensures the natural circulation capability to permit plant cooldown following a loss of all main reactor coolant pumps.

Flow diagrams of the steam and feedwater systems are shown in Drawings 33013-1231, 33013-1232, and 33013-1236. In the event that the condenser is not available to receive the steam generated by residual heat, the water stored in the feedwater system may be pumped into the steam generators and the resultant steam vented to the atmosphere. The preferred auxiliary feedwater system will supply water to the steam generators in the event that the main feedwater pumps are inoperative. The system is described in Section 10.5.

The safety injection system is described in Section 6.3. The residual heat removal system is described in Section 5.4.5.

5.1.8 SYSTEM INCIDENT POTENTIAL

The potential of the reactor coolant system as a cause of accidents is evaluated by investigating the consequences of certain credible types of component and control failures as discussed in Sections 15.1 through 15.4 and Section 15.6. Reactor coolant pipe rupture is evaluated in Section 15.6.4.

REFERENCES FOR SECTION 5.1

1. Westinghouse Electric Corporation, Rochester Gas and Electric Reactor Vessel Life Attainment Plan, March 1990.

2. Westinghouse Calculation Note, CN-SCS-05-1, "R.E. Ginna (RGE) 19.5% Uprate Program Plant Operability and Margin to Trip Analysis," Rev. 2.

3. Westinghouse Calculation Note, CN-PCWG-04-10, "Closeout of PCWG Open Items for Additional Best Estimate Performance Calculations based on Plant Data to Support the R.E. Ginna Unit 1 (RGE) Uprate Program," Rev. 0.

Table 5.1-1
REACTOR COOLANT SYSTEM PRESSURE SETTINGS

	Pressure (psig)
Design pressure	2485
Operating pressure	2235
Safety valves	2485
Power relief valves	2335
Spray valves (open)	2260
High-pressure trip	2377
High-pressure alarm	2310
Low-pressure trip	1873
Hydrostatic test pressure	3110

Table 5.1-2
REACTOR COOLANT PIPING DESIGN DATA

Reactor inlet piping, I.D., in.	27-1/2
Reactor outlet piping, I.D., in.	29
Coolant pump suction piping, I.D., in.	31
Pressurizer surge piping, in.[a]	10 - Schedule 140
Design/operating pressure, psig	2485/2235
Hydrostatic test pressure (cold), psig	3110
Design temperature, °F	650
Design temperature (pressurizer surge line), °F	680
Water volume, ft^3	552

a. Surge line fitted with a 14-in./10-in. adapter at the pressurizer.

Table 5.1-3
REACTOR COOLANT SYSTEM DESIGN PRESSURE DROP

	Pressure Drop (psi)[a]
Across pump discharge leg	1.25
Across vessel, including nozzles	42.60
Across hot leg	1.45
Across replacement steam generator	34.95
Across pump suction leg	2.85
Total pressure drop	83.10

a. Best estimate flow for 1775 MWt with T_{AVG}=573° and 0% SG tube plugging as calculated by *Reference 3*.

Table 5.1-4
THERMAL AND LOADING CYCLES

Transient Condition	Design Cycles[a]
Plant heatup at 100 °F/hr	200
Plant cooldown at 100 °F/hr	200
Plant loading at 5 % of full power per min[b]	6,460
Plant unloading at 5 % of full power per min[b]	6,460
Step load increase of 10 % of full power (but not to exceed full power)	2,000
Step load decrease of 10 % of full power	2,000
Step load decrease from 100 % to 50 % of full power	200
Partial loss of flow[c]	80
Loss of load[c]	80
Reactor trip	400
Hydrostatic test	
Pressure 3125 psia at 100 °F	5
Pressure 2500 psia at 400 °F	40

Steady-state fluctuations:

The reactor coolant average temperature for purposes of design is assumed to increase and decrease a maximum of 6 °F in 1 min. The corresponding reactor coolant pressure variation is less than 100 psig. It is assumed that an infinite number of such fluctuations will occur.

a. Estimated for equipment design purposes (40-year life) and not intended to be an accurate representation of actual transients or to reflect actual operating experience.

b. The number of cycles summarized for both plant loading and plant unloading at 5% of full power per minute is the most recent analyzed low-cycle fatigue results, as summarized in LTR-RIDA-12- 149, based on analyses performed for the baffle-former bolt replacement/inspection campaign per ECP-10-000422.

c. Not an original Ginna design basis load. Included in uprate assessments performed by Westinghouse to be consistent with the list of design transients included in WCAP-14460.

5.2 INTEGRITY OF THE REACTOR COOLANT PRESSURE BOUNDARY

5.2.1 COMPLIANCE WITH CODES

5.2.1.1 System Integrity

The reactor coolant system serves as a barrier preventing radionuclides contained in the reactor coolant from reaching the atmosphere. In the event of a fuel cladding failure the reactor coolant system is the primary barrier against the uncontrolled release of fission products. By establishing a system pressure limit, the continued integrity of the reactor coolant system is ensured. Thus, the safety limit of 2735 psig (110% of design pressure) has been established. This represents the maximum transient pressure allowable in the reactor coolant system under the ASME Code, Section III, for MODES 1 and 2 and anticipated transient events. Reactor coolant system pressure settings are given in Table 5.1-1.

Release of activity into the reactor coolant in itself does not constitute a significant hazard. Activity in the coolant could constitute a significant hazard only if the reactor coolant system barrier is breached, and then only if the coolant contains excessive amounts of activity which could be released to the environment. The chemical and volume control system maintains primary reactor coolant activity within acceptable levels, as defined in the Technical Specifications.

A rupture of a steam generator tube would allow reactor coolant to enter the secondary system. In this event, a portion of the reactor coolant system gaseous activity could be released to the atmosphere. The radiological consequences of the event are discussed in Section 15.6.3.

As part of the design control on materials, Charpy V-notch toughness test curves were conducted for all ferritic material used in fabricating pressure parts of the reactor vessel and pressurizer to provide assurance for hydrotesting and operation in the ductile region at all times. For the replacement steam generators (RSGs) pressure boundary materials comply with ASME Section II and III requirements. The RT NDT (Reference Temperature for Nil-Ductility Transition Temperature) is used to specify the RSG material toughness. This temperature for each RSG pressure boundary plate, forging or weld is equal to or less than 0°F; typically these range from -70°F to -20°F. This provides assurance that these materials remain ductile for hydrotesting and operation at all times. In addition, drop-weight tests were performed on the reactor vessel material. Reactor vessel materials are discussed in Section 5.3.1. Reactor coolant pressure boundary materials are discussed in Section 5.2.3.

As an assurance of system integrity, all components in the system were hydro-tested at 3110 psig prior to initial operation.

As part of the Systematic Evaluation Program (SEP) the NRC evaluated, in part, the stresses in reactor coolant system components under normal and accident conditions. In the NRC Safety Evaluation Report (*Reference 1*) it was concluded that the control rod drive mechanism, reactor coolant pumps, steam generator and tube supports, and pressurizer and reactor vessel supports were acceptably designed, with the stress analysis results within established limits.

5.2.1.2 Codes and Classifications

5.2.1.2.1 Code Requirements

All pressure-containing components of the reactor coolant system were originally designed, fabricated, inspected, and tested in conformance with the applicable codes listed in Table 5.2-1.

As part of the SEP, the codes, standards, and classifications to which the station was built were compared to current code requirements. It was generally concluded that changes between original and current code requirements do not affect the safety functions of the systems and components reviewed. Details of the review, which includes the reactor coolant system are presented in Section 3.2.

The reactor coolant system is classified as Seismic Category I, requiring that there will be no loss of function of such equipment in the event of the assumed maximum potential ground acceleration acting in the horizontal and vertical directions simultaneously, when combined with the primary steady state stresses.

Commencing in 1979, RG&E performed a reanalysis of Class I piping systems including the reactor coolant system for the seismic upgrade program. The analytical procedure used for the piping reanalysis is described in Section 3.7.3.7.5. The piping and thermal stresses were calculated using the formulas given in ANSI 31.1-1973, 1973 Summer Addenda requirements. The piping reanalysis is discussed in Section 3.9.2.1.8.

5.2.1.2.2 Quality Control

Quality control techniques used in the fabrication of the reactor coolant system were equivalent to those used in manufacture of the reactor vessel which conforms to Section III of the ASME Code.

Nuclear Piping Code B31.7 is derived from ASME III criteria. Thus, the added quality assurance requirements by Westinghouse to USAS B31.1.0-1967 procured reactor coolant piping ensured that the quality level of a Westinghouse plant was comparable to that of the Nuclear Piping Code USAS B31.7 as itemized below:

A. The material specifications were ASTM specifications approved for nuclear use in the various code cases.

B. The reactor systems materials were nondestructively examined to the levels required of Class A vessels - the same levels set forth in USAS B31.7.

C. Welding procedures and welders were required to be qualified to the requirements of Section IX of the ASME Code. The same requirement prevails in USAS B31.7.

D. All butt welds were examined to the same standards required in USAS B31.7.

E. All nozzle welds were required to be radiographically examined when the branch weld was in excess of 2-in. pipe size. This requirement exceeds that of USAS B31.7.

F. All nozzle, girth, and longitudinal welds were required to be liquid penetrant examined. This requirement is equivalent to USAS B31.7.

G. Hydrostatic testing was performed in completed systems. This requirement is equivalent to USAS B31.7.

5.2.1.2.3 Field Erection Procedures

Field erection and welding procedures were governed by Westinghouse specifications, which ensured that the field fabrication resulted in the same quality consistent with that exercised in the shop fabrication of the same piping. In these specifications for shop fabrication and field erection were references to portions of the ASME Code (Sections III, VIII, and IX), USAS Pressure Piping Code (B31.1) and Nuclear Code Cases N-7 and N-10, and ASTM Standards, as well as a number of Westinghouse documents.

During the erection, Westinghouse onsite personnel continually monitored all operations to ensure conformance to specifications, regulatory codes, and good construction practices. Adequate records are maintained onsite or at Westinghouse and include radiography reports and other nondestructive testing reports.

5.2.1.3 Seismic Loads

The seismic loading conditions were initially established by the design earthquake and maximum potential earthquake. The former was selected to be typical of the largest probable ground motion based on the site seismic history. The latter was selected to be the largest potential ground motion at the site based on seismic and geological factors and their uncertainties.

For the design earthquake loading condition, the nuclear steam supply system was designed to be capable of continued safe operation. Therefore, for this loading condition critical structures and equipment needed for this purpose are required to operate within normal design limits. The seismic design for the maximum potential earthquake was intended to provide a margin in design that ensures capability to shut down and maintain the nuclear facility in a safe condition. In this case, it was only necessary to ensure that the reactor coolant system components do not lose their capability to perform their safety function. This had come to be referred to as the no-loss-of-function criteria and the loading condition as the no-loss-of-function earthquake loading condition.

The analytical method employed in the design is described in Section 3.7 for Seismic Category I structures and components. The natural periods necessary for the determination of the loads were obtained by physical model testing.

The loading combinations and associated stress limits used for the piping systems which are part of the Seismic Piping Upgrade Program are discussed in Section 3.9.2.1.8.

The criteria adopted for allowable stresses and stress intensities in vessels and piping subjected to normal loads plus seismic loads are defined in Sections 3.9.2 and 5.4.11. These criteria ensure the integrity of the reactor coolant system under seismic loading.

For the combination of normal and design earthquake loadings, the stresses in the support structures are kept within the limits of the applicable codes.

For the combination of normal and no-loss-of-function earthquake loadings, the stresses in the support structures are limited to values as necessary to ensure their integrity and to contain the stresses in the reactor coolant system components within the allowable limits as previously established.

As part of the Ginna Station SEP the reactor coolant system has been reevaluated for the design-basis earthquake (safe shutdown earthquake) loadings wherein the ground acceleration is 0.2g. This reevaluation is discussed in Sections 3.7, 3.8, and 3.9.

5.2.2 OVERPRESSURIZATION PROTECTION

5.2.2.1 Normal Operation

During MODES 1, 2, and 3, the reactor coolant system is protected against overpressure by safety valves located on the top of the pressurizer. The safety valves on the pressurizer are sized to prevent system pressure from exceeding the design pressure by more than 10%, in accordance with Section III of the ASME Code. The capacity of the pressurizer safety valves is determined from considerations of (1) the Reactor Trip System (RTS) and (2) accident or transient conditions which may potentially cause overpressure.

The combined capacity of the safety valves is equal to or greater than the maximum surge rate resulting from complete loss of load without a direct reactor trip or any other control, except that the safety valves on the secondary plant are assumed to open when the steam pressure reaches the secondary plant safety valve setting. Details of the analysis are reported in Section 15.2.2. The pressurizer relief discharge system and safety valves are described in Sections 5.4.8.1 and 5.4.10.1.

5.2.2.2 Low Temperature Overpressure Protection (LTOP) System

Low temperature reactor vessel overpressure protection is provided by the two pressurizer power operated relief valves (PORVs) (Section 5.4.10) with a low-pressure setpoint as specified in the Pressure and Temperature Limits Report (PTLR). Whenever the reactor coolant system cold leg temperature is below the temperature setpoint specified for LTOP in the PTLR or the residual heat removal system is in operation, the low-pressure setpoint is manually enabled from the control room. Pressure transients caused by mass addition or heat addition are terminated below the limits of 10 CFR 50, Appendix G, as amended by ASME Code Case Cases N-640 and N-588, by automatic operation of the pressurizer power operated relief valves (PORVs). The system is designed to protect the reactor coolant system pressure boundary from the effects of operating errors during MODES 4, 5, and 6 (as applicable in the Technical Specifications) when the reactor coolant system is in a water-solid condition. The system also supplies protection for the residual heat removal system from overpressurization. The following sections give a more detailed discussion of the Low Temperature Overpressure Protection (LTOP) System.

5.2.2.2.1 Design Bases

The basic purpose of the Low Temperature Overpressure Protection (LTOP) System is to prevent reactor vessel pressure in excess of 10 CFR 50, Appendix G limits (ASME Code Cases N-640 and N-588). Specific criteria for system performance are:

5.2.2.2.3.1 Operator Action: No credit can be taken for operator action for 10 minutes after the opera- tor is aware of a transient.

5.2.2.2.3.1 Single Failure: The system must be designed to relieve the pressure transient given a single failure in addition to the failure that initiated the pressure transient.

5.2.2.2.3.1 Testability: The system must be testable on a periodic basis consistent with the systems employment.

5.2.2.2.3.1 Seismic Criteria: The system safety function is met by equipment categorized as Seismic Category I. The basic objective is that the system should not be vulnerable to a common failure that would both initiate a pressure transient and disable the overpressure mitigating system. Such events as loss of instrument air and loss of offsite power must be considered.

Two kinds of pressure transients are considered:

1. Mass input transients from injection sources such as charging pumps, safety injection pumps, or safety injection accumulators.
2. Heat input transients from sources such as steam generators or decay heat.

On Westinghouse designed plants, a common cause of overpressure transients is isolation of the letdown path (letdown during low-pressure operations is via a flow path through the residual heat removal system). Thus, isolation of the residual heat removal system can initiate a pressure transient if a charging pump is left running. Although other transients occur with lower frequency, those which result in the most rapid pressure increases are of main concern. The most limiting mass input transient is the charging-letdown mismatch with three charging pumps left running with letdown completely isolated. The most limiting thermal expansion transient is the start of a reactor coolant pump with a 50°F temperature difference between the water in the reactor vessel and the water in the steam generator.

The NRC considers the pressurizer power operated relief valve (PORV) with a manually enabled low-pressure setpoint to be an acceptable overpressure mitigating system. Detailed information on system design is contained in *References 2 through 4*.

5.2.2.2.2 System Description

The "Reference Mitigating System" concept developed by Westinghouse and the Westinghouse Owner's Group was originally adopted by Rochester Gas and Electric Corporation. This concept is acceptable to Ginna LLC. The actuation circuitry of the pressurizer power operated relief valves (PORVs) requires a low-pressure setpoint (setpoint provided in the Pressure and Temperature Limits Report (PTLR)) during startup and shutdown conditions. The low-pressure pressurizer power operated relief valve (PORV) actuation circuitry uses multiple pressure sensors, power supplies, and logic trains to improve system reliability. Each of the two pressurizer power operated relief valves (PORVs) is manually enabled using

two keylock switches, one to line up the nitrogen supply and the other to enable the low-pressure setpoint. When the reactor vessel is at low temperatures with the overpressure protection system enabled, a pressure transient is terminated below the Appendix G limit (ASME Code Cases N-640 and N-588) by automatic opening of the pressurizer power operated relief valves (PORVs). An enabling alarm monitors the reactor coolant system temperature, the position of the keylock switches (two per channel), and the upstream isolation valve position.
The overpressure protection system is required to be in operation during plant cooldown prior to reaching the temperature limit specified in the PTLR or on initiation of the residual heat removal system and it is disabled prior to exceeding 350°F during plant heatup. The enabling alarm alerts the operator in the event the reactor coolant system temperature is below the PTLR temperature limit and overpressure protection system valve or switch alignment has not been completed.

The Ginna pressurizer power operated relief valves (PORVs) are spring closed and air or nitrogen opened. Each of the two pressurizer power operated relief valves (PORVs) receives actuating gas from either the plant instrument air system or a backup nitrogen accumulator; however, only nitrogen is used for Low Temperature Overpressure Protection (LTOP) conditions. The accumulators are sized to provide sufficient actuating nitrogen for 10 min of pressurizer power operated relief valve (PORV) operation (about 40 cycles) without operator action during the most limiting transient and a loss of the plant instrument air system. Low-pressure alarms are installed in the control room to alert the operator to a low nitrogen accumulator pressure condition. See Drawing 33013-1263. Performance of secondary side hydrostatic tests are permitted without the pressurizer power operated relief valves (PORVs) or a reactor coolant system vent ≥ 1.1 sq. in. operable, however, no safety injection pump may be capable of injecting into the reactor coolant system during the tests.

An alarm monitors the position of the pressurizer power operated relief valve (PORV) isolation valves (515 and 516), along with the low setpoint enabling switch, to ensure that the overpressure mitigating system is properly aligned for shutdown conditions. An overpressure alarm which incorporates two setpoints is also provided. One setpoint is variable and follows the limit specified in the PTLR. The other setpoint alarms at a preprogrammed differential pressure. Both setpoints alarm and light on the plant process computer system.

The installed pressure and temperature instrumentation at Ginna Station will provide a permanent record over the full range of both pressure and temperature.

5.2.2.2.3 System Evaluation

5.2.2.2.3.1 General

Generic Letter 88-11, "NRC Position on Radiation Embrittlement of Reactor Vessel Materials and Its Impact on Plant Operations," required each licensee to reevaluate the effect of neutron radiation on reactor vessel material using the methods described in Regulatory Guide 1.99, Revision 2. The pressure-temperature limits resulting from the implementation of Regulatory Guide 1.99, Revision 2, required the reevaluation of the pressurizer power operated relief valve (PORV) setpoint. The setpoint reevaluation was performed by Westinghouse and is documented in *Reference 5*. This evaluation was superseded by *Reference 15* which incorporated the characteristics of the replacement steam generators, and the approval to use ASME

Code Case N-514 for Ginna. The use of ASME Code Case N-514 was disallowed by License Amendment 106. ASME Code Cases N-640 and N-588 are now used in place of Code Case N-514. The Appendix G limits were updated in *Reference 29*, and incorporated into *Reference 15*.

The Low Temperature Overpressure Protection system (LTOP) transient analyses were performed using the RELAP5/MOD2 B&W Version 20 (*Reference 16*) computer code. The plant model that was employed for the LTOP analyses is described in *Reference 15*.

For the limiting mass addition case, the primary system was initialized at 60°F and 315 psig with two reactor coolant pumps running. The primary and secondary systems were decoupled since there was no heat transfer in this case. The event was initiated by starting three charging pumps with a total capacity of 180 gpm. The analysis was terminated after 10 minutes when the operator was assumed to secure charging flow. The peak RCS pressure was compared with the acceptance criteria.

For the heat addition cases, the primary system was initialized to isothermal conditions with no reactor coolant flow. The secondary and primary fluid in the steam generators were initialized at a temperature 50 degrees above the primary system. The transient was initiated by starting a reactor coolant pump in the loop that contains the pressurizer. The analysis was run until the peak pressure was obtained. The peak pressures in the reactor vessel and the RHR system were compared with the acceptance criteria.

5.2.2.2.3.2 Mass Addition Case

The mass addition case was initialized at a primary temperature of 60°F and a primary pressure of 315 psig. Using the initial pressure of 315 psig assures that the transient is well defined by the time the power operated relief valve (PORV) is actuated. Two reactor coolant pumps were assumed running and the pressurizer was water solid. It was assumed that the residual heat removal (RHR) system was removing decay heat, so it was conservatively not modeled. The event was initiated by starting three pump charging flow (180 gpm or 25 lb/sec). The analysis was run for ten minutes. The sequence of events for this case is shown in Table 5.2-7.

The peak reactor vessel pressure was 587.4 psia. The allowable pressure, according to ASME Code Cases N-640 and N-588 at 60°F is 621 psig or 635.7 psia. Therefore, there is 48.3 psi margin to the Appendix G acceptance criterion.

To compare the peak pressure in the RHR system with the acceptance criterion, the pressure drop from the hot leg to the RHR pump discharge was added to the peak hot leg pressure. This case yielded a peak RHR pressure of 663.5 psia. The peak allowable pressure in the RHR system is 674.7 psia. This results in a 11.2 psi margin to the acceptance criterion.

5.2.2.2.3.3 Heat Addition at 60°F

The most limiting heat addition case was analyzed at a primary system temperature of 60°F and a primary pressure of 315 psig. The secondary system was assumed to be 50 degrees hotter than the primary system, so the temperature in the secondary system and the primary side of the steam generator was 110°F. Initially, the reactor coolant pumps were not running and cooling was assumed to be provided by the residual heat removal (RHR) system. The RHR system was operating. The pressurizer was water solid. There was no charging flow for this event. The event was initiated by starting the reactor coolant pump in the loop that contained the pressurizer. The transient was analyzed for 40 seconds. The sequence of events for this case is shown in Table 5.2-8.

The peak pressure in the reactor vessel for this case was 551.3 psia. The allowable pressure limit at this temperature is 635.7 psia. This yields a 84.5 psi margin. This case is the most limiting for Appendix G. (ASME Code Cases N-640 and N-588)

The peak pressure in the RHR System was 650.0 psia as compared with an acceptance criterion of 674.7 psia, for a margin of 24.7.

5.2.2.2.3.4 Heat Addition at 320°F

The heat addition case was also analyzed with steam generator secondary system temperatures of 370°F. This temperature is the maximum temperature, including instrument uncertainty, at which both reactor coolant pumps can be stopped. The primary system was assumed to be 50 degrees colder than the secondary system, so the temperature in the primary system was 320°F. Initially, the reactor coolant pumps were not running and cooling was assumed to be provided by the residual heat removal (RHR) system. The RHR system was operating. The pressurizer was water solid. There was no charging flow for this event. The transient was initiated by starting the reactor coolant pump in the loop that contained the pressurizer.

The peak pressure in the reactor vessel for this case was 563.8 psia. The allowable reactor vessel pressure limit at this temperature is > 2400 psia. This yields a > 1836 psi margin.

The peak pressure in the RHR system was 655.7 psia as compared with an acceptance criterion of 674.7 psia, for a margin of 19.0 psia.

5.2.2.2.3.5 Administrative Controls

To limit the magnitude of postulated pressure transients to within the bounds of the analysis, a defense-in-depth approach is adopted using administrative controls. Specific conditions required to ensure that the plant is operated within the bounds of the analysis are described in the bases for Technical Specification LCO 3.4.12.

A number of provisions for prevention of pressure transients are also contained in the Ginna operating procedures. These procedures require that an acceptable reactor coolant system temperature profile be achieved prior to startup of a reactor coolant pump with the reactor coolant system in a water-solid condition. In addition, plant shutdown and cooldown procedures call for one reactor coolant pump to be run until the reactor coolant system temperature has been lowered to 160°F, thus reducing the possibility of a significant reactor coolant system temperature asymmetry.

Also, plant procedures restrict water-solid operations to only those times when absolutely necessary. For example, the plant must be maintained in a water-solid condition during reactor coolant system filling and venting operations, during hydrostatic testing of the reactor coolant system, and during plant heatup prior to bringing the reactor coolant system within water chemistry specifications.

The cooldown procedures require the safety injection signal associated with the pressurizer and steam line low pressure be blocked at approximately 2000 psig. At less than 350°F psig, the high-head safety injection discharge valves to the reactor coolant system loops are shut and the high-head safety injection pumps are deenergized by placing their control switches in the "pull-stop" position. In the "pull-stop" position the safety injection pumps cannot automatically start. The safety injection pumps are not reenergized while the reactor coolant system is in a cold and shutdown condition unless special surveillance testing is in progress or a safety injection accumulator is to be filled when only one safety injection pump is energized.

The diesel-generator load and safeguards sequence test conducted during cold or MODE 6 (Refueling) shutdown operates each safeguards train (two pumps). However, the pump discharge valves are closed, the valve power supply breakers are open, and the breaker dc control fuses are removed. During other tests the safety injection pumps are prohibited from starting and, except during valve cycling tests, the discharge valves are shut.

5.2.2.2.4 Tests and Inspections

Operability of the Low Temperature Overpressure Protection (LTOP) System is verified prior to solid system, low temperature operation by use of the remotely operated isolation valve, and the enable/disable switches. The actuation circuitry is tested each MODE 6 (Refueling) outage. Testing requirements are included in the Technical Specifications.

5.2.3 REACTOR COOLANT PRESSURE BOUNDARY MATERIALS

5.2.3.1 Material Specifications

Each of the materials used in the reactor coolant system is selected for the expected environment and service conditions. The major component materials are listed in Table 5.2-2.

5.2.3.1.1 Nondestructive Examination of Materials and Components Prior to Operation

5.2.3.1.1.1 Quality Assurance Program

Table 5.2-3 summarizes the initial quality assurance program for all reactor coolant system components. In this table, all of the nondestructive tests and inspections required by Westinghouse specifications on reactor coolant system components and materials are specified for each component. All tests required by the applicable codes are included in this table. Westinghouse requirements, which were more stringent in some areas than those requirements specified in the applicable codes, are also included.

Table 5.2-3 also summarizes the quality assurance program with regard to inspections performed on primary system components. In addition to the inspections shown in Table 5.2-3, there were those that the equipment supplier performed to confirm the adequacy of material received, and those performed by the material manufacturer in producing the basic material. The inspections of reactor vessel, pressurizer, and steam generator were governed by ASME Code requirements. The inspection procedures and acceptance standards required on pipe materials and piping fabrication were governed by USAS B31.1 and Westinghouse requirements and were equivalent to those performed on ASME Code vessels.

Procedures for performing the examinations were consistent with those established in the ASME Code, Section III, and were reviewed by qualified Westinghouse engineers. These procedures were developed to provide the highest assurance of quality material and fabrication. They considered not only the size of the flaws, but equally as important, how the material was fabricated, the orientation and type of possible flaws, and the areas of most severe service conditions. In addition, the surfaces most subject to damage as a result of the heat treating, rolling, forging, forming, and fabricating processes, received a 100% surface inspection by magnetic particle or liquid penetrant testing after all these operations were completed. Although flaws in plates are inherently laminations in the center, all reactor coolant plate material is subject to shear as well as longitudinal ultrasonic testing to give maximum assurance of quality. (All forgings received the same inspection.) In addition, 100% of the material volume was covered in these tests as added assurance over the grid basis required in the code.

Westinghouse quality control engineers and RG&E engineers monitored the supplier's work, and witnessed key inspections not only in the supplier's shop but in the shops of subvendors of the major forgings and plate material. Normal surveillance included verification of records of material, physical and chemical properties, review of radiographs, performance of required tests, and qualification of supplier personnel.

5.2.3.1.1.2 Welding and Heat Treatment

Equipment specifications for fabrication required that suppliers submit the manufacturing procedures (welding, heat treating, etc.) to Westinghouse where they were reviewed by qualified Westinghouse engineers. This also was done on the field fabrication procedures to ensure that installation welds were of equal quality.

Section III of the ASME Code required that nozzles carrying significant external loads be attached to the shell by full penetration welds. This requirement was carried out in the reactor coolant piping, where all auxiliary pipe connections to the reactor coolant loop were made using full penetration welds.

Preheat requirements, nonmandatory under code rules, were performed on all weldments, including P1 and P3 materials which were the materials of construction in the reactor vessel, pressurizer, and steam generators. Preheat and postheat of weldments both serve a common purpose: the production of tough, ductile metallurgical structures in the completed weldment.

Preheating produces tough ductile welds by minimizing the formation of hard zones, whereas postheating achieves this by tempering any hard zones which may have formed due to rapid cooling. Thus, the reactor coolant system components were welded under procedures that required the use of both preheat and postheat.

5.2.3.1.2 Quality Assurance for Electroslag Welds

5.2.3.1.2.1 Piping Elbows

The 90-degree primary system elbows were electroslag welded. The following efforts were performed for quality assurance of these components:

a. The electroslag welding procedure employing one-wire technique was qualified in accordance with the requirements of ASME Code, Section IX and Code Case 1355, plus supplementary evaluations as requested by Westinghouse. The following test specimens were removed from a 5-in.-thick weldment and successfully tested. They were:

   1. Six transverse tensile bars - as welded.
   2. Six transverse tensile bars - 2050°F, H_2O quench.
   3. Six transverse tensile bars - 2050°F, H_2O quench + 750°F stress relief heat treatment.
   4. Six transverse tensile bars - 2050°F, H_2O quench, tested at 650°F.
   5. Twelve guided side bend test bars.

b. The casting segments were surface conditioned for 100% radiographic and penetrant inspections. The acceptance standards were ASTM E-186 severity level 2 (except no category D or E defectiveness was permitted) and USAS Code Case N-10, respectively.

c. The edges of the electroslag weld preparations were machined. These surfaces were penetrant inspected prior to welding. The acceptance standards were USAS Code Case N-10.

d. The completed electroslag weld surfaces were ground flush with the casting surface. Then the electroslag weld and adjacent base material were 100% radiographed in accordance with ASME Code Case 1355. Also, the electroslag weld surfaces and adjacent base material were penetrant inspected in accordance with USAS Code Case N-10.

e. Weld metal and base metal chemical and physical analyses were determined and certified.

f. Heat treatment furnace charts were recorded and certified.

5.2.3.1.2.2 Reactor Coolant Pump Casings

The Ginna reactor coolant pump casings were electroslag welded. The following efforts were performed for quality assurance of the components.

The electroslag welding procedure employing two-wire and three-wire techniques was qualified in accordance with the requirements of the ASME Code, Section IX and Code Case 1355, plus supplementary evaluations as requested by Westinghouse. The following test specimens were removed from an 8-in.-thick and from a 12-in.-thick weldment and successfully tested for both the two-wire and the three-wire techniques, respectively. They were as follows.

a. Two-wire electroslag process - 8-in.-thick weldment.

   1. Six transverse tensile bars - 750°F postweld stress relief.
   2. Twelve guided side bend test bars.

b. Three-wire electroslag process - 12-in.-thick weldment.

   1. Six transverse tensile bars - 750°F postweld stress relief.
   2. Seventeen guided side bend test bars.
   3. Twenty-one Charpy V-notch specimens.
   4. Full section macroexamination of weld and heat affected zone.
   5. Numerous microscopic examinations of specimens removed from the weld and heat affected zone regions.
   6. Hardness survey across weld and heat affected zone.

c. A separate weld test was made using the two-wire electroslag technique to evaluate the effects of a stop and restart of welding by this process. This evaluation was performed to establish proper procedures and techniques as such an occurrence was anticipated during production applications due to equipment malfunction, power outages, etc. The following test specimens were removed from an 8-in.-thick weldment in the stop-restart-repaired region and successfully tested.

   1. Two transverse tensile bars - as welded.
   2. Four guided side bend test bars.
   3. Full section macroexamination of weld and heat affected zone.

d. All of the weld test blocks in items a, b, and c above were radiographed using a 24-MeV betatron. The radiographic quality level obtained was between 0.5% to 1% (1-1T). There were no discontinuities evident in any of the electroslag welds.

   1. The casting segments were surface conditioned for 100% radiographic and penetrant inspections. The radiographic acceptance standards were ASTM E-186 severity level 2 (except no category D or E defectiveness was permitted for section thickness up to 4.5 in.) and ASTM E-280 severity level 2 for section thicknesses greater than 4.5 in. The penetrant acceptance standards were ASME Code, Section III, paragraph N-627.
   2. The edges of the electroslag weld preparations were machined. These surfaces were penetrant inspected prior to welding. The acceptance standards were ASME Code, Section III, paragraph N-627.
   3. The completed electroslag weld surfaces were ground flush with the casting surface. Then, the electroslag weld and adjacent base material were 100% radiographed in accordance with ASME Code Case 1355. Also, the electroslag weld surfaces and adjacent base material were penetrant inspected in accordance with ASME Code, Section III, paragraph N-627.
   4. Weld metal and base metal chemical and physical analyses were determined and certified.
   5. Heat treatment furnace charts were recorded and certified.

5.2.3.1.2.3 Reactor Coolant Pump Field Erection and Welding

Field erection and field welding of the reactor coolant system were performed so as to permit exact fit-up of the 31-in. I.D. closure pipe subassemblies between the steam generator and the reactor coolant pump. After installation of the pump casing and the steam generator, measurements were taken of the pipe length required to close the loop. Based on these measurements, the 31-in. I.D. closure pipe subassembly was properly machined and then erected and field welded to the pump suction nozzle and to the steam generator exit nozzle. Thus, upon completion of the installation, the system was essentially of zero stress in the installed position.

Cleaning of reactor coolant system piping and equipment was accomplished before and/or during erection of various equipment. Stainless steel piping was cleaned in sections as specific portions of the systems were erected. Pipe and units large enough to permit entry by personnel were cleaned by locally applying approved solvents (Stoddart solvent, acetone, and alcohol) and demineralized water, and by using a rotary disk sander or 18-8 wire brush to remove all trapped foreign particles. Standards for final physical and chemical cleanliness are defined in Section 14.1.1.2.2.

5.2.3.2 Compatibility With Reactor Coolant

All reactor coolant system materials that are exposed to the coolant are corrosion resistant. They consist of stainless steels and Inconel, and they are chosen for specific purposes at various locations within the system for their superior compatibility with the reactor coolant.

All external insulation of reactor coolant system components is compatible with the component materials. The cylindrical shell exterior and closure flanges to the reactor vessel are insulated with metallic reflective insulation. The closure head is insulated with low halide-content insulating material. All other external corrosion resistant surfaces in the reactor coolant system are insulated with low or halide-free insulating material as required.

The water chemistry is selected to provide the necessary boron content for reactivity control and to minimize corrosion of the reactor coolant system surface. Periodic analyses of the coolant chemical composition are performed to monitor the adherence of the system to the reactor coolant water quality listed in plant procedures. Concentration limits of lithium and lithium hydroxide as a function of boron concentration are determined from plant procedures. Maintenance of the water quality to minimize corrosion is performed by the chemical and volume control system and sampling system, which are described in Sections 9.3.4 and 9.3.2.

Generic Letter 88-05 (*Reference 6*) directed PWR licensees to have a program that addresses the corrosive effects of reactor coolant system leakage below Technical Specifications limits wherein the coolant containing dissolved boric acid comes in contact with and degrades low alloy carbon steel components. The concern is that concentrated boric acid solution or boric acid crystals, formed by evaporation of water from the leaking reactor coolant, is more corrosive than the coolant and will corrode the reactor coolant pressure boundary. The boric acid corrosion prevention program at Ginna Station addresses both reactor coolant system leaks and leaks from other systems containing boric acid that may contact any reactor coolant system carbon steel components. The program meets the intent of Generic Letter 88-05 (*Reference 7*).

5.2.4 INSERVICE INSPECTION AND TESTING OF THE REACTOR COOLANT SYSTEM PRESSURE BOUNDARY

5.2.4.1 Inservice Inspection Program

The Inservice Inspection Program for Ginna Station is designed to verify that the structural integrity of the reactor coolant pressure boundary is maintained throughout the life of the station. The program is scheduled for 10-year inspection intervals. The current 10-year inspection interval is specified in the Inservice Inspection Program document.

The inservice inspection program for the reactor vessel includes a visual examination of accessible internal surfaces, nozzles, and internal components of the reactor vessel and ultrasonic examinations of the vessel welds. The program is performed in accordance with Ginna Station procedures. The inservice inspection program for steam generator tubes was developed to meet the Ginna Technical Specifications, and the requirements of the Electric Power Research Institute PWR Steam Generator Program Guidelines. The program is described in the Ginna Station Engineering procedures. Special reporting requirements are described in the Steam Generator Program. The inservice inspection program for the reactor coolant pump flywheels was developed to meet the guidance provided in Regulatory Guide 1.14, Revision 1. The program is also described in the Ginna Station procedures, and is specified in the Inservice Inspection (ISI) Program document.

NRC Bulletin 88-09 (*Reference 8*) requested licensees to establish an inspection program to monitor thimble tube performance because of recently identified thimble tube thinning and leakage. Since no inservice inspection or testing requirements for thimble tubes existed, the NRC believed that this may have resulted in significant thimble tube degradation having gone undetected, creating a condition that may be adverse to safety. To comply with NRC Bulletin 88-09, a "Thimble Tube Inspection Program" has been established to ensure that the acceptance criterion of 65% through-wall wear is not exceeded and that appropriate corrective action is performed for any tube whose inspection indicates equal to or greater than 55% through-wall in the wear area as documented in *Reference 9*.

5.2.4.2 Inspection Areas and Components

<u>5.2.4.2.1 Accessible Components and Areas</u>

The following components and areas are available and accessible for visual and/or nondestructive examination:

1. Reactor vessel.
    a. Longitudinal and circumferential shell welds.
    b. Circumferential welds in bottom head. Replacement reactor vessel closure head provided by PCR 2001-0042 is a one piece forging.
    c. Vessel-to-flange circumferential welds.

d. Primary nozzle-to-vessel welds and inside nozzle section.

e. Penetrations, including control rod drive and instrumentation penetrations.

f. Nozzle-to-safe-end welds.

g. Closure head studs, nuts, washers, and pressure retaining bolts.

h. Integrally welded attachments.

i. Interior surface.

j. Core support structures.

k. Control rod drive housings.

2. Pressurizer.

   a. Longitudinal and circumferential welds.

   b. Nozzle-to-vessel welds and nozzle-to-vessel radiused section.

   c. Heater penetrations.

   d. Nozzle-to-safe-end welds.

   e. Bolts, studs, and nuts.

   f. Integrally welded attachments.

3. Steam Generators.

   a. Longitudinal and circumferential welds, including tubesheet-to-head or shell welds on the primary side.

   b. Nozzle-to-safe-end welds.

   c. Bolts, studs, washers, and nuts.

   d. Integrally welded attachments.

   e. Tubing.

4. Reactor Coolant Pumps.

   a. Pump casing welds.

   b. Supports.

   c. Bolts, studs, and nuts.

   d. Integrally welded attachments.

   e. Flywheel.

5. Pressure Boundary Piping.

   a. Safe-end to piping welds and safe-end in branch piping welds.

   b. Circumferential and longitudinal pipe welds.

   c. Branch pipe connection welds.

   d. Socket welds.

e. Supports.

f. Bolts, studs, and nuts.

g. Integrally welded attachments.

6. Pressure Boundary Valves.

a. Valve-body welds.

b. Supports.

c. Bolts, studs, and nuts.

d. Integrally welded attachments.

5.2.4.2.2 Accessible Areas During Refueling

The internal surface of the reactor vessel is inspected periodically using optical devices over the accessible areas. During refueling, the vessel cladding can be inspected in certain areas between the closure flange and the primary coolant inlet nozzles and, if deemed necessary by this inspection, the core barrel could be removed making the entire inside vessel surface accessible. Ultrasonic testing methods are employed as required. In order to facilitate this test program, critical areas of the reactor vessel were mapped during the fabrication phase to serve as a reference base for subsequent ultrasonic tests.

Externally, the control rod drive mechanism nozzles on the closure head, the instrument nozzles on the bottom of the vessel, and the extension spool pieces on the primary coolant outlet nozzles are accessible for visual, magnetic particle, or dye penetrant inspection during refuelings.

The closure head is examined visually during each refueling. Optical devices permit a selective visual inspection of the cladding, control rod drive mechanism nozzles, and the gasket seating surface. The knuckle transition piece, which is the area of highest stress of the closure head, also is accessible on the outer surface for inspection by visual and dye penetrant means.

The closure studs are inspected periodically using magnetic particle tests and/or ultrasonic tests. Additionally, it is possible to perform strain tests during the tensioning, which assists in verifying the material properties.

5.2.4.3 Accessibility

The considerations that are incorporated into the reactor coolant system design to permit these inspections are as follows:

A. All reactor internals are completely removable. The tools and storage space required to permit these inspections are provided.

B. The closure head is stored dry on the reactor operating deck during MODE 6 (Refueling) to facilitate direct visual inspection.

C. All reactor vessel studs, nuts, and washers are removed to dry storage during refueling.

D. Removable plugs are provided in the primary shield just above the coolant nozzles, and the insulation covering the nozzle welds is readily removable.

E. Access holes are provided in the lower internals barrel flange to allow remote access to the reactor vessel internal surfaces between the flange and the nozzles without removal of the internals.

F. A removable plug is provided in the lower core support plate to allow access for inspection of the bottom head without removal of the lower internals.

G. The storage stands that are provided for storage of the internals allow for inspection access to both the inside and outside of the structures.

H. The station that is provided for changeout of control rod clusters from one fuel assembly to another is especially designed to allow inspection of both fuel assemblies and control rod clusters.

I. The control rod mechanism is especially designed to allow removal of the mechanism assembly from the reactor vessel head.

J. Manways are provided in the steam generator, steam drum, and channel head to allow access for internal inspection.

K. A manway is provided in the pressurizer top head to allow access for internal inspection.

L. Insulation on the primary system components (except the reactor vessel) and piping (except for the penetration in the primary shield) included in the inservice inspection program is removable.

5.2.4.4 Examination Methods

The reactor coolant pressure boundary areas and components identified in Section 5.2.4.2 will be examined by the required visual, surface, or volumetric methods. These examinations will include one or a combination of visual, liquid penetrant, magnetic particle, ultrasonic, eddy-current, or radiographic examination. These methods will be in accordance with the rules of IWA-2000 of the ASME Code, Section XI as specified in the Inservice Inspection (ISI) Program document.

Steam generator tubes will be examined by a volumetric method (e.g., eddy-current) or an alternative acceptable method. In response (*References 12* and *13*) to Generic Letter 95-03 (*Reference 14*), RG&E provided information to the NRC about techniques which were used (and will be used) in the performance of eddy-current testing of the replacement steam generators. In response (*References 18* and *19*) to Generic Letter 97-05 (*Reference 20*), RG&E provided additional information to the NRC regarding steam generator inspection techniques used at Ginna Station.

Reactor coolant pump flywheels will be examined by the required surface and volumetric methods in accordance with the requirements of IWA-2200 of the ASME Code, Section XI.

The edition and addenda of the ASME Code sections cited in UFSAR Sections 5.2.4.4 through 5.2.4.8 are as specified in the Inservice Inspection (ISI) Program document.

In 1981, RG&E performed a 10-year inservice inspection of the reactor coolant pump bowl successfully utilizing the portable radiographic linear accelerator prototype MINAC, developed by the Electric Power Research Institute, and a manipulator/control system developed by RG&E. The system was placed onto the reactor coolant pump and a radiographic examination was made of the middle weld (ranging in thickness from 5 in. to 9 in.), bottom weld (ranging in thickness from 8.5 in. to 9 in.), and the top weld (ranging in thickness from 10.25 in. to 10.5 in.). A sensitivity level of 1T was obtained in most exposures and all radiographs were acceptable. Video enhancement equipment was used in conjunction with the MINAC head-mounted camera during the visual examination of the inside surface of the welds and also as an aid to verify the position of the ground weld and MINAC head alignment for each of the exposures of the three welds.

5.2.4.5 Evaluation of Examination Results

The evaluation of nondestructive examination results will be in accordance with Article IWB-3000 of the ASME Code, Section XI and the Inservice Inspection (ISI) Program document. All reportable indications will be subject to comparison with previous data to aid in their characterization and in determining their origin.

The evaluation of the nondestructive examination results from the steam generator tube examination will dictate certain action in terms of resumption of operation and corrective measures, depending on the type and extent of degradation. Specific criteria are included in the Steam Generator Program.

5.2.4.6 Repair Requirements

Repair of reactor coolant pressure boundary components will be performed in accordance with the applicable subsections of the ASME Code, Section XI. Examinations associated with repairs or replacements will meet the applicable design and code requirements described in the Inservice Inspection (ISI) Program document.

Repair of steam generator tubes that have unacceptable defects will be performed by using a tube plugging technique or sleeving. Steam generator tube and sleeve repair criteria are included in the Technical Specifications and in the Inservice Inspection Program. Repair of a reactor coolant pump flywheel that has unacceptable defects will be performed in accordance with Regulatory Guide 1.14, Revision 1 and the Inservice Inspection (ISI) Program document.

5.2.4.7 Pressure Testing

The reactor coolant system pressure test will be conducted in accordance with the Inservice Inspection (ISI) Program document.

5.2.4.8 Exemptions

In accordance with paragraphs IWB-1220 and IWC-1220 of the ASME Code, Section XI, components may be exempt from examinations where certain conditions exist. Detailed descriptions of the exemptions at Ginna Station appear in the Inservice Inspection (ISI) Plan document. The majority of exemptions cover areas where a later edition of the ASME Code provides better assurance and is more practicable; in these cases relief from the earlier version has been approved by the NRC.

5.2.5 DETECTION OF LEAKAGE THROUGH REACTOR COOLANT PRESSURE BOUNDARY

5.2.5.1 Leakage Detection Methods

The existence of leakage from the reactor coolant system to the containment, regardless of the source of leakage, is detected by one or more of the following conditions:

A. Two radiation sensitive instruments provide the capability for detection of leakage from the reactor coolant system. The containment air particulate monitor (R-11) is quite sensitive to low leak rates. The rate of leakage to which the instrument is sensitive is 0.018 gpm within 20 minutes, assuming the presence of noble gas progeny. The containment radiogas monitor (R-12) is much less sensitive but can be used as a backup to the air particulate monitor. The sensitivity range of the instrument is approximately 7 gpm within 1 hour. Operability of both monitors is addressed in the Technical Specifications.

B. An increase in containment sump A level (LT-2039 and LT-2044) and sump pump actuation monitoring are means of detecting increases in unidentified leakage and can measure approximately a 2.0 gpm leak in 1 hour. Operability of these monitors is addressed in the Technical Specifications.

C. An increase in the amount of coolant makeup water which is required to maintain normal level in the pressurizer is apparent from monitoring the volume control tank level.

D. A leakage detection system is installed which determines leakage losses from all water and steam systems within the containment including that from the reactor coolant system. This system collects and measures moisture condensed from the containment atmosphere by the cooling coils of the containment recirculation fan cooler (CRFC) units. It relies on the principle that all leakages up to sizes permissible with continued plant operation will be evaporated into the containment atmosphere. This system provides a dependable and accurate means of measuring integrated total leakage, including leaks from the cooling coils themselves which are part of the containment boundary. This system can detect leakage from approximately 1 gpm to 30 gpm within 1 hour.

E. Other alternative instruments used in leak detection are the humidity detectors. These provide a backup means of measuring overall leakage from all water and steam systems within the containment but furnish a less sensitive measure. The humidity monitoring method provides backup to the radiation monitoring methods. The sensitivity range of these instruments is from approximately 2 gpm to 10 gpm.

F. Additional indication of leakage can be obtained from the containment atmosphere temperature (TE-6031, 6035, 6036, 6037, 6038, and 6045) and pressure (PI-944) monitors.

Table 5.2-5 lists the leakage-detection systems available to monitor reactor coolant pressure boundary leakage to the containment. Table 5.2-6 lists the leakage-detection systems used for intersystem leakage.

5.2.5.1 Leakage Limitations

Reactor coolant system components are manufactured to exacting specifications which exceed normal code requirements (as outlined in Section 5.2.1.2). In addition, because of the welded construction of the reactor coolant system and the extensive nondestructive testing to which it is subjected (as outlined in Section 5.2.3), it is considered that leakage through metal surfaces or welded joints is very unlikely.

However, some leakage from the reactor coolant system is permitted by the reactor coolant pump seals. Also, all sealed joints are potential sources of leakage even though the most appropriate sealing device is selected in each case. Thus, because of the large number of joints and the difficulty of ensuring complete freedom from leakage in each case, a small integrated leakage is considered acceptable.

The Electric Power Research Institute (EPRI) established a program in 1984 that identified two improvements in valve stem packing to reduce leakage. These included replacement of woven asbestos packing with die-formed flexible graphite and the addition of live (spring) loading of packing gland followers. Ginna Station modified the valve stem packing of several valves to include these improvements.
Leakage from the reactor coolant system is collected in the containment or by other closed systems. These closed systems are the steam and feedwater system, the waste disposal system, and the component cooling water (CCW) system. Assuming the existence of the maximum allowable activity in the reactor coolant (see the Technical Specifications), the rate of 1 gpm unidentified leakage, also given in the Technical Specifications, is a conservative limit on what is allowable before the guidelines of 10 CFR 20 would be exceeded. This is shown as follows. If the reactor coolant activity is $100/\bar{E}$ $\mu Ci/cm^3$-MeV ($\bar{E}$ = average beta + gamma energy per disintegration in MeV) and 1 gpm of primary system leakage is assumed to be discharged through the air ejector, the yearly whole-body dose resulting from this activity at the site boundary, using an annual average $X/Q = 2.63 \times 10^{-6}$ sec/m^3, is 0.024 R/yr as compared with the 10 CFR 20 guideline of 0.5 R/yr.

With the limiting reactor coolant activity and assuming initiation of a 1-gpm leak from the reactor coolant system to the component cooling system, the radiation monitor in the component cooling system would annunciate in the control room and initiate closure of the vent line from the surge tank in the component cooling system, within less than 1 minute. In the case of failure of the closure of the vent line and resulting continuous discharge to the atmosphere via the component cooling surge tank vent, the resultant dose at the site boundary would be 0.024 R/yr.

Leakage directly into the containment indicates the possibility of a breach in the coolant envelope. The limitation of 1 gpm for a source of leakage not identified is sufficiently above the minimum detectable leakage rate to provide a reliable indication of leakage. The 1-gpm limit is well below the capacity of one coolant charging pump (60 gpm).

When the source of leakage has been identified, the situation can be evaluated to determine if operation can safely continue. Under these conditions, an allowable leakage rate of 10 gpm has been established which is also well within the capacity of one charging pump and makeup would be available even under the loss of offsite power condition.

5.2.5.2 Locating Leaks

Methods of leak location that can be used during plant shutdown include visual observation for escaping steam or water or for the presence of boric acid crystals near the leak. The boric acid crystals are transported outside the reactor coolant system in the leaking fluid and then left behind by the evaporation process.

Periodic reactor coolant system leakage surveillance is conducted pursuant to plant procedures.

5.2.5.3 Leakage Detection System Descriptions

5.2.5.4.1 Containment Air Particulate and Radiogas Monitor

5.2.5.4.1.1 Air Particulate Monitor

The containment air particulate monitor (R-11) is the most sensitive instrument of those available for detection of reactor coolant leakage into the containment.

This instrument is capable of detecting particulate radioactivity in concentrations as low as 5 x 10^{-10} μCi/cm^3 of containment air.

5.2.5.4.1.2 Sensitivity Assumptions

The sensitivity of the air particulate monitor to primary system leakage is determined by making the following initial assumptions:

a. Containment volume - 970,000 ft^3 = 2.7 x 10^{10} cm^3.

b. Maximum air recirculation rate - 166,800 ft^3/min.

c. Average minimum noble gas (Xe-138/Kr-88) activity in reactor coolant system is 0.05 μCi/cm^3 in a 4:1 ratio of Xe-138/Kr-88.

d. Detector sensitivity threshold 5 x 10^{-10} μCi/cm^3 of sampled air, for average beta energies of Cs-138/Rb-88.

Using the mass balance equation

$$A - Qc = V \times (dc/dt)$$ (Equation 5.2-1)

where:	A =	Leak rate (μCi/min)
	Q =	Recirculation flow (cm^3/min)
	c =	Concentration in containment (μCi/cm^3)
	V =	Containment volume (cm^3)
	t =	Time (min)

Rearranging Equation 5.2-1 gives

$dc/(A-Qc) = 1/V\ dt$ (Equation 5.2-2)

Now, for a constant leak rate, i.e., A = constant,

$d(A-Qc) = -Qdc$

$dc = d(A-Qc)/-Q$ (Equation 5.2-3)

Rearranging Equation 5.2-2 gives

$d(A-Qc)/(A-Qc) = -(Q/V)dt$ (Equation 5.2-4)

Integrate Equation 5.2-4 gives

$A - Qc = K\ e^{-Qt/V}$ (Equation 5.2-5)

where $c = C_0$ and $K = A - QC_0$ at t = O

thus

$A - Qc = (A - Q\ C_0)\ e^{-Qt/V}$

or

$c = (A/Q) - [(A/Q) - C_0]\ e^{-Qt/V}$ (Equation 5.2-6)

Equation 5.2-6 is solved assuming various leak rates of reactor coolant with a noble gas activity of $0.05\mu Ci/cm^3$ which is the minimum average Kr-88/Xe-138 activity in the reactor coolant system. The results are plotted in Figure 5.2-3.

The sensitivity indicated by Figure 5.2-3 does not take into account the following advantages or disadvantages.

Advantages

i. The air particulate monitor filter paper can be fixed; the resulting sensitivity would afford earlier detection for a given leak rate.

ii. The air recirculation rate can be lower (here we have assumed the maximum), thus giving a more rapid increase in containment air activity.

iii. Other particulate activity released in an RCS leak (eg. NA-24, Co-58, Mo-99, C-11) would increase sensitivity to leak detection.

Disadvantages

i. The effect of partition factor in regions where leakage occurs.

ii. The absence of volatile radioactive particulate (absence of iodine isotope).

5.2.5.4.1.3 Leakage Detection Threshold

The sensitivity of the air particulate monitor is greatest when baseline leakage is low, as has been demonstrated by the experience of Indian Point Unit 1, Yankee Rowe, and Dresden Unit 1. Where containment air particulate activity is below the threshold of detection (5.0 x 10^{-10} μCi/cm^3), the sensitivity of the monitor can be improved by fixing the filter paper in the monitor. In this case, there will be an accumulation of activity at the rate of flow of the sample. For example, if a sample flow rate of R_S cm^3/min is assumed, the accumulation of activity A_D at the detector will be governed by the following relationship:

$$A_D = R_S \int_0^t C(t)dt$$

(Equation 5.2-7)

where A_D is in μCi.

C(t) is given by a modified form of Equation 5.2-6 where V and Q are equal to the volume and recirculation terms applicable.

Hence,

$$A_D = R_S \left[K_1 t + K_2 \cdot e^{\frac{-Q}{V}t} \right]_0^t$$

(Equation 5.2-8)

The evaluation of the above equation for a given leakage would depend upon the characteristics and response time for a given detector.

Assuming a low background of containment air particulate radioactivity, a reactor coolant noble gas with particle progeny activity of 0.5 μCi/cm^3 (a value consistent with little or no fuel cladding leakage), and complete dispersion of the leaking radioactive gas into the containment air, the air particulate monitor is capable of detecting leaks as small as approximately 0.018 gpm (70 cm^3/min) within 20 min after they occur. If only 10% of the particulate activity is actually dispersed in the air, leakage rates of the order of 0.18 gpm (700 cm^3/min) are well within the detectable range.

For cases where baseline reactor coolant leakage falls within the detectable limits of the air particulate monitor, the instrument can be adjusted to alarm on leakage increases from two to five times the baseline value.

5.2.5.4.1.1 Radiogas Monitor

The containment radiogas monitor (R-12) is inherently less sensitive (threshold at 10^{-6} μCi/ cm^3) than the containment air particulate monitor. With typical RCS activity, R-12 is able to identify a 7 gpm leak within 1 hour from the liquid space of the RCS. A leak from the gas space or during periods of high make-up or failed fuel will increase the sensitivity to 2-4gpm. Because of the lower sensitivity, this instrument is a useful back up to the particulate monitor.

5.2.5.4.2 Humidity Detector

The humidity detection instrumentation offers another means of detection of leakage into the containment. This instrumentation has not nearly the sensitivity of the air particulate monitor but has the advantage of being sensitive to vapor originating from all sources, including the reactor coolant and steam and feedwater systems. Plots of containment air dewpoint variations above a baseline maximum established by the cooling water temperature to the air coolers should be sensitive to incremental leakage equivalent to 2.0 to 10 gpm.

The sensitivity of this method depends on cooling water temperature, containment air temperature variation, and containment air recirculation rate. The containment humidity information is displayed on the plant computer.

5.2.5.4.3 Condensate Measuring System

The principle that the condensate collected by the cooling coils matches, under equilibrium conditions, the leakage of water and steam from systems within the containment applies because conditions within the containment promote complete evaporation of leaking water from hot systems. The air and internal structure temperatures are normally held at 125°F or less, the air is dry (i.e., not saturated with water vapor), and the cooling coils provide the only significant surfaces at or below the dewpoint temperature.

The containment cooling coils are designed to remove the sensible heat generated within the containment. The resulting large coil surface area means that the exit air from the coils has a dewpoint temperature which is very nearly equal to the cooling water temperature at the air exit.

Measurement of the condensate drained from the cooling coils is made to determine collection rate and thus leak rate. About one-half hour after the occurrence of a leak, the equilibrium condition is established in which the amount of the leakage change is matched by a change in the cooling coil condensation rate.

The condensate from each of the four containment cooling coils drains to a condensate collector (drain pan) that is equipped with a standpipe that is approximately 200 in. long. The condensate collector level instrumentation provides a signal proportional to the water level in the standpipe indicating that the collector is from 0 to 100% full with an uncertainty of less than +3%. Readouts of collector water level and a hi-hi level alarm are provided in the control room. The hi-hi level alarm is actuated when the standpipe is 80% +3% full for three of the collectors and 66% +3% full for the fourth collector at which point the collector is dumped to the containment sump.

Condensate flows from approximately 1 gpm to 30 gpm can be measured by the condensate collection system. Flows less than 1 gpm can be measured by periodic observation of the level changes in the condensate collection system.

5.2.5.4.4 Liquid Inventory in Process Systems and Containment Sumps

Leaks can also be detected by unscheduled increases in the amount of reactor coolant makeup water, which is required to maintain the normal level in the pressurizer. Based on the frequency of the inventory balance, and the volume control tank level instrumentation, it is estimated that the charging system inventory method of leak detection can detect a 0.25-gpm leak.

Gross leakage will cause a rise in the containment sumps water levels. Sump A water level rise will be alarmed in the control room upon auto-start of either of the sump A pumps. Water level in containment sump B is indicated in the control room by a series of five lights actuated by redundant signal contacts evenly spaced along the height of the sump.

5.2.5.5 Leakage Detection System Evaluation

Detection of leakage from the reactor coolant pressure boundary was reviewed as part of the NRC Systematic Evaluation Program (SEP Topic V-5). The results of the review are documented in *References 10* and *11*. The review was based on the requirements of 10 CFR 50, Appendix A, General Design Criterion 30, as implemented by Regulatory Guide 1.45 and SRP Section 5.2.5, which specify the types and sensitivity of the systems, as well as their seismic, indication, and testability criteria necessary to detect leakage of primary reactor coolant to the containment or to other interconnected systems.

The NRC concluded the following:

A. Ginna Station has all three systems required by Regulatory Guide 1.45. Two of the three systems meet the sensitivity requirements. The third system (sump A level monitoring) can measure approximately a 2-gpm leak in 1 hour. In addition to the three leakage detection systems, Ginna also incorporates six other diverse systems. Taking all these systems into consideration, a 1-gpm leak from the reactor coolant pressure boundary to the containment can be detected within 1 hr, as required by the Regulatory Guide.

B. Ginna has, as one of the diverse systems, the sump B level monitoring system, which is Seismic Category I and can measure a 10.5-gpm leak within 1 hour. Therefore, the plant adequately meets the leak detection needs following a seismic event, including the safe shut-down earthquake.

C. Provisions are made to monitor reactor coolant inleakage to interconnected systems (component cooling water (CCW) system and secondary system).

D. The Ginna Technical Specifications meet the intent of the Standard Technical Specifications concerning the operability of the leakage detection systems to monitor leakage to the primary containment. There is a difference in the number of required systems, which is not a significant safety factor because of the various diverse leakage detection systems available to the plant operators.

REFERENCES FOR SECTION 5.2

1. Letter from D. M. Crutchfield, NRC, to J. E. Maier, RG&E, Subject: SEP Safety Topics III-6, Seismic Design Consideration and III-11, Component Integrity, dated January 29, 1982.

2. Letter from L. D. White, Jr., RG&E, to A. Schwencer, NRC, Subject: Overpressure Protection, dated February 24, 1977.

3. Letter from L. D. White, Jr., RG&E, to A. Schwencer, NRC, Subject: Overpressure Protection, dated March 31, 1977.

4. Letter from L. D. White, Jr., RG&E, to A. Schwencer, NRC, Subject: Overpressure Protection, dated July 29, 1977.

5. Westinghouse Electric Corporation, R. E. Ginna Low Temperature Overpressure Protection System (LTOP) Setpoint Phase II Evaluation Final Report, October 1990 (Proprietary) and February 1991 (Non-Proprietary) (Attachment C to letter from R. C. Mecredy, RG&E, to A. R. Johnson, NRC, Subject: Rochester Gas and Electric Corporation, R. E. Ginna Nuclear Power Plant, Docket 50-244, dated February 15, 1991).

6. Generic Letter 88-05, Boric Acid Corrosion of Carbon Steel Reactor Pressure Boundary Components in PWR Plants, dated March 17, 1988.

7. Letter from Allen Johnson, NRC, to R. C. Mecredy, RG&E, Subject: Prevention of Boric Acid Corrosion at Ginna Nuclear Power Plant, dated August 10, 1990.

8. NRC Bulletin 88-09, Thimble Tube Thinning in Westinghouse Reactors, dated July 26, 1988.

9. Letter from R. C. Mecredy, RG&E, to A. R. Johnson, NRC, Subject: NRC Bulletin 88-09, dated April 8, 1993.

10. Letter From D. M. Crutchfield, NRC, to J. E. Maier, RG&E, Subject: SEP Topic V-5, Reactor Coolant Pressure Boundary Leakage Detection, dated February 8, 1982.

11. U.S. Nuclear Regulatory Commission, Integrated Plant Safety Assessment, Systematic Evaluation Program, R. E. Ginna Nuclear Power Plant, NUREG 0821, December 1982.

12. Letter from R. C. Mecredy, RG&E, to A. R. Johnson, NRC, Subject: Generic Letter (GL) 95-03, Response to NRC Request for Additional Information, dated March 2, 1996.

13. Letter from R. C. Mecredy, RG&E, to G. Vissing, NRC, Subject: Response to Generic Letter 95-03, dated July 25, 1996.

14. Generic Letter 95-03, Circumferential Cracking of Steam Generator Tubes, dated April 28, 1995.

15. B&W Nuclear Technologies, LTOP Report for Ginna, 86-1234820-03, approved 9/19/97.

16. B&W Nuclear Technologies, RELAP5/MOD2-B&W - An Advanced Computer Program for Light Water Reactor LOCA and Non-LOCA Transient Analysis, Code Topical Report BAW-10164P, Revision 2, dated August 1992.

17. Deleted.

18. Letter from R. C. Mecredy, RG&E, to G. S. Vissing, NRC, Subject: Response to NRC Generic Letter 97-05, Steam Generator Tube Inspection Techniques, dated March 23, 1998.

19. Letter from R. C. Mecredy, RG&E, to G. S. Vissing, NRC, Subject: Response to Request for Additional Information Related to Generic Letter 97-05, dated September 16, 1998.

20. Generic Letter 97-05, Steam Generator Tube Inspection Techniques, dated December 17, 1997.

Table 5.2-1
REACTOR COOLANT SYSTEM CODE REQUIREMENTS

Component	Codes
Reactor Vessel	ASME III[a]Class A[b]
Rod drive mechanism housing	ASME III[a]Class A[b]
Replacement steam generators	
Tube side	ASME III[c]Class 1
Shell side[d]	ASME III[c]Class 2
Reactor coolant pump volute	ASME III[a]Class A
Pressurizer	ASME III[a]Class A
Pressurizer relief tank	ASME III[a]Class C
Pressurizer safety valves	ASME III[a]
Reactor coolant piping	USAS B31.1[e](1955)
Reactor coolant valves	ASA B16.5 (1961)
System valves, fittings, and piping	USAS B31.1[e](1955) ASA B16.5 (1961)

a. ASME Boiler and Pressure Vessel Code, Section III, Nuclear Vessels (1965).

b. The replacement reactor vessel closure head and replacement equivalent control rod drive mechanism housings supplied by PCR 2001-0042, were supplied as ASME Section III Class 1 appurtenances to the 1995 edition with 1996 addenda of the ASME Boiler and Pressure Vessel Code in accordance with the Ginna Station Section XI Repair and Replacement Program.

c. ASME Boiler and Pressure Vessel Code, Section III, Division 1, 1986.

d. The shell side of the steam generator conforms to the requirements for Class 1 vessels and is so stamped as permitted under the rules of Section III.

e. USAS B31.1 Code for Pressure Piping.

Table 5.2-2
MATERIALS OF CONSTRUCTION OF THE REACTOR COOLANT SYSTEM COMPONENTS

Component	Section	Materials
Replacement steam generator	Pressure plate (manway covers)	SA-533 Tp B Cl 1
	Cladding, stainless weld	SFA 5.9 ER 308L/309L
	Cladding for tubesheets and seatbar	SFA 5.14 ER NiCr-3
	Nozzle dam retention rings	SB-166 N06690
	Tubes	SB-163 Alloy 690
	Channel head tubesheet and nozzles	SA-508 Cl 3
	Primary nozzle safe-ends	SA-336 316N/316LN
	Divider plate	SB-168 N06690
Pressurizer	Shell	SA-302, grade B
	Heads	SA-216 WCC
	External plate	SA-302, grade B
	Cladding, stainless	Type 304 equivalent
	Internal plate	SA-240 type 304
	Internal piping	SA-376 type 316
Piping	Pipes	A-376 type 316
	Fittings	A-351, CF8M
	Nozzles	A-182, F316
Pumps	Shaft	Type 304
	Impeller	A-251, CF8
	Casing	A-351, CF8M

Table 5.2-3
REACTOR COOLANT SYSTEM QUALITY ASSURANCE PROGRAM

Component	RT	UT	PT	MT	ET
Replacement steam generator					
Tubesheet					
Forging		X		X	
Cladding		X	X		
Channel head					
Forgings		X		X	
Cladding		X	X		
Secondary shell and head					
Plates and forgings		X		X	
Tubes		X			X
Nozzles (Forgings)		X		X (or PT)	
Weldments					
Secondary shell, longitudinal	X	X		X	
Secondary shell, circumferential	X	X		X	
Cladding		X	X		
Nozzle to shell	X	X		X	
Support brackets				X	
Tube-to-tube sheets			X		
Instrument connections (primary and secondary)			X		
Temporary attachments after removal				X (or PT)	
After hydrostatic test (all welds)				X	
Nozzle safe ends (if forgings)	X	X	X (or MT)		
Nozzle safe ends (if weld deposit)		X			
Pressurizer					
Heads					
Castings	X			X	
Cladding			X		
Shell					

Component	RT	UT	PT	MT	ET
Plates		X			
Cladding			X		
Heaters					
Tubings		X	X		
Centering of Element	X				
Nozzle		X	X		
Piping					
Fittings (castings)	X		X		
Fittings (forgings)		X	X		
Pipe		X	X		
Weldments					
Longitudinal	X		X		
Circumferential	X		X		
Nozzle to run pipe	X		X		
Instrument connections			X		
Pumps					
Castings	X		X		
Forgings		X	X		
Weldments					
Circumferential	X		X		
Instrument connections			X		
Reactor vessel					
Forgings					
Flanges		X		X	
Studs		X		X	
Head adapters	X		X		
Plates		X		X	
Weldments					
Main steam	X			X	
Control rod drive head adapter connection (W85)	X	X	X		X

Component	**RT**	**UT**	**PT**	**MT**	**ET**
Instrumentation tube			X		
Main nozzles	X			X	
Cladding			X		
Nozzle safe ends	X		X	X	

Notes:
RT Radiographic
UT Ultrasonic
PT Dye penetrant
MT Magnetic particle
ET Eddy current

Table 5.2-4

Table DELETED

Table DELETED

Table 5.2-5
REACTOR COOLANT PRESSURE BOUNDARY TO CONTAINMENT LEAKAGE DETECTION SYSTEMS

System	Leak Rate Sensitivity	Time Required to Achieve Sensitivity	Control Room Indication for Alarms and Indicators	Testable During Normal Operation (MODES1 and 2)
Sump A level (LT-2039 and LT-2044) monitoring (inventory)	2 gpm	1 hour	Yes	Yes
Sump A pump actuations monitoring (time meters)	2 gpm	1 hour	Yes	Yes
Airborne particulate radioactivity (R-11) monitoring	1 gpm[a]	NA	Yes	Yes
Airborne gaseous radioactivity (R-12) monitoring	7 gpm	1 hr	Yes	Yes
Condensate flow rate from air coolers	1-30 gpm	1 hr	Yes	Yes
Containment atmosphere pressure (PI-944) monitoring	NA	1 hr	Yes	Yes
Containment atmosphere humidity monitoring	2-10 gpm	NA	No	Yes
Containment atmosphere temperature (TE-6031, 6035, 6036, 6037, 6038, and 6045) monitoring	NA	NA	No	Yes
Chemical and volume control system	0.25 gpm	1 hr	Yes	Yes

NOTE:—NA = Not available

a. 0.018 gpm within 20 min assuming the presence of noble gas with particle progeny.

Table 5.2-6
REACTOR COOLANT PRESSURE BOUNDARY INTERSYSTEM LEAKAGE DETECTION SYSTEMS

Systems Which Interface With Reactor Coolant Pressure Boundary	Methods to Measure Reactor Coolant Pressure Boundary Inleakage	Leak Rate Sensitivity	Time Required to Achieve Sensitivity	Control Room Indication for Alarms and Indicators	Testable During Normal operation (MODES 1 and 2)
Secondary system	Condensate air ejector radiation monitor	0.02 gpm[a]	1 minute	Yes	Yes
Secondary system	Blowdown monitor	0.0025 gpm[b]	1 hour	Yes	Yes
Component cooling water (CCW) system	Surge tank level	NA	NA	Yes	Yes
Component cooling water (CCW) system	Radiation monitor	0.16 gpm	NA	Yes	Yes

a. Primary-to-secondary leakage of 1 gpd (leakage ≥ 0.0007 gpm) can be detected by R-47 with non-defected fuel.

b. Total leakage of 0.5 gal necessary for indication.

Table 5.2-7
SEQUENCE OF EVENTS - MASS ADDITION CASE

EVENT	TIME, SECONDS
Charging pumps started	0.0
Charging pumps reach full flow	1.0
Peak pressure of 587.4 psia reached in the bottom of the reactor vessel	7.45
Peak pressure of 525.5 psia reached at suction point for the residual heat removal system	7.45

Table 5.2-8
HEAT ADDITION AT 60°F - SEQUENCE OF EVENTS

EVENT	TIME, SECONDS
Reactor coolant pump started in loop that contains the pressurizer	0.0
Reactor coolant pump reaches full flow	17.4
Power operated relief valve opening signal for the first time	46.0
Peak pressure reached in the reactor vessel	46.0
Peak pressure reached in the residual heat removal system	46.0

Table 5.2-9
HEAT ADDITION AT 320°F - SEQUENCE OF EVENTS

EVENT	TIME, SECONDS
Reactor coolant pump started in loop that contains the pressurizer	0.0
Power operated relief valve opening signal for the first time	8.81
Peak pressure reached at the residual heat removal pump outlet	10.5
Reactor coolant pump reaches full flow	17.4
Peak pressure reached in the reactor vessel	21.3

5.3 REACTOR VESSEL

5.3.1 REACTOR VESSEL MATERIALS

5.3.1.1 Reactor Vessel Description

The Ginna reactor vessel was designed and fabricated by Babcock and Wilcox Company in accordance with Westinghouse specifications and the requirements of ASME Code, Section III, 1965 Edition. The governing specifications are listed in Table 5.3-1.

A replacement reactor vessel closure head was installed at Ginna Station during the Fall 2003 refueling outage. The replacement closure head was procured by PCR 2001-0042 through Babcock & Wilcox (B&W) of Canada, Ltd. in accordance with technical specification BWG-TS-2915 in accordance with ASME Section III 1995 Edition, with 1996 Addenda, Class 1 requirements.

The replacement closure head eliminated the use of the existing Alloy 600 control rod drive mechanism (CRDM) housings and weld material and replaced them with Alloy 690 TT (thermally treated) CRDM housings and Alloy 52 weld material.

The reactor vessel is cylindrical in shape with a hemispherical bottom and a flanged and gasketed removable upper head. Coolant enters the reactor vessel through two inlet nozzles in a plane just below the vessel flange and above the core. The coolant flows downward through the annular space between the vessel wall and the core barrel into a plenum at the bottom of the vessel where it reverses direction. Approximately 95% of the total coolant flow is effective for heat removal from the core. The remainder is considered as bypass flow as it is not fully effective for removing heat generated in the core. This bypass flow includes the flow through the rod cluster control guide thimbles, the flow between the core baffle and barrel, the leakage across the outlet nozzles, the flow deflected into the head of the vessel for cooling the upper flange, and the excess flow in the flow cells surrounding the rod cluster control guide thimbles. The bypass coolant and core coolant unite and mix in the upper plenum, and the mixed coolant stream then flows out of the vessel through two exit nozzles located on the same plane as the inlet nozzles. Figure 5.3-1, Sheets 1 and 2, is a schematic of the reactor vessel.

A one-piece thermal shield, concentric with the reactor core, is located between the core barrel and the reactor vessel. The shield, which is cooled by the coolant on its downward pass, protects the vessel by attenuating much of the gamma radiation and some of the fast neutrons which escape from the core. This shield minimizes thermal stresses in the vessel which result from heat generated by the absorption of gamma energy. The shield is further described in Section 3.9.5.1.1.

Thirty-six core instrumentation nozzles are located on the lower head.

The reactor closure head and the reactor vessel flange are joined by forty-eight 6-in. diameter studs. Two metallic O-rings seal the reactor vessel when the reactor closure head is bolted in place. A leakoff connection is provided between the two O-rings to monitor leakage across the inner O-ring. In addition, a leakoff connection is also provided beyond the outer O-ring seal.

The vessel is insulated with metallic, reflective type insulation supported from the nozzles. Insulation panels are provided for the reactor closure head and are supported on the MODE 6 (Refueling) seal ledge and vent shroud support rings.

The reactor vessel internals are designed to direct the coolant flow, support the reactor core, and guide the control rods in the withdrawn position. The reactor vessel contains the core support assembly, upper plenum assembly, fuel assemblies, control rod cluster assemblies, surveillance specimens, and incore instrumentation.

The reactor internals are described in Sections 3.9.5 and 4.2.1 and the general arrangement of the reactor vessel and internals is shown in Figures 3.9-9 and 3.9-10.

Reactor vessel design data is listed in Table 5.3-2.

The reactor vessel is the only component of the reactor coolant system which is exposed to a significant level of neutron irradiation and it is therefore the only component which is subject to material radiation damage effects. The nil ductility transition temperature (NDTT) shift of the vessel material and welds, due to radiation damage effects during service, is monitored by a radiation damage surveillance program, as described in Section 5.3.3.

5.3.1.2 Material Specifications

The materials of construction of the reactor vessel are given in Table 5.3-3. A detailed listing of the reactor vessel core region forgings and welds is given in Table 5.3-4, along with the heat treatment history. The chemistry of all the materials is given in Table 5.3-5 and the mechanical properties are given in Table 5.3-6. The location of the reactor vessel beltline material is shown in Figure 5.3-2.

The cylindrical section of the reactor vessel is comprised of three cylindrical forgings (SA-508, class 2). The top and bottom dome sections are made from plate material (SA-533, grade A). The shell course, flanges, and nozzles are made from forgings (SA-508, class 2).

The forgings were processed by the mandrel forging technique. Prior to mandrel forging, the rough forging was upset and the center section removed. The forged section weld locations use the same inservice test techniques as those used for plate vessel welds.

The fracture toughness properties of forgings are comparable to plates in the unirradiated condition and the irradiated condition. Mechanical property tests for shell course forgings were taken at a one-fourth thickness location and at a minimum distance of one thickness from the quenched edge. Test locations complied with ASME Code, Section III, requirements.

The reactor vessel materials opposite the core were purchased to a specified Charpy V-notch impact energy of 30 ft-lb or greater at a corresponding NDTT of 40°F or less. The materials were subsequently tested (drop weight) to verify conformity to specified NDTT requirements. In addition, the plate sections were 100% volumetrically inspected by an ultrasonic test using both longitudinal and shear wave methods. The remaining material in the reactor vessel meets the appropriate design code requirements and specific component functional requirements.

The reactor vessel material is heat-treated specifically to obtain good notch-ductility which ensures a low NDTT, and thereby gives assurance that the finished vessel can be initially hydrostatically tested and operated as near to room temperature as possible without restrictions. The stress limits established for the reactor vessel are dependent upon the temperature at which the stresses are applied. As a result of fast neutron irradiation in the region of the core, the material properties will change, including an increase in the NDTT.

There are two welds in the beltline region: the nozzle shell to intermediate shell (SA-1101) and the intermediate shell to lower shell (SA-847). Both are circumferential welds made by the submerged arc process. Based on radiation exposure and chemical composition, weld SA-847 is the limiting vessel material.

5.3.1.3 Testing and Surveillance

Westinghouse required, as part of its reactor vessel specification, that certain special tests which were not specified by the applicable codes be performed. These tests are listed below:

Ultrasonic Testing

Westinghouse required that a 100% volumetric ultrasonic test of reactor vessel plate for shear wave be performed. The 100% volumetric ultrasonic test is a severe requirement, but it ensures that plate used for the reactor vessel is of the highest quality.

Radiation Surveillance Program

In the surveillance program, the evaluation of the radiation damage is based on pre-irradiation and post-irradiation testing of Charpy V-notch, tensile and wedge opening loading test specimens. These programs are directed toward evaluation of the effect of radiation on the fracture toughness of reactor vessel steels based on the transition temperature approach and the fracture mechanics approach, and are in accordance with ASTM E185, Recommended Practice for Surveillance Tests on Structural Materials in Nuclear Reactors. The surveillance program for the RG&E reactor vessel is described in Section 5.3.3.

5.3.2 PRESSURE-TEMPERATURE LIMITS

5.3.2.1 Thermal and Pressure Loadings

Reactor vessel design is based on the transition temperature method of evaluating the possibility of brittle fracture of the vessel material resulting from operations such as leak testing and plant heatup and cooldown. To establish the service life of the reactor coolant system components as required by the ASME Code Section III for Class A vessels, the unit operating conditions which involve the cyclic application of loads and thermal conditions have been established for the 40-year design life. The number of thermal and loading cycles used for design purposes are listed in Table 5.1-4.

The stress level of material in the reactor vessel, or in other reactor coolant system components, is a combination of stresses caused by internal pressures and by thermal gradients. The latter are significant as they may result from a rate of change of reactor coolant temperature and change the location of the limiting stress between heatup and cooldown. During cooldown, the thermal stress varies from tensile at the inner wall to compressive at the outer wall. The internal pressure superimposes a tensile stress on this thermal stress pattern, increasing the stress at the inside wall and relieving the stress at the outside wall. Therefore, the location of the limiting stress is always at the inside wall surface; however, for heatup the thermal stress is reversed so the location of the limiting stress is a function of the heatup rate. Operating restrictions are imposed to limit the combined stresses to 20% of minimum yield stress when at the design transition temperature. The design transition temperature is defined as the initial NDTT plus the increase in NDTT due to irradiation experienced plus 60°F. This stress limit (20% of yield stress) is reduced linearly to a value of 10% of yield at a temperature of 200°F below design transition temperature. Curves which define the operating limits are incorporated in the Pressure and Temperature Limits Report (PTLR).

5.3.2.2 Pressure-Temperature Limits

Pressure-temperature limits for reactor vessel operation provide a means of ensuring vessel integrity throughout its operating life. Operation in accordance with the curves ensures that, in the normal operating range, the vessel will operate in the upper-shelf region of its material toughness. This also provides assurance that the fracture toughness of vessel materials during heatup and cooldown transients will be adequate to prevent rapid crack propagation (brittle fracture).

Pressure-temperature limits for inservice testing, heatup and cooldown, and core operation are required to be in compliance with the rules of Appendix G to 10 CFR Part 50, Fracture Toughness Requirements. When first published in 1971, Appendix G used a transition temperature approach to establish safe operating limits. Appendix G was revised in 1973 to require a fracture mechanics approach, which usually gives more conservative operating limits.

The fracture mechanics approach relies on a fracture mechanics characterization of the material and its stress environment. Using this characterization, the stress in any portion of the vessel, in conjunction with any assumed flaw, can be compared with the stressed-flaw tolerance of the material, a material parameter such as K_{IC} (the plane strain fracture toughness of a material). Using this parameter, the stress in the vessel can be limited such that, in the presence of an assumed flaw size so large as to ensure detection, no rapid crack propagation can occur. Above NDTT, the fracture toughness of the materials used in the nuclear reactor vessels increases greatly. Thus, the crack tolerance of the material at the normal operating temperatures is high. Under this system of fracture control, prevention of rapid fracture is ensured by the control of stresses and flaw sizes. For nuclear vessel materials of normal shelf fracture toughness (according to Appendix G, 10 CFR 50, a Charpy upper-shelf energy of 50 ft-lb is required), very large cracks would be required to cause the onset of rapid crack propagation at operating temperature and pressure. In regions of high local stresses, such as nozzle corners, ductile tearing could commence at smaller cracks or lower pressure but, as the tear extended into a region of lower nominal stress such as the vessel wall, rapid fracture would again require very large cracks.

5.3.2.3 Pressure-Temperature Limit Calculation

The specific methods to calculate the pressure-temperature operating limits are contained in Appendix G to ASME Code, Section III. For regions remote from discontinuities (the beltline region), the stress intensity factors calculated in the development of these operating limits are based on a postulated sharp surface flaw penetrating to a depth of one-fourth of the vessel wall thickness and having a length one-and-one-half times the section thickness. Since the maximum size flaw that might escape detection in a preservice or inservice inspection is much smaller than this assumed flaw size, the combination of inspections and conservative pressure-temperature limits provides a high degree of assurance for vessel integrity throughout service life. For nozzles, flanges, and shell regions near discontinuities, a smaller defect size may be used. The smaller defect size must be justified and non-destructive examination methods must be sufficiently reliable and sensitive to detect these smaller defects. The procedures to calculate the stress intensity factors for these regions provide margins of safety comparable to those required for the beltline region. Appendix G provides methods to calculate stress intensities for membrane tension stress, bending stress, and stresses resulting from thermal gradients, and lists the safety factors to be applied to these stress intensities.

As a result of the extended power uprate to 1775 MWt, the impact of increased neutron fluence on the existing Ginna P/T limits curves was evaluated *(Reference 26)*. The evaluation determined that for the existing Ginna methodology for determining P/T limits, the integrated neutron fluence after uprate did not exceed the fluence projections used to develop the pre-uprate P-T limit curves for both 28 and 32 EFPY *(Reference 27)*. Consequently, the pre-uprate P-T limits used in the Ginna PTLR remained valid for up to 32 EFPY of reactor operation. For plant operation up to 53 EFPY new P-T limit curves for the Ginna PTLR were developed consistent with the methodology specified in Ginna Technical Specifications *(Reference 29)*.

The impact of the increased end of life neutron fluence due to uprate on the Upper Shelf Energy (USE) for reactor vessel pressure boundary materials was also evaluated *(Reference 26)*. The USE evaluation was re-performed following material analysis of Capsule N *(Reference 30)*. All beltline materials are expected to have a USE greater than 50 ft-lb through the end of plant life in 2029 except for the intermediate-to-lower shell girth weld and the intermediate-to-nozzle shell girth weld. As required by 10CFR50 Appendix G, an equivalent margins analysis (EMA) reflecting uprated conditions for these two weld locations was performed which demonstrated that sufficient USE margin existed at the uprated power level.

5.3.2.4 Irradiation Effect on Pressure-Temperature Limit

Irradiation degrades material toughness causing RT_{NDT} to increase. Since the pressure-temperature limits are based on a temperature above RT_{NDT}, these limits must be revised periodically to reflect the changes in toughness. Since the postulated flaw penetrates to one-fourth the wall thickness, the increase in RT_{NDT} is based on the fluence at the one-fourth thickness location. Increases in RT_{NDT} are usually obtained from the results of the vessel material surveillance program. If these results are for some reason not considered applicable or valid, the staff uses Regulatory Guide 1.99, Revision 2, to obtain conservative radiation damage values.

5.3.2.5 Heatup and Cooldown Rates

The reactor coolant system temperature and pressure and heatup and cooldown rates (with the exception of the pressurizer) are limited in accordance with the Pressure and Temperature Limits Report (PTLR). The actions to follow if the limits are exceeded are included in the Technical Specifications. The heatup and cooldown rates shall not exceed 60°F per hour and 100°F per hour, respectively. For the pressurizer, the heatup and cooldown rates do not exceed 100°F per hour and 200°F per hour, respectively. The pressurizer spray is not to be used if the temperature difference between the pressurizer and the spray fluid is greater than 320°F.

The normal system heating rate is 50°F per hour. Sufficient electrical heaters are installed in the pressurizer to permit a heatup rate of 55°F/hr, starting with a minimum water level.

The administrative limit for plant cooldown is 90°F/hr. The fastest cooldown rates which result from the hypothetical case of a main steam line break are discussed in Section 15.1.5.

A maximum temperature difference of 200°F between the pressurizer and reactor coolant system is specified to maintain thermal stresses within the surge line below design limits.

Temperature requirements for pressurization of the pressurizer and steam generators correspond with the design transition temperature measured for the material of each component.

The rates of temperature change are applied as total change in temperature in any 1-hour period.

5.3.3 REACTOR VESSEL INTEGRITY

5.3.3.1 Safety Factors

The reactor vessel has a 132-in. I.D., which is within standard size limits for which there is a good deal of operating experience. A stress evaluation of the reactor vessel was carried out in accordance with the rules of Section III of the ASME Code. The evaluation demonstrated that stress levels were within the stress limits of the code. Table 5.3-7 presents a summary of the results of the stress evaluation.

A summary of fatigue usage factors for components of the reactor vessel is given in Table 5.3-8.

The cycles specified for the fatigue analysis are the results of an evaluation of the expected plant operation coupled with experience from operational nuclear power plants. These cycles include five heatup and cooldown cycles per year, a conservative selection since the vessel would not complete more than one cycle per year during MODES 1 and 2.

The vessel design pressure is 2485 psig, while the normal operating pressure is 2235 psig. The resulting operating membrane stress is therefore amply below the code-allowable membrane stress to account for operating pressure transients.

The vessel closure contains forty-eight 6-in. studs. The stud material is ASTM A-540 and code case 1335-2 which has a minimum yield strength of 104,400 psi at design temperature.

The membrane stress in the studs when they are at the steady-state operational condition is approximately 37,500 psi. This means that 18 of the 48 studs have the capability of withstanding the hydrostatic end load on vessel head without the membrane stress exceeding yield strength of the stud material at design temperature.

The method to perform analyses to guard against fast fracture in the reactor vessel is included in Appendix G to Section III of the ASME Code. The method utilizes fracture mechanics concepts and is based on the reference nil ductility temperature, RT_{NDT}.

RT_{NDT} is defined as the greater of the drop weight NDTT (per ASTM E-208) or the temperature 60°F less than the 50 ft-lb temperature (or 35-mil lateral expansion temperature if this is greater), as determined from Charpy specimens oriented normal to the working direction of the material. The RT_{NDT} of a given material is used to index that material to a reference stress intensity factor curve, K_{IR} curve, which appears in Appendix G of the ASME Code. The K_{IR} curve is a lower bound of dynamic, crack arrest, and static fracture toughness results obtained from several heats of pressure vessel steel. When a given material is indexed to the K_{IR} curve, allowable stress intensity factors can be obtained for this material as a function of temperature. Allowable operating limits can then be determined utilizing these allowable stress intensity factors.

The RT_{NDT} and, in turn, the operating limits of the reactor are adjusted to account for the effects of radiation on the reactor vessel material properties. The radiation embrittlement or changes in mechanical properties of the pressure vessel steel are monitored by the material surveillance program as described in Section 5.3.3.2. The increase in the Charpy V-notch 50 ft-lb temperature (delta RT_{NDT}) due to irradiation is added to the original RT_{NDT} to adjust the RT_{NDT} for radiation embrittlement. This adjusted RT_{NDT} (RT_{NDT} initial + delta RT_{NDT}) is used to index the material to the K_{IR} curve and, in turn, to set operating limits for the plant which take into account the effects of irradiation on the reactor vessel materials.

As part of the plant operator training program, supervisory and operating personnel are instructed in reactor vessel design, fabrication, and testing, as well as precautions necessary for pressure testing and MODES 1 and 2. The need for recordkeeping is stressed, such records being helpful for future summation of time at power level and temperature which tends to influence the irradiated properties of the material in the core region. These instructions are incorporated into the operating manuals.

5.3.3.2 Material Surveillance Program

The material surveillance program for Ginna was previously described in WCAP 7254 (*Reference 1*). The program was designed to meet the requirements of 10 CFR 50, Appendix H, and ASTM E-185-73. Capsules withdrawn after July 26, 1983, will be tested and the results reported in accordance with the 1982 revision of ASTM E-185 as required by 10 CFR 50, Appendix H. It consists of six surveillance capsules (V, R, T, P, S, and N) positioned in the reactor vessel between the thermal shield and the reactor vessel wall as shown in Figure 5.3-3. The vertical center of each capsule is opposite the vertical center of the core. Each capsule contains tensile, Charpy V-notch, and wedge opening loading specimens from the forgings (heats 125P666 and 125S255) and weld metal, and Charpy V-notch specimens from heat-affected zone material and from an A-302, Grade B correlation material furnished by U.S. Steel Corporation. Data on the correlation material gives an indication of radiation damage in a commercial reactor vessel compared to a test reactor vessel, and also gives an indication of the accuracy of the neutron fluence calculations.

The material surveillance program for Ginna is described in BAW 1543 (*Reference 2*). BAW 1543 describes the Master Integrated Reactor Vessel Material Surveillance Program for Babcock & Wilcox-fabricated PWR reactor vessels containing seam welds fabricated by the automatic submerged arc process using copper-plated magnesium-molybdenum-nickel steel filler metal and Linde 80 flux. BAW 1543 describes the approach that the Babcock & Wilcox vessel owners will use in addressing the "Linde 80" welds. In addition to the six supplementary capsules that were previously added to the program, eight irradiation capsules are included, which further expand the fracture toughness data base for this class of materials and include life extension and annealing considerations. The Master Integrated Reactor Vessel Material Surveillance Program, therefore, includes a total of 17 plant specific reactor vessel surveillance programs and 14 supplementary material irradiation capsules. These reactor vessels include eight Babcock & Wilcox-designed 177 fuel assembly plants and nine Westinghouse-designed plants with Babcock & Wilcox-fabricated reactor vessels. The information obtained from all of these sources is coordinated and shared to maximize the usefulness of the data.

All surveillance specimens were machined from the one-fourth thickness location of the forgings. The specimens represent material that was taken at least one forging thickness away from the quenched end of the forging. All Charpy V-notch and tensile specimens were oriented with the longitudinal axis of each specimen parallel to the hoop direction (strong direction) of the forgings. The wedge opening loading specimens were machined with the simulated crack of each specimen perpendicular to the surfaces and the hoop direction of the forgings.

The surveillance capsules contain dosimeter wires of copper, nickel, and aluminum-cobalt. They also contain cadmium-shielded dosimeters of Neptunium-237 and Uranium-238. The dosimeters permit evaluation of the neutron flux seen by the various specimens. Surveillance capsules V, R, T, S, and N have been removed and tested in accordance with Technical Specifications and test results documented in *References 3 through 6* and *28* respectively. Test results are analyzed, the shift in transition temperature is compared to the predicted shift, and pressure-temperature limit curves (Section 5.3.2) are revised accordingly. Surveillance capsule P will be removed in the future. See Section 5.3.3.3.

5.3.3.3 Surveillance Program Analysis

Capsule V was removed and tested in 1971, capsule R in 1974, capsule T in 1980, and capsule S in 1993. The insertion and withdrawal schedules for capsules P and N have been prepared in accordance with ASTM E-185-82 and the criteria for integrated surveillance programs of 10 CFR 50, Appendix H, paragraph II.C, and reside in the Master Integrated Reactor Vessel Material Surveillance Program (*Reference 2*). The NRC staff has determined that the material surveillance program at Ginna satisfies Appendix H to 10 CFR 50 (*Reference 8*). *Reference 9* documented acceptability of BAW 1543. All capsules in the Ginna reactor vessel surveillance program contain SA-1036 weld material, which is a surrogate for SA-847, a beltline material in Ginna and Point Beach Unit 1. SA-1135 weld material is also a surrogate for SA-847.

As a result of the extended power uprate to 1775 MWt the integrated neutron fluence on the vessel at the end of the 60 year plant life increased. The effect of the increased integrated neutron fluence on the Ginna reactor vessel surveillance withdrawal schedule was evaluated as part of the extended power uprate *(Reference 26)*. The only changes to the capsule withdrawal schedule due to uprate are related to i) the capsule fluence values, ii) lead factors and iii) timing of future capsule withdrawals. The total number of capsules that are required to be removed over the life of the plant was unaffected by implementation of the extended power uprate.

Surveillance capsule N was withdrawn at the refueling outage following achievement of a neutron fluence shortly after the equivalent of 60 calendar years of operation. Capsule N was removed during the spring 2008 refueling outage and the test report submitted to the NRC by *Reference 28*.

Capsule P, the last surveillance capsule, is to be removed shortly after it accumulates a fluence equivalent to 80 years of operation, however, it is planned that dosimetry monitors will be reinstalled, such that the neutron flux could continue to be monitored throughout the period of extended operation (until year 2029).

5.3.3.3.1 Results Summary

The analysis of the reactor vessel materials contained in surveillance capsule N, the fifth capsule to be removed from the reactor vessel, was reported to the NRC by *Reference 28*. The analysis led to the following conclusions:

A. The capsule received an average fast neutron fluence (E > 1.0 MeV) of 5.80 x 10^{19} n/cm^2 after 30.5 effective full power years of plant operation.

B. Irradiation of the reactor vessel lower forging 125P666 Charpy specimens, oriented with the longitudinal axis of the specimen parallel to the major rolling direction (longitudinal direction), to 5.80 x 10^{19} n/cm^2 (E > 1.0 MeV) resulted in a 30 ft-lb transition temperature increase of 91.1°F and a 50 ft-lb transition temperature increase of 93.3°F. This results in an irradiated 30 ft-lb transition temperature of 44.9°F and an irradiated 50 ft-lb transition temperature of 78.4°F for the longitudinally oriented specimens.

C. Irradiation of the reactor vessel intermediate shell forging 125S255 Charpy specimens, oriented with the longitudinal axis of the specimen parallel to the major rolling direction (longitudinal orientation), to 5.80 x 10^{19} n/cm^2 (E > 1.0 MeV) resulted in a 30 ft-lb transition temperature increase of 76.4°F and a 50 ft-lb transition temperature increase of 100.0°F. This results in an irradiated 30 ft-lb transition temperature of 47.5°F and an irradiated 50 ft-lb transition temperature of 102.8°F for the longitudinally oriented specimens.

D. Irradiation of the weld metal Charpy specimens to 5.80 x 10^{19} n/cm^2 (E > 1.0 MeV) resulted in a 30 ft-lb transition temperature increase of 216.9°F and a 50 ft-lb transition temperature increase of 261.0°F. This results in an irradiated 30 ft-lb transition temperature of 182.2°F and an irradiated 50 ft-lb transition temperature of 276.0°F.

E. Irradiation of the weld heat affected zone metal Charpy specimens to 5.80 x 10^{19} n/cm^2 (E > 1.0 MeV) resulted in a 30 ft-lb transition temperature increase of 107.7°F and a 50 ft-lb transition temperature increase of 74.5°F. This results in an irradiated 30 ft-lb transition temperature of 43.0°F and an irradiated 50 ft-lb transition temperature of 58.4°F.

F. The average upper shelf energy of intermediate shell forging 125P666 (longitudinally orientation) resulted in an energy decrease of 32.3 ft-lb after irradiation to 5.80 x 10^{19} n/cm^2 (E > 1.0 MeV). This results in an irradiated average upper shelf energy of 142.3 ft-lb for longitudinally oriented specimens.

G. The average upper shelf energy of intermediate shell forging 125S255 (longitudinally orientation) resulted in an energy decrease of 5.7 ft-lb after irradiation to 5.80 x 10^{19} n/cm^2 (E > 1.0 MeV). This results in an irradiated average upper shelf energy of 134.3 ft-lb for longitudinally oriented specimens.

H. The average upper shelf energy of the weld metal Charpy specimens resulted in an energy decrease of 27.1 ft-lb after irradiation to 5.80 x 10^{19} n/cm^2 (E > 1.0 MeV). This results in an irradiated average upper shelf energy of 51.9 ft-lb for the weld metal specimens.

I. The calculated end-of-life (53 effective full power years) maximum neutron fluence (E > 1.0 MeV) for the reactor vessel is as follows:

Vessel inner radius [a] = 5.56 x 10^{19} n/cm^2

Vessel 1/4 thickness = 3.76 x 10^{19} n/cm^2

Vessel 3/4 thickness = 1.73 x 10^{18} n/cm^2

J. The average upper shelf energy of the weld heat affected zone metal decrease of 1.7 ft-lb after irradiation to 5.80 x 10^{19} n/cm^2 (E > 1.0 MeV). This results in an irradiated upper shelf energy of 88.3 ft-lb for the weld heat affected zone metal.

K. A comparison of the surveillance Capsule N test results with the Regulatory Guide 1.99, Revision 2, predictions led to the following conclusions:

- The measured 30 ft-lb shift in transition temperature values of the Intermediate Shell Forging 125S255 and Lower Shell Forging 125P666 specimens contained in capsule N are greater than the Regulatory Guide 1.99, Revision 2 predictions.
- The measured 30 ft-lb shift in transition temperature value of the Surveillance Weld Heat # 61782 specimens contained in capsule N is less than the Regulatory Guide 1.99, Revision 2 prediction.
- The measured percent decrease in upper shelf energy for all forging and weld surveillance materials in Capsule N are less than the Regulatory Guide 1.99, Revision 2 predictions.

a. Clad / base metal interface.

The summary of all five surveillance capsule results appears in Table 5.3-9.

Generic Letter 88-11, "NRC Position on Radiation Embrittlement of Reactor Vessel Materials and Its Impact on Plant Operations," required each licensee to reevaluate the effect of neutron radiation on reactor vessel material using the methods described in Regulatory Guide 1.99, Revision 2. This reevaluation was performed by Westinghouse and is documented in *Reference 10*. Based on the Westinghouse reevaluation, the heatup and cooldown limit curves in effect at the time were considered to be appropriate for use up to 21 effective full power years of operation. The most recent curves are in the Pressure and Temperature Limits Report (PTLR).

5.3.3.3.2 Charpy V-Notch Impact Test Results

Irradiation of the reactor vessel lower shell forging 125P666 Charpy specimens oriented with the longitudinal axis of the specimen parallel to the major rolling direction of the forging (longitudinal orientation) to 5.80 x 10^{19} n/cm^2 (E > 1.0 MeV) at 550°F to 574°F resulted in a 30 ft-lb transition temperature increase of 91.1°F and a 50 ft-lb transition temperature increase of 93.3°F. This results in an irradiated 30 ft-lb transition temperature of 44.9°F and an irradiated 50 ft-lb transition temperature of 78.4°F (longitudinal orientation).

The average upper shelf energy of the lower shell forging 125P666 Charpy specimens (longitudinal orientation) resulted in an energy decrease of 32.3 ft-lb after irradiation to 5.80 x 10^{19} n/cm^2 (E > 1.0 MeV) at 550°F to 574°F. This results in an irradiated average upper shelf energy of 142.3 ft-lb.

Irradiation of the reactor vessel intermediate shell forging 125S255 Charpy specimens oriented with the longitudinal axis of the specimen parallel to the major rolling direction of the forging (longitudinal orientation) to 5.80 x 10^{19} n/cm^2 (E > 1.0 MeV) at 550°F to 574°F resulted in a 30 ft-lb transition temperature increase of 76.4°F and a 50 ft-lb transition temperature increase of 100.0°F. This results in an irradiated 30 ft-lb transition temperature of 47.5°F and an irradiated 50 ft-lb transition temperature of 102.8°F (longitudinal direction).

The average upper shelf energy of the intermediate shell forging 125S255 Charpy specimens (longitudinal direction) resulted in an energy decrease of 5.7 ft-lb after irradiation to 5.80 x 10^{19} n/cm^2 (E > 1.0 MeV) at 550°F to 574°F. This results in an irradiated average upper shelf energy of 134.3 ft-lb.

Irradiation of the surveillance weld metal Charpy specimens to 5.80 x 10^{19} n/cm^2 (E > 1.0 MeV) at 550°F to 574°F resulted in a 30 ft-lb transition temperature shift of 216.9°F and a 50 ft-lb transition temperature increase of 261.0°F. This results in an irradiated 30 ft-lb transition temperature of 182.2°F and an irradiated 50 ft-lb transition temperature of 276.0°F.

The average upper shelf energy of the surveillance weld metal resulted in an energy decrease of 27.1 ft-lb after irradiation to 5.80 x 10^{19} n/cm^2 (E > 1.0 MeV) at 550°F to 574°F. This results in an irradiated average upper shelf energy of 51.9 ft-lb.

Irradiation of the reactor vessel weld heat affected zone metal Charpy specimens to 5.80 x 10^{19} n/cm^2 (E > 1.0 MeV) at 550°F to 574°F resulted in a 30 ft-lb transition temperature increase of 107.7°F and a 50 ft-lb transition temperature increase of 74.5°F. This results in an irradiated 30 ft-lb transition temperature of 43.0°F and an irradiated 50 ft-lb transition temperature of 58.4°F.

The average upper shelf energy of the weld heat affected zone metal resulted in an energy decrease of 1.7 ft-lb after irradiation to 5.80 x 10^{19} n/cm^2 (E > 1.0 MeV) at 550°F to 574°F. This results in an irradiated average upper shelf energy of 88.3 ft-lb.

A comparison of the 30 ft-lb transition temperature increases and upper shelf energy decreases for the various R. E. Ginna surveillance materials with predicted values using the methods of Regulatory Guide 1.99, Revision 2, is presented in Table 5.3-10. This comparison indicates that the capsule N surveillance materials are in good agreement with the Regulatory Guide 1.99, Revision 2, predictions.

5.3.3.3.3 Tension Test Results

The results of the tension tests performed on the lower shell forging 125P666 (longitudinal orientation) indicated that irradiation to 5.80 x 10^{19} n/cm^2 (E > 1.0 MeV) at 550°F to 574°F caused a 13.1 ksi increase in the 0.2% offset yield strength and a 8.3 ksi increase in the ultimate tensile strength when compared to unirradiated data.

The results of the tension tests performed on the intermediate shell forging 125S255 (longitudinal orientation) indicated that irradiation to 5.80 x 10^{19} n/cm^2 (E > 1.0 MeV) at 550°F to 574°F caused a 17.7 ksi increase in the 0.2% offset yield strength and a 11.3 ksi increase in the ultimate tensile strength when compared to unirradiated data.

The results of the tension tests performed on the surveillance weld metal indicated that irradiation to 5.80 x 10^{19} n/cm^2 (E > 1.0 MeV) at 550°F to 574°F caused a 24.5 ksi increase in the ultimate tensile strength when compared to unirradiated data.

5.3.3.3.4 Radiation Analysis and Neutron Dosimetry

The radiation analysis and neutron dosimetry methods employed in the Surveillance Program Analysis are described in detail in *Reference 30*.

5.3.3.4 Analysis of Effects of Loss of Coolant and Safety Injection on the Reactor Vessel

The analysis of the effects of injecting safety injection water into the reactor coolant system following a postulated loss-of-coolant accident was performed by Westinghouse for the initial licensing with the following results.

5.3.3.4.1 Reactor Vessel

For the reactor vessel, three modes of failure were considered, including the ductile mode, brittle mode, and fatigue mode.

A. Ductile Mode. The failure criterion used for this evaluation was that there shall be no gross yielding across the vessel wall using the material yield stress specified in Section III of the ASME Boiler and Pressure Vessel Code. The combined pressure and thermal stresses during injection through the vessel thickness as a function of time were calculated and compared to the material yield stress at the times during the safety injection transient.

   The results of the analyses showed that local yielding may occur in approximately the inner 12% of the base metal and in the cladding.

B. Brittle Mode. The possibility of a brittle fracture of the irradiated core region was considered from both a transition temperature approach and a fracture mechanics approach. The failure criterion used for the transition temperature evaluation was that a local flaw cannot propagate beyond any given point where the applied stress would remain below the critical propagation stress at the applicable temperature at that point.

   The results of the transition temperature analysis showed that the stress-temperature condition in the outer 65% of the base metal wall thickness remains in the crack arrest region at all times during the safety injection transient. Therefore, if a defect were present in the most detrimental location and orientation (i.e., a crack on the inside surface and circumferentially directed), it could not propagate any further than approximately 35% of the wall thickness, even considering the worst case assumptions used in the analysis.

   The results of the fracture mechanics analysis, considering the effects of water temperature, heat transfer coefficients, and fracture toughness of the material as a function of time, temperature, and irradiation were considered. Both a local crack effect and a continuous crack effect were considered with the latter requiring the use of a rigorous finite element axisymmetric code.

C. Fatigue Mode. The failure criterion used for the failure analysis was the one presented in Section III of the ASME Boiler and Pressure Vessel Code. In this method the piece was assumed to fail once the combined usage factor at the most critical location for all transients applied to the vessel exceeds the code allowable usage factor of one.

   The results of the analysis showed that the combined usage factor never exceeded 0.2, even after assuming that the safety injection transient occurred at the end of plant life.

In order to promote a fatigue failure during the safety injection transient at the end of plant life, it has been estimated that a wall temperature of approximately 1100°F is needed at the most critical area of the vessel (instrumentation tube welds in the bottom head).

The design basis of the safety injection system ensures that the maximum cladding temperature does not exceed the Zircaloy-4 or ZIRLO™ melt temperature. This is achieved by prompt recovery of the core through flooding, with the passive accumulator and the injection systems. Under these conditions, a vessel temperature of 1100°F is not considered a credible possibility and the evaluation of the vessel under such elevated temperatures is for a hypothetical case.

For the ductile failure mode, such hypothetical rise in the wall temperature would increase the depth of local yielding in the vessel wall.

The results of these analyses show that the integrity of the reactor vessel is never violated.

5.3.3.4.2 Safety Injection Nozzles

The safety injection nozzles have been designed to withstand ten postulated safety injection transients without failure. This design and associated analytical evaluation was made in accordance with the requirements of Section III of the ASME Boiler and Pressure Vessel Code.

The maximum calculated pressure plus thermal stress in the safety injection nozzle during the safety injection transient was calculated to be approximately 55,400 psi. This value compares favorably with the code allowable stress of 80,000 psi.

These ten safety injection transients were considered along with all the other design transients for the vessel in the fatigue analysis of the nozzles. This analysis showed the usage factor for the safety injection nozzles was 0.219 which is well below the code allowable value of 1.0.

The safety injection nozzles are not in the highly irradiated region of the vessel and thus they are considered ductile during the safety injection transient.

5.3.3.4.3 Fuel Assembly Grid Springs

The effect of the safety injection water on the fuel assembly grid springs was evaluated. Due to the fact that the springs have a large surface area to volume ratio, being in the form of thin strips, and are expected to follow the coolant temperature transient with very little lag, no thermal shock is expected and the core cooling is not compromised.

5.3.3.4.4 Core Barrel and Thermal Shield

Evaluations of the core barrel and thermal shield have also shown that core cooling is not jeopardized under the postulated accident conditions.

5.3.3.4.5 Subsequent Analyses of Reactor Vessel

Subsequent analyses on the reactor vessel integrity were submitted to the NRC by *Reference 11* (WCAP 10019). This report, submitted in response to NUREG 0737, Item II.K.2.13, provides classical mechanics analyses of design-basis accidents, which demonstrate that there are no immediate reactor vessel integrity concerns. The basis for the thermal stress and fracture analyses of *Reference 11* was also used in the evaluation of the reactor vessel integrity, performed after the Ginna steam generator tube rupture incident (*Reference 12*).

In 1992, the NRC issued Revision 1 of Generic Letter 92-01 (*Reference 13*) to obtain information needed to assess compliance with requirements and commitments regarding reactor vessel integrity in view of certain concerns raised in the NRC staff's review of reactor vessel integrity for the Yankee Rowe Nuclear Power Station. In 1995, the NRC issued Revision 1, Supplement 1 of Generic Letter 92-01 (*Reference 15*). RG&E's responses to Revision 1 and Revision 1, Supplement 1 of Generic Letter 92-01 are contained in *References 14*, *16*, and *17*. The NRC in *Reference 18* stated that since RG&E had provided the requested information and indicated that previous submittals remained valid, the NRC considers the reactor pressure vessel integrity data for Ginna to be complete and closed out Generic Letter 92-01, Revision

1, Supplement 1. Since the closure, additional reactor vessel integrity correspondence (*References 19 through 21*) has been sent to or from the NRC.

5.3.3.5 Pressurized Thermal Shock

The issue of pressurized thermal shock arises because in pressurized water reactors transients and accidents can occur that result in severe overcooling of the reactor vessel, concurrent with or followed by repressurization. The issue is a concern after the reactor vessel has lost its toughness properties and is embrittled by neutron irradiation. The rate of decrease of the fracture resistance of the reactor vessel material is dependent on the metallurgical composition of the vessel walls and welds.

In accordance with the requirements of 10 CFR 50.61, Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events, Ginna Station submitted projected RT_{PTS} values for the reactor vessel beltline materials from the present to the expiration date of the operating license to the NRC (*Reference 22*). The projected values were below the screening criteria for the expiration date and beyond 32 effective full power years. The NRC by *Reference 23*, which included the safety evaluation report, reported that Ginna Station met the requirements of the pressurized thermal shock rule (10 CFR 50.61).

On March 22, 1996, the NRC issued a Safety Evaluation Report (*Reference 24*), that concluded that the Ginna reactor vessel is projected to be below the PTS screening criteria at the expiration of its license. The Safety Evaluation Report contained the following conclusions:

1. The RG&E method for determining the credibility of the Ginna surveillance data did not conform to the guidance in Regulatory Guide 1.99, Rev. 2. However, the NRC evaluation of the data indicates that the Ginna surveillance data complies with the credibility criteria of Regulatory Guide 1.99, Rev. 2.
2. Since the Ginna surveillance data complies with the credibility criteria of Regulatory Guide 1.99, Rev. 2, the surveillance data should be used to determine the chemistry factor for the limiting Ginna vessel weld.
3. RG&E's and the NRC's calculated values of RT_{PTS} at the expiration of the Ginna license are within 3°F (265°F and 268°F) and are well below the 300°F screening criterion specified in 10 CFR 50.61 for circumferential welds. Since this conclusion is dependent upon the available chemistry data and surveillance data, it is subject to change when new data becomes available.

As a result of the extended power uprate to 1775 MWt, the impact of increased neutron fluence on the pre-uprate Pressurized Thermal Shock (PTS) analyses was evaluated *(Reference 26)*. PTS was re-evaluated following the pulling of Capsule N for extended plant operation out to 53 EFPY *(Reference 29)*. The evaluation assessed the impact of increased fluence on all of the reactor vessel beltline materials using the rules from 10CFR50.61. The limiting materials for PTS is the intermediate-to-lower shell girth weld. For this material the increased fluence from the uprate caused the end of plant life reference temperature to increase slightly from 270.6°F to 273°F, and increased slightly following Capsule N analysis to 275°F out to 53 EFPY, which is below the 300°F allowable temperature for circumferential weld materials. Consequently, at uprated plant conditions, the reactor vessel beltline materials continue to comply with the PTS screening criteria requirements of 10CFR50.61.

REFERENCES FOR SECTION 5.3

1. Westinghouse Electric Corporation, Rochester Gas and Electric R. E. Ginna Unit 1 Reactor Vessel Radiation Surveillance Program, WCAP 7254, May 1969.

2. Babcock & Wilcox, Master Integrated Reactor Vessel Material Surveillance Program, BAW 1543, Revision 4, February 1993.

3. Westinghouse Electric Corporation, Analysis of Capsule V from the Rochester Gas and Electric, R. E. Ginna Unit No. 1 Reactor Vessel Radiation Surveillance Program, FP-RA-1, April 1973. (Submitted by RG&E to the AEC by letter from G. E. Green to D. K. Skovholt, dated April 25, 1973.)

4. Westinghouse Electric Corporation, Analysis of Capsule R from the Rochester Gas and Electric, R. E. Ginna Unit No. 1 Reactor Vessel Radiation Surveillance Program, WCAP 8421, November 1974. (Submitted by RG&E to the NRC as an enclosure to Application for Amendment to the Operating License for the R. E. Ginna Nuclear Power Plant, dated March 6, 1975.)

5. Westinghouse Electric Corporation, Analysis of Capsule T from the Rochester Gas and Electric Corporation of R. E. Ginna Nuclear Plant Reactor Vessel Radiation Surveillance Program, WCAP 10086, April 1982. (Submitted by RG&E to the NRC as an enclosure to the Application for Amendment to the Operating License for the R. E. Ginna Nuclear Power Plant, dated December 8, 1982.)

6. Westinghouse Electric Corporation, Analysis of Capsule S from the Rochester Gas and Electric Corporation R. E. Ginna Reactor Vessel Radiation Surveillance Program, WCAP 13902, December 1993. (Submitted by RG&E to the NRC in a letter dated March 29, 1994.)

7. Letter from R. W. Kober, RG&E, to W. A. Paulson, NRC, Subject: Reactor Vessel Surveillance Capsule T, dated August 8, 1984.

8. Letter from A. R. Johnson, NRC, to R. C. Mecredy, RG&E, Subject: Issuance of Amendment 44 to Facility Operating License (Section 3.1.2), dated August 8, 1991.

9. Letter from J. N. Hannon, NRC, to J. H. Taylor, B&W Owners Group, Subject: B&W Report BAW 1543, Rev. 3, Master Integrated Reactor Vessel Material Surveillance Program (TAC No. 75131), dated June 11, 1991.

10. Westinghouse Electric Corporation, Rochester Gas and Electric Reactor Vessel Life Attainment Plan, March 1990.

11. T. Mayer, Summary Report on Reactor Vessel Integrity for Westinghouse Operating Plants, WCAP 10019, December 1981.

12. Letter from J. E. Maier, RG&E, to D. M. Crutchfield, NRC, Subject: Incident Evaluation Steam Generator Tube Rupture Incident, dated April 12, 1982.

13. NRC Generic Letter 92-01, Revision 1, Subject: Reactor Vessel Structural Integrity, 10 CFR 50.54(f), dated March 6, 1992.

14. Letter from R. C. Mecredy, RG&E, to A. R. Johnson, NRC, Subject: Reactor Vessel Structural Integrity, 10 CFR 50.54(f), Response to Generic Letter 92-01, Revision 1, dated July 2, 1992.

15. Generic Letter 92-01, Revision 1, Supplement 1, Subject: Reactor Vessel Structural Integrity, dated May 19, 1995.

16. Letter from R. C. Mecredy, RG&E, to A. R. Johnson, NRC, Subject: Response to NRC Generic Letter 92-01, Revision 1, Supplement 1, Reactor Vessel Structural Integrity, dated August 11, 1995.

17. Letter from R. C. Mecredy, RG&E, to A. R. Johnson, NRC, Subject: Six Month Response to NRC Generic Letter 92-01, Revision 1, Supplement 1, Reactor Vessel Structural Integrity, dated November 20, 1995.

18. Letter from G. S. Vissing, NRC, to R. C. Mecredy, RG&E, Subject: Closeout for RG&E Response to Generic Letter 92-01, Revision 1, Supplement 1 (TAC No. M92679), dated August 1, 1996.

19. Letter from R. C. Mecredy, RG&E, to G. S. Vissing, NRC, Subject: Use of Ratio Procedure in Determining RT_{PTS} Values for the R. E. Ginna Reactor Pressure Vessel, dated December 30, 1996.

20. Letter from G. S. Vissing, NRC, to R. C. Mecredy, RG&E, Subject: Request for Additional Information Regarding Reactor Pressure Vessel Integrity (TAC No. MA0546), dated April 3, 1998.

21. Letter from R. C. Mecredy, RG&E, to G. S. Vissing, NRC, Subject: Response to Request for Additional Information (RAI) Related to Reactor Pressure Vessel Integrity (TAC No. MA0546), dated August 25, 1998.

22. Letter from R. W. Kober, RG&E, to G. E. Lear, NRC, Subject: Pressurize Thermal Shock Rule - Six Month Submittal, dated January 13, 1986.

23. Letter from G. E. Lear, NRC to R. W. Kober, RG&E, Subject: Projected Values of Material Properties for Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events, R. E. Ginna Nuclear Power Plant, dated November 17, 1986.

24. Letter from A. R. Johnson, NRC, to R. C. Mecredy, RG&E, Subject: Pressurized Thermal Shock Evaluation (TAC No. M93827), dated March 22, 1996.

25. Dominion Engineering Inc. R-4814-00-01 Revision 1.0, Oct. 2006, "Reactor Vessel Tensioning Optimization Stress Report, Ginna Nuclear Power Plant."

26. Letter from M. Korsnick, CEG, to US NRC Document Control Desk, Subject: R.E. Ginna Nuclear Power Plant, License Amendment Request Regarding Extended Power

Uprate, (Attachment 5 — Licensing Report), dated July 7, 2005 (CMIS record ID "1001353").

27. WCAP-14684, R.E. Ginna Heatup and Cooldown Limit Curves for Normal Operation, June 1996.

28. Letter from T. Harding, Constellation Energy to U. S. NRC Document Control Desk, Subject: Reactor Vessel Surveillance Capsule "N" Results, Westinghouse Report WCAP-17036-NP, "Analysis of Capsule N from the R. E. Ginna Reactor Vessel Radiation Surveillance Program," Revision 0, dated May 29, 2009 (WPLNRC-1002144).

29. WCAP-17214-NP, R. E. Ginna Heatup and Cooldown Limit Curves for Normal Operation and Pressurized Thermal Shock Evaluation, Revision 0, dated July 26, 2010.

30. WCAP-17036-NP, Analysis of Capsule N from the R. E. Ginna Reactor Vessel Radiation Surveillance Program, Revision 1, dated September 14, 2010.

Table 5.3-1
REACTOR VESSEL SPECIFICATIONS

No.	Specification
1.	The American Society of Mechanical Engineers, Boiler and Pressure Vessel Code, Section III, Rules for Construction of Nuclear Vessels, 1965, and applicable code cases for Class A vessels. Code Cases: Upper Shell Course - 1332-1 Shell is fabricated of SA-336 manganese-molybdenum steel. Lower Head Ring - 1332-1 Ring is fabricated of SA-336 manganese-molybdenum steel.
2.	The American Society of Mechanical Engineers, Boiler and Pressure Vessel Code, Section IX, Welding Qualifications, 1965.
3.	ASA B31.1, Code for Pressure Piping, Section VI, Chapter 3, 1955.
4.	Westinghouse Atomic Power Division Equipment Specification 676206 except as amended by Westinghouse Electric Corporation, Atomic Power Division Contract No. 54-Q-49758-BP, dated November 15, 1965.
5.	The Babcock & Wilcox Company, Quality Control Department Specifications covering the topics of welding, nondestructive testing, heat treating, cleaning, and testing.
6.	Replacement reactor vessel closure head provided by PCR 2001-0042 was designed in accordance with ASME Boiler and Pressure Vessel Code Section III, 1995 edition with 1996 addenda.

Table 5.3-2
REACTOR VESSEL DESIGN DATA

Design/operating pressure, psig	2485/2235
Hydrostatic test pressure, psig	3110
Design temperature, °F	650
Overall height of vessel and closure head, ft-in.	39-1.3
Water volume (with core and internals in place), ft^3	2473
Minimum thickness of insulation, in.	3.0
Number of reactor closure head studs	48
I.D. of flange, in.	121.81
Inlet Nozzle I.D., in.	27.47
I.D. at shell, in.	132.0
Outlet nozzle, I.D., in.	28.97
Core flooding water, nozzle, in.	3.5
Minimum clad thickness, in.	0.156
Minimum lower head thickness, in.	4.125
Minimum vessel beltline thickness, in.	6.5
Closure head thickness, in.	5.375

Table 5.3-3
REACTOR VESSEL MATERIALS

Section	Materials
Dome plate (bottom)	SA-533, grade A
Cylindrical forgings	SA-508, Class 2
Shell course, flanges, and nozzle forgings	SA-508, Class 2
Cladding (stainless weld rod)	Type 304 equipment
Thermal shield and internals	A-240, type 304
Replacement reactor vessel closure head (PCR 2001-0042)	SA-508, grade3, class 1 forging with cladding, 1st layer: 309L, subsequent layers: 308L

Table 5.3-4
IDENTIFICATION OF BELTLINE MATERIALS

WELDS

Weld Location	Weld Process	Weld Control Number	Weld Wire Type	Flux Type	Postweld Heat Treatment
Nozzle shell to intermediate shell	Submerged arc	SA-1101	Mn-Mo-Ni	Linde 80	1100-1125°F-48 hr-FC
Intermediate shell to lower shell	Submerged arc	SA-847	Mn-Mo-Ni	Linde 80	1100-1125°F-48 hr-FC
Surveillance weld	Submerged arc	SA-1036	Mn-Mo-Ni	Linde 80	1100°F-11-1/4 hr-FC

FORGINGS

					Heat Treatment	
Component	Forging Number	Material Specs	Supplier	Austenitize	Temper	Stress Relief
Nozzle shell	123P118VA1	A336	Bethlehem	1550°F-11 hr-WQ	1220°F-22 hr-AC	1125°F-30 hr-FC
Intermediate shell	125S255VA1	A508 CL2	Bethlehem	1550°F-15-1/2 hr-WQ	1210°F-18 hr-AC	1125°F-30 hr-FC
Lower shell	125P666VA1	A508 CL2	Bethlehem	1550°F-9 hr-WQ	1220°F-12 hr-AC	1125°F-30 hr-FC
Surveillance	125S255VA1	A508 CL2	Bethlehem	1550°F-15-1/2 hr-WQ	1210°F-18 hr-AC	1100°F-11-1/4 hr-FC
Forgings	125P666VA1	A508 CL2	Bethlehem	1550°F-9 hr-AC	1220°F-12 hr-AC	1100°F-11 hr-FC
Inlet nozzle	ZT 2254-2	A508 CL2	Midvale Hepenstall Co	Not available		
Forgings	ZT 2289-2	A508 CL2	Midvale Hepenstall Co.	Not available		

AC air cooled
WQ water quenched
FC furnace cooled

Table 5.3-5
BELTLINE MATERIAL CHEMICAL COMPOSITION (WEIGHT PERCENT)

Forging Number	C	P	S	Mn	Si	Mo	Ni	Cr	Cu	V
123P118VA1	0.19	0.010	0.009	0.65	0.23	0.60	0.69	0.42	---	---
125S255VA1	0.18	0.010	0.007	0.66	0.23	0.58	0.69	0.33	0.07	0.02
125P666VA1	0.19	0.010	0.011	0.67	0.20	0.57	0.69	0.37	0.05	0.02
ZT-2254-2	0.19	0.012	0.014	0.59	0.21	0.58	0.71	0.37	0.09	---
ZT-2289-2	0.20	0.011	0.014	0.66	0.20	0.60	0.69	0.30	0.09	---
Weld Control Number										
SA-1101	0.07	0.021	0.014	1.28	0.52	0.37	0.60	0.16	0.26	---
SA-847	0.080	0.012	0.012	1.34	0.45	0.38	0.54	0.08	0.25	---
Surveillance weld (SA-1036)	0.075	0.012	0.016	1.31	0.59	0.36	0.56	0.59	0.23	---

Table 5.3-6a
MECHANICAL PROPERTIES OF BELTLINE MATERIALS - FORGINGS

Parameter					
Forging Number	123P118VA1	125S255VA1	125P666VA1	125S255VA1 Surveillance test results	125P666VA1 Surveillance test results
T_{NDT} °F	30	20	40	20	40
$R_{T\ NDT}$ °F[a]	30	20	40	20	40
Upper Shelf Energy (ft-lb[a]**)**	117	106	114	91	120
Yield strength ksi	66.87	67.25	63.50	78.22	62.72
Ultimate tensile strength ksi	88.00	88.25	85.00	97.19	83.65
Elongation (%)	25.50	26.25	26.25	23.30	26.35
RA (%)	73.50	70.10	71.05	66.85	70.75

a. Estimated based on NRC Standard Review Plan Section 5.3.2 and MTEB 5-2.

Table 5.3-6b
MECHANICAL PROPERTIES OF BELTLINE MATERIALS

Parameter			
Weld Control Number	SA-1484	SA-1101	Surveillance weld
Weld Wire Type	Mn-Mo-Ni	Mn-Mo-Ni	---
Flux Type	Linde 80	Linde 80	---
T_{NDT} °F[a]	0	0	0
Energy at 10°F ft-lb	45, 45, 46	58, 60, 36	54, 66.5, 71[b]
$R_{T\ NDT}$ °F	0[a]	0[a]	-19.5[c]
Shelf Energy (ft-lb)	---	---	79.0
Yield strength ksi	68.63	67.00	73.52
Ultimate tensile strength ksi	84.26	81.88	87.35
Elongation (%)	28.5	29.5	22.8
RA (%)	---	0	62.0

a. Estimated based on NRC Standard Review Plan Section 5.3.2 and MTEB 5-2.
b. Energy at 60°F.
c. Mean value from data in BAW-1803, Revision 1 and BAW-1920P

Table 5.3-7
SUMMARY OF PRIMARY-PLUS-SECONDARY STRESS INTENSITY FOR COMPONENTS OF THE REACTOR VESSEL

Location	Maximum Range of Stress Intensity (ksi)	
	Stress Intensity (ksi)	Allowable Stress $3S_m$ at Operating Temperature (ksi)
Closure Head at Flange	52.9[ab]	80.1
Vessel at Flange	52.3[a]	80.1
Closure Studs	97.2[a]	104.1[ac]
CRDM Nozzle CRDM Nozzle J-weld	79.0/35.6[d] 63.8	69.9[d] 69.9
Vent Nozzle Vent Nozzle J-weld	31.9 41.9	41.1 69.9
Outlet Nozzle Safe End Nozzle Support Pad	 39.9 49.2 n/a[e]	 49.2 80.1 n/a[e]
Inlet Nozzle Safe End Nozzle Support Pad	 35.8 38.8 n/a[e]	 49.2 80.1 n/a[e]
Safety Injection Nozzle	55.4	80.1
Vessel Wall Transition	32.2	80.1
Bottom Head to Shell Juncture	28.6	80.1<
Bottom Head Instrumentation Nozzle	37.1	69.9
Core Support Pad	52.5	69.9
External Support Bracket	41.2	80.1

a. Values reported from the Reactor Vessel Tensioning Optimization Stress Report, (Reference 25).

b. "Closure Head at Flange" historic location reported is at the junction of closure dome plate section to the forged flange ring section. The original closure head design included a weld at this location. The replacement Reactor Vessel Closure Head for the Ginna Reactor installed under PCR 2001-0042, is a one piece forging and the weld at the dome to flange location was eliminated. The value reported herein, is taken from the Reactor Vessel Tensioning Optimization Stress Report at the closure head flange mating surface and is reported as bounding for all stress levels in the head model used for the tensioning optimization stress report.

c. Value reported is at 2.7 Sm.

d. For the CRDM tube, the allowable range of stress is exceeded (79.0) but this is permissible since the range excluding thermal bending (35.6) is below $3S_m$.

e. The nozzle at the support pad is considered a peak stress location and consequently, only fatigue is considered at that location.

Table 5.3-8
SUMMARY OF CUMULATIVE FATIGUE USAGE FACTORS FOR COMPONENTS OF THE REACTOR VESSEL

Location	Cumulative Fatigue Usage Factor
Closure Head at Flange Vessel at Flange Closure Studs	0.386[a] 0.264[a] 0.972[a]
CRDM Nozzle Nozzle J-Weld	0.580 0.742
Vent Nozzle Nozzle J-Weld	0.009 0.494
Outlet Nozzle Safe End Nozzle Forging Support Pad	 n/a[b] 0.044 0.386
Inlet Nozzle Safe End Nozzle Forging Support Pad	 n/a[b] 0.033 0.061
Safety Injection Nozzles	0.219
Vessel Wall Transition	0.003
Bottom Head to Shell Juncture	0.002
Bottom Head Instrumentation Nozzle	0.228
Core Support Guides	0.132
External Support Brackets	0.979

a. Revised cumulative usage factors are provided from Reactor Vessel Tensioning Optimization work, (Reference 25).

b. Cumulative fatigue usage factors were not reported for the safe ends of the outlet and inlet nozzles because the nozzle-to-shell junction, not the safe end, was found to be the worst fatigue location.

Table 5.3-9
SUMMARY OF SURVEILLANCE CAPSULE RESULTS

Material	Copper (wt. %)	30 ft-lb Temperature Shift After Fluence				
		[a] 5.87×10^{18} n/cm^2	[b] 1.02×10^{19} n/cm^2	[c] 1.69×10^{19} n/cm^2	[d] 3.64×10^{19} n/cm^2	[e] 5.8 x 1019 n/cm2
Weld SA-1036	0.23	146.7 °F	156.2 °F	149.7 °F	212.2 °F	216.9°F
Forging 125P666VA1	0.05	34.7 °F	57.5 °F	33.6 °F	45.8 °F	91.1°F
Forging 125S255VA1	0.07	0 °F	20.1 °F	0 °F	76.8 °F	76.4°F

a. Analysis of Capsule V, *Reference 30*.

b. Analysis of Capsule R, *Reference 30*

c. Analysis of Capsule T, *Reference 30*.

d. Analysis of Capsule S, *Reference 30*

e. Analysis of Capsule N, *Reference 30*

Table 5.3-10
COMPARISON OF SURVEILLANCE MATERIAL 30 FT-LB TRANSITION TEMPERATURE SHIFTS AND UPPER SHELF ENERGY DECREASES WITH REGULATORY GUIDE 1.99, REVISION 2, PREDICTIONS

			30 ft-lb Transition Temperature Shift		Upper Shelf Energy Decrease	
Material	**Capsule**	**Fluence ($x 10^{19} n/cm^2$)**	**Predicted[a] (°F)**	**Measured (°F)[b]**	**Predicted[a] (%)**	**Measured (%)**
Intermediate Shell Forging 125S255 (Longitudinal)	V	0.587	37.4	0.0[c]	14.5	3.7
	R	1.02	44.2	20.1	16.5	-1.6
	T	1.69	50.4	0.0[c]	19	-8.8
	S	3.64	58.8	76.8	23	0.7
	N	5.8	62.9	76.4	26	4.1
Lower Shell Forging 125P666 (Longitudinal)	V	0.587	26.4	34.7	13	10.1
	R	1.02	31.2	57.5	15	15.4
	T	1.69	35.5	33.6	17	18.4
	S	3.64	41.4	45.8	21	18.4
	N	5.8	44.3	91.1	23	18.5
Weld Metal (Heat # 61782)	V	0.587	135.2	146.7	32.5	30.1
	R	1.02	159.7	156.2	37	38.1
	T	1.69	181.8	149.7	40.5	33.3
	S	3.64	212.1	212.2	49	33.9
	N	5.8	227.2	216.9	54	34.3
HAZ Material	V	0.587	---	30.7	---	-50.0
	R	1.02	---	58.6	---	8.0
	T	1.69	---	41.0	---	-30.8
	S	3.64	---	38.9	---	-15.0
	N	5.8	---	107.7	---	1.9

a. Based on Regulatory Guide 1.99, Revision 2, methodology using mean wt % values of copper and nickel.

b. Calculated by CVGraph Version 5.3 using measured Charpy data

c. ΔRT_{NDT} value was determined to be negative, but physically a reduction should not occur, therefore a conservative valueof zero is not used.

5.4 COMPONENT AND SUBSYSTEM DESIGN

5.4.1 REACTOR COOLANT PUMPS

5.4.1.1 General Description

5.4.1.1.1 Centrifugal Pump

Each reactor coolant loop contains a vertical single-stage centrifugal type pump, which employs a controlled leakage seal assembly. A view of a controlled leakage pump is shown in Figure 5.4-1 and the principal design parameters for the pumps are listed in Table 5.4-1. The reactor coolant pump estimated performance and net positive suction head characteristics are shown in Figure 5.4-2. The performance characteristics are common to all of the higher specific speed centrifugal pumps and the "knee" at about 45% design flow introduces no operational restrictions, since the pumps operate at full speed.

The reactor coolant pump performance characteristics were updated as a result of the conversion to an 18 month fuel cycle to incorporate the currently installed internals and impellers. The updated characteristics are illustrated on Figures 5.4-2a through 5.4-2d for both hot and cold conditions. The tabular data upon which the curves are based is shown as Tables 5.4-3 and 5.4-4, as submitted by *Reference 56*.

Reactor coolant is pumped by the impeller attached to the bottom of the rotor shaft. The coolant is drawn up through the impeller, discharged through passages in the diffuser and out through a discharge nozzle in the side of the casing. The motor-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 austenitic stainless steel or equivalent corrosion resistant materials.

5.4.1.1.2 Controlled Leakage Shaft Seal

The pump employs a controlled leakage seal assembly to restrict leakage along the pump shaft, as well as a secondary seal that directs the controlled leakage out of the pump, and a third seal that minimizes the leakage of water and vapor from the pump into the containment atmosphere.

A portion of the high-pressure water flow from the charging pumps is injected into the reactor coolant pump between the impeller and the controlled leakage seal. The shaft seal arrangement is shown in Figure 5.4-3. Part of the flow enters the reactor coolant system through a labyrinth seal in 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. A very small amount that leaks through the secondary seal is also collected and removed from the pump.

The original seal material for the reactor coolant pump controlled leakage (number one) seal assembly was aluminum oxide. These seals were replaced during pump maintenance during the 1991 and 1997 refueling outages. The new seal assemblies contain silica nitrate. The new material was an improvement over aluminum oxide since silica nitrate seals are capable of surviving minor rubbing of the seal faces with no degradation in seal performance. New high temperature o-rings have also been installed on both reactor coolant pump seal assemblies.

Component cooling water (CCW) is supplied to the motor bearing oil coolers and the thermal barrier cooling coil. Motor bearing lube-oil level indication is provided in the control room.

Reactor coolant pump seal operation requires one of two water sources for operation. The normal supply to the seals is cooled and filtered seal injection water from the charging system. If seal injection were lost, the reactor coolant system water would pass up through the labyrinth seal and thermal barrier to the number one seal. This water is unfiltered and is cooled by the thermal barrier by component cooling water. Since it is unlikely that both seal injection and component cooling water would be terminated, if termination did occur, in most cases, the reactor would be shutdown and the reactor coolant pumps tripped.

Essential services for reactor coolant pump operation are available during a containment isolation signal unless a safety injection signal occurs with a loss of offsite power. Seal injection from the chemical and volume control system is terminated by a charging pump trip upon receipt of a safety injection signal. However, component cooling water (CCW) services to the reactor coolant pump remain in operation independent of the safety injection and/or containment isolation signals, unless offsite power is lost. A loss of offsite power coincident with a safety injection signal will trip the component cooling water (CCW) pumps, thereby terminating component cooling water (CCW) flow to the reactor coolant pumps. Since the reactor coolant pumps operate from offsite power, the reactor coolant pumps will also be tripped and will not be available while offsite power is lost.

An extensive test program was conducted for several years to develop the controlled leakage shaft seal for pressurized water reactor applications. Long-term tests were conducted on less than full scale prototype seals as well as on full size seals. At the time of initial operation of Ginna Station, operating experience with large size, controlled leakage shaft seal pumps was available from plants such as San Onofre Unit 1 and Connecticut Yankee.

5.4.1.1.3 Pump Motor

The squirrel cage induction motor driving the pump is air cooled and has oil-lubricated thrust and radial bearings. A water-lubricated bearing provides radial support for the pump shaft.

5.4.1.1.4 Vibration Measurement

Each pump is equipped with two vibration pickups (seismic displacement) mounted at the bottom of the motor casings and two shaft (non-contact) pickups mounted below the coupling on the seal housing, located 90 degrees apart, to determine pump shaft vibration. Vibration levels are checked periodically or whenever an abnormal condition is expected. A keyphasor proximity transducer is provided to supply vibration information for diagnostic testing. The data includes dynamic and static vibration data, which is digitized and transmitted to the general purpose computer where it can be reduced and displayed in the form of plots, alarm lists, reports, and logs.

5.4.1.1.5 Lube Oil Leakage Collection System

A system of drip pans, splash guards, and enclosures for collecting reactor coolant pump lube-oil leakage is installed on each reactor coolant pump to reduce the potential for fire caused by lube-oil contacting and igniting on hot reactor coolant system components. Drain piping from seven collection points on each reactor coolant pump directs leakage to an oil collection system storage tank.

5.4.1.2 Pump Flywheel Integrity

5.4.1.2.1 Pump Overspeed

Precautionary measures, taken to preclude missile formation from primary coolant pump components, ensure that the pumps will not produce missiles under any anticipated accident condition.

The primary coolant pumps run at 1189 rpm and may operate briefly at overspeeds up to 109% (1295 rpm) during loss of external load. At 1189 rpm, the bore stress due to rotation is 14,000 psi. For conservatism, however, 125% of operating speed was selected as the design overspeed for the primary coolant pumps. The maximum pump overspeed on loss of external load is 118% based on turbine overspeed with failure of the turbine steam control valve.

For the overspeed condition, which would not persist for more than 30 sec, pump operating temperatures would remain at about the design value. Furthermore, the probability of attaining a post-loss-of-coolant accident overspeed sufficient to cause loss of flywheel integrity is very remote. This probability would be the product of the conditional probabilities of (1) the break of a large primary coolant pipe, (2) the failure of associated pipe restraints such that the break could become a double-ended guillotine break (calculations show a significantly smaller overspeed for a realistic break), and (3) the loss of electric power to the pump motor such that there is no electric braking effect and the pump is permitted to accelerate freely. Also, the pump would have to remain free-spinning; seizure of the shaft or motor components could prevent overspeed.

Each component of the primary pumps has been analyzed for missile generation. Any fragments would be contained by the heavy stator. The same conclusion applies to the impeller because the small fragments that might be ejected would be contained by the heavy casing. At an overspeed of 1486 rpm, the maximum tangential stress reaches 21,500 psi, which is less than 50% of the minimum yield strength (50,000 psi) at the operating temperature.

5.4.1.2.2 Pump Flywheel Design and Fabrication

The primary coolant pump flywheels are shown in Figure 5.4-4. As for the pump motors, the most adverse operating condition of the flywheels is visualized to be the loss-of-load situation. The following conservative design operating conditions preclude missile production by the pump flywheels. The wheels are fabricated from rolled, vacuum-degassed, ASTM A-533 steel plates. Flywheel blanks are flame-cut from the plate, with allowance for exclusion of flame affected metal. A minimum of three Charpy V-notch tests are made from each plate parallel and normal to the rolling direction, to determine that each blank satisfies design requirements. A nil ductility transition temperature (NDTT) less than +10°F is specified.

The flywheel material has a minimum yield strength of 50,000 psi and tensile strength of 80,000 psi. The finished flywheels are subjected to 100% volumetric ultrasonic inspection. The finished machined bores are also subjected to magnetic particle or liquid penetrant examination. The pump flywheels are mounted on a shaft of radius 4.2 in. and consist of two large steel disks bolted together. The disks are 75 in. and 65 in. in diameter.

These design-fabrication techniques yield flywheels with primary stress at operating speed (shown in Figure 5.4-5) less than 50% of the minimum specified material yield strength at room temperature (100°F to 150°F). Bursting speed of the flywheels has been calculated by Ginna, on the basis of Griffith-Irwin's results, (*Reference 1*) to be 3900 rpm. The NRC staff has independently determined the bursting speed for the flywheel to be 3400 rpm (*Reference 2*). Regulatory Guide 1.14 requires that the margin against ductile failure relative to the minimum specified yield strength be 3 and 1.5 at normal operating speed and design overspeed, respectively. For the Ginna Station flywheels, the margin is 3.7 at normal operating speed and 2.33 at a design overspeed of 125%. Therefore, they have a wide margin of safety against ductile failure and the requirements of Regulatory Guide 1.14 are satisfied.

5.4.1.2.3 Flywheel Design Evaluation

A fracture mechanics evaluation was made on the reactor coolant pump flywheel. This evaluation considered the following assumptions:

A. Maximum tangential stress at an assumed design overspeed of 125% compared to a maximum expected overspeed of 109%.

B. A crack through the thickness of the flywheel at the bore.

C. 400 cycles of startup operation in 40 years.

Using critical stress intensity factors and crack growth data attained on flywheel material, the critical crack size for failure was greater than 17 in. radially and the crack growth data was 0.030 in. to 0.060 in. per 1000 cycles.

The NRC staff performed an independent fracture mechanics evaluation to determine the speed at which unstable crack propagation would occur for a 4-in. crack emanating from the keyway. The results of the evaluation showed that a 4-in. crack would remain stable at speeds up to 3000 rpm. Therefore, a very large crack, on the order of 10 in., would remain stable at an overspeed of 1486 rpm (*Reference 2*).

5.4.1.2.4 Pump Seismic Design

The original design specifications for the reactor coolant pumps include as a design condition the stresses generated by a maximum hypothetical earthquake ground acceleration of 0.2g. Within the scope of SEP Item III-6, the reactor coolant pump was analyzed with respect to stresses induced by 0.8g horizontal and 0.54g vertical loadings. It was shown that the stresses are lower than the ASME Code allowable and that the reactor coolant pumps have acceptable seismic resistance.

5.4.1.2.5 Inservice Inspection Program

The inservice inspection program for the installed reactor coolant pump (RCP) flywheels consists of either an ultrasonic (UT) examination over the volume from the inner bore of the flywheel to the circle of one-half the outer radius or conduct an ultrasonic (UT) and a surface (MT and/or PT) examination of exposed surfaces defined by the volume of the disassembled flywheels once every 20 years *(Reference 66)*. The Ginna Station reactor coolant pump flywheels meet the requirements of Regulatory Guide 1.14. Additional information on the overall inservice inspection program is contained in the Inservice Inspection (ISI) Program document.

5.4.1.2.6 Conclusion

Following a hypothetical bearing seizure, the flywheel is not expected to twist off. Therefore, it has been concluded that the reactor coolant pumps are not sources of missiles and the engineered safety features are not in jeopardy.

5.4.2 STEAM GENERATORS

Each loop contains a vertical shell and U-tube steam generator. A steam generator of this type is shown in Figure 5.4-6. Principal design parameters are listed in Table 5.4-2. The steam generators are designed and manufactured in accordance with Section III of the ASME Boiler and Pressure Vessel Code (see Section 3.2).

Reactor coolant enters the inlet side of the channel head at the bottom of the steam generator through the inlet nozzle, flows through the U-tubes to an outlet channel, and leaves the generator through another bottom nozzle. The inlet and outlet channels are separated by a partition. Manways are provided to permit access to the U-tubes and moisture separating equipment.

Feedwater to the steam generator enters just above the top of the U-tubes through a feedwater ring. The water flows downward through an annulus between the tube wrapper and the shell and then upward through the tube bundle where part of it is converted to steam. The steam-water mixture from the tube bundle passes through centrifugal type separators which impart a centrifugal motion to the mixture and separate the water particles from the steam. The water leaves the separator through the separator return cylinder and combines with the feedwater for another pass through the tube bundle. The steam rises through additional separators which limit the moisture content of the steam to 0.1% or less at design load conditions.

Steam generator performance gradually deteriorates over time. A pressure transmitter is installed in each steam generator to monitor pressure in the upper portion of the generator. The pressure signals are monitored by the plant process computer system and used to trend and analyze degradation of steam generator performance.

A loose-parts monitoring system was installed for each steam generator in 1982. The system is designed to indicate the presence of potentially damaging foreign objects in either the primary channel head or the secondary side of the tubesheet. Information specific to the secondary side of the steam generator is contained in Section 10.3.2.

5.4.2.1 Replacement Steam Generator Materials

The steam generator pressure boundary is constructed of ferritic steel, either carbon steel or low alloy. The heat transfer tubes are SB-163 Alloy 690 as permitted by ASME Code Case N-20-3. The interior surfaces of the channel heads and nozzles are clad with austenitic stainless steel, and the side of the tubesheet in contact with the reactor coolant is clad with Alloy 600 weld metal. The primary head is a single forging of SA-508 Cl 3 material with integrally forged manways and inlet/outlet nozzles. There are stainless steel safe ends of SA-336-316N/316LN welded to each of these nozzles. The tubesheet is also a SA-508 Cl 3 forging. The divider plate is machined from SB-168 Alloy 690 plate and welded around its entire periphery to either the primary head or the tubesheet. The tubes are welded to the cladding on the tubesheet face after which each tube is hydraulically expanded through the full tubesheet thickness. The tubes are supported by lattice bars and U-bend flat bars made of SA-240 Type 410S stainless steel.

5.4.2.2 Steam Generator Inservice Inspection

Inservice inspection of steam generators is conducted in accordance with the Inservice Inspection (ISI) Program document. A program of periodic steam generator inspections, designed to meet the Ginna Technical Specifications and EPRI PWR Steam Generator Program documents, is conducted to provide assurance of acceptable steam generator performance. The inservice inspection program for the reactor coolant pressure boundary is discussed in Section 5.2.4. As part of the response (*Reference 64*) to NRC Generic Letter 97-06 (*Reference 65*), RG&E committed to develop a secondary side inspection program to ensure that degradation of steam generator internals does not adversely affect tube integrity, as defined within the Steam Generator Program.

5.4.2.3 Replacement Steam Generator Design Evaluation

Structural and seismic evaluation of replacement steam generators primary and secondary side pressure boundaries demonstrate that these components satisfy ASME III, Division 1, Class I design requirements for service levels A, B, C, and D (normal, upset, emergency and faulted conditions, respectively). Steam generator internal components are not governed by the ASME Boiler & Vessel Code. However, ASME III Subsection NB for Class 1 components is used as a guide for structural analysis of RSG internal components. RSG internal components are required to withstand all specified loadings to maintain heat transfer capability during and following a design basis earthquake. In addition, tubes must be shown not to deform as a result of a design basis earthquake. This helps to ensure that safe shutdown capability is maintained. The RSG structural evaluation is documented in a Code Stress Report.

The structural analysis demonstrates that for an instantaneous full rupture of the steam line downstream of the steam outlet nozzle occurring during normal full power operation, the tube integrity is maintained. The structural evaluation of the tubing for level D is in accordance with the ASME Boiler and Pressure Vessel Code Section III requirements. Tube Integrity must also be maintained for a small steam line break to Level C criteria.

A flow-induced vibration (FIV) analysis is performed to confirm that the tube bundle is adequately supported to avoid significant levels of tube vibration. An FIV Analysis Report and a

Wear Analysis Report verify that the vibration of the RSG internals does not result in excessive wear or fatigue throughout the tube bundle and U-bend regions.

The three pertinent cross-flow FIV mechanisms in the RSG are vortex shedding resonance, random turbulence excitation, and fluid elastic instability. The FIV analysis verifies that excessive tube vibration from these sources is avoided. Particular areas of emphasis are the tube bundle entrance and the U-bend region.

5.4.2.4 High Cycle Fatigue Failure of Original Steam Generator Tubes

This section was applicable to the original Westinghouse Model 44 steam generators which were replaced in 1996 and does not apply to the BWI replacement steam generators. This section is retained for historical purposes.

NRC Bulletin 88-02 (*Reference 11*) requested that holders of operating licenses or construction permits for Westinghouse reactors with carbon steel support plates implement actions to minimize the potential for a steam generator tube rupture event caused by a rapidly propagating fatigue crack, such as occurred at North Anna Unit 1 on July 15, 1987. North Anna experienced a circumferential tube rupture at the top of the top tube support plate, which was attributed to limited displacement, fluid elastic instability. The unstable condition was caused by tube denting at the support plate. The Ginna steam generator tubes were examined and analyzed to determine their susceptibility to high cycle fatigue failure (*References 50* and *51*). Those tubes identified as being potentially susceptible to high cycle fatigue or susceptible to the consequences of fatigue failure were stabilized unless they had been previously plugged. This action was found acceptable by the NRC (*Reference 12*). Subsequent to the initial evaluation, Westinghouse performed a re-evaluation based on updated information and, as a result, additional Ginna Station steam generator tubes were plugged and stabilized during the 1992 MODE 6 (Refueling) outage (*References 13* and *52*).

The NRC concluded that the actions taken in response to Bulletin 88-02 were acceptable as long as administrative controls were developed that ensured updated stress ratio and fatigue usage calculations were performed in the event of any significant changes to the steam generator operating parameters (*Reference 53*). As a result, during startup core physics testing coming out of a MODE 6 (Refueling) outage, steam generator pressure was verified to be greater than 675 psig and if steam generator pressure fell below 675 psig, power escalation was terminated. In addition, during plant coastdown operations, steam generator pressure was maintained above 660 psig (675 psia).

5.4.3 REACTOR COOLANT PIPING

5.4.3.1 General

<u>5.4.3.1.1 General Description</u>

The general arrangement of the reactor coolant system piping is described in Section 5.1.1. Piping design data are presented in Table 5.1-2.

The austenitic stainless steel reactor coolant piping and fittings which make up the loops are 29 in. I.D. in the hot legs, 27-1/2 in. I.D. in the cold legs, and 31 in. I.D. between the steam generators and the reactor coolant pump inlet. The pressurizer relief line, which connects the pressurizer safety and pressurizer power operated relief valves (PORVs) to the discharge nozzle flange on the pressurizer relief tank, is constructed of carbon steel.

Smaller piping, including the pressurizer surge and spray lines, drains, and connections to other systems, is austenitic stainless steel. All joints and connections are welded except for stainless steel flange connections to the carbon steel pressurizer relief tank and the connections at the relief and safety valves.

Thermal sleeves are installed at the following locations where high thermal stresses could otherwise develop due to rapid changes in fluid temperature during MODES 1 and 2 transients:

- Return line from the residual heat removal loop.
- Both ends of the pressurizer surge line.
- Pressurizer spray line connection to the pressurizer.
- Charging lines and alternate charging line connections.

All valve surfaces in contact with reactor coolant are austenitic stainless steel or equivalent corrosion resistant materials. Connections to stainless steel piping are welded.

All relief valves used in systems handling radioactive fluids are of the closed bonnet design and are constructed of stainless steel.

5.4.3.1.2 Pressure Isolation of Low-Pressure Systems

Three systems have a direct interface with the reactor coolant system pressure boundary but have a design pressure rating below that of the reactor coolant system. These systems are the chemical and volume control system (Section 9.3.4), the safety injection system (Section 6.3), and the residual heat removal system (Section 5.4.5). The isolability of the low-pressure systems from the reactor coolant system is discussed in the respective sections.

In response to Generic Letter 87-06, *Reference 16* lists all pressure isolation valves (PIV) that separate the high pressure reactor coolant system from attached lower pressure systems and the periodic tests or other measures performed to ensure the integrity of the isolation valves as an independent barrier at the reactor coolant pressure boundary.

5.4.3.2 Reactor Coolant System Vents

5.4.3.2.1 General

The requirements for reactor coolant system high point vents are stated in 10 CFR 50.44, Standards for Combustible Gas Control Systems in Light Water Cooled Power Reactors. They are further described in Standard Review Plan Section 5.4.12, Reactor Coolant System High Point Vents, and in Item II.B.1 of NUREG 0737, Clarification of TMI Action Plan Requirements. In response to these and previous requirements, RG&E has submitted information in *References 17 through 21* in support of the vent system at Ginna Station.

The function of the high point vent system is to vent noncondensible gases from the high points of the reactor coolant system to ensure that core cooling during natural circulation will not be inhibited. The Ginna Station reactor vessel head vent system provides venting capability from the reactor vessel head while the pressurizer can be vented through the existing pressurizer power operated relief valves (PORVs). The noncondensible gases, steam, and/or liquids vented from the reactor vessel head are piped and discharged directly to the refueling cavity and the discharges from the pressurizer are piped to the pressurizer relief tank. The reactor vessel head vent system is designed to vent a volume of gas at least equal to one half of the reactor coolant system volume in 1 hour. Flow restriction orifices in the reactor vessel head vent system paths, however, limit the flow from a pipe rupture or from inadvertent actuation of the vent system to less than the capability of the reactor coolant makeup system. Compliance with 10 CFR 50.46, Acceptance Criteria for Emergency Core Cooling Systems (ECCS) for Light Water Nuclear Power Reactors, is not affected by the addition of the reactor vessel head vent system.

5.4.3.2.2 Reactor Head Vent System Description

The reactor vessel head vent system consists of two redundant vent paths from the reactor vessel head to the refueling cavity, each containing a manually operated valve followed by two solenoid-operated valves in series that are remotely controlled from the main control room. The two paths are connected to a single 3/4-in. reactor head vent pipe downstream of a manually operated valve. A degree of redundancy has been provided by powering each reactor vessel head vent system vent path from a separate emergency bus to ensure that reactor coolant system venting capability from the reactor vessel head is maintained. Reactor vessel head vent system valve seat leakage is detected, together with other unidentified reactor coolant system leakage, by way of containment radiation (R-11 and R-12) and containment sump A level monitoring (LT-2039, LT-2044, and sump pump actuation) in accordance with the Technical Specifications. The manual valves in each vent path provide a means of isolating that path in the event of leakage of the normally closed solenoid valves. The pressurizer power operated relief valves (PORVs), used to vent the pressurizer, function as a part of the automatic reactor coolant system pressure control system but they can also be manually controlled from the main control room. The pressurizer power operated relief valves (PORVs) and block valves receive power from emergency buses and have positive valve position indication in the main control room. The portion of each reactor vessel head vent system path up to and including the second normally closed valve forms a part of the reactor coolant pressure boundary and thus must meet reactor coolant pressure boundary requirements. Therefore, the piping out to the flow restriction orifices is ASME Code, Section III, Class 1, and the system beyond the orifices to the second vent valves is ASME Code, Section III, Class 2, in compliance with 10 CFR 50.55a and Regulatory Guide 1.26. The entire reactor vessel head vent system is designated Seismic Category I in compliance with Regulatory Guide 1.29. The reactor vessel head vent system is designed for pressures and temperatures corresponding to the reactor coolant system design pressure and temperature.

In addition, the vent system materials are compatible with the reactor coolant chemistry and were fabricated and tested in accordance with ASME Code, Section III, subsections NB, NC, and NF, and plant specifications. The reactor vessel head vent system and the pressurizer power operated relief valve (PORV) vent system are separated and protected from missiles and the dynamic effects of postulated piping ruptures. The design of the portions of the reactor vessel head vent system up to and including the second normally closed valve conforms to all reactor coolant pressure boundary requirements, including 10 CFR 50.55a and the applicable portions of General Design Criteria 1, 2, 4, 14, 30, and 31. The essential operation of other safety-related systems will not be impaired by postulated failures of reactor vessel head vent system components.

The reactor vessel head vent system design has been reviewed to ensure an acceptably low probability for inadvertent or irreversible actuation of the vent system. Each vent path has two solenoid-operated globe valves in series, and each valve has a separate key-locked control switch that is locked closed during normal reactor operation. The valves are powered by 125-V dc emergency power supplies and fail to the closed position in the event of loss of power. No single active component failure or human error should result in inadvertent opening or failure to close after intentional opening of the reactor vessel head vent system.

The locations where the reactor vessel head vent system normally discharges to the containment atmosphere in the vicinity of the refueling cavity are in areas that ensure good mixing with the containment atmosphere to prevent the accumulation or pocketing of high concentrations of hydrogen in compliance with 10 CFR 50.44, Standards for Combustible Gas Control System in Light Water Cooled Power Reactors. Additionally, these locations are such that the operation of safety-related systems would not be adversely affected by the discharge of the anticipated mixtures of steam, liquids, and noncondensible gases.

The reactor vessel head vent system valves are exercised periodically and proper valve position is visually verified. Operability testing of the pressurizer power operated relief valves (PORVs) and block valves is specified in the Ginna Pump and Valve Inservice Testing Program and is in accordance with the Code for Operation and Maintenance of Nuclear Power Plants.

Reactor vessel head vent system parameters are given in Table 5.4-5.

5.4.4 MAIN STEAM LINE ISOLATION SYSTEM

Each steam line has a fast-closing Main Steam Isolation Valve (MSIV) and a Main Steam non-return check valve. These four valves prevent blowdown of more than one steam generator for any break location even if one valve fails to close (Section 15.1.5. See also Sections 10.3.2.6 and 10.3.2.7).

The main steam isolation valves are 30-in. pipe size, 24-in. seat diameter, ANSI 600-lb rating, Atwood and Morrill Company, Inc., swing-disk check valves. The open position of the disk is at full horizontal held open against the steam flow by an air cylinder. The valves have stainless steel disks and disk arms. The stiffness of the disk arms is designed to reduce valve strains developed during closure following a postulated downstream pipe break. The disks and disk arms are also designed to uniformly transfer the kinetic energy from the disk to the valve body during impact. The valve disks and disk arms are stainless steel in order to better withstand the local strains in the contact region. The overall design of the valve will reduce the likelihood of damage due to spurious closure and will prevent excessive degradation of the valves during normal service.

The main steam isolation valves serve to limit an excessive reactor coolant system cooldown rate and resultant reactivity insertion following a main steam line break incident. Their ability to close upon signal is verified at periodic intervals. A closure time of 5 sec was selected as being consistent with expected response time for instrumentation as described in the steam line break incident analysis.

The purpose of the fast acting valves is to prevent continuous blowdown from more than one steam generator following any steam line rupture even with failure of any single check or isolation valve. Flow from a second steam generator for up to 7 sec (including 2 sec for instrument response time) following a steam line break has a negligible effect on the peak core power eventually attained from continuous blowdown of one steam generator. The main effect of flow from the second steam generator is to reduce the pressure faster during the initial portion of the transient, thereby causing safety injection flow to occur earlier. Flow from the second steam generator has little effect on the reactivity insertion rate, which occurs after the reactor pressure has fallen to the safety injection pump shutoff head, since by this time the isolation valve has closed.

It should be noted that 5 sec is the maximum allowable closure time for the valves with no flow passing through them. Tests with no flow have shown that closure time to be less than 5 sec. With the flow, which will exist through a valve following a steam line rupture, the closure time will be considerably faster than 5 sec since flow will tend to force the valve to its closed position.

The steam line break accident analysis is presented in detail in Section 15.1.5. The main steam isolation valves are tested in accordance with the Technical Specifications.

5.4.5 RESIDUAL HEAT REMOVAL (RHR) SYSTEM

5.4.5.1 Design Bases

The residual heat removal loop is designed to remove residual and sensible heat from the core and reduce the temperature of the reactor coolant system during the second phase of plant cooldown. During the first phase of cooldown, the temperature of the reactor coolant system is reduced by transferring heat from the reactor coolant system to the steam and power conversion system.

All active loop components which are relied upon to perform their function are redundant, except as described in Section 5.4.5.3.4.

The loop design provides means to detect radioactivity migration to the ultimate heat sink environment and includes provisions, which initiate adequate action for continued core cooling when required, in the event radioactivity limits are exceeded.

The loop design precludes any significant reduction in the overall design reactor shutdown margin when cooling water is introduced into the core for decay heat removal during the emergency core cooling recirculation mode of operation.

The loop design includes provisions to enable periodic hydrostatic testing to applicable code test pressure.

Loop components, whose design pressure and temperature are less than the reactor coolant system design limits, are provided with redundant isolation means and overpressure protective devices.

5.4.5.2 System Design

The residual heat removal loop consists of two heat exchangers, two pumps, piping, and the associated valves and instrumentation (Drawing 33013-1247). After the steam generators have been used to reduce the reactor coolant temperature to 350°F, decay heat cooling is initiated by aligning the residual heat removal pumps to take suction from the reactor coolant system loop A hot leg and discharge through the residual heat removal heat exchangers to the loop B cold leg. With both pumps and heat exchangers in operation, residual heat removal flow is adjusted to maintain a cooldown rate of less than 80°F/hr. If only one pump and heat exchanger are available, cooldown is accomplished at a lower rate.

The heat from the residual heat removal heat exchangers is transferred to the component cooling water (CCW) system (Section 9.2.2), and from the component cooling water (CCW) system to the service water (SW) system (Section 9.2.1). The minimum pump head on the residual heat removal pumps is 150 psig, the component cooling water system operating pressure is 80 psig, and the service water (SW) system operating pressure is 75 psig; therefore, in the event of a residual heat removal heat exchanger tube leak, the flow of impurities should be away from the primary system.

During plant shutdown, the cooldown rate of the reactor coolant is controlled by regulating the flow through the tube side of the residual heat removal heat exchangers. A bypass line and control valve around the residual heat exchangers are used to maintain a constant flow through the residual heat removal loop. To minimize the potential for flow-induced vibration in the residual heat removal heat exchangers, as of 1994 component cooling water (CCW) flow has been limited to approximately 1800 gpm through the shell side of each exchanger. See Section 9.2.2.4.1.6.

The pumps and heat exchangers are each half-capacity for the normal heat removal function; however, they are full capacity for their alternative function, which is low-head safety injection during loss-of-coolant accident conditions.

The residual heat removal pumps are driven by drip-proof type motors with either Class B PMR (protective moisture resistant) insulation or Class H insulation to be capable of operation in high humidity conditions. They are also equipped with Seismic Category I splash barriers to protect the motors in the event of a pipe line break in the area, which could possibly spray and wet the motors. Two access doors have been installed on each motor splash barrier near the radial and thrust bearing vibration transducer buttons to improve access for motor vibration measurements to be taken with hand-held transducers.

The residual heat removal pumps are powered from separate safeguards buses. Emergency power for these buses is available from either of two separate emergency diesel generators (Section 8.3). Two reactor coolant drain pumps, also powered from separate safeguards buses, can be used to back up the residual heat removal pumps for core cooling. The loss of residual heat removal pumps and loss of reactor coolant drain pumps are addressed by emergency procedures.

Double, remotely operated valving is provided to isolate the residual heat removal loop from the reactor coolant system (Section 5.4.5.3). During reactor operation all equipment of the low-head injection and residual heat removal loop is idle, and the associated isolation valves are closed. During an accident condition fission products are recirculated through the exterior piping system. To obtain the total radiation dose to the public due to leakage for this system, the potential leaks have been evaluated and discussed in Sections 6.2 and 15.6.4.

5.4.5.2.1 Codes and Classifications

All piping and components were designed to the applicable codes and standards listed in Table 3.2-1. Austenitic stainless steel piping is used in the residual heat removal loop, which contains reactor coolant, and in the spent fuel pool (SFP) cooling system, which contains water without corrosion inhibitor.

Pressure retaining components (or compartments of components) through which reactor coolant circulates at pressures and temperatures significantly less than the reactor operating conditions at rated power comply with the following codes:

A. System pressure vessels - ASME Boiler and Pressure Vessel Code, Section III (1965), Class C, including paragraph N-2113.

B. System valves, fittings, and piping - USAS B31.1 (1965), including nuclear code cases.

A comparison of the requirements of the original design codes and standards to the current requirements is presented in Section 3.2.2.

5.4.5.2.2 Components

Residual heat removal system component design data is given in Table 5.4-6.

5.4.5.2.2.1 Heat Exchangers

The two residual heat removal heat exchangers are of the shell and U-tube type with the tubes welded to the tubesheet. Reactor coolant circulates through the tubes, while component cooling water (CCW) circulates through the shell side. The tubes and other surfaces in contact with reactor coolant are austenitic stainless steel and the shell is carbon steel.

5.4.5.2.2.2 Pumps

The two residual heat removal pumps are horizontal, centrifugal units with special seals to prevent reactor coolant leakage to the atmosphere. All pump parts in contact with reactor coolant are austenitic stainless steel or equivalent corrosion resistant material.

5.4.5.2.2.3 Valves

The valves used in the residual heat removal loop are constructed of austenitic stainless steel or equivalent corrosion resistant material.

Manual stop valves are provided to isolate equipment for maintenance. Throttle valves are provided for remote and manual control of residual heat exchanger tube side flow, and for automatic control of bypass flow. Check valves prevent reverse flow through the residual heat removal pumps.

Overpressure in the residual heat removal loop is relieved through a check valve to the low-pressure letdown stream in the chemical and volume control system.

Manually operated valves have backseats to facilitate repacking and to limit the stem leakage when the valves are fully opened to the back-seated position.

5.4.5.2.2.4 Piping

All residual heat removal loop piping is austenitic stainless steel. The piping is welded except for flanged connections at the pumps, flow orifices, and flow control valves 624, 625, and 626.

5.4.5.3 Performance Evaluation

Basic functional requirements for the residual heat removal system are contained in NRC Branch Technical Position RSB 5-1, Design Requirements of the Residual Heat Removal System. Although the Position was issued after the design of Ginna Station, the following paragraphs provide a comparison of the Ginna design to these guidelines.

5.4.5.3.1 Isolation Requirement

5.4.5.3.1.1 Isolation Valve Description

The residual heat removal suction and discharge valves connecting this system to the primary coolant system are shown in Drawing 33013-1247. The reactor coolant system suction supply to the residual heat removal pumps is from the hot leg of loop A through motor-operated valves MOV-700 and MOV-701 in series. The residual heat removal pump discharge return to the loop B cold leg of the reactor coolant system is through two in-series motor-operated valves, MOV-720 and MOV-721. There are no check valves in series with MOV-720 and MOV-721.

Permissive interlocks required to open the four residual heat removal system isolation valves are listed below.

1. MOV-700
    a. Reactor coolant system pressure must be less than 410 psig.
    b. Residual heat removal suction valves MOV-850A and MOV-850B from the containment sump must be closed.
2. MOV-701
    a. Residual heat removal suction valves MOV-850A and MOV-850B from the containment sump must be closed.
    b. The valve is operated by a key switch.

3. MOV-720

   No interlocks exist but the valve is operated by a key switch.

4. MOV-721

   Reactor coolant system pressure must be less than 410 psig.

No interlocks are associated with valve closure. There are no automatic functions that close the valves and no alarms generated by the valves (*Reference 22*). The valves fail "as is" upon loss of power supply and have remote position indication in the control room.

The residual heat removal system discharge line is not used for an Emergency Core Cooling System (ECCS) function that would require MOV-720 or MOV-721 to open; however, a branch of the residual heat removal discharge line provides low-pressure safety injection to the reactor vessel via parallel lines with one normally closed motor-operated valve (MOV-852A or B) and one check valve (CV-853A or B) in each line. The check valves are periodically tested. The motor-operated valve position indication is provided in the control room and these valves receive an open signal coincident with the safety injection signal.

5.4.5.3.1.2 Deviations From Branch Technical Position RSB 5-1

Based on the above description, the residual heat removal system deviates from the following provisions of Branch Technical Position RSB 5-1:

a. The outboard residual heat removal discharge and suction isolation valves (MOVs 701 and 720) do not have independent diverse interlocks to prevent opening the valves until reactor coolant system pressure is below 410 psig. The outboard valves are manually controlled with key-locked switches. By procedure, MOV-701 and MOV-720 are not opened until reactor coolant system pressure is less than 410 psig.

b. The power-operated valves (MOVs 852A and B) in the low-pressure safety injection lines open on a safety injection signal before reactor coolant system pressure drops below residual heat removal design pressure.

c. The residual heat removal isolation valves have no interlock feature to close them when reactor coolant system pressure increases above the design residual heat removal pressure.

RG&E has concluded that the deviation regarding the independent, diverse interlocks to prevent opening of the outboard residual heat removal isolation valves (MOVs 701 and 720) until pressure is below 410 psig is acceptable. The outboard residual heat removal isolation valves will open against a differential pressure of greater than 500 psid. However, the inboard isolation valves (MOVs 700 and 721) are provided with a pressure interlock. By administrative procedure, the outboard residual heat removal valves (MOVs 701 and 720) are key-locked closed, with power removed. In addition, a relief valve (RV-203) with a capacity of 70,000 lb/hr, set at 600 psig, is available. Power would have to be restored, the key-locked switch enabled, and MOV 701 or 720 opened in violation of procedures and, in addition, interlocked valve MOV 700 or 721 would have to fail to allow significant leakage for a potential residual heat removal system overpressurization to occur. MOVs 700 and 721 are in the Ginna Pump and Valve Inservice Test Program and are leak-tested on a refueling basis.

Therefore it is concluded that the probability of an intersystem loss-of-coolant accident is acceptably low.

The deviation regarding the low-pressure safety injection isolation valves (MOVs 852A and B) is considered acceptable (*Reference 23*), since the check valve testing provides sufficient assurance that these valves (CVs 853A and B) will perform their isolation function until reactor coolant system pressure decreases below residual heat removal system pressure.

The deviation regarding lack of automatic closure for the residual heat removal isolation valves on increasing pressure is acceptable (*Reference 23*) based on the administrative controls which are provided for the operation of these valves, coupled with the residual heat removal system high pressure alarm at 550 psig and the reactor coolant system interlock pressure alarm at 410 psig. These alarms provide adequate assurance that the operator action required by procedure will be taken to shut the isolation valves when reactor coolant system pressure is increasing towards the residual heat removal design pressure.

5.4.5.3.2 Residual Heat Removal Overpressure Protection

5.4.5.3.2.1 Design Basis

The residual heat removal relief valve has a nominal setpoint of 600 psig and a capacity of 70,000 lb/hr. The residual heat removal system is provided with a 550 psig high-pressure alarm and a reactor coolant system interlock pressure alarm at 410 psig. The residual heat removal system is connected to the loop A hot leg on the suction side and the loop B cold leg on the discharge side. The design pressure and temperature of the residual heat removal system are 600 psig and 400°F. The design basis with regard to overpressure protection for the Ginna Station residual heat removal system is to prevent opening of the residual heat removal isolation valves when reactor coolant system pressure exceeds 450 psig and to provide relief capacity sufficient to accommodate thermal expansion of water in the residual heat removal system and/or leakage past the system isolation valves.

5.4.5.3.2.2 Analysis

An analysis of incidents which might lead to overpressurizing the residual heat removal system was performed (*Reference 24*). Three events were considered in the analysis:

a. With reactor coolant system in solid condition and residual heat removal and charging pumps operating, the letdown line from the reactor coolant system is isolated.
b. During cooldown using two residual heat removal trains, one residual heat removal train suffers a failure at a time when the core heat generation rate exceeds the heat removal capability of one train.
c. Pressurizer heaters are energized with residual heat removal in operation and reactor coolant system solid.

The results of these analyses showed that the residual heat removal system is provided adequate relief capacity when appropriate procedural steps are in place.

There is no safety relief valve at the suction side of the residual heat removal system to protect the residual heat removal system from potential overpressurization; thus the Low-Temperature Overpressure Protection (LTOP) system (Section 5.2.2) also protects the residual heat removal system from overpressurization when the residual heat removal system is connected to the reactor coolant system. Westinghouse performed an evaluation of the design-basis transients for mass input and heat input (*Reference 24*), which was subsequently updated in support of the steam generator replacement project (*Reference 54*). The design-basis transient for the mass input case is the charging-letdown mismatch with three positive displacement charging pumps in operation. The design-basis transient for the heat input case is the start of a reactor coolant pump with the steam generator secondary-side water and primary-side tube water 50°F higher than the rest of the reactor coolant system. It was determined that the allowable peak reactor coolant system pressure is more limiting for the residual heat removal system protection than that for the protection against the 10 CFR 50 Appendix G reactor pressure vessel limits.

The Technical Specifications (LCO 3.4.12) require that no safety injection pump be capable of injecting into the reactor coolant system whenever overpressure protection is provided by the pressurizer power operated relief valves (PORVs). The PORV setpoints contained in the Pressure and Temperature Limits Report (PTLR) provide overpressure protection for both the residual heat removal system and the reactor vessel 10 CFR 50 Appendix G limits for both the mass and heat input events. Also, the Technical Specifications allow that no more than one safety injection pump be capable of injecting when the overpressure protection is provided by a reactor coolant system vent equal to or greater than 1.1 in.^{2} Mass addition from the inadvertent operation of a safety injection pump will not result in residual heat removal system pressure exceeding allowable limits when overpressure protection is being provided by a reactor coolant system vent equal to or greater than 1.1 in.^{2}

The Technical Specifications requirements discussed above were originally approved by *Reference 15*, and later by *Reference 55*. The analysis was subsequently updated in support of the steam generator replacement project (*Reference 54*) and approved by the NRC in *Reference 57*.

The ability of the Ginna Low Temperature Over-Pressure Protection (LTOP) System to provide over-pressure protection for both the Reactor Coolant System (RCS) and the RHR system following a plant uprate to 1775 MWt was reviewed as part of the uprate project. As discussed in Section 2.8.4.3 of *Reference 69*, the bounding LTOP mass addition and heat addition analyses for RCS and RHR over-pressure protection are not affected by loss of decay heat cooling and therefore are not affected by the power uprate. Consequently, the existing over-pressure protection of the RHR System provided by LTOP is acceptable for uprate.

5.4.5.3.2.3 Effect of Stuck Open Relief Valve

Fluid discharged through the 2-in. residual heat removal relief valve (RV-203) is directed to the pressure relief tank inside the reactor containment. The pressure relief tank has a rupture disk which is designed to rupture at 100 psig and allow the contents of the tank to overflow to the containment sump, where it would be available for recirculation. Should flow from a stuck open residual heat removal relief valve cause the rupture disk to rupture, the consequences to safety-related equipment would be less severe than the consequences of post-loss-of-coolant accident containment flooding which has been previously analyzed and found acceptable (*Reference 23*).

If RV-203 were to stick open in a post-loss-of-coolant-accident event, residual heat removal flow to the reactor coolant system for both low-head recirculation and low-head safety injection modes would be affected. This is because a flow path would exist from the residual heat removal system to RV-203 via valves HCV-133 and V-703 in either of these residual heat removal operating modes. HCV-133 fails shut following loss of instrument air on containment isolation following a loss-of-coolant accident, but a flow path would still exist to RV-203 via the 0.75-in. locked open manual valve 703. The effect of this flow diversion would not reduce the capability of the Emergency Core Cooling System (ECCS) below that needed to mitigate the consequences of a postulated loss-of-coolant accident. This is because the design flow rate through RV-203 (70,000 lb/hr, which is a conservative number in this case since HCV-133 is shut) is much less than the flow rate of a residual heat removal pump in the low-pressure safety injection mode (776,000 lb/hr). Each residual heat removal pump has the capacity to provide 100% of the required low-pressure safety injection flow. Therefore, the leakage through RV-203 would not be as severe an event as the loss of a residual heat removal pump which has been postulated as a single failure in the Emergency Core Cooling System (ECCS) analysis.

5.4.5.3.3 Residual Heat Removal Pump Protection

The features designed into the Ginna Station residual heat removal system to prevent damage to the system centrifugal pumps are provisions for pump cooling, a pump mini-flow recirculation flow path, and system design to prevent loss of net positive suction head.

The component cooling water (CCW) system provides cooling for the residual heat removal pumps to prevent damage from overheating. The residual heat removal pumps are provided with a recirculation line to recycle a portion of the pump discharge fluid to the pump suction. This prevents overheating during pump operation when the residual heat removal system is not delivering flow to the reactor coolant system. Net positive suction head calculations were performed for the residual heat removal pumps, and the residual heat removal system operation was evaluated for normal plant shutdown cooling, low-pressure safety injection, and post-loss-of-coolant accident recirculation. Although recirculation operation developed the most limiting net positive suction head requirements, the calculations indicated that an acceptable net positive suction head margin is available. See Section 6.3.3.9.

NRC Bulletin 88-04 expressed concern about the possibility of residual heat removal pump damage during parallel pump operation feeding a common discharge header under low flow conditions. Slight differences in their performance characteristics could result in the stronger pump forcing the weaker pump’s discharge check valve closed, thereby creating a zero-flow or deadhead condition and damaging the pumps by overheating.

The Ginna residual heat removal pumps are each provided with a recirculation path to prevent pump damage from overheating. Each pump is provided with a 3-in. recirculation line with manual isolation valves on either end. The lines tap off from the pump discharge line between the heat exchanger and check valve and return to the residual heat removal pump suction line just downstream of the outlet check valve for the refueling water storage tank (RWST). The check valves (697A, 697B) isolate the pump recirculation paths from each other. Each 3-in. recirculation line contains a 200-gpm orifice plate. Each residual heat removal pump thus has a minimum flow recirculation line that is independent of the opposite train and that provides sufficient recirculation flow to prevent damage when the pump discharge path is isolated. Each recirculation line is equipped with relief valves, located downstream of each 200 gpm orifice plate. The relief valves function during pump recirculation to ensure that pump suction pressure does not prevent the pump suction isolation valves (MOV 850A and MOV 850B) from opening, due to thermal expansion of the recirculating fluid. In MODES 4, 5, and 6, when the system takes suction from the hot leg, the relief valves are manually isolated. Pressure, temperature, and flow instrumentation is provided for each recirculation train. Therefore, it has been determined that the safety concerns raised in NRC Bulletin 88-04 have been resolved (*Reference 25*).

The residual heat removal pumps are provided continuously with component cooling water (CCW) flow for the pump thrust and radial bearing housings using a water jacket that surrounds the oil bath. Component cooling water (CCW) is also provided continuously through a water jacket within the residual heat removal pump head that encloses the mechanical seal. In addition, the mechanical seal includes a pumping ring that pumps process fluid from the seal area through an external heat exchanger (cooled by component cooling water), and back to the seal area. During the injection phase post-accident, the water source for the pump is the refueling water storage tank (RWST). Component cooling water (CCW) is assumed not to be available during the injection phase. Since the temperature of the water source (RWST) is less than 104°F during this period, the residual heat removal pump remains fully operable without component cooling water (CCW). During the recirculation phase, component cooling water (CCW) is made available to the pump. Because the temperature of the water source (from containment sump B) is expected to be much higher during the recirculation phase, component cooling water (CCW) is needed for cooling the mechanical seal. Cooling for the bearing housing water jacket is expected to be available, since the same cooling lines provide this CCW flow. This cooling flow would improve reliability, but is not required. Component cooling water (CCW) is required to be operable by Technical Specifications while the plant is operating in MODES 1, 2, 3, and 4, and operable (as a “necessary support system”) during MODES 5 and 6 when the residual heat removal pump is operating. During the transfer to the sump recirculation phase post-accident, cooling water would be immediately delivered to the residual heat removal pumps upon start of the component cooling water (CCW) pumps. During normal plant cooldown or heatup, when the residual heat removal system is in operation and the pumped fluid is taken directly from the reactor coolant system, component cooling water (CCW) is necessary for the residual heat removal pump mechanical seal and bearing housing water jackets. However, during normal plant shutdown cooling operation, once the water temperature is stable and less than 120°F, component cooling water (CCW) is no longer considered necessary to maintain residual heat removal pump operability, but is desired from the standpoint of reliability.

5.4.5.3.4 Single-Failure Considerations

The single residual heat removal cooling suction line from the reactor coolant system and single discharge line to the reactor coolant system render the residual heat removal susceptible to single failure of the in-line suction valves (700, 701) in the closed position and passive failures of either suction or discharge lines. (Valves 700 and 701, which are inside containment, can be manually operated to overcome a motor operator or power supply failure.) Although these failures would render the residual heat removal mode of decay heat removal inoperable, the alternate means of decay heat removal using the steam generators is still available as a backup. For the case of a failure of valves 700 or 701 or a pipe break downstream of these valves, an alternative flow path for core cooling is available via the residual heat removal cooling discharge line and the high-pressure safety injection pumps. Other means of core decay heat removal have a low heat removal capability but could be used to supplement steam generator heat removal until the decay heat rate was low enough. These methods are heat removal via the chemical and volume control system nonregenerative and excess letdown heat exchangers (requires component cooling water (CCW)) and cooldown flow from the pressurizer to the containment via the pressurizer safety valves with coolant injections from the safety injection or chemical and volume control systems. If a pipe break upstream of valves 700 and 701 should occur (i.e. a loss-of-coolant accident), the core could be adequately cooled by means of the residual heat removal sump recirculation mode.

The residual heat removal system contains a bypass line which is normally isolated during operation at power. During cooldown, the bypass line functions to control the total flow through the residual heat removal loops. A redundant bypass line is unnecessary in the system design, since the line can be manually isolated and the decay heat removal rate manually controlled in the event of a failure.

5.4.5.3.5 Leakage Provisions

The two residual heat removal pumps are located below the basement floor of the auxiliary building in a room provided with two environmentally qualified 50-gpm sump pumps, which discharge to the waste holdup tank. Environmentally qualified level switches control operation of the auxiliary building sump pumps and provide high level alarms on the plant process computer system. A single sump pump is capable of handling a leak rate from a residual heat removal pump seal failure, conservatively assumed to be 50 gpm. It is assumed that this passive failure could be isolated within 30 minutes. Consequently the waste holdup tank is required to operate at a level that will provide a holdup capability of 1500 gal for this postulated event during postaccident recirculation. Each auxiliary building sump pump starts automatically upon receiving a high-water level signal from one of two level instruments in the room.

From the standpoint of system reliability and availability in the unlikely event of failure of both auxiliary building sump pumps and assuming a conservative leak rate of 50-gpm, sufficient time is available (approximately 2 hr) to isolate the leaking residual heat removal pump before the water level in the pump room would flood the residual heat removal pump motors. The residual heat removal pumps are on separate pipe lines in the room in which they are located. Each pipe line contains a motor-operated valve, which could be closed remotely to isolate the leakage should the seal failure occur during postaccident recirculation. The residual heat removal pumps are driven by drip-proof type motors capable of operation in high humidity conditions and are provided with splash barriers. (See Section 5.4.5.2.)

5.4.5.3.6 Boron Concentration

One or more reactor coolant pumps or the residual heat removal system is in operation when a reduction is made in the boron concentration of the reactor coolant. At least one reactor coolant pump must be in operation for a planned transition from one reactor operating mode to another involving an increase in the boron concentration of the reactor coolant, except for emergency boration. When the boron concentration of the reactor coolant system is to be changed, the process must be uniform to prevent sudden reactivity changes in the reactor. Mixing of the reactor coolant is sufficient to maintain a uniform boron concentration if at least one reactor coolant pump or one residual heat removal pump is running (except as noted above) while the change is taking place. One residual heat removal pump will circulate the reactor coolant system volume in approximately 0.5 hour.

5.4.5.4 Residual Heat Removal at Reduced Coolant Inventory

5.4.5.4.1 Generic Letter 88-17 Requirements

Generic Letter 88-17 identified actions to be taken to preclude loss of decay heat removal during nonpower operations. These actions included operator training and the development of procedures and hardware modifications as necessary to prevent the loss of decay heat removal during reduced reactor coolant inventory operations, to mitigate accidents before they progress to core damage, and to control radioactive material if a core damage accident should occur. Procedures and administrative controls were required to ensure containment closure prior to the time that a core uncovery could result from a loss of decay heat removal coupled with an inability to supply alternative cooling or addition of water to the reactor coolant system inventory. Procedures were required that cover reduced inventory operations and ensure that all hot legs are not blocked by nozzle dams unless a vent path is provided that is large enough to prevent pressurization and loss of water from the reactor vessel. Instrumentation was required to provide continuous core exit temperature and reactor water level indication. Sufficient equipment was required to be maintained in an operable or available status so as to mitigate loss of the residual heat removal cooling or loss of reactor coolant system inventory should they occur.

Westinghouse provided thermal hydraulic evaluations of the loss of the residual heat removal system in the reduced inventory condition in *Reference 26*. *Reference 26* analyzed five configurations the plant could be in while the reactor coolant system (RCS) is in the reduced inventory mode. Ginna Station has committed in *Reference 58* not to enter two of those configurations: configuration #4 (cold side opening exists and nozzle dam not installed) and configuration #5 (cold side opening exists and nozzle dam installed).

Ginna Station had committed to maintain configuration #3 (hot side vent path exists when the RCS is being drained) for entry into the reduced inventory mode where the RCS will be opened for maintenance activities.

The RCS can be filled by two methods. The first is a conventional fill and vent with several starts of the reactor coolant pumps (RCPs) to dynamically vent the steam generators. This method uses the hot side vent path as the reduced inventory configuration. Operating procedures have alternate steps to perform a conventional RCS fill and vent to bring the RCS to a solid plant condition.

Ginna specific analysis were performed in *Reference 67* and *68*. The results of these analyses were used by Ginna to form the basis for the required operator actions, which are implemented in procedures and administrative controls, and for the equipment required to be available for providing core cooling in the event residual heat removal cooling is lost.

The second fill method, RCS vacuum vent and fill, utilizes configuration #2 (intact RCS with water in the secondary side in the narrow range of a steam generator) for the final fill of the reactor coolant system. RCS vacuum vent equipment and temporary hoses are setup prior to the final fill of the RCS. When all nozzle dams have been removed, all primary manways have been installed on both steam generators, and at least one steam generator secondary has been filled to the narrow indication range, the plant is made ready to transition to configuration #2 (intact RCS).

The configuration is established when one or more of the power operated relief valves (PORVs) are opened and the pressurizer hot side vent is closed. At this time the vacuum vent and fill of the RCS can begin. Operating procedures maintain all the reduced inventory controls during the vacuum venting process while the RCS is being filled to a solid condition.

5.4.5.4.2 Containment Closure

Generic Letter 88-17 allows Westinghouse plants to take up to 2 hr to close containment when operating in the reduced inventory condition with openings totaling greater than 1 in^2 in the cold legs if a vent path exists that is sufficiently large that core uncovery cannot occur due to pressurization resulting from boiling in the core. Ginna procedures provide for control of containment penetrations and the capability to establish containment closure within 2 hr while in the reduced inventory condition during the period following reactor shutdown when the decay heat rate is high enough to cause core uncovery. However, since RCS pressure is not large enough to prevent gravity fill from the RWST, the core will not uncover and the Generic Letter 88-17 2 hour containment closure criteria is applicable. As an improvement to achieve containment closure within 2 hr, containment penetration number 2 was modified to provide access into the containment for the steam generator inspection and maintenance cabling, which had been previously routed through the equipment hatch during the annual inspection and outage (see Section 6.2.4.4.6). Thus, the hatch can be closed and containment isolated within the 2-hr time limit. The 2-hr time limit is not applicable at the end of a planned MODE 6 (Refueling) outage when operating in the reduced inventory condition because the time to reach saturation and core uncovery are extended. *Reference 27* provides plant-specific curves covering the reactor coolant system response to a loss of residual heat removal cooling with the reactor coolant system partially filled for all anticipated plant configurations. Ginna procedures provide for establishment of a large hot-side reactor coolant system vent path by removing the pressurizer manway before RCS inventory is reduced for mid-loop operation.

The use of configuration #2 (intact RCS) (see Section 5.4.5.4.1) during RCS vacuum vent and fill does not change the containment closure allowance time of 2 hours. Generic Letter 88-17 specifies that a closure time of 2.5 hours is acceptable provided there are no openings in the cold legs, reactor coolant pumps, and crossover legs (RCS intact). This configuration is entered following refueling when the time after shutdown is extended and decay heat is reduced.

5.4.5.4.3 Instrumentation for Reduced Inventory Operation

Ginna has instrumentation that is designed to aid operators in trending parameters important to maintaining residual heat removal operation and to detect abnormalities prior to a condition that could lead to a loss of residual heat removal cooling. The concern was that when using the residual heat removal system for shutdown cooling with a reduced reactor coolant system inventory, residual heat removal pump net positive suction head (NPSH) could be lost. The Ginna residual heat removal system has been provided with instrumentation to continuously monitor residual heat removal system performance whenever the system is being used for cooling the reactor coolant system and the coolant inventory is reduced. The instrumentation measures pump suction pressure, pump motor current, pump suction temperature, and pump discharge flow. The pump suction pressure, temperature, and flow signals are provided to the plant process computer system, which calculates pump NPSH from these inputs. The residual heat removal pump motor current and suction pressure also permit trending of current and pressure fluctuations associated with vortexing at the junction of the residual heat removal suction pipe and the reactor coolant loop. The plant process computer system can display and trend pump suction pressure and temperature, discharge flow, motor current, and margin to loss of NPSH for each residual heat removal pump. The plant process computer system provides an audible alarm on reaching the set low limit of margin for loss of NPSH. The plant process computer system also has a rate-of-change alarm on pump motor current. Loop level instrumentation is provided that accurately measures reactor coolant system loop level during reduced inventory conditions. The range is 0 to 100 in. Zero in. corresponds to a level 4 in. above the bottom of the hot leg and 100 in. is approximately 16 in. above the reactor vessel flange. The level sensing line for reactor coolant loop A is tied into the reactor coolant loop A hot leg via the residual heat removal suction lines. The sensing line for the reactor coolant loop B hot leg is tapped directly off the hot leg. The loop level instrumentation directly senses the head of water existing in the reactor coolant system and converts it to proportional electrical signals for transmission to the display and processing systems. The loop level instrumentation is designed for use when the plant is shut down and the reactor coolant system depressurized.

Local sightglass indication of loop level for the B loop is available in the containment basement. The sightglass (polycarbonate tube) with graduated level indication markings ranging from 0 to 144 in. of water is installed and used only during MODE 6 (Refueling) outages and is removed and stored prior to commencing power operations. The 0 in. marking corresponds to a level 4 in. above the bottom of the hot leg. Ten in. equals the mid-loop condition (centerline of the reactor coolant system hot leg). The sightglass is tied into the B loop level instrumentation tap. Permanently installed stainless steel tubing, valves (2), and supports accommodate the removable sightglass.

5.4.5.4.4 Available Equipment to Mitigate Loss of Residual Heat Removal Cooling

Generic Letter 88-17 recommends that at least two available or operable means of adding inventory to the reactor coolant system be provided in addition to the residual heat removal system during reduced inventory operations. These means should include at least one high-pressure injection pump. Ginna will have three methods available during reduced inventory operations. The preferred method is by gravity feed from the refueling water storage tank (RWST) directly to the loop A hot leg through valves MOV-856, MOV-701, and MOV-700 (see Drawing 33013-1247). Procedures provide for a large hot-side vent and allowable time constraints prior to entering a reduced inventory condition where the reactor coolant system will be opened for maintenance activities. The gravity feed method will be effective as long as a sufficient vent path exists and time constraints are adhered to, as defined in *Reference 27*. Gravity feed will raise the water level well above the top of the hot leg and allow restart of the residual heat removal pump. Charging pumps will be available as the second method of inventory addition. After uprate two charging pumps are required for the time period from 48 to 70 hours after shutdown to provide sufficient reactor coolant system (RCS) water addition to match the steam boil off rate. After a shutdown time of 70 hours only one charging pump is required to match the steam boil off rate. The flow path will be from the refueling water storage tank (RWST) to the loop B cold leg (normal charging path). For situations where a loop B cold-side opening exists, charging will be shifted to the loop A alternative charging pump line prior to opening the loop B cold side. (See Drawing 33013-1265, Sheets 1 and 2.) The adequacy of the charging pump method to the intact cold leg has been demonstrated in *Reference 26* by equating charging pump flow to core boil off rate.

The third method of recovery will be an available safety injection pump taking suction from the refueling water storage tank (RWST) and delivering to the loop A or B cold legs.

The fourth method of recovery will be an available safety injection pump taking suction from the refueling water storage tank (RWST) and delivering to the loop A hot leg if safety injection pump B is used or to the loop B hot leg if safety injection pump A is used. This method is also used if at any time core boiling is imminent or occurring as determined by core exit thermocouple indication or steam escaping from any reactor coolant system vents. (See Drawing 33013-1262, Sheets 1 and 2).

Ginna procedures require that the preferred flow paths and equipment be available prior to draindown with power to the appropriate components.

5.4.5.4.5 Reduced Inventory Procedures

Ginna procedures provide for the following during reduced reactor coolant inventory operations:

- Require a large vent path (i.e., pressurizer manway) sufficient to limit pressurization and subsequent loss of inventory, which could subsequently lead to core uncovery if unmitigated, whenever the reactor coolant system is to be opened for maintenance activities.
- Two core exit thermocouples powered from separate trains remain connected during reduced inventory operations.

- Control the removal and installation of steam generator manways and nozzle dams so that the hot leg manways and nozzle dams are removed first and installed last in the sequencing of steam generator maintenance.
- Provide control of containment penetrations and the capability to control containment closure.
- Provide capability to establish containment closure condition within the 2-hr limit.
- Require residual heat removal flow to be reduced and maintained at 800 gpm or less when operating at a level between 6 in. above loop centerline to loop centerline.
- Reduce residual heat removal flow to approximately 500 gpm or less for operation below loop centerline (necessary to perform resistance temperature detector maintenance).
- Reduced inventory condition will not be entered until reactor coolant system cold-leg water temperature has been reduced to less than 140°F and until at least 48 hours after shutdown.
- Require preferred flow paths and equipment be available with power to the appropriate components prior to draindown for means of adding inventory to the reactor coolant system in the event of loss of residual heat removal cooling.

Administrative controls implemented based on reduced inventory considerations:

- Prohibiting cold-side openings with the reactor coolant system unvented.
- Stationing an individual inside containment when water level is below the top of the hot leg to vent the residual heat removal system if necessary.
- Use of a volumetric measurement of reactor coolant system inventory during draindown to ensure that the appropriate volume of water has been drained prior to steam generator manway removal.
- Minimizing the time while operating at reduced inventory consistent with accomplishing required tasks during this condition and in consideration of overall plant safety.
- The hot leg vent is not required after the nozzle dams are removed and manways are installed if the intact reactor coolant system (RCS) configuration, with at least one steam generator filled with water to the narrow range taps (e.g. tubes covered with water) has been established. The power operated relief valves (PORVs) can then be opened and the pressurizer manway installed. This transition is only performed during the RCS vacuum vent and fill process.

5.4.5.4.6 Analyses

Plant-specific analyses were conducted to provide the evaluations for expected nuclear steam supply system behavior for all phases of non-power operations after the plant uprate to 1775 MWt. Results of the analyses include the following:

- Determined the plant-specific curves for time to reach saturation as a function of time after shut down for reactor coolant system initial temperatures of 100°F and 140°F.
- Calculated the boil-off rate following loss of residual heat removal for the above. These results were used to determine required makeup flow to prevent core uncovery.

- Calculated the reactor coolant system pressurization following loss of residual heat removal with nozzle dams installed and as a function of time after shut down. This analysis was used to justify the use of the pressurizer manway vent path and demonstrate gravity fill would be available.
- Calculated the time to core uncovery as a function of time after shutdown assuming the pressurizer manway opening for the following scenarios:

  1. All nozzle dams installed.
  2. No nozzle dams installed.
  3. Cold-leg opening nozzle dams not installed.
  4. Effects of surge line flooding on reactor coolant system pressurization.

5.4.5.5 Tests and Inspections

The residual heat removal pumps flow instrument channels are calibrated periodically.

Regulatory Guide 1.68, Initial Test Programs for Water-Cooled Reactor Power Plants, was not in existence when the Ginna Station preoperational and initial startup testing was accomplished. However, tests have been performed to confirm that cooldown under natural circulation can be achieved. The core flow rates achieved under natural circulation were more than adequate for decay heat removal. The calculated core flow at approximately 2% reactor power was 4.2% of nominal full power flow. At approximately 4% reactor power, calculated core flow was 5.2% of nominal. Flow rates of this magnitude provide adequate mixing of boron added to the reactor coolant system during cooldown.

Rochester Gas and Electric Corporation has implemented a valve test program in response to a generic NRC requirement (*Reference 28*) associated with the issue of the isolability of low-pressure systems from interfacing high-pressure systems. As implemented in the Technical Specifications, check valves 853A, 853B, 867A, 867B, 877A, 877B, 878F, 878G, 878H, and 878J, and motor operated valves 878A and 878C are tested to 0.5 gpm or less per nominal inch of valve size up to 5.0 gpm leakage.

5.4.6 MAIN STEAM AND FEEDWATER PIPING

The main steam piping has an inner diameter of 28 in. Steam flow is measured by monitoring dynamic head in nozzles inside the main steam piping. The nozzles, which have an inner diameter of 16 in., are located inside containment near the steam generators and serve to limit the maximum steam flow for any main steam line break further downstream. Note that in 1996, Replacement Steam Generators (RSGs) with integral main steam nozzle flow restrictors were installed. These restrictors limit maximum steam flow for all main steam line breaks. The main steam system is discussed in Chapter 10.

The main feedwater piping is ASTM A106 grade C seamless pipe with ASTM A234 grade WPB fittings (except as noted below), and was fabricated to the requirements of the ASA Code for Pressure Piping, B31.1-1955. Replacement Steam Generators (RSGs) were installed in 1996. The RSG feedwater nozzles are forged SA-508 Class 3. The RSG feedwater nozzles include a forged Inconel (SB166, UNS NO6690) safe-end transition between the nozzle and feedwater piping. This safe-end also provides the connection to the RSG welded thermal sleeve for the RSG internal feedwater distribution piping and feed ring. The internal thermal sleeve, distribution piping and feed ring are SA-355, GR P22. The feed ring is equipped with Inconel J-nozzles (SB-167, UNS NO6690).

In 1979, several pressurized water reactors, Ginna Station included, experienced feedwater pipe cracking in the vicinity of the feedwater to steam generator nozzles. At Ginna Station, stress-assisted corrosion and corrosion fatigue cracking were found in the feedwater piping-to-nozzle elbow welds just upstream of the nozzles. In response to IE Bulletin 79-13 (*References 29 through 31*), the welds were repaired and documented by reports (*References 32* and *33*) submitted to the NRC. Also in 1979, the 18-in. elbows at the steam generator nozzles were replaced with elbows of ASTM A234 grade WP-11. To facilitate Steam Generator Replacement, these elbows were again replaced in 1996. The 1996 replacement elbows are SA234, GR WP11. Additionally, the internal feedwater distribution piping for the RGSs employs a gooseneck design between the feedwater nozzle and feed ring. The gooseneck limits the volume of horizontal piping. This minimizes fill time and, therefore, reduces the thermal stratification, temperature distributions, and thermal stresses which contribute to the stress-assisted corrosion and corrosion fatigue cracking experienced previously.

5.4.7 PRESSURIZER

5.4.7.1 System Description

The general arrangement of the pressurizer is shown in Figure 5.4-8 and the design characteristics are listed in Table 5.4-7.

The pressurizer maintains the required reactor coolant pressure during steady-state operation, limits the pressure changes caused by coolant thermal expansion and contraction during normal load transients, and prevents the pressure in the reactor coolant system from exceeding the design pressure. The pressurizer vessel contains replaceable direct immersion heaters, multiple safety and pressurizer power operated relief valves (PORVs) (Section 5.4.10), a spray nozzle, and interconnecting piping, valves, and instrumentation.

There are 78 heaters separated into a control/variable group and a backup group. The heaters are made of nichrome wire with a magnesium oxide insulator. The heater terminals are hermetically sealed and designed to withstand the design pressure and temperature of the pressurizer. The heaters are located in the lower section of the vessel and pressurize the reactor coolant system by keeping the water and steam in the pressurizer at saturation temperature. The heaters are capable of raising the temperature of the pressurizer and contents at approximately 55°F/hr during startup of the reactor. Of the 78 heaters installed, 74 heaters are currently available for use. The 74 heaters have a total capacity of approximately 760 kW. (The original 78 heaters had a total capacity of approximately 800 kW).

In the event of a loss of offsite power, pressurizer heaters can be manually loaded onto emergency power sources.

The pressurizer is designed to accommodate positive and negative surges caused by load transients. The surge line, which is attached to the bottom of the pressurizer, connects the pressurizer to the hot leg of the B reactor coolant loop. During a positive surge, caused by a decrease in plant load, the spray system, which is fed from the cold leg of a coolant loop, condenses steam in the vessel to prevent the pressurizer pressure from reaching the setpoint of the pressurizer power operated relief valves (PORVs). Power-operated spray valves on the pressurizer limit the pressure during load transients. In addition, the spray valves can be operated manually by a controller in the main control room.

Two separate, automatically controlled spray valves with remote-manual overrides are used to initiate pressurizer spray. A manual throttle valve in parallel with each spray valve permits a small continuous flow through each spray line to reduce thermal stresses and thermal shock when the spray valves open. The throttle valve flow also helps maintain uniform temperature and water chemistry in the pressurizer. Two separate spray valves and spray line connections are provided so that the spray will operate when only one reactor coolant pump is operating.

A flow path from the chemical and volume control system is also provided to the pressurizer spray line. This flow path provides auxiliary spray to the vapor space of the pressurizer during cooldown when the reactor coolant pumps are out of service. Thermal sleeves on the pressurizer spray connection and spray piping are designed to withstand the thermal stresses resulting from the introduction of cold spray water.

During a negative pressure surge, caused by an increase in plant load, flashing of water to steam and generation of steam by automatic actuation of the heaters keep the pressure above the minimum allowable limit. Heaters are also energized on high water level during positive surges to heat the subcooled surge water entering the pressurizer from the reactor coolant loop.

The pressurizer is a vertical, cylindrical vessel with hemispherical top and bottom heads, constructed of carbon steel with internal surfaces clad with austenitic stainless steel. The heaters are sheathed in austenitic stainless steel. The pressurizer is insulated to minimize heat loss from the pressurizer vessel. The insulation consists of reflective panels that are removable to permit visual examination of the pressurizer as required by the Inservice Inspection (ISI) Program document.

The pressurizer vessel surge nozzle is protected from thermal shock by a thermal sleeve. A thermal sleeve also protects the pressurizer spray nozzle connection.

5.4.7.2 Seismic Evaluation

Within the scope of SEP Topic III-1 [Classification of Structures, Components, and Systems (Seismic and Quality)] the seismic resistance of the pressurizer was evaluated. Based on analyses of a heavier, 1800 ft^3, model (but with identical support skirts to the Ginna 800 ft^3 model) and utilizing a finite element model it was concluded that the Ginna pressurizer is adequately supported for the 0.2g safe shutdown earthquake.

5.4.8 PRESSURIZER RELIEF DISCHARGE SYSTEM

5.4.8.1 System Description

The pressurizer safety and pressurizer power operated relief valves (PORVs), described in Section 5.4.10, discharge to the pressurizer relief tank.

Principal design parameters of the pressurizer relief tank are given in Table 5.4-8. A diagram of the tank is shown in Figure 5.4-9.

Steam and water discharged from the pressurizer safety valves and pressurizer power operated relief valves (PORVs) pass to the pressurizer relief tank which is partially filled with water at or near ambient containment conditions. The cool water condenses the discharged steam and the condensate is drained to the waste disposal system. The tank normally contains water in a predominantly nitrogen atmosphere, although provisions have been made to periodically analyze the tank gas for accumulation of hydrogen and oxygen. Nitrogen pressure is normally maintained at 3 psig. The tank is equipped with a spray and drain which are operated to cool the tank following a discharge.

The tank size is based on the requirement to condense and cool a discharge equivalent to 110% of the full power pressurizer steam volume. Assuming an initial tank water temperature of 125°F, the tank is capable of absorbing an amount of heat such that the final water temperature is no greater than 200°F. If the temperature in the tank rises above 120°F during plant operation, the tank is cooled by spraying in cool reactor makeup water and draining out the warm mixture to the reactor coolant drain tank.

The spray rate is designed to cool the tank from 200°F to 120°F in approximately 1 hr following the design discharge of pressurizer steam. The volume of nitrogen gas in the tank is selected to limit the maximum pressure to 50 psig following a design discharge.

The tank is protected against a discharge exceeding the design value by a rupture disk which discharges into the reactor containment. The rupture disk on the relief tank has a relief capacity in excess of the combined capacity of the pressurizer safety valves. The tank design pressure (and the rupture disk setting) is twice the calculated pressure resulting from the maximum safety valve discharge described above, i.e., the tank design pressure is 100 psig. This margin is to prevent deformation of the disk. The tank and rupture disk holder are also designed for full vacuum to prevent tank collapse if the tank contents cool without nitrogen being added.

The impact of plant uprate to 1775 MWt on the design of the pressure relief tank was assessed by determining the amount of steam discharged to the tank from the limiting loss of electrical load transient at the uprate power level. As described in Section 2.5.2.2.2 of *Reference 69*, the amount of steam discharged to the tank from the limiting uprate external load transient is less than the amount of steam assumed to be discharged to the tank for the original tank design basis. Therefore, the original pressure relief tank discharge design basis is still satisfied at the plant uprate power level of 1775 MWt.

The impact of an elevated containment temperature of 5°F (from 120°F to 125°F) on the design of the pressurizer relief tank was assessed in Reference 71 for the limiting loss of electrical load transient at the uprate power level. This assessment demonstrated that the original pressurizer relief tank discharge design basis is still satisfied.

Pressure relief tank pressure is indicated in the control room on the main control board on a narrow range (0-7.5 psig) and wide range (0-150 psig) meter. This allows the control room operator to monitor pressure relief tank pressure up to the rating of the rupture disk.

The discharge piping from the safety and pressurizer power operated relief valves (PORVs) to the relief tank is sufficiently large to prevent backpressure at the safety valves from exceeding 20% of the setpoint pressure at full flow.

The pressurizer relief tank, by means of its connection to the waste disposal system, provides a means for removing any noncondensible gases from the reactor coolant system which might collect in the pressurizer vessel. The tank is constructed of carbon steel with a corrosion resistant coating on the internal surface. A flanged nozzle is provided on the tank for the pressurizer discharge line connection. The pressurizer discharge line, the nozzle, and the sprayer inside the tank are austenitic stainless steel.

5.4.8.2 System Analysis

In response to NUREG 0737, Section II.D.1, and the NRC plant-specific submittal request for piping evaluation, Westinghouse performed an analysis of the Ginna pressurizer safety and relief valve discharge piping system (see Section 3.9.2.1.4). It was determined that the operability and structural integrity of the system were ensured for all applicable loadings and load combinations including all pertinent safety and relief valve discharge cases.

5.4.9 VALVES

5.4.9.1 Original Valve Design

All the valves originally installed in the nuclear steam supply system had stems with back seats to prevent ejection of valve stems. If it were assumed that the stem threads fail, the upset required for the back seat prevents penetration of the bonnet as shown by analysis, thereby preventing the stem from becoming a missile. The stems of air- and motor-operated valves included similar interference.

Valves with nominal diameter larger than 2 in. were designed to prevent bonnet-body connection failure and subsequent bonnet ejection. The means of prevention included (a) using the design practice of ASME Section VIII, which limits the allowable stress of bolting material to less than 20% of its yield strength, (b) using the design practice of ASME Section VIII for flange design, and (c) controlling the load during the bonnet-body connection stud-tightening process.

The pressure containing parts, except the flange and studs, were designed per criteria established by the USAS B16.5. Flanges and studs were designed in accordance with ASME Section VIII. Materials of construction for these parts were procured per ASTM A182, F316, or A351, GR CF8M.

Stud and nut material was ASTM A193-B7 and A194-2H. The bonnet-body studs and nut material were later upgraded with 17-4PH ASTM A-564 TP 630 and ASTM A-194-8M TP316 material, respectively. The proper stud torquing procedures and the use of a torque wrench, with indication of the applied torque, limited the stress of the studs to the allowable limits established in the ASME Code, i.e., 20,000 psi. This stress level was far below the material yield, i.e., about 105,000 psi. The complete valves were hydrotested per USAS B16.5 (1500-lb USAS valves were hydrotested to 5400 psi). The cast stainless steel bodies and bonnets were radiographed and dye penetrant tested to verify soundness.

Valves with nominal diameter of 2 in. or smaller were forged and had screwed bonnets with canopy seals. The canopy seal was the pressure boundary while the bonnet threads were designed to withstand the hydrostatic end force. The pressure containing parts were designed to the criteria established by the USAS B16.5 specification.

5.4.9.2 Valve Wall Thickness

An engineering review of nuclear valves was conducted during the 1974-1975 time period as required by *Reference 34*. The review was the first phase of a program to demonstrate acceptable wall thickness on certain valves important to nuclear safety.

The engineering review of valves identified 55 valves with greater than a 1-in. nominal pipe size within the Ginna Station reactor coolant pressure boundary. These valves were 1500-lb pressure class valves designed for reactor coolant system design pressure of 2485 psi and design temperature of 650°F. The valves were originally purchased to either ASA B16.5, MSS SP-66, or ASME Section III. The valves varied in size from 2-in. to 10-in. nominal pipe size.

Physical or ultrasonic inspections were conducted to verify adequate wall thickness on all valves described above. The measurement program was based on design and manufacturing requirements in ANSI B16.5 or MSS SP-66. The valves were either found to meet requirements or, in the case of one valve, repaired to meet requirements. Valve wall thickness measurements were made on all spare nuclear valves then in stock. Specifications were prepared requiring measurement and manufacturer's certification of adequate valve wall thickness for all valves to be subsequently purchased for use in Ginna Station Seismic Category I systems.

5.4.9.3 Motor-Operated Valve Program

<u>Generic Letters 89-10 and 96-05</u>

The Ginna Station motor-operated valve program was established in response to IE Bulletin 85-03 (*Reference 35*). The program was later expanded to address the recommendations of Generic Letter 89-10 (*Reference 36*) and Generic Letter 96-05 (*Reference 61*) to include all motor-operated valves in safety-related systems that are not blocked from inadvertent operation from either the control room, motor control center, or the valve itself. The following safety-related systems are included in the program:

- High-head safety injection - injection mode.
- Low-head safety injection - injection mode.
- High-head safety injection - recirculation mode.
- Low-head safety injection - recirculation mode.
- Auxiliary feedwater.

- Standby auxiliary feedwater.
- Containment spray.
- Component cooling water (CCW) - safety injection and residual heat removal pump cooling; sump recirculation cooling.
- Service water (SW) - nonessential load isolation.

The motor-operated valves in the above systems are tested at design pressure when practicable; otherwise, alternative methods are used to ensure motor-operated valve operability. The motor-operated valve program is described in the Ginna Station Motor-Operated Valve Qualification Program Plan. The motor-operated valve program is used to establish torque switch and limit switch settings for safety-related ac and dc motor-operated valves and to demonstrate valve operability during normal and abnormal design-basis events. The program also includes periodic and post maintenance and repair testing to verify continued valve operability. This program includes periodic verification of motor-operated valve capability and trending of motor-operated valve problems. The motor-operated valve program and Ginna Station procedures are designed to ensure that the switch settings of the motor-operated valves in the program are selected, set, and maintained correctly to accommodate the maximum differential pressures expected across the valves during both normal and abnormal design-basis events throughout the life of the plant. In response to Generic Letter 96-05 (*Reference 62*), the program was enhanced to include provisions for continually monitoring valve performance for degradation and periodic verification of program effectiveness. In *Reference 59*, RG&E provided closure notification to the NRC and in *Reference 60*, the NRC closed out its review of Generic Letter 89-10. In *Reference 63*, the NRC stated that RG&E has established an acceptable program to verify periodically the design-basis capability of all safety-related motor-operated valves at Ginna Station, and is adequately addressing the actions requested in Generic Letter 96-05.

As discussed in Section 2.2.4.2.2 of *Reference 69*, the impact of a plant uprate to 1775 MWt on the Ginna MOV Program was evaluated. The evaluation determined that although there were minor changes to flows, temperatures and differential pressures for some of the valves within the Ginna MOV Program, the changes did not affect the ability of the Ginna MOVs to comply with the requirements of Generic Letter 89-10 and Generic Letter 95-06.

Generic Letter 95-07

In response to Generic Letter 95-07 (*Reference 48*), Pressure Locking and Thermal Binding of Safety-Related Power-Operated Gate Valves, RG&E considered the safety-related motor-operated gate valves, including all valves within the GL 89-10 program, that could be potentially susceptible to this phenomena, and performed assessments, analyses or identified previous valve modifications to justify continued operability of the valves. The assessments of each valve were based upon the operational configurations and conditions imposed.

A number of valves received analysis that demonstrated that the developed valve thrust is capable of overcoming the imposed loads. These included: valves 860A, 860B, 860C, and 860D, (discharge isolation valves from containment spray pumps); and 871A and 871B (discharge valves from safety injection pump C to reactor coolant system loops A and B). Valves

852A and 852B, residual heat removal supply valves to the reactor vessel deluge, were modified in 1999 with flexible wedges that have vent holes. Valves 515 and 516, the pressurizer power operated relief (PORV) block valves, were modified in 1989 with upstream discs that have vent holes, and valves 850A and 850B, the residual heat removal suction valves from containment sump B, were modified in 1970 to include bonnet vents to the residual heat removal pump suction side of the valves. Valves 857A, 857B, and 857C, the discharge valves from residual heat removal pumps to safety injection pumps, were modified in 1996 to install a bonnet pressure relieving hole in the designated valve disc relieving pressure to the residual heat removal side of the valves. The balance of the valves identified were justified based upon the operational configuration and conditions imposed. These included: valves 738A and 738B, (component cooling water supply valves to the residual heat removal heat exchanger); 3504A and 3505A, (main steam supply valves to the turbine driven auxiliary feedwater pump); 704A and 704B, (suction isolation valves to the residual heat removal pumps); 1815A and 1815B, (suction isolation valves for safety injection pump C); and 4615 and 4616 (service water isolation valves to auxiliary building loads). RG&E's response to the generic letter is contained in *Reference 49*.

As discussed in Section 2.2.4.2.2 of *Reference 69*, the impact of a plant uprate to 1775 MWt on the Ginna valve pressure locking and thermal bounding analyses was evaluated. The evaluation determined that the plant uprate had no impact on any of the Ginna pressure locking or thermal binding analyses.

5.4.10 SAFETY AND PRESSURIZER POWER OPERATED RELIEF VALVES (PORVS)

5.4.10.1 System Description

The reactor coolant system is protected against overpressure (Section 5.2.2) by control and protective circuits such as the two high-pressure code safety valves and the two pressurizer power operated relief valves (PORVs) connected to the top head of the pressurizer. The valves discharge into the pressurizer relief tank, which condenses and collects the valve effluent. The schematic arrangement of the relief devices is shown in Drawings 33013-1258 and 33013-1260.

The pressurizer power operated relief valves (PORVs) and spring-loaded code safety valves are provided to protect against pressure surges that are beyond the pressure limiting capacity of the pressurizer spray. The pressurizer discharge lines leading to each pressurizer power operated relief valve (PORV) contain a motor-operated block valve to be used if the pressurizer power operated relief valve (PORV) opens inadvertently or fails to close following an overpressurization transient. The block valves are remote manually controlled from the control room. Leakage limits for the block valves are included in the reactor coolant system (RCS) operational limits in the Technical Specifications. Design parameters of the safety, relief, and blocking valves are given in Table 5.4-9.

At least one pressurizer code safety valve is in service whenever the reactor is subcritical and the reactor coolant system is in MODE 4 (Hot Standby), except during hydrostatic tests. Both pressurizer code safety valves are in service during MODE 3 (Hot Shutdown) and prior to criticality.

Each of the two pressurizer code safety valves is designed to relieve 288,000 lb/hr of saturated steam at the valve setpoint. Below 350°F and 350 psig in the reactor coolant system, the residual heat removal system can remove residual heat and thereby control system temperature and pressure. >For the original licensed power of 1520 MWt if no residual heat were removed by any of the means available the amount of steam which could be generated at safety valve relief pressure would be less than half the valves capacity. Since the plant uprate to 1775MWt increased reactor power and the corresponding decay heat by approximately 17%, the amount of steam generated if no residual heat was removed would still be less than the flow capacity of one safety valve. Therefore, one valve provides adequate defense against overpressurization. In addition, the low temperature overpressure protection system (LTOP) is placed in service prior to the RCS system being cooled below the LTOP enable temperature or the residual heat removal system being placed in service. The LTOP system and its operators are described in detail in Section 5.2.2.

A resistance temperature detector located in the discharge pipe of each code safety valve provides indication of valve movement or significant seat leakage. Actuation of a safety valve will cause a rapid rise in discharge temperature, which is sensed by the resistance temperature detector and indicated/alarmed in the control room. Also linear voltage differential transducers on the pressurizer safety valves provide a direct indication of valve position.

The pressurizer power operated relief valves (PORVs) have direct stem position indication in the control room. An alarm is provided in conjunction with the indication.

5.4.10.2 Performance Testing and Evaluation

Under NUREG 0737, Item II.D.1, Performance Testing of BWR and PWR Relief and Safety Valves, all operating plant licensees and applicants were required to conduct testing to qualify the reactor coolant system relief and safety valves under expected operating conditions for design-basis transients and accidents. In addition to the qualification of valves, the functional ability and structural integrity of the as-built discharge piping and supports were also required to be demonstrated on a plant-specific basis.

In response to these requirements, a program for the performance testing of pressurized water reactor safety and pressurizer power operated relief valves (PORVs) was formulated by EPRI. The primary objective of the test program was to provide full-scale test data confirming the functional ability of the reactor coolant system pressurizer power operated relief valves (PORVs) and safety valves for expected operating and accident conditions. The second objective of the program was to obtain sufficient piping thermal hydraulic load data to permit confirmation of models which may be utilized for plant unique analysis of safety and relief valve discharge piping systems.

The valves, piping arrangements, and fluid inlet conditions used in the EPRI tests confirmed the ability of the Ginna Station safety valves, pressurizer power operated relief valves (PORVs), and block valves to open and close under expected conditions. Power-operated relief and block valves were found to fulfill their design functions with neither the valves nor the control circuitry being subjected to a harsh environment.

The operability and structural integrity of the Ginna Station configuration was also verified on a plant-specific basis by Westinghouse for all applicable loadings and load combinations, including pertinent safety valve and relief valve discharge cases. See Section 3.9.2.1.4 for a discussion of the analysis. The NRC Safety Evaluation Report (*Reference 37*) concluded that Ginna Station had provided an acceptable response to the requirements of NUREG 0737, Item II.D.1, provided that plant procedures are adopted for inspecting the relief and safety valves after each lift involving the loop seal or water discharge.

5.4.11 COMPONENT SUPPORTS

5.4.11.1 Design Criteria

5.4.11.1.1 General

The classification of all components, systems, and structures for the purposes of seismic design are given in Section 3.7.1. The definition of the three original seismic Classes is given in Section 3.7.1.1.

All components of the reactor coolant system and associated systems were designed to the standards of the applicable ASME Code or USAS Code. The loading combinations that were originally employed in the design of Seismic Category I components of these systems, i.e., vessels, piping, supports, vessel internals, and other applicable components, are given in Table 3.9-1. This table also indicates the stress limits that were used in the design of the listed equipment for the various loading combinations.

To be able to perform their function, i.e., allow core shutdown and cooling, the reactor vessel internals had to satisfy deformation limits that were more restrictive than the stress limits shown in Table 3.9-1. For this reason the reactor vessel internals were treated separately (see Section 3.9.5).

In general, modifications or additions to piping systems at Ginna Station since initial operation have been seismically qualified using dynamic analyses. Some small piping has been seismically qualified using equivalent analysis or spacing table techniques. Specific cases are discussed in Section 3.9.2.1.

As a result of the SEP preliminary seismic review of Ginna Station, IE Bulletin 79-14, and other NRC seismic requirements, Ginna initiated a seismic upgrade program after the completion of piping support modifications required by IE Bulletin 79-14. The loading combinations and associated stress limits used for the piping systems that are part of the seismic upgrade program are discussed in Section 3.9.2.1.8 and appear in Table 3.9-8.

5.4.11.1.2 Asymmetric Loss-of-Coolant Accident Loading

In January 1978, all licensees of pressurized-water reactor plants were required by the NRC to provide an assessment of the adequacy of the reactor vessel supports and other affected structures to withstand combinations of response to asymmetric loss-of-coolant accident loads and the safe shutdown earthquake. In response, *References 38 through 41* were submitted to the NRC for the Westinghouse Owners Group plants in the form of Topical Reports relating to the "leak-before-break" concept. The NRC evaluation (*Reference 42*) of the above references concluded that an acceptable basis had been provided so that the asymmetric blowdown loads resulting from double-ended pipe breaks in main coolant loop piping need not be considered as a design basis for the Westinghouse Owners Group plants, provided that leakage detection systems exist to detect postulated flaws utilizing guidance from Regulatory Guide 1.45.

By *Reference 43* Ginna provided information to the NRC concerning the capability of the leakage detection systems installed at Ginna Station to detect a 1.0-gpm leak within 4 hours. By *Reference 44* the NRC reported that the NRC met the criteria specified in *Reference 42* and that the asymmetric blowdown loads resulting from double-ended pipe breaks in main coolant loop piping need not be considered as a design basis for Ginna Station.

In the SER provided by the NRC, *Reference 69* concluded, in Section 2.1.6, that the Ginna analyses were still valid after the plant uprate to 1775MWt.

5.4.11.1.3 Lamellar Tearing

During the mid-1970s the NRC raised a number of questions about the potential for lamellar tearing and low fracture toughness of materials used in steam generator supports and reactor coolant pump supports; Ginna addressed this issue in *References 45* and *46*. It was concluded that adequate fracture toughness exists for the supports at Ginna Station and that lamellar tearing was not an issue for the Ginna Station design and installation.

5.4.11.2 Support Structures

See also Section 3.9.3.2.

5.4.11.2.1 Reactor Vessel Supports

The vessel is supported on six individual pedestals. Each pedestal rests upon plates that are in turn supported upon the circular concrete primary shield wall.

The reactor vessel has six supports comprising four support pads located one on the bottom of each of the primary nozzles and two gusset support pads. One of the reactor inlet nozzles is centered approximately 2 degrees counterclockwise from the 90-degree axis and the other is centered approximately 2 degrees counterclockwise from the 270-degree axis.

Each support bears on a support shoe, which is fastened to the support structure. The support shoe is a structural member that transmits the support loads to the supporting structure. The support shoe is designed to restrain vertical, lateral, and rotational movement of the reactor vessel, but allows for thermal growth by permitting radial sliding at each support, on bearing plates.

The seismic resistance of the reactor vessel supports was evaluated as part of SEP Topic III-6. It was concluded, based on experience for nozzle-supported vessels, that the seismically induced stresses in the nozzles and adjacent shells are very small and that the governing element for reactor vessel support is the concrete shield wall. The shield wall was considered to be adequate to withstand the 0.2g safe shutdown earthquake according to the NRC review (*Reference 47*).

5.4.11.2.2 Steam Generator Supports

Each steam generator is supported on a structural system consisting of four vertical support columns and two (upper and lower) support systems. The vertical columns, which are pin connected to the steam generator support feet, serve as vertical restraint for operating weights, pipe rupture, and seismic considerations while permitting movement in the horizontal plane. The support systems, by using a combination of stops, guides, and snubbers, prevent rotation and excessive movement of the steam generator in any vertical plane. Thermal expansion is permitted in the support systems by a key arrangement. (See Section 3.9.3.2.2.)

5.4.11.2.3 Reactor Coolant Pump Supports

The reactor coolant pump is supported by a structural system consisting of three vertical columns and a system of stops. The vertical columns are bolted to the pump support feet and permit movement in the horizontal plane to accommodate reactor coolant pipe expansion. Horizontal restraint is accomplished by a combination of tie rods and stops which limit horizontal movement for pipe rupture and seismic effects.

5.4.11.2.4 Pressurizer Supports

The pressurizer is supported on a heavy concrete slab spanning between the concrete shield walls for the steam generator compartment. The pressurizer is a bottom skirt support vessel.

5.4.11.2.5 Reactor Coolant Piping Supports

The reactor coolant piping layout is designed on the basis of providing floating supports for the steam generator and reactor coolant pump in order to permit the thermal expansion from the fixed or anchored reactor vessel. A comprehensive thermal analysis was performed to ensure that stresses induced by linear thermal expansion were within code limits.

Two shock suppressors (snubbers) are provided on each steam generator to ensure piping structural integrity during and following a seismic event or other event initiating dynamic loads.

5.4.11.2.6 Inspection and Testing

The inspection and testing of all safety-related hydraulic and mechanical shock suppressors (snubbers) shall be implemented and performed in accordance with the "Snubber Inspection and Testing Program", to ensure the required operability of these snubbers during and following a seismic or other event, initiating dynamic loads. Station procedures include a listing of safety-related hydraulic snubbers that must be operable, limiting conditions of operations relative to these snubbers, and an inspection and testing program for snubbers. The inspection program includes all safety-related snubbers and snubbers installed on non-safety-related systems whose failure or failure of the system on which they are installed could have an adverse effect on a safety-related system (see Section 3.9.3.3.5).

REFERENCES FOR SECTION 5.4

1. Ernest L. Robinson, "Bursting Tests of Steam-Turbine Disk Wheels," Transactions of the ASME, July 1974.

2. Letter from D. M. Crutchfield, NRC, to J. E. Maier, RG&E, Subject: Completion of Topic III-10.B, Pump Flywheel Integrity, dated June 22, 1981.

3. Deleted

4. Deleted

5. Deleted

6. Deleted

7. Deleted

8. Deleted

9. Deleted

10. Deleted

11. NRC Bulletin No. 88-02, Rapidly Propagating Fatigue Cracks in Steam Generator Tubes, dated February 5, 1988.

12. Letter from C. Stahle, NRC, to R. C. Mecredy, RG&E, Subject: Closeout of Bulletin 88-02, Issues on Ginna (TAC 67305), dated March 30, 1989.

13. Letter from J. F. Hofscher, Westinghouse, to P. Gorski, RG&E, Subject: S/G Tube Fatigue Evaluation Update, dated July 16, 1991 (RGE-91-579).

14. Deleted

15. Letter from D. L. Ziemann, NRC, to L. D. White, RG&E, Subject: Amendment No. 26 to the Provisional Operating License, dated April 18, 1979.

16. Letter from R. W. Kober, RG&E, to C. Stahle, NRC, Subject: Periodic Verification of Leak Tight Integrity of Pressure Isolation Valves (PIV) (Generic Letter 87-06), dated June 11, 1987.

17. Letter from L. D. White, Jr., RG&E, to D. L. Ziemann, NRC, Subject: Followup Actions Resulting from the NRC Staff Reviews Regarding the Three Mile Island Unit 2 Accident, dated October 17, 1979.

18. Letter from L. D. White, Jr., RG&E, to D. L. Ziemann, NRC, Subject: Three Mile Island Lessons Learned - Short Term Requirements, dated December 28, 1979.

19. Letter from L. D. White, Jr., RG&E, to D. M. Crutchfield, NRC, Subject: Short Term Lessons Learned, Reactor Coolant System Venting, dated June 2, 1980.

20. Letter from J. E. Maier, RG&E, to D. M. Crutchfield, NRC, Subject: NUREG 0737 Requirements, dated July 1, 1981.

21. Letter from J. E. Maier, RG&E, to D. M. Crutchfield, NRC, Subject: Reactor Coolant System Vents (TMI Item II.B.1), dated May 7, 1982.

22. Letter from L. D. White, RG&E, to A. Schwencer, NRC, Subject: Reactor Vessel Overpressurization, dated February 24, 1977.

23. Letter from D. M. Crutchfield, NRC, to J. E. Maier, RG&E, Subject: SEP Topics V-10.B, V-11.B, and VII-3, dated September 29, 1981.

24. Westinghouse Electric Corporation, R. E. Ginna Low Temperature Overpressure Protection System (LTOP) Setpoint Phase II Evaluation Final Report, October 1990 (Proprietary) and February 1991 (Non-Proprietary), (Attachment C to letter from R. C. Mecredy, RG&E, to A. R. Johnson, NRC, Subject: Rochester Gas and Electric Corporation, R. E. Ginna Nuclear Power Plant, Docket 50-244, dated February 15, 1991).

25. Letter from A. R. Johnson, NRC, to R. C. Mecredy, RG&E, Subject: NRC Bulletin 88-04, Potential Safety-Related Pump Loss, dated August 16, 1989.

26. WCAP 11916, Loss of Residual Heat Removal Cooling While the RCS is Partially Filled, Revision 0, dated July 1988.

27. Ginna Design Analysis DA-NS-2006-019, Revision 0, entitled "Loss of RHR Cooling During Mid-Loop for EPU", dated October 17, 2006.

28. Letter from D. M. Crutchfield, NRC, to J. Maier, RG&E, Subject: Order for Modification of License Concerning Primary Coolant System Pressure Isolation Valves (PIV), dated April 20, 1981.

29. U.S. Nuclear Regulatory Commission, "Cracking in Feedwater System Piping," IE Bulletin 79-13, June 25, 1979.

30. U.S. Nuclear Regulatory Commission, "Cracking in Feedwater System Piping," IE Bulletin 79-13, Revision 1, August 30, 1979.

31. U.S. Nuclear Regulatory Commission, "Cracking in Feedwater Piping," IE Bulletin 79-13, Revision 2, October 17, 1979.

32. Letter from L. D. White, Jr. RG&E, to B. H. Grier, NRC, Subject: Cracking in Feedwater Piping, dated July 27, 1979.

33. Westinghouse Electric Corporation, Metallurgical Investigation of the Steam Generator Feedwater Piping Cracks at the Robert Emmett Ginna Nuclear Power Generating Station, WCAP 9563, August 1979.

34. Letter from J. P. O"Reilly, AEC, to RG&E, Subject: Valve Wall Thickness, June 22, 1972.

35. U.S. Nuclear Regulatory Commission, "Motor-Operated Valve Common Mode Failures During Plant Transients Due to Improper Switch Settings," IE Bulletin 85-03, November 15, 1985.

36. U.S. Nuclear Regulatory Commission, "Safety-Related Motor-Operated Valve Testing and Surveillance" Generic Letter 89-10, June 28, 1989.

37. Letter from C. Stahle, NRC, to R. W. Kober, RG&E, Subject: Safety Evaluation on the Performance Testing of the Ginna Relief and Safety Valves Conducted in Accordance With NUREG 0737 Requirements, dated August 20, 1987.

38. Westinghouse Electric Corporation, Mechanistic Fracture Evaluation of Reactor Coolant Pipe Containing a Postulated Circumferential Throughwall Crack, WCAP 9558, Revision 2 (Proprietary), WCAP 9570 (Non-Proprietary), May 1981.

39. Westinghouse Electric Corporation, Westinghouse Owners Group Asymmetric LOCA Load Evaluation - Phase C, W9 (Non-Proprietary), June 1980.

40. Westinghouse Electric Corporation, Tensile and Toughness Properties of Primary Piping Weld Metal for Use in Mechanistic Fracture Evaluation, WCAP 9787, Revision 1, May 1981.

41. Letter Report from E. P. Rahe, Westinghouse, to D. G. Eisenhut, NRC, Subject: Westinghouse Response to Questions and Comments Raised by Members of ACRS Subcommittee on Metal Components During the Westinghouse Presentation on September 15, 1981, NS-EPR-2519, dated November 10, 1981.

42. U.S. Nuclear Regulatory Commission, "Safety Evaluation of Westinghouse Topical Reports Dealing With Elimination of Postulated Pipe Breaks in PWR Primary Main Loops," Generic Letter 84-04, February 1, 1984.

43. Letter from R. W. Kober, RG&E, to W. A. Paulson, NRC, Subject: Generic Issue A-2, Elimination of Postulated Pipe Breaks, dated October 17, 1984.

44. Letter from Dominic C. DiIanni, NRC, to R. W. Kober, RG&E, Subject: Generic Letter 84-04, dated September 9, 1986.

45. Letter from L. D. White, Jr., RG&E, to D. L. Ziemann, NRC, Subject: Steam Generator and Reactor Coolant Pump Material, dated April 5, 1978.

46. Westinghouse Electric Corporation, Fracture Toughness and Design Considerations for Addressing Lamellar Tearing of Steam Generator and Reactor Coolant Pump Support Materials, R. E. Ginna Nuclear Power Plant, May 1978.

47. U.S. Nuclear Regulatory Commission, Seismic Review of the Robert E. Ginna Nuclear Power Plant as Part of the Systematic Evaluation Program, NUREG/CR 1821, dated November 1980.

48. U.S. Nuclear Regulatory Commission, “Pressure Locking and Thermal Binding of Safety-Related Power-Operated Gate Valves”, Generic Letter 95-07, dated August 17, 1995.

49. Letter from R.C. Mecredy, RG&E, to A.R. Johnson, NRC, Subject: 180-day Response to NRC Generic Letter 95-07, dated February 16, 1996.

50. Letter from B. A. Snow, RG&E, to W. T. Russell, NRC, Subject: NRC Bulletin 88-02: Rapidly Propagating Fatigue Cracks in Steam Generator Tubes, dated March 25, 1988.

51. Letter from R. C. Mecredy, RG&E, to W. T. Russell, NRC, Subject: Additional Information Relative to NRC Bulletin 88-02: Rapidly Propagating Fatigue Cracks in Steam Generator Tubes, dated March 3, 1989.

52. Letter from R. C. Mecredy, RG&E, to A. R. Johnson, NRC, Subject: Transmittal of Westinghouse Reassessment of IEB 88-02 for R. E. Ginna, dated March 2, 1992.

53. Letter from A. R. Johnson, NRC, to R. C. Mecredy, RG&E, Subject: Westinghouse Reassessment of MPA X802 (Bulletin 88-02), dated December 22, 1992.

54. Letter from R. C. Mecredy, RG&E, to A. R. Johnson, NRC, Subject: Application for Amendment to Facility Operating License, Methodology for Low Temperature Overpressure Protection (LTOP) Limits, dated February 9, 1996.

55. Letter from A. R. Johnson, NRC, to R. C. Mecredy, RG&E, Subject: Issuance of Amendment 48 to Facility Operating License No. DPR-18, dated March 6, 1992.

56. Letter from K.C. Hoskins, Westinghouse, to R.W. Eliasz, RG&E, Subject: Reactor Coolant Pump Performance Curves, NTD-NSRLA-OPL-94-301, dated October 10, 1994.

57. Letter from G. S. Vissing, NRC, to R. C. Mecredy, RG&E, Subject: Issuance of Amendment No. 64 to Facility Operating License No. DPR-18, R. E. Ginna Nuclear Power Plant, (TAC No. M94770), dated May 23, 1996.

58. Letter from R. C. Mecredy, RG&E, to C. Stahle, NRC, Subject: Loss of Decay Heat Removal (Generic Letter 88-17), dated January 4, 1989.

59. Letter from R. C. Mecredy, RG&E, to G. S. Vissing, NRC, Subject: Closure of NRC Generic Letter 89-10, Safety-Related Motor-Operated Valve Testing and Surveillance, dated March 3, 1998.

60. Letter from E. M. Kelly, NRC, to R. C. Mecredy, RG&E, Subject: NRC Motor Operated Valve Inspection 50-244/98-06, dated June 12, 1998.

61. U.S. Nuclear Regulatory Commission, "Periodic Verification of Design-Basis Capability of Safety-Related Motor-Operated Valves," Generic Letter 96-05, dated September 18, 1996.

62. Letter from R. C. Mecredy, RG&E, to G. S. Vissing, NRC, Subject: Response to NRC Generic Letter 96-05, dated March 3, 1998.

63. Letter from G. S. Vissing, NRC, to R. C. Mecredy, RG&E, Subject: Safety Evaluation Regarding the Licensee's Response to Generic Letter 96-05 (TAC No. M97050), dated December 27, 1999.

64. Letter from R. C. Mecredy, RG&E, to G. S. Vissing, NRC, Subject: Response to NRC Generic Letter 97-06, Degradation of Steam Generator Internals, dated March 30, 1998.

65. Generic Letter 97-06, Degradation of Steam Generator Internals, dated December 30, 1997.

66. Letter from Herbert N. Berkow, NRC, to Robert H. Bryan, Chairman Westinghouse Owners Group, TVA, Subject: Safety Evaluation of Topical Report WCAP-15666, "Extension of Reactor Coolant Pump Motor Flywheel Examination," dated May 5, 2003.

67. Westinghouse Report; "GOTHIC Mid-loop Analysis to Support Ginna Plant," June 2006.

68. Westinghouse Report: "GOTHIC Mid-loop Analysis to Support Ginna Plant," October 2006.

69. Letter from M. Korsnick (Ginna) to Document Control Desk, NRC, "License Amendment Request Regarding Extended Power Uprate," (Letter No. 1001353) dated July 7, 2005.

70. NRC Letter from P. Milano to M. Korsnick (Ginna), "R.E. Ginna Nuclear Power Plant - Amendment RE: 16.8% Power Uprate (TAC No. MC7382)," dated July 11, 2006.

71. Westinghouse Letter, "Pressurizer Relief Tank Level Setpoints for Elevated Containment Temperature Conditions at R.E. Ginna Unit 1," LTR-SEE-III-13-26, Revision 0, dated March 4, 2014.

Table 5.4-1
REACTOR COOLANT PUMP DESIGN DATA

Number of pumps	2
Pump model	93
Design pressure/operating pressure, psig	2485/2235
Hydrostatic test pressure (cold), psig	3110
Design temperature (casing), °F	650
Nameplate rating, rpm	1189
Suction temperature, °F	556
Developed head, ft	252
Net positive suction head, ft	170
Capacity, gpm	90,000
Seal-water injection, gpm	8
Seal-water return, gpm	3
Pump discharge nozzle I.D., in.	27-1/2
Pump suction nozzle I.D., in.	31
Overall unit height, ft	28.22
Water volume, ft^3	192
Pump-motor moment of inertia, $lb\text{-}ft^2$	80,000
Motor data	
Type	ac induction single speed
Voltage	4000
Phase	3
Frequency, cps	60
Starting	
Input (hot reactor coolant), kW	4000
Input (cold reactor coolant), kW	5300
Power, hp (nameplate)	6000

Table 5.4-2
REPLACEMENT STEAM GENERATOR DESIGN DATA

Normal pressure, reactor coolant/steam outlet, psig	2235/755
Design pressure, reactor coolant/steam, psig	2485/1085
Reactor coolant, hydrostatic test pressure (tube side cold), psig	3310
Normal temperature, reactor coolant, °F[a]	540 - 611.9
Design temperature, reactor coolant/steam, °F	650/556
Reactor coolant flow, lb/hr (total)	64.8×10^6
Heat transferred, Btu/hr (total)[a]	6201×10^6
Steam conditions at full load, outlet nozzle	
Steam flow, lb/hr[a]	3.94×10^6
Steam temperature, °F[a]	521.5
Steam pressure, psia[a]	823
Feedwater temperature, °F[a]	435
Overall height, ft-in	63 - 1.63
Shell O.D., upper/lower, in.	166/127.5
Reactor coolant water volume, ft^3 [b]	969.6
Secondary side volume, ft^3 [b]	4513
Supplier	Babcock and Wilcox International
Number of tubes per steam generator	4765
Tube size	0.750 in. O.D., 0.043 in. average wall thickness

a. Start-up conditions with RCS T_{AVG}=576°F and reactor power of 1811 MWt (102% of full power)

b. Volumes at both 1525 Mwt and Zero Power

Table 5.4-3
REACTOR COOLANT PUMP COMPOSITE HOT PERFORMANCE CURVE DATA

Flow (GPM)	Total Head (FT)	BHP	Hydraulic Efficiency (%)
0	475.6	4667.1	0.00
0	478.8	4577.1	0.00
5647	464.3	4611.3	10.70
5797	460.3	4701.1	10.68
11294	449.7	4642.0	20.60
11595	445.1	4729.6	20.54
16941	435.3	4671.6	29.71
17392	430.0	4755.2	29.60
22588	420.9	4705.3	38.03
23190	414.9	4783.2	37.86
28234	414.5	4779.2	46.09
28987	407.8	4850.4	45.87
33881	418.3	4957.6	53.81
34785	410.7	5022.3	53.55
39528	425.5	5142.8	61.56
40582	417.1	5199.3	61.28
45175	421.2	5174.9	69.21
46380	412.0	5218.4	68.93
50822	411.4	5345.1	73.63
52177	401.5	5376.5	73.35
56469	401.1	5537.4	76.98
57974	390.5	5555.3	76.70
62116	387.7	5641.1	80.36
63772	376.5	5641.9	80.09
67763	371.4	5689.7	83.26
69569	359.5	5670.5	83.01
73410	352.7	5708.9	85.37
75367	340.1	5667.4	85.13
79056	332.8	5693.6	86.99

Flow (GPM)	Total Head (FT)	BHP	Hydraulic Efficiency (%)
81164	319.6	5627.2	86.76
84703	310.3	5646.4	87.62
86962	296.4	5552.6	87.37
90350	285.4	5549.2	87.45
92759	270.8	5425.3	87.16
95997	257.1	5389.1	86.21
98557	242.0	5232.0	85.80
101644	227.1	5161.0	84.19
104354	211.4	4967.5	83.59
107291	194.1	4810.8	81.49
110152	177.8	4576.8	80.56

Table 5.4-4
REACTOR COOLANT PUMPS COLD PERFORMANCE CURVE DATA FOR INDIVIDUAL IMPELLERS

IMPELLER S/N 1619				IMPELLER S/N 340			
Flow (gpm)	Total Head (ft)	BHP	Hyd. Eff.[a](%)	Flow (gpm)	Total Head (ft)	BHP	Hyd. Eff.[a](%)
0	471.7	5948	0.00	0	468.6	6066	0.00
5553	457.3	5992	10.70	5701	453.5	6111	10.68
11105	443.0	6032	20.60	11402	438.5	6148	20.54
16658	428.8	6071	29.71	17104	423.6	6181	29.60
22211	414.6	6115	38.03	22805	408.8	6217	37.86
27763	408.3	6211	46.09	28506	401.7	6305	45.87
33316	412.1	6442	53.81	34207	404.7	6528	53.55
38869	419.2	6683	61.56	39908	411.0	6758	61.28
44421	414.9	6725	69.21	45610	405.9	6783	68.93
49974	405.3	6946	73.63	51311	395.6	6989	73.35
55527	395.1	7196	76.98	57012	384.7	7221	76.70
61079	381.9	7331	80.36	62713	370.9	7334	80.09
66632	365.9	7394	83.26	68414	354.1	7371	83.01
72185	347.4	7419	85.37	74115	335.1	7367	85.13
77737	327.9	7399	86.99	79817	314.9	7314	86.76
83290	305.7	7338	87.62	85518	292.0	7218	87.37
88843	281.1	7211	87.45	91219	266.8	7052	87.16
94395	253.3	7003	86.21	96920	238.4	6801	85.80
99948	223.7	6707	84.19	102621	208.3	6457	83.59
105501	191.2	6252	81.49	108323	175.2	5949	80.56

a. Hydraulic Efficiency

Table 5.4-5
REACTOR VESSEL HEAD VENT EQUIPMENT PARAMETERS

VALVES

Solenoid-operated globe valves	Cv = 2.0
	1 in. Schedule 160S connections
	Active valve per Regulatory Guide 1.48
	Operating design pressure - 2500 psig
	Design temperature - 680 °F
	Design humidity - 100%
	Radiation environment - post-accident 10^8 rads (beta) and 1.43 x 10^7 (gamma)
	Design code - ASME Section III, 1974, Class 2
	Seismic Category I
	Quality Group B
	Fail closed
	Red/green main control board status lights (Reed switches)
Manual globe valve 500 and 500B	Design pressure - 2500 psia
	Design temperature - 650°F
	Material - austenitic stainless steel
	Design code - ASME - Section III 1995 edition with 1996 addenda, Safety Class I
	Seismic Category I
	Cv = 4.0
	Quality Group A
Manual globe valves 592A and 593A	Design pressure - 2500 psig
	Design temperature - 650°F
	Material - austenitic stainless steel
	Design Code - ASME Section III, Safety Class 2
	Seismic Category I
	Nonactive valves
	Quality Group B

PIPING

Existing vent pipe	3/4 in. Schedule 80S
	Code compliance - ANSI B31.1
New piping (to head vent system)	3/4 in. and 1 in. Schedule 160S
	Code compliance - ASME Section III, 1977, Classes 1 and 2
PIPING SUPPORTS AND SUPPORT STRUCTURES	Code compliance - ASME, Section III, 1977, Subsection NF for new supports and structures

Table 5.4-6
RESIDUAL HEAT REMOVAL SYSTEM COMPONENT DESIGN DATA

Reactor coolant temperature at startup of decay heat removal, °F	350
Time to cool reactor coolant system from 350 °F to 140 °F, hour	73/101[a]
Refueling water storage temperature, °F	Ambient
Decay heat generation at 20 hours after shutdown condition, Btu/hr	37.4×10^6
Reactor cavity fill time, hour	1
Reactor cavity drain time, hour	4
H_3BO_3 concentration in refueling water storage tank (RWST), ppm boron	2750-3050
RESIDUAL HEAT REMOVAL PUMPS	
Quantity	2
Type	Horizontal centrifugal
Design rated capacity (each), gpm	1560
Head at rated capacity, ft H_2O	280
Motor horsepower	200
Material	Stainless steel
Design pressure, psig	600
Design temperature, °F	400
SUMP PUMPS (AUXILIARY BUILDING)	
Quantity	2
Type	Vertical, duplex
Capacity, gpm	50
Head, ft	55
Motor horsepower	1.5
Material (wetted surface)	Stainless steel

RESIDUAL HEAT REMOVAL HEAT EXCHANGERS

Quantity	2
Type	Shell and U-tube
Heat transferred, Btu/hr	[b]24.15×10^6
Reactor coolant flow, lb/hr (tube side)	763,000
Cooling water flow (each), gpm (shell side)	2780[b]
Cooling water inlet temperature, °F	100
Material, shell/tube	Carbon steel/stainless steel
Design pressure, shell/tube, psig	150/600
Design temperature, °F	350/400

a. The 20 hour cooldown times are for 80°F/85°F lake temperatures and two functional CCW and RHR heat exchangers. Times also assume RCP heat addition from one RCP until 160°F.

b. To minimize the potential for flow induced vibration in the residual heat removal heat exchangers, as of 1994 component cooling water flow has been limited to approximately 1800 gpm through the shell side of each heat exchanger. See Section 9.2.2.4.1.6.

Table 5.4-7
PRESSURIZER DESIGN DATA

Design/operating pressure, psig	2485/2235
Hydrostatic test pressure (cold), psig	3110
Design/operating temperature, °F	680/653
Water volume,[a]	475
Steam volume, full power, ft^3	325
Surge line diameter, in	10
Spray lines (2) diameter, in.	3
Spray flow, maximum, gpm per valve	200
Surge line nozzle diameter, in./pipe schedule	14 / Sch. 140
Shell I.D., in./calculated minimum shell thickness, in.	84/4.1
Minimum clad thickness, in.	0.188
Electric heaters capacity, kW	800
Heatup rate of pressurizer using heaters only, °F/hr	55 (approximately)
POWER-OPERATED RELIEF VALVES (PORV)	
Number	2
Set pressure (open), psig	2335
Capacity, lb/hr saturated steam/valve	179,000
SAFETY VALVES	
Number	2
Set pressure, psig	2485
Capacity, lb/hr saturated steam/valve	288,000 at 2485 psig + 3% accumulation

a. Based on full power pressurizer level of 61.2% at an RCS T_{AVG}=576°F.

Table 5.4-8
PRESSURIZER RELIEF TANK DESIGN DATA

Design pressure, psig	100
Rupture disk release pressure, psig	100
Design temperature, °F	340
Normal water temperature, °F	Containment ambient
Total volume, ft^3	800
Rupture disk relief capacity, lb/hr	7.20×10^5

Table 5.4-9
VALVE AND PIPING INFORMATION

SAFETY VALVE INFORMATION	
Number of valves	2
Manufacturer	Crosby Valve and Gauge
Type	Self-actuated
Size	4K26
Steam flow capacity, lb/hr/valve	288,000
Design pressure, psig	2485
Design temperature, °F	650
Set pressure, psig	2485
Accumulation	3% of set pressure
Blowdown	5% of set pressure
Original valve procurement specification	E-676279
RELIEF VALVE INFORMATION	
Number of valves	2
Manufacturer	Copes-Vulcan
Type	Pressurizer power-operated relief
Size	3 in. - NPS
Steam flow capacity, lb/hr/valve	179,000
Design pressure, psi	2485
Design temperature, °F	680
Opening pressure, psig	2335
Closing pressure, psig	2315
SAFETY AND RELIEF VALVE INLET PIPING INFORMATION	
Design pressure, psig	2485
Design temperature, °F	650
Loop seal volume, ft^3	0.18

SAFETY AND RELIEF VALVE DISCHARGE PIPING INFORMATION

Design pressure, psig	600
Design temperature, °F	650
Pressurizer relief tank design pressure, psig	100
Backpressure, normal, psig	3 to 5
Backpressure, developed, psig	350

BLOCK VALVE INFORMATION

Number of valves	2
Manufacturer	Anchor Darling
Type	Motor-operated double-disk gate
Size	3 in.
Steam flow capacity, lb/hr/valve	179,000
Design pressure, psi	2485
Design temperature, °F	650
Leakage limit, water/hr/in. diameter	10 cm^3
Stroke time, open or close	≈12 seconds
Motor operator	Limitorque SMB-00-15