

6 ENGINEERED SAFETY FEATURES

6.1 ENGINEERED SAFETY FEATURES INTRODUCTION AND MATERIALS

6.1.1 INTRODUCTION

The central safety objective in reactor design and operation is control of reactor fission products. The methods used to ensure this objective are:

- A. Core design to preclude release of fission products from the fuel (Chapter 4).
- B. Retention of fission products by the reactor coolant system boundary wherever leakage from the fuel occurs (Chapter 5).
- C. Retention of fission products by the containment for operational and accidental releases beyond the reactor coolant boundary (Section 6.2).
- D. Optimizing fission product dispersal to minimize population exposure (Chapters 2 and 11).

The engineered safety features are the provisions in the plant which embody methods 2 and 3 above to prevent the occurrence or to ameliorate the effects of serious accidents.

The engineered safety features in this plant are (1) the containment systems (Section 6.2), which include the containment recirculation fan cooler (CRFC) and containment post-accident charcoal systems, containment isolation valves, the containment spray system and the hydrogen recombiners, and (2) the Emergency Core Cooling System (ECCS) which includes the high and low-pressure safety injection systems and the accumulators described in Section 6.3.

Evaluations of techniques and equipment used to accomplish the central objective, including accident cases, are described in Chapters 6 and 15.

The criteria applying to the engineered safety features or related systems at the time of licensing of Ginna Station were part of the Atomic Industrial Forum (AIF) version of proposed criteria issued by the AEC for comment on July 10, 1967. These criteria (AIF-GDC 37, 38, 41, 42, and 43) are discussed in Section 3.1.1 and in the sections where the engineered safety features or related systems are discussed. Compliance of the design to the 1972 version of 10 CFR 50, Appendix A, General Design Criteria, is discussed in Section 3.1.2. Evaluations, such as the Systematic Evaluation Program (SEP) evaluations, that have been performed since 1972 against General Design Criteria appear in the design evaluation sections for the specific engineered safety feature or related system.

6.1.2 ENGINEERED SAFETY FEATURES MATERIALS

6.1.2.1 Postaccident Environmental Conditions

6.1.2.1.1 General

An evaluation of the suitability of materials of construction for use in the containment system has been performed considering the following:

- A. The integrity of the materials of construction of engineered safety features equipment when exposed to post design-basis accident conditions.
- B. The effects of corrosion and deterioration products from both engineered safety features (vital equipment) and other (non-vital) equipment, on the integrity and operability of the engineered safety features equipment.

The post design-basis accident environment conditions of temperature, pressure, radiation, and chemical composition are described in the following sections. The time-temperature-pressure cycle used in the materials evaluation is most conservative, since it considers only partial safeguards operation during the design-basis accident. The spray and core cooling solutions include both the design chemical compositions and the design chemical compositions contaminated with deterioration products and fission products that may conceivably be transferred to the solution during recirculation through the various containment engineered safety features systems.

In addition, within the scope of the SEP, Topic VI-1, the postaccident chemistry and suitability of organic materials in containment were evaluated by the NRC. The conclusions of this evaluation are discussed in Section 6.1.2.4 regarding the postaccident chemistry and Section 6.1.2.9 regarding organic materials.

6.1.2.1.2 Design-Basis Accident Temperature-Pressure Cycle

Figures 6.1-1 and 6.1-2 present the temperature and pressure profiles following the design-basis accident for Ginna Station. The figures represent environmental design-basis accident conditions considering partial engineered safety features operation; that is, operation with one of the two spray pumps, two of the four containment fans, one of the two residual heat removal pumps, and two of the three safety injection pumps. An overlay of the EQ temperature profile (Figure 6.1-1) and the EPU accident profile (Figure 6.2-5) shows that peak containment temperature is bounded by the EQ profile except for the first 5 sec. and for 2.3 days after 24 hr.

The first 5 sec. is irrelevant because of the thermal lag of energy transfer across the boundaries between the containment atmosphere and the surface of the equipment.

At 24 hours, containment temperature is 9°F above the EQ temperature profile and drops below the profile after approximately 2.3 days. The impact of the 9°F was evaluated with an efficiency aging analysis. The results of the analysis concluded that the EPU accident profile is bounded by the EQ profile with respect to long term aging. Therefore, the postaccident operability time of the EQ components remain valid for EPU operation. In reality, the long-term temperature of 152°F is a conservative bounding value. The EPU long-term temperature is calculated to be approximately 125°F.

Figure 6.1-3 presents the materials evaluation conditions for the containment and core environment.

Environmental qualification of safety-related equipment, including materials evaluations, were performed for conditions that either simulated the time-temperature and pressure conditions of Figures 6.1-1 and 6.1-2 or higher temperature and pressure conditions for

longer periods. The basis for each qualification or evaluation is described with the discussion of its particular suitability in the following sections.

6.1.2.1.3 Design-Basis Accident Radiation Environment

Evaluation of materials for use in containment included a consideration of the radiation stability requirements for the particular materials application. Figures 6.1-4 and 6.1-5 present the post design-basis accident containment atmosphere total gamma dose and total beta dose, respectively. These data were calculated on the basis of a core meltdown and assuming the following fission product releases, consistent with the TID 14844 model; and Regulatory Guide 1.89, Revision 1, that is, an instantaneous release of 100% noble gases, 50% halogens, and 1% other isotopes. The 1% other isotopes are assumed to be transported with the primary coolant directly to the containment sump. Additional detail is provided in Section 3.11.3.1.

6.1.2.1.4 Design Chemical Composition of the Emergency Core Cooling Solution

Present system designs provide for use of either boric acid solution or alkaline-adjusted boric acid solution as the spray and core cooling fluid.

Unadjusted boric acid solution is anticipated as the spray and core cooling liquid where other means (for example, charcoal filters) are provided for fission product removal from the containment atmosphere and for postulated small-break-type accident conditions wherein spray additive, although available, is not required.

Boric acid solution containing approximately 2750 to 3050 ppm boron is pumped from the refueling water storage tank (RWST) to the containment by means of the safety injection system pumps, the residual heat removal pumps, and the containment spray pumps. Figure 6.1-6 presents the variation of boric acid solution pH versus boron concentration. The solution pH value at 24°C for 2750 and 3050 ppm boron as boric acid solutions are approximately 4.4 and 4.32.

Plant designs which utilize the spray solution for fission product iodine removal, as well as containment cooling, include provisions for injection of chemical additive (sodium hydroxide) to the Emergency Core Cooling System (ECCS). The original containment spray system design requirements dictated a sprayed liquid pH range of 9.0 to 10.0 during the injection phase following a loss-of-coolant accident (LOCA); the lower value being associated with iodine removal and the higher value associated with material compatibility. A minimum pH of 8.5 for the containment sump liquid was originally used in the design to preclude re-evolution of iodine. This minimum value was later revised to a value of 7.0 as approved in Standard Review Plan 6.5.2. To maintain this spray pH range with a refueling water storage tank (RWST) boron concentration of 2750 to 3050 ppm, it was determined that the sodium hydroxide concentration in the spray additive tank must be maintained between 30 wt % and 35 wt % (*Reference 16*). This conclusion was based on a sodium hydroxide eductor suction flow of 20 gpm for a containment spray flow from 1320 gpm at 60 psig containment pressure to 1560 gpm at 0 psig containment pressure per containment spray pump. See UFSAR Section 6.5.2 for additional background information on NaOH and sump pH.

The sump pH has been calculated (*Reference 19*) for an RWST boron concentration of 2750 ppm to 3050 ppm, an accumulator concentration of 2550 ppm to 3050 ppm, and reactor coolant system operating concentrations. During the injection phase of a design-basis accident, with a spray additive tank concentration of 30 wt % to 35 wt %, the sump liquid pH range was determined to exceed the minimum required value of 7.0 as specified in Branch Technical Position MTEB 6-1, and to be less than 10.5 as specified in Standard Review Plan 6.5.2. This evaluation also considered chemical effects limits (See Section 6.3.2.1.1).

The solutions were considered aerated throughout the entire exposure period as in the case of the pure boric acid spray solution.

6.1.2.1.5 Trace Composition of Emergency Core Cooling Solution

During spraying and recirculation, the emergency core cooling solution will wash over virtually all of the exposed components and structures in the reactor containment. The emergency core cooling solution is recirculated through a common sump and hence, any contamination deposited in or leached by the solution from the exposed components and structures will be uniformly mixed in the solution.

The materials compatibility discussion includes consideration of the effects of trace elements which are identified as conceivably being present in the emergency core cooling solution during recirculation.

To identify the trace elements in containment which may have a deleterious effect on engineered safety features equipment, one must first establish which elements are potentially harmful to the materials of construction of the engineered safety features equipment and second, ascertain the presence of these elements in forms which can be released to the emergency core cooling solution following a design-basis accident. Table 6.1-1 presents a listing of the major periodic groups of elements. Elements which are known to be harmful to various metals are noted and potential sources of these elements are identified.

The concentration of the trace contaminants in the emergency core cooling solution will vary with individual plant construction as well as with the chemical composition of the emergency core cooling solution itself.

6.1.2.2 Materials of Construction in the Containment

All materials in the containment were reviewed from the standpoint of ensuring the integrity of the equipment constructed of these materials and to ensure that deterioration products of some materials do not seriously aggravate the accident condition. In essence, therefore, all materials of construction in containment must exhibit resistance to the postaccident environment or, at worst, contribute only insignificant quantities of trace contaminants which have been identified as potentially harmful to vital engineered safety features equipment.

Table 6.1-2 lists typical materials of construction used in the reactor containment system. Examples of equipment containing these materials are included in the table.

Corrosion testing, described in Section 6.1.2.3, showed that of all the metals tested only aluminum alloys were found incompatible with the alkaline sodium borate solutions.

Aluminum was observed to corrode at a significant rate, with the generation of hydrogen gas. Since hydrogen generation can be hazardous to containment integrity, a detailed survey was conducted to identify all aluminum components in the containment.

Table 6.1-3 lists the aluminum inventory in the Ginna Station containment. Included in the table is the mass of metal and exposed surface area of each component. The 6000 series aluminum alloys are the major types found in the containment.

The reactor vessel insulation foil is made of thin, 1100 series aluminum foil, which, for the purposes of hydrogen production calculations, is consumed immediately following the event.

All the metals of construction in containment including aluminum are compatible with unadjusted boric acid solution at the design-basis accident condition.

6.1.2.3 Corrosion of Metals of Construction in Design-Basis Emergency Core Cooling Solution

6.1.2.3.1 Corrosion Resistance

Emergency core cooling components are austenitic stainless steel and, hence, are quite corrosion resistant to the alkaline sodium borate solution, as demonstrated by corrosion tests reported in WCAP 7153 (*Reference 1*). The general corrosion rate for types 304 and 316 stainless steels was found to be 0.01 ml/month in pH 10 solution at 200°F. Data on corrosion rates of these materials in the alkaline sodium borate solution have also been reported by Oak Ridge National Laboratory (*References 2 and 3*) to confirm the low values.

Extensive testing was also performed on other metals of construction that are found in the containment. Testing was performed on these materials to ascertain their compatibility with the spray solution at design postaccident conditions and to evaluate the extent of deterioration product formation, if any, from these materials.

Metals tested included zircaloy, Inconel, aluminum alloys, cupro-nickel alloys, carbon steel, galvanized carbon steel, and copper. The results of the corrosion testing of these materials are reported in detail in *Reference 1*. In addition, ZIRLO® and Optimized ZIRLO™ material has been corrosion tested as documented in *References 17, 20, and 21*, respectively. Of the materials tested, only aluminum was found to be incompatible with the alkaline sodium borate solution. Aluminum corrosion is discussed in Section 6.1.2.5. A summary of the corrosion data obtained on various materials of construction exposed for several weeks in aerated alkaline (pH 9.3 to 10.0) sodium borate solution at 200°F is shown in Table 6.1-4. The exposure condition is considered conservative since the test temperature (200°F) is considerably higher than the long-term design-basis accident temperature (152°F).

Tests conducted at Oak Ridge National Laboratory (*References 2 and 3*) have also verified the compatibility of various materials of construction with alkaline sodium borate solution. In tests conducted at 284°F, 212°F, and 130°F, stainless steels, Inconel, cupronickels, Monel, and Zircaloy-2 experienced negligible changes in appearance and negligible weight loss.

The Ginna Station containment recirculation fan cooler (CRFC) coils are fabricated of copper Turbex plate fins vertically oriented on stainless steel (AL-6XN) tubes.

Corrosion tests at both PWRD and Oak Ridge National Laboratory have shown copper suffers only slight attack when exposed to the alkaline sodium borate solution at design-basis accident conditions. The corrosion rate of copper in alkaline sodium borate solution at 200°F is approximately 0.015 ml/month (*Reference 1*). The corrosion of copper in an alkaline sodium borate environment under spray conditions at 284°F and 212°F have been reported by Oak Ridge National Laboratory. Corrosion penetrations of less than 0.02 ml were observed after 24-hour exposure at 284°F (see *Reference 3*, Table 3.13) and a corrosion rate of less than 0.3 ml/month was observed at 212°C (see *Reference 2*, Table 3.6).

It can be seen, therefore, that the corrosion of copper in the postaccident environment will have a negligible effect on the integrity of the material. The corrosion product formed during exposure to the solution appears tightly bound to the metal surface and so will not be released to the emergency core cooling solution.

6.1.2.3.2 Caustic Stress Cracking Resistance

Consideration was given to possible caustic corrosion of austenitic stainless steels by the alkaline solution. Data presented by Swandby (*Reference 4*) (Figure 6.1-8) show that these steels are not subject to caustic stress cracking at the temperature (285°F and below) and caustic concentrations (less than 1 wt %) of interest. It can be seen in Figure 6.1-8 that the stress cracking boundary temperature as defined by Swandby is considerably above (approximately 80°F) the long-term postaccident design temperature of 152°F. A temperature in excess of 500°F is required to produce stress corrosion cracking at sodium hydroxide concentrations greater than 85%, as seen in Figure 6.1-8.

It should be noted when considering the possibility of caustic cracking of stainless steels that the sodium hydroxide-boric acid solution is a buffer mixture wherein no free caustic exists at the temperatures of interest even if the solution is concentrated locally through evaporation of water. Therefore, the above consideration is somewhat hypothetical with regard to the Ginna Station postaccident environment.

6.1.2.4 Corrosion of Metals of Construction by Trace Contaminants in Emergency Core Cooling Solution

Trace Elements

Of the various trace elements which could occur in the emergency core cooling solution in significant quantities, only chlorine (as chloride) and mercury are adjudged potentially harmful to the materials of construction of the engineered safety features equipment.

Mercury

The use of mercury or mercury bearing items, however, has been prohibited in the Ginna Station containment. This includes mercury vapor lamps, fluorescent lighting, and instruments which employ mercury for pressure and temperature measurements and for electrical equipment. Potential sources of mercury, therefore, have been excluded from containment and hence no hazard from this element is recognized.

Chlorides

The possibility of chloride stress corrosion of austenitic stainless steels has also been evaluated. It is believed that corrosion by this mechanism will not be significant during the postaccident period for the following reasons:

6.1.2.4.1 Low Temperature of Emergency Core Cooling Solution

The temperature of the emergency core cooling solution is reduced after a relatively short period of time (i.e., a few hours) to about 150°F. While the influence of temperature on stress corrosion cracking of stainless steel has not been unequivocally defined, significant laboratory work and field experience indicate that lowering the temperature of the solution decreases the probability of failure. Hoar and Hines (*Reference 5*) observed this trend with austenitic stainless steel in 42 wt % solutions of magnesium chloride with a temperature decrease from 310°F to 272°F. Staehle and Latanision (*Reference 6*) present data which also show the decreased probability of failure with decreasing solution temperature from about 392°F to 302°F. Staehle and Latanision (*Reference 6*) also report the data of Warren (*Reference 7*) which showed the significant change with decrease in temperature from 212°F to 104°F. The work of Warren, while pertinent to the present consideration in that it shows the general relationship of temperature to time to failure, is not directly applicable in that the chloride concentration (1800 ppm chlorine) believed to have affected the failure was far in excess of the reasonable chloride contamination which may occur in the emergency core cooling solution.

6.1.2.4.2 Low Chloride Concentration of Emergency Core Cooling Solution

It is anticipated that the chloride concentration of the emergency core cooling solution during the postaccident period will be low. Throughout plant construction, surveillance has been maintained to ensure that the chloride inventory in the containment would be maintained at a minimum. Controls on use of chloride bearing substances in the containment include the following:

- A. Restriction in chloride content of water used in concrete.
- B. Prohibition of use of chloride in cleaning agents for stainless steel components and surfaces.
- C. Prohibition of use of chloride in concrete etching for surface preparation.
- D. Use of non-chloride bearing protective coatings in containment.
- E. Restriction of chloride concentration in safety injection solution, 0.15 ppm chloride maximum.

The effect of decreasing chloride concentration on decreasing the probability of failure of stressed austenitic stainless steel has been shown by many experimenters. Staehle and Latanision (*Reference 6*) present data of Staehle which show the decrease in probability of failure with a decrease in chloride concentration at 500°F. Edeleanu (*Reference 8*) shows the same trend at chloride concentrations from 40% to 20% as magnesium chloride and reported no failures in this experiment at less than about 5% magnesium chloride.

Instances of chloride cracking at representative emergency core cooling solution temperatures and at low solution chloride concentration have generally been on surfaces on which concentration of the chloride occurred. In the Emergency Core Cooling System (ECCS), concentration of chlorides is not anticipated since the solution will operate subcooled with respect to the containment pressure and the containment atmosphere will be 100% relative humidity.

6.1.2.4.3 Alkaline Nature of the Emergency Core Cooling Solution

The emergency core cooling solution will have a solution pH of between 8.0 and 9.76 after the addition of spray additive (sodium hydroxide) as discussed in Section 6.1.2.1.4. Numerous investigators have shown that increasing the solution pH decreases the probability of failure. Thomas, et al., (*Reference 9*) showed that the failure probability decreases with increasing pH of boiling solutions of magnesium chloride. More directly applicable, Scharfstein and Brindley (*Reference 10*) showed that increasing the solution pH to 8.8 by the addition of sodium hydroxide prevented the occurrence of chloride stress corrosion cracking in a 10 ppm chlorine (as sodium chloride) solution at 185°F. Thirty stressed stainless steel specimens were tested: including type 304 as received, type 347 as received, and type 304 sensitized. No failures were observed. Other test runs by Scharfstein and Brindley showed the influence of solution pH on higher chloride concentrations, up to 550 ppm chlorine. However, in these tests the pH adjusting agents were either sodium phosphate or potassium chromate. The authors express the opinion, however, that in the case of the chromate solution, chloride cracking inhibition was simply due to the hydrolysis yielding pH 8.8 and not to an influence of the chromate anion. A similar hydrolysis will occur in the borate solution.

6.1.2.4.4 Summary

In summary, therefore, it is concluded that exposure of the stainless steel engineered safety features components to the emergency core cooling solution during the post-accident period will not impair its operability from the standpoint of chloride stress corrosion cracking. The environment of low temperature, low chlorides, and high pH which will be experienced during the postaccident period will not, it is believed, be conducive to chloride cracking.

As part of SEP Topic VI-1, the NRC staff independently evaluated the pH for the containment sump solution, which results from mixing of the containment spray solution with the reactor coolant and emergency core cooling fluids in the sump during recirculation. The NRC verified (*Reference 11*) that sufficient sodium hydroxide is available to raise the pH of the containment sump solution above the minimum level of 7.0, consistent with the guidance of Branch Technical Position MTEB 6-1, to reduce the likelihood of stress corrosion cracking of stainless steel components. It was also verified that the sump maximum pH will not exceed a value of 10.5 as specified in Standard Review Plan 6.5.2.

6.1.2.5 Corrosion of Aluminum Alloys

Corrosion testing showed that aluminum alloys are not compatible with alkaline borate solutions. The alloys generally corrode fairly rapidly, at the postaccident condition temperatures, with the liberation of hydrogen gas. A number of corrosion tests were conducted in the PWRD laboratories and at Oak Ridge National Laboratory facilities. The

design corrosion rates at various temperatures are shown in Figure 6.1-9. The time-temperature cycle (Table 6.1-5) considered in the calculation of aluminum corrosion is a conservative step-wise representation of the containment pressure response to the primary coolant break described in Section 6.2.1.2.2.6.

Figures 6.1-10 and 6.1-12 present the hydrogen generation from aluminum corrosion in containment. The calculated hydrogen generation is conservative since it considers all the aluminum inventory to be in contact with the spray solution for the entire postaccident period. No credit is taken in the calculation for anodizing, shielding, or coating of any aluminum components. See Section 6.2.5.3.2 for a discussion of the effect of additional aluminum, associated with the reactor vessel insulation, that was not included in the analysis of record.

6.1.2.6 Compatibility of Protective Coatings With the Postaccident Environment

The investigation of materials compatibility in the post-accident design-basis environment also included an evaluation of protective coatings for use in the containment (*Reference 12*).

The results of the protective coatings evaluation showed that several inorganic zincs, modified phenolics, and epoxy coatings are resistant to an environment of high temperature (320°F maximum test temperature) and alkaline sodium borate. Long-term tests included exposure to spray solution at 150°F to 175°F for 60 days, after initially being subjected to the design basis accident cycle. The protective coatings, which were found to be resistant to the test conditions (that is, exhibited no significant loss of adhesion to the substrate or formation of deterioration products), comprise virtually all of the protective coatings used in the Ginna containment. Hence, the protective coatings will not add deleterious products to the core cooling solution. Essentially all carbon steel surfaces are coated with Carbozinc-11^a (inorganic zinc primer) and Phenoline 305 (modified phenolic top coat). Phenoline 305 protective coating is also used on concrete surfaces.

Several test panels of the types of protective coatings used at Ginna Station were exposed for two design-basis accident cycles and showed no deterioration or loss of adhesion with the substrate.

6.1.2.7 Evaluation of the Compatibility of Concrete-Emergency Core Cooling Solution in the Postaccident Environment

Concrete specimens were tested in boric acid and alkaline sodium borate solutions at conditions conservatively (320°F maximum and 200°F steady-state) simulating the post design basis accident environment.

The purpose of this study was to establish:

- A. The extent of debris formation by solution attack of the concrete surfaces.
- B. The extent and rate of boron removal from the emergency core cooling solution through boron-concrete reaction.

a. These coatings are products of the Carboline Co., St. Louis, Missouri.

Tests were conducted in an atmospheric pressure reflux apparatus to simulate long-term exposure conditions, and in a high-pressure autoclave facility to simulate the design-basis accident short-term, high-temperature transient.

The total surface area of concrete in the Ginna containment which may be exposed to the emergency core cooling solution following a design-basis accident is estimated at 6.3×10^4 ft². This value includes both coated and uncoated surfaces. The emergency core cooling solution volume is approximately 313,000 gallons and the surface to volume ratio from these values is approximately 29 in²/gal. The surface to volume ratios for the concrete-boron tests were between 28 and 78 in²/gal of solution. Table 6.1-6 presents a summary of the data obtained from the concrete-boron tests series.

Testing of uncoated concrete specimens in the postaccident environment showed that attack by both boric acid and the alkaline boric acid solution is negligible and the amount of deterioration product formation is insignificant. Other specimens covered with modified phenolic and epoxy protective coatings showed no deterioration product formation. These observations are in agreement with Orchard (*Reference 13*) who lists the following resistances of portland cement concrete to attack by various compounds:

Boric acid	Little or no attack
Alkali hydroxide solution under 10%	Little or no attack
Sodium borate	Mild attack
Sodium hydroxide over 10%	Very little attack

Exposure of uncoated concrete to spray solution between 320°F and 210°F has shown a tendency to remove boron very slowly, presumably precipitating an insoluble calcium salt. The rate of change of boron in solution was measured at about 130 ppm/month with a pH 9 solution at 210°F for an exposed surface of about 36 in.²/gal of solution (much greater than any potential exposure in the containment). The boron loss during the high-temperature transient test (320°F maximum) was about 200 ppm. Figure 6.1-11 shows a representation of the boron loss from the emergency core cooling solution versus time, by a boron-concrete reaction following a design-basis accident. The time period from 0 to 6 hours shows the loss during a conservative high-temperature transient test, ambient to 320°F to 285°F. The data from 6 hours to 30 days are based on 210°F data.

A depletion of boron at this rate poses no threat to the safety of the reactor because of the large shutdown margin and the feasibility of adding more boron solution should sample analysis show a need for such action.

Furthermore, essentially all of the concrete in the containment which will be exposed to the spray solution has been coated with one of the products shown to be resistant to the solution at the design-basis accident condition.

6.1.2.8 Miscellaneous Materials of Construction

6.1.2.8.1 Sealants

Candidate sealant materials for use in the reactor containment system were evaluated in simulated design-basis accident environments. Cured samples of various sealants were exposed in alkaline sodium borate solution, pH 10.0, 3000 ppm, to a maximum temperature of 320°F.

Table 6.1-7 presents a summary of the sealant materials tested, together with a description of the panels' appearance after testing. Three generic types of sealants were tested: butyl rubber, silicone, and polyurethane. Each of the materials was the "one package" type, that is, no mixing of components was necessary prior to application. The materials were applied on stainless steel and allowed to cure well in excess of the manufacturer's recommended time prior to testing.

The test results showed that the silicone sealants tested were chemically resistant to the design-basis accident environment and were acceptable for use in the containment. Sealant 780 by Dow Corning Corporation was selected for use at Ginna Station. The major application of this sealant is for use on the containment liner insulation panels. Sealant 780 will contribute no deterioration products to the emergency core cooling solution during the post design-basis accident period and will maintain its structural integrity.

6.1.2.8.2 Containment Recirculation Fan Cooler (CRFC) Materials

Samples of the following containment recirculation fan cooler (CRFC) materials were exposed in an autoclave facility to the design-basis accident temperature-pressure cycle:

- Moisture separator pad.
- High efficiency particulate air filter media.
- Pleated asbestos separators.
- Adhesive for joining separator pads and high efficiency particulate air filter media corners.
- Neoprene gasketing material.

The materials were exposed in both the steam phase and liquid phase of a solution of sodium tetraborate (15 ppm boron) to simulate the concentrations expected downstream of the containment recirculation fan cooler (CRFC) cooling coils. Examination of the specimens after exposure showed the following:

- A. Moisture separator pads were somewhat bleached in color but maintained their structural form and showed good resiliency in both liquid and steam phase exposure. These loss-of-coolant accident (LOCA) qualified moisture separator pads were replaced during the 1993 MODE 6 (Refueling) outage. The replacement pads contain the same filter media but the binder material was modified slightly. Although the replacement pad binder material was not tested in an autoclave, an analysis was performed which concluded that the replacement binder material will withstand loss-of-coolant accident conditions.
- B. High efficiency particulate air filter media maintained their structural integrity in both the liquid and steam phase with no apparent change.

- C. Pleated asbestos separators showed some slight color bleaching; however, both steam and liquid phase samples maintained their structural integrity with only slight loss of rigidity. All of these loss-of-coolant accident qualified waterproof asbestos separators were replaced during the 1993 refueling outage and subsequent outages, with the last separators being replaced during the 1999 refueling outage. The replacement pleated separators are made of vinyl-coated aluminum. The vinyl-coated aluminum material was tested in an autoclave and an analysis was performed that concluded that this material will withstand loss-of-coolant accident conditions.
- D. Adhesive material for the high efficiency particulate air/separator pad edges showed no deterioration or embrittlement and maintained its adhesive property.
- E. Neoprene gasketing material is also satisfactory in both the steam and liquid phase. The material showed only weight gain and a shrinkage of 15% to 30% based on a superficial, one flat side area. The gasket thickness decreased about 10%. The gasket material was unrestrained during the exposure; therefore, the dimensional changes experienced are greater than those which would result in the containment recirculation fan cooler (CRFC).

6.1.2.8.3 Polyvinyl Chloride Protective Coating

Tests were conducted to determine the stability of the polyvinyl chloride protective coating used on the neutron detector aluminum conduit, Table 6.1-3, in the design-basis accident environment. Samples of polyvinyl chloride exposed to alkaline sodium borate solutions at design-basis accident conditions showed no loss in structural rigidity and no change in weight or appearance.

A sample of polyvinyl-chloride-coated aluminum conduit (1 in. O.D. x 8 in. length) was irradiated by means of a cobalt-60 source, at an average dose rate of 3.2×10^6 rad/hr to a total accumulated dose of 9.1×10^7 rad. The specimen was immersed in alkaline sodium borate solution (pH 10, 3000 ppm boron) at 70°F. Visual examination of the coating after the test showed no evidence of cracking, blistering, or peeling and the specimen appeared completely unaffected by the gamma exposure. Chemical analysis of the test solution indicated that some bond breakage had occurred in the polyvinyl chloride coating as evidenced by an increase in the chloride concentration. The gamma exposure of approximately 10^8 rad resulted in a release to the solution of 26 mg of chloride/ft² of exposed polyvinyl chloride surface.

Considering the total surface area of polyvinyl chloride coating present in containment (approximately 500 ft²) and the emergency core cooling solution volume of 313,000 gal, the chloride concentration increase in the emergency core cooling solution due to irradiation of the coating, would be approximately 0.01 ppm.

It is concluded, therefore, that the polyvinyl chloride protective coating will be stable in the design-basis accident environment. It should be pointed out, however, that no credit was assumed for this coating in the determination of hydrogen buildup from aluminum corrosion (Section 6.1.2.5). This is most conservative since the neutron detector conduit contributes about 25% of the total calculated aluminum corrosion hydrogen inventory.

6.1.2.8.4 Vinylcel Insulation

The inner surface of the containment building is insulated with panels of Vinylcel, a rigid cross-linked polyvinyl chloride foam. The total weight of insulating foam in the containment building is about 15,000 lb.

An analysis of the decomposition effects of Vinylcel insulation in a design-basis accident environment was conducted by Wyle Laboratories, and a report was submitted to the NRC (*Reference 14*).

It was concluded in the report that the only gas produced in more than trace quantities would be hydrogen chloride. No hydrogen chloride would be expected until radiation levels exceed 5×10^6 rad. The postulated total integrated dose of 2×10^8 rad was assumed to result in the release of all of the chloride content of the foam over a period of several weeks resulting in about 4 metric tons of hydrogen chloride gas. The calculation indicated that this much hydrogen-chloride gas would be effectively buffered to a pH of about 8.5 by the alkaline sodium hydroxide additions to the containment spray system. The corrosion effects of dry hydrogen-chloride gas were determined to be insignificant. Corrosion of the carbon-steel containment liner could occur if aqueous hydrogen chloride contacts its surface; however, the protective coating is a barrier to direct hydrogen chloride/carbon steel contact.

The corrosion effect of the 0.019-in. stainless steel facing that covers the containment liner insulation was also considered by Wyle Laboratories (*Reference 15*). Some corrosion of the stainless steel panels could occur within 1.4 months. This corrosion would not cause failure of the panels, however. Panel failure due to corrosion could occur by one of two methods: either the panel could corrode around the bolt area, or the bolts themselves could corrode. Both of these cases were examined.

Using an assumption of long-term containment temperature of 150°F, it was calculated that through-wall panel corrosion around the bolts could occur at about 7.5 months. Sufficient bolt corrosion to cause shear failure could occur at about 1.9 years. A more realistic estimate of the long-term containment temperature would be about 100°F. Using the resulting

extrapolated corrosion rate of about 13 mils/year i.e., $\left(10 \times 2^{\frac{11.1}{30}} = 12.93 \text{ mils / yr} \right)$

would change these estimates to 1.46 years and 4.38 years, respectively. These estimates show that potential panel failure would not be expected to occur for a long time following a postulated loss-of-coolant accident.

At a time many months into the accident, the water in the containment would be quiescent. Only a few hundred gallons per minute would be drawn through the sump for use in long-term postaccident recirculation. Any panels which might become detached from the containment wall would simply sink to the floor. Since the sump is about 20 ft from the containment wall and the panels are 44 in. by 84 in., they would not fall near the sump. There would be no forcing mechanism to draw the panels toward the containment sump. Any containment liner insulation which might become detached, being of very light material (about 4 pcf and of low

moisture absorptivity, would float. Therefore, RG&E does not believe that there is any potential for sump clogging due to corrosion and detachment of the containment liner insulation and facing.

6.1.2.9 Organic Materials Evaluation

The organic materials suitability for the postaccident environment was evaluated under SEP Topic VI-1.1 (*Reference 11*). It was concluded that the organic materials used in the plant and described in Sections 6.1.2.6 and 6.1.2.8 are acceptable and will not interfere with the operation of engineered safety features under accident conditions, and that they will maintain their integrity and remain in serviceable condition after exposure to the severe environmental conditions of a design-basis accident. It was also concluded that insignificant quantities of organic gases and hydrogen would be generated under these conditions.

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Table 6.1-1
REVIEW OF SOURCES OF VARIOUS ELEMENTS IN CONTAINMENT AND THEIR
EFFECTS ON MATERIALS OF CONSTRUCTION

<u>Group</u>	<u>Representative Elements</u>	<u>Corrosivity of Elements</u>	<u>Sources of Elements</u>
0	He, Ne, Kr, Xe	No effect on any materials of construction	Fission product release
I a	Li, Na, K	Generally corrosion inhibitive properties for steels and copper alloys; harmful to aluminum	Li - coolant pH adjusting agent Na - spray additive concrete leach product K - concrete leach product
II a	Mg, Ca, Sr, Ba	Generally not harmful to steel or copper alloys	Concrete leach products - deteriorated insulation
III a	Y, La, Ac	Not considered harmful in low concentrations	Fission product release
IV a	Ti, Zr, Hf	Not considered harmful to any materials	Fuel rod cladding, control rod material, alloying constituent
V a	V, Nb, Ta	Not considered harmful to any materials	Alloying constituents in low concentration
VI a	Cr, Mo, W	Not considered harmful to any materials	Alloying constituents in equipment
VII a	Mn, Tc, Re	Not considered harmful constituent	Mn - alloy
VIII	Fe, Ni, Cr, Os	Fe, Ni, Cr not harmful to any materials	Fe, Ni, Cr, alloy constituents - others have no identifiable sources
I b	Cu, Ag, Au	Not harmful to any materials	Cu present as material of construction and alloying constituent
II b	Zn, Cd, Hg	Hg - harmful to stainless steel, Cu alloys, aluminum Zn - unknown Cd - unknown	Hg has been entirely excluded from use in the containment. Cd finish plating on components. Zn galvanizing and alloying constituent.
III b	B, Al, Ga, In	Not harmful to materials	B - neutron poison additive Al - materials of construction

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<u>Group Representative Elements</u>		<u>Corrosivity of Elements</u>	<u>Sources of Elements</u>
IV b	C, Si, Sn, Pb	C, Si, Sn not harmful to materials Pb considered harmful to nickel alloys	Si - concrete leach product Pb - alloy constituent in some brazes
V b	N, P, As, Sb, Bi	No effect from N unless ammonia is formed - others unknown	N - containment air - others not identified in significant materials

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<u>Group</u>	<u>Representative Elements</u>	<u>Corrosivity of Elements</u>	<u>Sources of Elements</u>
VI b	O, S, Se, Te	S possibly harmful to nickel alloys	S - oils, greases, insulating materials Te - fission product
VII b	F, Cl, Br, I	F considered potentially harmful to Zircaloy Cl potentially harmful to stainless steel materials Br and I not generally harmful	F - organic Cl - concrete leach product, general contamination Br and I - fission products, low concentration

Table 6.1-2
MATERIALS OF CONSTRUCTION IN GINNA STATION CONTAINMENT

<u>Material</u>	<u>Equipment Application</u>
300 Series stainless steel	Reactor coolant system, residual heat removal loop, spray system, reactor compartment coolers
400 Series stainless steel	Valve materials, lattice grid flat bars
AL-6XN stainless steel	Tubing for containment recirculation fan cooler (CRFC) coils and CRFC motor coolers
Inconel (600, 718, 690)	Steam generator tubing, reactor vessel nozzles, core supports, and fuel rod grids
Galvanized steel	Ventilation duct work, control rod drive mechanism shroud material instrumentation and control conduit
Aluminum	See Table 6.1-3
Copper	CRFCs: coil fins and motor cooler fins; reactor compartment coolers: Fins
Carbon steel	Component cooling loop, structural steel, main steam piping, etc.
Monel	Possibly instrument housings
Brass	Possibly instrument housings
Polyvinyl chloride	Conduit sheathing, electrical insulation, containment liner insulation
Protective coatings	General use on carbon steel structures and equipment, concrete
Inorganic zincs	
Epoxy	
Modified phenolics	
Phenyl formaldehyde resin	High efficiency filter binders in air handling system
Resorcinol formaldehyde resin	Moisture separator pad binders in air handling system
Gasketing and sealants	Ventilation duct work gasketing, containment liner insulation sealant

Table 6.1-3
INVENTORIES OF ALUMINUM INSIDE CONTAINMENT BUILDING

<u>Item</u>	<u>Weight (lbs)</u>	<u>Areas (ft²)</u>
Reactor Vessel insulation		
foil	129	very high
angle	302	354
CRDM Cooling Shroud Door Handles	5	2
Flux map drive system	137	57
Nuclear instrumentation system	280	95
Rod Position Indicators	116	75
Miscellaneous valves	230	86
Control rod drive mechanism connectors	117	26
Refueling machine	28	5
Equipment Hatch Job Crane	3	4.5
Roto-jet outlets	420	690
Canes air outlet	35	23
Damper operators	48	31
Lighting cables	350	400
Lighting fixtures	310	26
Hoffman Enclosure Covers	9	12
Neutron detector conduits	1000	497
Connectors	15	small
Transmitters, TRDs, indicators, radiation monitors	132	38
Platform to Reactor Head	81	90
Ladder and Platform to CRD	57	45

Table 6.1-4
CORROSION OF MATERIALS IN SODIUM BORATE SOLUTION

<u>Material</u>	<u>Maximum Observed Corrosion Rate (ml/month)</u>
Carbon steel	0.003
Zircaloy-4	0.004
Inconel-718	0.003
Copper	0.015
90-10 Cu-Ni	0.02
70-30 Cu-Ni	0.006
Galvanized carbon steel	0.051
Brass	0.01

Table 6.1-5
GINNA Post-LOCA CONTAINMENT TEMPERATURES

<u>Time Interval (sec)</u>			<u>Temperature (°F)</u>
0	-	1,000	286
1,000	-	8,000	250
8,000	-	20,000	200
20,000	-	86,400	190
86,400	-	8,640,000	153 ^a
		(100 days)	

- a. Note that final temperature is fixed at 153°F in order to maintain the long-term corrosion rate of aluminum at 200 mils/yr per USNRC Regulatory Guide 1.7.

**Table 6.1-6
CONCRETE SPECIMEN TEST DATA**

<u>Concrete - Boron Test</u>	<u>Total Exposure</u>	<u>Surface/ Volume</u>	<u>Exposed Weight</u>	<u>Initial Specimen</u>	<u>Visual Examination</u>
<u>Number</u>	<u>Period (days)</u>	<u>(in.²/ gal)</u>	<u>Change (grams)</u>	<u>Weight (grams)</u>	
1	24	28	-22.4	560.0	No apparent change
3	28	20	+21.5	404.0	Light, yellowish deposit on specimen
4 ^a	72	38	0	641.2	No apparent change, coating adhesion excellent
5	72	43	-0.2	769.5	Light, hard deposit on specimen
6	4 ^b	54	---	601.4	No apparent change, small amount of sand particles in test can
7	175	23	+11.0	457.0	No apparent change
8 ^a	175	38	+26.5	751.0	No apparent change, coating adhesion excellent
9 ^a	5 ^b	78	+4.0	702.0	No apparent change, coating adhesion excellent

- a. These specimens coated with Phenoline 305. All others were uncoated.
- b. These tests were at high temperature design-basis accident transient conditions. All others at 195°F to 205°F.

Table 6.1-7
EVALUATION OF SEALANT MATERIALS FOR USE IN THE CONTAINMENT

<u>Sealant Type</u>	<u>Manufacturer</u>	<u>Post-test Appearance</u>
Butyl rubber	A	Unchanged, somewhat flexible
Silicone	B	Unchanged, flexible
Silicone	B	Unchanged, flexible
Polyurethane	C	Sealant bubbled and became very soft. Solution permeated into bubbles
Polyurethane	C	Sealant swelled and became soft, solution permeated into material
Polyurethane	C	Sealant swelled, very soft and tacky, solution permeated into material

6.2 CONTAINMENT SYSTEMS

The containment systems include the containment system structure, the containment heat removal systems, the containment isolation system, and the containment combustible gas control system.

6.2.1 CONTAINMENT SYSTEM STRUCTURE

The reactor containment structure is a reinforced-concrete vertical right cylinder with a flat base and a hemispherical dome. A welded steel liner is attached to the inside face of the concrete shell to ensure a high degree of leaktightness.

The cylindrical reinforced-concrete walls are 3 ft 6 in. thick and the concrete hemispherical dome is 2 ft 6 in. These thicknesses are established to satisfy the requirements of the structural criteria as well as shielding requirements. These thicknesses are nominal values. The true relevant engineering values are dependent on the specific location in the structure and the loading condition that is present. The concrete base slab is 2 ft thick with an additional thickness of concrete fill of 2 ft over the bottom liner plate. The containment cylinder is founded on rock by means of post-tensioned rock anchors which ensure that the rock then acts as an integral part of the containment structure. The cylinder wall is prestressed vertically with 160 tendons which are coupled to the rock anchors.

Details on the containment structure design are given in Section 3.8.1.4.

6.2.1.1 Design Basis

The safety design basis for the containment is that the containment must withstand the pressures and temperatures of the design-basis accident without exceeding the design leak rate. In conjunction with the other containment systems and engineered safety features, the release of radioactive material subsequent to a design-basis accident does not result in doses exceeding the guideline values of 10 CFR 50.67. The containment is designed so that the discharge of reactor coolant through a double-ended rupture of the main loop piping (referred to as a loss-of-coolant accident or LOCA), followed by operation of only those engineered safety features which can be run simultaneously with power from one emergency onsite diesel generator (two high-head safety injection pumps, one residual heat removal pump, two containment recirculation fan cooler (CRFC) units, and one containment spray pump), results in a sufficiently low radioactive materials leakage from the containment structure so that there is no undue risk to the health and safety of the public.

The reactor containment completely encloses the entire reactor and reactor coolant system and ensures that an acceptable upper limit for leakage of radioactive materials to the environment is not exceeded even if gross failure of the reactor coolant system occurs. The structure provides biological shielding for both normal and accident situations.

Access to the containment structure during operation is provided by two containment air locks, (i.e., an equipment hatch and personnel hatch). Each air lock is equipped with two personnel access doors designed with an interlocked door opening feature. The containment

air locks are leak testable at containment design pressure between the two doors. A single control room alarm exists for the open status of any of the four access doors.

The maximum containment leak rate allowed by the Technical Specifications is 0.2 wt % per day. This leakage rate ensures that public exposure in the event of a design-basis accident will be maintained well below the 10 CFR 50.67 guidelines.

6.2.1.1.1 Principal Design Criteria

6.2.1.1.1.1 General

The following design criteria 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 (see Section 3.1.1). Conformance with 1972 General Design Criteria (GDC) of 10 CFR 50, Appendix A, is discussed in Section 3.1.2. The criteria discussed in Section 3.1.2 as they apply to containment systems include 16, 36, 38, 39, 40, 41, 42, 43, 50, 51, 52, 53, 54, 55, 56, and 57. Evaluations performed since 1972 against these GDC appear in the design evaluation sections for the containment systems.

6.2.1.1.1.2 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 containment system structure 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 governing the above features conforms to the applicable provisions of recognized codes and good nuclear practice. The concrete structure of the reactor containment conforms to the applicable portions of ACI-318-63. Further elaboration on quality standards of the reactor containment is given in Section 3.8.1.

6.2.1.1.1.3 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 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 components and supporting structures of the reactor containment are designed so that there is no loss of function of such equipment in the event of maximum potential ground acceleration acting in the horizontal and vertical directions simultaneously. The dynamic response of the structure to ground acceleration, based on the site characteristics and on the structural damping, is included in the design analysis.

The reactor containment is defined as a Seismic Category I structure (see Section 3.8.1). Its structural members have sufficient capacity to accept a combination of normal operating loads, functional loads due to a loss-of-coolant accident (LOCA), and the loadings imposed by the maximum potential earthquake, without exceeding specified stress limits.

6.2.1.1.1.4 *Fire Protection*

CRITERION: A reactor facility shall be designed to ensure that the probability of events such as fires and explosions and the potential consequences of such events will not result in undue risk to the health and safety of the public. Noncombustible and fire resistant materials shall be used throughout the facility wherever necessary to preclude such risk, particularly in areas containing critical portions of the facility such as containment, control room, and components of engineered safety features (AIF-GDC 3).

The reactor containment system is designed to maintain its capability in case of fire to safely shut down and isolate the reactor.

The containment ventilation systems are operable from the control room. Smoke or heat (ionization) detectors and control room alarms are provided for the post-accident charcoal filter banks and auxiliary filter charcoal banks. The postaccident charcoal filters are not required following a design-basis accident, however, manually actuated sprays are installed in the event of a coincident beyond design-basis fire in the filters.

The containment liner thermal insulation does not support combustion.

6.2.1.1.1.5 *Records Requirement*

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, construction, and testing of the containment are maintained throughout the duration of the facility operating license.

6.2.1.1.1.6 *Reactor Containment*

CRITERION: The containment structure shall be designed (a) to sustain without undue risk to the health and safety of the public the initial effects of gross equipment failures, such as a large reactor coolant pipe break, without loss of required integrity, and (b) together with other engineered safety features as may be necessary, to retain for as long as the situation requires, the functional capability of the containment to the extent necessary to avoid undue risk to the health and safety of the public (AIF-GDC 10).

The containment structure is a reinforced-concrete vertical cylinder with vertical prestressed tendons in the vertical wall, a reinforced-concrete ring anchored to bedrock, and a reinforced hemispherical dome. See Section 3.8.1.

The design pressure of the containment exceeds the peak pressure occurring as the result of the complete blowdown of the reactor coolant through any rupture of the reactor coolant system up to and including a loss-of-coolant accident (LOCA) as well as a postulated main steam line break.

The containment structure and all penetrations are designed to withstand, within design limits, the combined loadings of the design-basis accident and design seismic conditions.

All piping systems which penetrate the containment are anchored in the penetration sleeve or the structural concrete of the Containment Building. The penetrations for the main steam, feedwater, blowdown, and sample lines are designed so that the penetration is stronger than the piping system and the containment will not be breached due to a postulated pipe rupture. The liner thickness in the vicinity of typical penetrations is increased to a minimum of 3/4 in. A typical analysis of a penetration including the liner to sleeve connection is given in Section 3.8.1.5. The pipe capacity in flexure is assumed to be limited to the plastic moment capacity based upon the ultimate strength of the pipe material. All lines connected to the primary coolant system that penetrate the containment are also anchored in the secondary shield walls (i.e., walls surrounding the steam generators and reactor coolant pumps) and are each provided with at least one valve between the anchor and the reactor coolant system. These anchors are designed to withstand the thrust, moment, and torque resulting from a postulated rupture of the attached pipe.

All containment isolation valves are supported to withstand, without impairment of valve operability, the combined loadings of the design-basis accident and design and maximum potential seismic conditions.

The design pressure is not exceeded during any subsequent long-term pressure transient determined by the combined effects of heat sources, such as residual heat and metal-water

reactions, with minimum operation of the emergency core cooling and the containment recirculation fan cooling and containment spray systems.

6.2.1.1.1.7 *Reactor Containment Design Basis*

CRITERION: The reactor containment structure, including openings and penetrations, and any necessary containment heat removal systems, shall be designed so that the leakage of radioactive materials from the containment structure under conditions of pressure and temperature resulting from the largest credible energy release following a loss-of-coolant accident, including the calculated energy from metal-water or other chemical reactions that could occur as a consequence of failure of any single active component in the Emergency Core Cooling System (ECCS) will not result in undue risk to the health and safety of the public (AIF-GDC 49).

The following general criteria are used to ensure conservatism in computing the required structural load capacity:

- a. In calculating the containment pressure, rupture sizes up to and including a double-ended severance of a reactor coolant pipe (i.e., LOCA) or main steam line are considered.
- b. In considering post-accident pressure effects, various malfunctions of the emergency systems are evaluated consistent with the single failure criteria.
- c. The pressure and temperature loadings obtained by analyzing various accidents, when combined with operating loads and maximum wind or seismic forces, do not exceed the load-carrying capacity of the structure, its access opening, or penetrations.

6.2.1.1.1.8 *Nil Ductility Transition Requirement for Containment Material*

CRITERION: The selection and use of containment materials shall be in accordance with applicable engineering codes (AIF-GDC 50).

The selection and use of containment materials comply with the applicable codes and standards tabulated in Section 3.8.1.2.5.

The concrete containment is not susceptible to a low temperature brittle fracture.

The containment liner is enclosed within the containment and thus is not exposed to the outside temperature extremes. The containment average ambient temperature during operation is between 50°F and 125°F. The minimum service metal temperature of the containment liner is well above the nil ductility transition temperature +30°F for the liner material. Containment penetrations which can be exposed to the environment are also designed to the nil ductility transition +30°F criterion.

6.2.1.1.2 Supplementary Accident Criteria

Systems relied upon to operate under post-accident conditions, which are located external to the containment and communicate directly with the containment atmosphere, are considered to be extensions of the leakage-limiting boundary.

The pressure-retaining components of the containment structure are designed for the maximum potential earthquake ground motion of the site combined with the simultaneous loads of the design-basis accident as follows:

- A. The liner is designed to ensure that no average strains greater than the strain at the guaranteed yield point occur at the factored loads. The structural integrity of the liner is discussed in Sections 3.8.1.4.7 and 3.8.2.3. The liner will maintain its leaktight integrity under postulated pressure and thermal loads created by a LOCA or a main steam line break (see Section 3.8.2.3).
- B. The prestressed concrete is designed on the basis of a resultant concrete compression or zero tension due to primary and secondary membrane forces resulting from the factored loads.
- C. The mild steel reinforcement is designed to ensure that no strains greater than the strain at the guaranteed yield point occur at the factored loads.

The pressure-retaining components of containment subject to deterioration or corrosion in service are provided with appropriate protective means or devices, e.g., protective coatings.

6.2.1.1.3 Energy and Material Release

The design pressure is not exceeded during any subsequent long-term pressure transient determined by the combined effects of heat sources such as residual heat and metal-water reactions, structural heat sinks, and the operation of other engineered safety features. The containment functional design evaluation is discussed in the sections that follow.

6.2.1.2 Containment Integrity Evaluation

6.2.1.2.1 Systematic Evaluation Program (SEP) Evaluation

6.2.1.2.1.1 Introduction

As part of the Systematic Evaluation Program (SEP), the containment functional design capability was reevaluated. The evaluation performed by the NRC compared Ginna Station with the criteria used in the early 1980s by the regulatory staff for licensing new facilities. Specifically, the areas reevaluated were (1) the containment pressure and heat removal capability and (2) the mass and energy release from possible pipe breaks inside containment.

The review criteria used in the evaluation are contained in the following documents:

- a. 10 CFR 50, Appendix A, General Design Criteria.
 - GDC 16 - Containment Design.
 - GDC 38 - Containment Heat Removal.
 - GDC 50 - Containment Design Basis.
- b. 10 CFR 50.46, Acceptance Criteria for Emergency Core Cooling System (ECCS).
- c. 10 CFR 50, Appendix K, Emergency Core Cooling System (ECCS) Evaluation Models.
- d. NUREG 75/087, Standard Review Plan, Section 6.2.1, Containment Functional Design.

The review of the containment analysis as described in the original FSAR indicated two differences from the criteria used at that time. First, for the LOCA analysis, the cold-leg pump suction break location, the core reflood phase of mass and energy release, and the release of secondary system energy were not considered. Second, the main steam line break analysis was not performed in detail.

6.2.1.2.1.2 *NRC Analyses*

To assess the significance of these two differences, the NRC performed independent LOCA and main steam line break analyses. It was concluded (*Reference 1*), overall, that in the original LOCA analysis the design-basis pressure envelopes the NRC results, and the design-basis temperature profile exceeds the NRC results, except in the range between 10,000-20,000 seconds after the design-basis event. In this range, the design-basis temperature profile was revised for the purposes of environmental qualification of equipment (see Section 3.11.3.1.1). Regarding the main steam line break analysis, the NRC concluded that the calculated peak containment pressure is less than the containment design pressure and the temperature profile as revised for the LOCA case was acceptable for use in equipment qualifications.

6.2.1.2.1.3 *Summary*

Along with the SEP evaluations of the containment functional design, the response to concerns raised by IE Bulletin 80-04 was also evaluated. The concerns related to continued addition of water to the affected steam generator in the event of a main steam line break and the possibility of containment over-pressurization. The NRC concluded (*Reference 2*) that there is no potential for over-pressurization resulting from a main steam line break at Ginna because (1) the main feedwater system is automatically isolated and the preferred auxiliary feedwater system limits flow to the affected steam generator, (2) the motor driven auxiliary feedwater pumps (MDAFW) are protected from the effects of runout flow and therefore can be expected to carry out their intended function during a main steam line break event, and (3) all potential water sources are identified and, although a reactor return-to-power is predicted, there is no violation of the specified acceptable fuel design limits.

6.2.1.2.2 Mass and Energy Release Safety Analysis

6.2.1.2.2.1 *Loss-of-Coolant (LOCA) Mass and Energy Releases*

The uncontrolled release of pressurized high-temperature reactor coolant, termed a loss-of-coolant accident (LOCA), will result in release of steam and water into the containment. This, in turn, will result in increases in the local subcompartment pressures, and an increase in the global containment pressure and temperature. There are both long and short-term issues relative to a postulated LOCA transient that must be considered for a complete containment integrity analysis for the R. E. Ginna Power Station.

The containment long-term LOCA mass and energy releases, addressed in this section are utilized as input to the containment analysis (see subsection 6.2.1.2.6). The containment analysis demonstrates the acceptability of the containment safeguards systems to mitigate the consequences of a hypothetical large-break LOCA (LBLOCA) the long-term LOCA mass and energy releases were analyzed to 30 days and used as input to the containment integrity analysis. The containment safeguards systems must be capable of limiting the peak

containment pressure to less than the design pressure, and maintaining the Environmental Qualification (EQ) conditions within acceptable limits. The long-term LOCA mass and energy releases analyzed for Ginna are comprised of two parts. The Westinghouse LOCA mass and energy release evaluation model described in *Reference 47* is used for the blowdown, reflood and post-reflood phases out to 3600 seconds, i.e., the time at which all energy in the primary heat structures and steam generator secondary is assumed to be released / depressurized to atmospheric pressure, (i.e., 14.7 psia and 212°F). Then the long-term post-one-hour releases (boil-off from the core at the decay heating rate) are calculated by the GOTHIC code. The long-term post-one-hour mass and energy release was optimized to reduce conservatism in the long-term containment temperature calculation. GOTHIC was used to calculate the mass and energy boil-off rate, via interactive iteration on the calculated Residual Heat Removal (RHR) heat exchanger conditions. The pre-3600-second and post-3600 second mass and energy releases were inputs to the containment integrity analysis (discussed in UFSAR Section 6.2.1.2.6, Long-Term LOCA Containment Response).

The short-term LOCA-related mass and energy releases were used as input to the subcompartment analyses. Those analyses were performed to ensure that the walls of a subcompartment can maintain their structural integrity during the short pressure pulse (generally less than 3 seconds) accompanying a high-energy line pipe rupture within that subcompartment. Short-term mass and energy release calculations are performed to support analysis of reactor coolant loop (RCL) compartments (Section 6.2.1.3.2), the concrete around and under the reactor vessel (Section 6.2.1.3.4), and the concrete structures around the steam generator (Section 6.2.1.3.4). Since Ginna is approved for leak-before-break (LBB), the LBB methodology was used to qualitatively demonstrate that any changes associated with the EPU are off-set by the LBB benefit (i.e., of using the smaller reactor coolant system (RCS) nozzle breaks). This demonstrates that the current licensing bases for these subcompartments remain bounding. The critical mass flux correlation utilized in the SATAN computer program (*Reference 48*) was used to conservatively estimate the impact of the changes in RCS temperatures on the short-term release. The evaluation showed that the design basis releases would remain bounding due to LBB. Section 6.2.1.3, Short-Term LOCA Mass and Energy Releases, discusses the short-term LOCA mass and energy releases generated for the EPU program relative to Section 6.2.1.3, Evaluation of Containment Internal Structures.

Loss-of-Coolant (LOCA) Long-Term Mass and Energy Releases

This section discusses the long-term LOCA mass and energy releases generated for the Ginna EPU program. The long-term LOCA mass and energy releases rates described in this section form the basis of further computations to evaluate the containment response (containment integrity peak pressure and the long-term containment temperature calculations) in support of the environmental equipment qualification (EQ) analysis following the postulated LOCA (UFSAR Section 6.2.1.2.6) and to ensure that containment design margin is maintained. Long-Term LOCA mass and energy releases for the hypothetical double-ended pump suction (DEPS) rupture with minimum and maximum safeguards are discussed in this section. The double-ended hot (DEHL) rupture break case mass and energy release which is limiting for the blowdown portion of the LOCA transient (approximately < 16 seconds) is also discussed.

The EPU analyses were performed using the Westinghouse LOCA Mass and Energy Release Model for Containment Design, March 1979 Version, described in WCAP-10325-P-A (*Reference 47*). The NRC review and approval letter is included with *Reference 47*. This methodology has been used and approved on many plant-specific dockets. The application of LOCA mass and energy release methodologies is appropriate to produce mass and energy release for the long-term containment EQ environmental conditions (*Reference 49*). UFSAR Section 6.2.1.2.6, Long-Term LOCA Containment Response, discusses containment integrity/ response analysis relative to containment peak pressure, temperature, long-term containment temperature and containment heat removal. The model as re-run for Reference 60 incorporated corrections identified in Reference 62.

6.2.1.2.2.2 *Input Parameters and Assumptions*

The mass and energy release analysis is sensitive to the characteristics of various plant systems, in addition to other key modeling assumptions. Where appropriate, bounding inputs are utilized and instrumentation uncertainties are included. For example, the RCS operating temperatures are chosen to bound the highest average coolant temperature range of all operating cases and a temperature uncertainty allowance of (+4.0°F) is then added. Nominal parameters are used in certain instances. For example, the RCS pressure in this analysis is based on a nominal value of 2250 psia plus an uncertainty allowance (+60.0 psi).

All input parameters are chosen consistent with accepted analysis methodology. Some of the most critical items are the RCS initial conditions, core decay heat, safety injection flow, and primary and secondary metal mass and steam generator heat release modeling. Tables 6.2-1 through 6.2-4 present key data assumed in the analysis.

The core rated power of 1811 MWt adjusted for calorimetric error (i.e., 102% of 1775 MWt) was used in the analysis. As previously noted, the use of RCS operating temperatures to bound the highest average coolant temperature range were used as bounding analysis conditions. The use of higher temperatures is conservative because the initial fluid energy is based on coolant temperatures that are at the maximum levels attained in steady-state operation. Additionally, an allowance to account for instrument error and deadband is reflected in the initial RCS temperatures. As previously discussed, the selection of 2310 psia (2250 psia nominal value + 60 psi uncertainty allowance) as the limiting pressure is considered to affect the blowdown phase results only, since this represents the initial pressure of the RCS. The RCS rapidly depressurizes from this value until the point at which it equilibrates with containment pressure.

The rate at which the RCS blows down is initially more severe at the higher RCS pressure. Additionally, the RCS has a higher fluid density at the higher pressure (assuming a constant temperature) and subsequently has a higher RCS mass available for releases. Thus, 2250 psia plus uncertainty was selected for the initial pressure as the limiting case for the long-term mass and energy release calculations.

The selection of the fuel design features for the long-term mass and energy release calculation is based on the need to conservatively maximize the energy stored in the fuel at the beginning of the postulated accident (that is, to maximize the core-stored energy). The core-stored energy that was selected for the 14x14 422V+ fuel product bounds the core-stored energy for the 14x14 optimized fuel assembly (OFA) fuel product and also the transition core. The core-

stored energy is based on the time in life for maximum fuel densification. The assumptions used to calculate the fuel temperatures for the core-stored energy calculations account for appropriate uncertainties associated with the models in the PAD code (such as calibration of the thermal model, pellet densification model, or clad creep model). In addition, the fuel temperatures for the core-stored energy calculation account for appropriate uncertainties associated with manufacturing tolerances (such as pellet as-built density). The total uncertainty for fuel temperature calculation is a statistical combination of these effects and is dependent upon fuel type, power level, and burnup. Thus, the analysis very conservatively accounts for the stored energy in the core.

The nominal RCS volume is increased by 3 percent, (which is composed of 1.6-percent allowance for thermal expansion and 1.4-percent allowance for uncertainty) for the LOCA mass and energy release calculation. This assumption helps maximize the initial RCS mass and energy.

A uniform steam generator tube plugging (SGTP) level of zero percent (0%) was modeled. This assumption maximizes the reactor coolant volume and fluid release by considering the RCS fluid in all steam generator (SG) tubes. During the post-blowdown period, the steam generators are active heat sources, as significant energy remains in the secondary metal and secondary mass that has the potential to be transferred to the primary side. The zero percent SGTP assumption maximizes the heat transfer area and therefore, the transfer of secondary heat across the SG tubes. Additionally, this assumption reduces the reactor coolant loop resistance, which reduces the pressure drop (i.e., ΔP) upstream of the break for the pump suction breaks and increases break flow. Thus, the analysis very conservatively accounts for the effects related to SGTP.

The secondary-to-primary heat transfer is maximized by assuming conservative heat transfer coefficients. This conservative energy transfer is ensured by maximizing the initial internal energy of the inventory in the steam generator secondary side. This internal energy is based on full-power operation plus uncertainties. The BWI replacement SG initial fluid mass was calculated at full power (100%), based on the nominal SG level of 52% Narrow Range Span (NRS), with an uncertainty of +8% NRS (+4% uncertainty plus a +4% bias), and then further increased by 10% to cover uncertainties.

Following the large break LOCA blowdown inside containment, the safety injection system (SIS) operates to reflood the RCS. Regarding safety injection flow, the mass and energy release calculation considered configurations, component failures, and offsite power assumptions to conservatively bound respective alignments. The first phase of the SIS operation is the passive accumulator injection. Two accumulators are assumed available to inject. When the RCS depressurizes below 714.7 psia the accumulators begin to inject. The accumulator injection temperature was conservatively modeled high at 125°F. Relative to the active pumped emergency core cooling system (ECCS) operation, the M&E release calculation considered configurations, component failures, and offsite power assumptions to conservatively bound respective alignments. The cases include a minimum safeguards case (two high-head SI (HHSI) pumps and one low-head SI (LHSI) pump, see Table 6.2-2), and a maximum safeguards case, (three HHSI pumps and two LHSI pumps, see Table 6.2-3). In addition, the containment backpressure is assumed to be equal to the containment design

pressure. This assumption was shown in *Reference 47* to be conservative for the generation of M&E energy releases.

Since the minimum safeguards case was the most limiting, only the results for this case will be presented herein. In summary, the following assumptions were employed to ensure that the mass and energy releases are conservatively calculated, thereby maximizing energy release to containment:

1. The nominal RCS volume is increased by 3 percent (1.6-percent allowance for thermal expansion, and 1.4-percent allowance for uncertainty).
2. The reactor is assumed to be operating at full core power of 1811 MWt; which includes an allowance for calorimetric error of 2.0 percent of power.
3. Core-stored energy is based on the time in life for maximum fuel densification. The assumptions used to calculate the fuel temperatures for the core-stored energy calculation account for appropriate uncertainties associated with the models in the PAD code (e.g., calibration of the thermal model, pellet densification model, cladding creep model, etc.) (*Reference 53*). In addition, the fuel temperatures for the core-stored energy calculation account for appropriate uncertainties associated with manufacturing tolerances (e.g., pellet as-built density). The total uncertainty for the fuel temperature calculation is a statistical combination of these effects and is dependent upon fuel type, power level, and burn-up.
4. The RCS is assumed to be at the maximum expected full power operating temperature and an allowance for temperature measurement uncertainty (+4.0°F) is added. These uncertainties conservatively include both deadband and bias.
5. The RCS is assumed to be at the nominal RCS pressure and an allowance for pressure measurement uncertainty (+60 psi) is added.
6. Conservatively high heat transfer coefficients (i.e., steam generator primary/secondary heat transfer, and RCS metal heat transfer) are modeled.
7. A maximum containment backpressure equal to design pressure (60.0 psig). This assumption determines the end of the blowdown phase and minimizes the safety injection flow rate during the reflood phase.
8. A uniform SGTP level of 0% is assumed. This assumption:
 - Maximizes reactor coolant volume and fluid release,
 - Maximizes heat transfer area across the SG tubes,
 - Reduces reactor coolant loop resistance, which reduces the ΔP upstream of the break for the pump suction breaks and increase break flow
9. The SG initial fluid mass was calculated at full power (100%), and then further increased by 10% to cover uncertainties.
10. Main feedwater addition is modeled. The feedwater control valve closure time is based on time to reach the safety injection (SI) signal, plus electronic delay and valve stroke time.

Thus, based on the previously discussed conditions and assumptions, an analysis of the Ginna Station was performed for the release of mass and energy from the RCS in the event of a large break LOCA at 1811 MWt core power.

Decay Heat Model

The American Nuclear Society (ANS) Standard 5.1 (*Reference 50*) has been used for the determination of decay heat energy in the LOCA M&E release model for the Ginna EPU Program. This standard was balloted by the Nuclear Power Plant Standards Committee (NUP-PSCO) in October 1978 and subsequently approved. The official standard was issued in August 1979. Table 6.2-4 lists the decay heat curve used in the Ginna EPU Program M&E release analysis. Significant assumptions in the generation of the decay heat curve for use in the LOCA M&E release analysis include the following:

- The decay heat sources considered are fission product decay and heavy element decay of U-239 and Np-239.
- The decay heat power from fissioning isotopes other than U-235 is assumed to be identical to that of U-235.
- The fission rate is constant over the operating history of maximum power level.
- The factor accounting for neutron capture in fission products has been taken from *Reference 50*.
- The fuel has been assumed to be at full power for 10^8 seconds.
- The total recoverable energy associated with one fission has been assumed to be 200 MeV/fission.
- Two sigma uncertainty (two times the standard deviation) has been applied to the fission product decay.

Based upon NRC staff review, Safety Evaluation Report of the March 1979 evaluation model, use of the ANS Standard 5.1 (*Reference 50*) decay heat model was approved for the calculation of mass and energy releases to the containment following a LOCA.

Application of Single-Failure Criterion

The mass and energy release calculation assumes a complete loss of all offsite power coincident with the LOCA. The emergency diesel generators (EDGs) are actuated to provide power for the safety injection system. The combination of signal delay plus diesel delay and additional delays (i.e., electronic delays) subsequently delays the operation of the safety injection system (SIS) that is required to mitigate the transient, and results in the delivery of safety injection (SI) after the end of blowdown. Since none of the powered safety systems were assumed to be operational during the initial blowdown phase, this is not an issue for the blow-down phase, thus application of single-failure criteria would not impact the DEHL break (See subsection 6.2.1.2.2.3).

Two cases have been analyzed to assess the effects of a single failure. The first case assumes a single failure of one of the emergency diesel generators, resulting in the loss of one train of safeguards equipment. This assumption results in the loss of one train of safeguards

equipment. Thus, the remaining train was conservatively modeled as: two HHSI pumps and one LHSI pump. The second case assumed the single failure in the containment spray system i.e. with a maximum safeguards SI flow based on no postulated failures that could impact the amount of ECCS flow. The maximum safeguards case was modeled as: three HHSI pumps and two LHSI pumps. The analysis of the cases described provided confidence that the effect of credible single failures is bounded. Only the results of the first case are presented since it is the most limiting.

Acceptance Criteria

Although Ginna is not a Standard Review Plan (SRP) plant, for completeness, the SRP long-term cooling criterion is also examined. A large LOCA is classified as an ANS Condition IV event, an infrequent fault. To satisfy the NRC acceptance criteria presented in the SRP Section 6.2.1.3, the relevant requirements are as follows:

- 10CFR50, Appendix A
- 10CFR50, Appendix K, paragraph I.A

To meet these requirements, the following must be addressed:

- Sources of energy
- Break size and location
- Calculation of each phase of the accident

6.2.1.2.2.3 *Description of Analyses*

The evaluation model used for the long-term LOCA mass and energy release calculations is the March 1979 model described in WCAP-10325-P-A (*Reference 47*). This evaluation model has been reviewed and approved generically by the Nuclear Regulatory Commission (NRC). The approval letter is included with *Reference 47*.

This report section presents the long-term LOCA M&E releases generated in support of the Ginna EPU program. These M&E releases were used in the containment integrity analysis and environmental qualification temperature evaluation (Section 6.2.1.2.6, Long-Term LOCA Containment Response). Even though this is a first-time application for the R. E. Ginna Station, the March 1979 model described in *Reference 47* has been utilized and approved on the plant-specific dockets for other Westinghouse pressurized water reactors (PWRs).

The M&E release rates described in this section form the basis of further computations to evaluate the containment following the postulated accident. Discussed in this section are the long-term LOCA M&E releases for the hypothetical double-ended pump suction (DEPS) rupture cases considering both minimum safeguards, and maximum safeguards and also the DEHL rupture case. The DEPS break cases were analyzed with two service water pumps in operation. Only the results for the limiting peak pressure case and the limiting long-term EQ case are presented. The M&E releases and applicable transient data for the limiting case is shown in Tables 6.2-5 through 6.2-13, 6.2-21, and 6.2-22.

LOCA Mass and Energy Release Phases

The containment system receives mass and energy releases following a postulated rupture in the RCS. These releases continue over a time period, which, for the LOCA mass and energy analysis, is typically divided into four phases:

- Blowdown - the period of time from accident initiation (when the reactor is at steady-state operation) to the time that the RCS and containment reach an equilibrium state.
- Refill - the period of time when the lower plenum is being filled by accumulator and Emergency Core Cooling System (ECCS) water. At the end of blowdown, a large amount of water remains in the cold legs, downcomer, and lower plenum. To conservatively consider the refill period for the purpose of containment mass and energy releases, it is assumed that this water is instantaneously transferred to the lower plenum along with sufficient accumulator water to completely fill the lower plenum. This allows an uninterrupted release of mass and energy to containment. Thus, the refill period is conservatively neglected in the mass and energy release calculation.
- Reflood - begins when the water from the lower plenum enters the core and ends when the core is completely quenched.
- Post-reflood (Froth) - describes the period following the reflood phase. For the pump suction break, a two-phase mixture exits the core, passes through the hot legs, and is super-heated in the steam generators prior to exiting the break as steam. After the broken loop steam generator cools, the break flow becomes two phase.

Computer Codes

The *Reference 47* mass and energy release evaluation model is comprised of mass and energy release versions of the following codes: SATAN VI, WREFLOOD, FROTH, and EPITOME. These codes were used to calculate the long term LOCA mass and energy releases for the Ginna Extended Power Uprate Program.

SATAN VI calculates blowdown, the first portion of the thermal-hydraulic transient following break initiation, including pressure, enthalpy, density, mass and energy flow rates, and energy transfer between primary and secondary systems as a function of time.

The WREFLOOD code addresses the portion of the LOCA transient where the core reflooding phase occurs after the primary coolant system has depressurized (blowdown) due to the loss of water through the break and when water supplied by the ECCS refills the reactor vessel and provides cooling to the core. The most important feature of WREFLOOD is the steam/water mixing model, discussed in subsection 6.2.1.2.2.4.

FROTH models the post-reflood portion of the transient. The FROTH code calculates the heat release from the energy stored in the secondary fluid and metal masses, excluding the upper internals and upper elliptical head. This part of the steam generator metal mass is not actively cooled by the two-phase fluid circulating through the steam generators and takes longer to cooldown.

EPITOME continues the FROTH post-reflood portion of the transient from the time at which the secondary equilibrates to containment design pressure to the end of the transient. It also compiles a summary of data on the entire transient, including formal instantaneous mass and energy release tables and mass and energy balance tables with data at critical times.

Break Size and Location

Generic studies (Reference 46, Chapter 3) have been performed to determine the effect of postulated break size on the LOCA mass and energy releases. The double-ended guillotine break has been found to be limiting due to larger mass flow rates during the blowdown phase of the transient. During the reflood and froth phases, the break size has little effect on the releases.

Three distinct locations in the reactor coolant system can be postulated for a pipe rupture for mass and energy release purposes:

- Hot leg (between vessel and steam generator)
- Cold leg (between pump and vessel)
- Pump suction (between steam generator and pump)

The double ended hot leg (DEHL) break location yields the highest blowdown mass and energy release rates (*Reference 46*, Section 3.3). Although the core flooding rate would be the highest for this break location, the amount of energy released from the steam generator secondary is minimal because the majority of the fluid that exits the core vents directly to containment bypassing the steam generators. As a result, the reflood mass and energy releases are reduced significantly as compared to either the pump suction or cold-leg break locations where the core exit mixture must pass through the steam generators before venting through the break. Generic studies have confirmed that there is no reflood peak (i.e., from the end of the blowdown period the containment pressure would continually decrease) for the hot leg break. The mass and energy releases for the blowdown phase of the hot-leg break are calculated and used in the containment peak pressure and temperature response calculation. Therefore, with respect to long-term heat removal the hot leg break is not limiting and no further evaluation is necessary.

Studies have determined that the blowdown transient for the double-ended cold leg (DECL) break is, in general, less limiting than that for the pump suction break (*Reference 46*, Section 3.3). The cold leg blowdown is faster than that of the pump suction break, and more mass is released into the containment. However, the core heat transfer is greatly reduced, and this results in a considerably lower energy release into containment. The flooding rate during the reflood phase is greatly reduced, and the energy release rate into the containment is reduced. Therefore, the cold leg break is bounded by other breaks and no further evaluation is necessary.

The pump suction break combines the effects of the relatively high core flooding rate, as in the hot leg break, and the addition of the stored energy in the steam generators. As a result, the pump suction break yields the highest energy flow rates during the post-blowdown period by including all of the available energy of the RCS in calculating the releases to containment.

Therefore, the break locations that were analyzed for this program were the DEPS rupture (10.48 ft²) and the DEHL rupture (9.174 ft²). LOCA mass and energy releases have been calculated for the blowdown, reflood, and post-reflood phases for the DEPS cases. For the DEHL case, the releases were calculated only for the blowdown phase with this methodology.

6.2.1.2.2.4 *Mass and Energy Release Data*

Blowdown Mass and Energy Release Data

The SATAN-VI code is used for computing the blowdown transient. The code utilizes the control volume (element or nodal) approach with the capability for modeling a large variety of plant specific thermal fluid system configurations. The fluid properties are considered uniform, and thermodynamic equilibrium is assumed in each element. A point kinetics model is used with weighted feedback effects. The major feedback effects include moderator density, moderator temperature, and Doppler broadening. A critical flow calculation for subcooled (modified Zaloudek), two-phase (Moody), or superheated break flow is incorporated into the analysis. The methodology for the use of this model is described in *Reference 47*.

Table 6.2-5 presents the calculated mass and energy release for the blowdown phase of the DEHL break. For the hot leg break mass and energy release tables, break path 1 refers to the mass and energy exiting from the reactor vessel side of the break; and break path 2 refers to the mass and energy exiting from the steam generator side of the break.

Table 6.2-8 presents the calculated mass and energy releases for the blowdown phase of the DEPS break with minimum safeguard. For the pump suction breaks, break path 1 in the mass and energy release tables refers to the mass and energy exiting from the steam generator side of the break; and break path 2 refers to the mass and energy exiting from the pump side of the break.

Reflood Mass and Energy Release Data

The WREFLOOD code is used for computing the reflood transient. The WREFLOOD code consists of two basic hydraulic models - one for the contents of the reactor vessel and one for the coolant loops. The two models are coupled through the interchange of the boundary conditions applied at the vessel outlet nozzles and at the top of the downcomer. Additional transient phenomena, such as pumped safety injection and accumulators, reactor coolant pump performance, and steam generator releases are included as auxiliary equations that interact with the basic models as required. The WREFLOOD code permits the capability to calculate variations during the core reflooding transient of basic parameters such as core flooding rate, core and downcomer water levels, fluid thermodynamic conditions (pressure, enthalpy, density) throughout the primary system, and mass flow rates through the primary system. The code permits hydraulic modeling of the two flow paths available for discharging steam and entrained water from the core to the break, that is, the path through the broken loop and the path through the unbroken loops.

A complete thermal equilibrium mixing condition for the steam and ECCS injection water during the reflood phase has been assumed for each loop receiving ECCS water. This is consistent with the usage and application of the (*Reference 47*) M&E release evaluation model

in recent analyses, for example, D. C. Cook Docket Unit 1 (*Reference 51*). Even though the Reference 47 model credits steam/water mixing only in the intact loop and not in the broken loop, the justification, applicability, and NRC approval for using the mixing model in the broken loop has been documented (*Reference 51*). Moreover, this assumption is supported by test data and is further discussed below.

The model assumes a complete mixing condition (i.e., thermal equilibrium) for the steam/water interaction. The complete mixing process, however, is made up of two distinct physical processes. The first is a two-phase interaction with condensation of steam by cold ECCS water. The second is a single-phase mixing of condensate and ECCS water. Since the steam release is the most important influence to the containment pressure transient, the steam condensation part of the mixing process is the only part that need be considered. (Any spillage directly heats only the sump.)

The most applicable steam/water mixing test data have been reviewed for validation of the containment integrity reflood steam/water mixing model. This data was generated in 1/3 scale tests (*Reference 52*), which are the largest scale data available and thus most clearly simulates the flow regimes and gravitational effects that would occur in a pressurized water reactor (PWR). These tests were designed specifically to study the steam/water interaction for PWR reflood conditions.

A group of 1/3 scale steam/water mixing tests discussed in *Reference 52* corresponds directly to containment integrity reflood conditions. The injection flow rates for this group cover all phases and mixing conditions calculated during the reflood transient. The data from these tests were reviewed and discussed in detail in *Reference 47*. For all of these tests, the data clearly indicate the occurrence of very effective mixing with rapid steam condensation. The mixing model used in the containment integrity reflood calculation is, therefore, wholly supported by the 1/3 scale steam/water mixing data.

Additionally, the following justification is also noted. The post-blowdown limiting break for the containment integrity peak pressure analysis is the pump suction double-ended rupture break. For this break, there are two flow paths available in the RCS by which mass and energy may be released to containment. One is through the outlet of the steam generator, the other via reverse flow through the reactor coolant pump. Steam that is not condensed by ECCS injection in the intact RCS loops passes around the downcomer and through the broken loop cold leg and pump in venting to containment. This steam also encounters ECCS injection water as it passes through the broken loop cold leg, complete mixing occurs and a portion of it is condensed. It is this portion of steam that is condensed that is taken credit for in this analysis. This assumption is justified based upon the postulated break location, and the actual physical presence of the ECCS injection nozzle. A description of the test and test results are contained in *References 47* and *51*.

Table 6.2-9 presents the calculated mass and energy releases for the reflood phase of the double ended pump suction rupture with minimum safeguards.

The transient responses of the principal parameters during reflood are given in Table 6.2-10 for the DEPS minimum safeguards case.

Post-Reflood Mass and Energy Release Data

The FROTH code (*References 47 and 48*) is used for computing the post-reflood transient. The FROTH code calculates the heat release rates resulting from a two-phase mixture present in the steam generator tubes. The mass and energy releases that occur during this phase are typically superheated due to the depressurization and equilibration of the broken loop and intact loop steam generators. During this phase of the transient, the RCS has equilibrated with the containment pressure. However, the steam generators contain a secondary inventory at an enthalpy that is much higher than the primary side. Therefore, there is a significant amount of reverse heat transfer that occurs. Steam is produced in the core due to core decay heat. For a pump suction break, a two-phase fluid exits the core, flows through the hot legs, and becomes superheated as it passes through the steam generator. Once the broken loop cools, the break flow becomes two-phase. During the FROTH calculation, ECCS injection is addressed for both the injection phase and the recirculation phase. The FROTH code calculation stops when the secondary side equilibrates to the saturation temperature (T_{sat}) at the containment design pressure. After this point, the EPITOME code completes the steam generator depressurization. The methodology for the use of this model is described in *Reference 47*. (See subsections 6.2.1.2.2.4 and 6.2.1.2.2.5 for additional information.) The mass and energy release rates are calculated by FROTH and EPITOME until the time of containment depressurization. After containment depressurization (14.7 psia), the mass and energy release available to containment is generated directly from core boil-off/decay heat.

Table 6.2-11 presents the two-phase post-reflood mass and energy release data for the minimum safeguards pump suction double ended break case.

Post-Reflood Mass and Energy Release Data-Steam Generator Equilibrium and Depressurization

Steam generator equilibration and depressurization is the process by which secondary-side energy is removed from the steam generators in stages. The FROTH computer code calculates the heat removal from the secondary mass until the secondary temperature is the saturation temperature (T_{sat}) at the containment design pressure. After the FROTH calculations, the EPITOME code continues the FROTH calculation for steam generator cooldown removing steam generator secondary energy at different rates (i.e., first and second-stage rates). The first-stage rate is applied until the steam generator reaches T_{sat} at the user specified intermediate equilibration pressure, when the secondary pressure is assumed to reach the actual containment pressure. Then the second-stage rate is used until the final depressurization, when the secondary reaches the Reference temperature of T_{sat} at 14.7 psia, or 212°F. The heat removal of the broken loop and intact loop steam generators are calculated separately.

During the FROTH calculations, steam generator heat removal rates are calculated using the secondary side temperature, primary side temperature and a secondary side heat transfer coefficient determined using a modified McAdams correlation. Steam generator energy is removed during the FROTH transient until the secondary side temperature reaches saturation temperature at the containment design pressure. The constant heat removal rate used during the first heat removal stage is based on the final heat removal rate calculated by FROTH. The steam generator energy available to be released during the first stage interval is determined by

calculating the difference in secondary energy available at the containment design pressure and that at the (lower) user specified intermediate equilibration pressure, assuming saturated conditions. The intermediate equilibrium pressures are chosen as discussed in *Reference 47*, Section 2.3 and 3.3. This energy is then divided by the first stage energy removal rate, resulting in an intermediate equilibration time. At this time, the rate of energy release drops substantially to the second stage rate. The second stage rate is determined as the fraction of the difference in secondary energy available between the intermediate equilibration and final depressurization at 212°F, and the time difference from the time of the intermediate equilibration to the user-specified time of the final depressurization at 212°F. With current methodology, all of the secondary energy remaining after the intermediate equilibration is conservatively assumed to be released by imposing a mandatory cooldown and subsequent depressurization down to atmospheric pressure at 3600 seconds, i.e., 14.7 psia and 212°F (labeled as "Available Energy").

6.2.1.2.2.5 *Long-Term Mass and Energy Releases*

The long-term post-one-hour mass and energy releases (boil-off from core at the decay heating rate) are performed through user defined input functions in the GOTHIC code. This method of determining the long-term mass and energy releases is consistent with past application of Westinghouse methodology. See subsection 6.2.1.2.6 for discussion of long-term mass and energy calculations.

Sources of mass and Energy

The sources of mass considered in the LOCA mass and energy release are given in Tables 6.2-6 and 6.2-12. The sources are:

- The RCS water
- Accumulator water (two accumulators injecting)
- Pumped safety injection water

The energy inventories considered in the LOCA mass and energy release analysis are presented in Tables 6.2-7 and 6.2-13. The energy sources are listed below:

- Reactor coolant system water
- Accumulator water (two accumulators injecting)
- Pumped safety injection water
- Decay heat
- Core-stored energy
- Reactor coolant system metal (includes the reactor vessel and internals, hot and cold leg piping, steam generator inlet and outlet plenums, and steam generator tubes)
- Steam generator metal (includes transition cone, shell, wrapper, and other internals)
- Steam generator secondary energy (includes fluid mass and steam mass)

- Secondary transfer of energy (feedwater into and steam out of the steam generator secondary; feedwater coastdown due to closure of the flow control valve)

The analysis used the following energy reference points:

- Available energy: 212°F; 14.7 psia (energy available that could be released)
- Total energy content: 32°F; 14.7 psia (total internal energy of the RCS)

The mass and energy inventories are presented at the following times, as appropriate:

- Time zero (initial conditions)
- End of blowdown time
- End of refill time
- End of reflood time
- Time of broken loop steam generator equilibration to pressure setpoint
- Time of intact loop steam generator equilibration to pressure setpoint
- Time of full depressurization (3600 seconds)

The energy release from the zirc-water reaction is considered as part of the *Reference 47* methodology. Based on the way that the energy in the fuel is conservatively released to the vessel fluid, the fuel cladding temperature does not increase to the point where the zirc-water reaction is significant. This is in contrast to the 10CFR50.46 analyses, which are biased to calculate high fuel-rod-cladding temperatures and therefore a non-significant zirc-water reaction. For the LOCA M&E calculation, the energy created by the zirc-water reaction value is small and is not explicitly provided in the energy balance tables. The energy that is determined is part of the M&E releases, and is therefore already included in the LOCA M&E release.

The sequence of events for the LOCA transients are shown in Tables 6.2.1-20, 6.2.1-21, and 6.2.1-22 (for the DEHL and DEPS minimum, and maximum safeguards cases, respectively).

6.2.1.2.2.6 Long-Term LOCA Containment Response

The evaluation of the design basis LOCA event relative containment peak pressure and temperature response was completed to demonstrate the acceptability of the containment heat removal system to mitigate the consequences of a LOCA inside containment and to support the EPU program operation. This evaluation is documented in the subsections below.

The containment response analysis demonstrates the acceptability of the containment heat removal systems to mitigate the consequence of a large LOCA inside containment. The impact of LOCA M&E releases on the containment pressure and temperature are addressed to assure that the containment pressure and temperature remain below their respective design limits. The systems must also be capable of maintaining the Environmental Qualification (EQ) parameters to within acceptable limits at the EPU program conditions.

The Ginna LOCA containment response analysis considered a spectrum of cases as discussed in Section 6.2.1.2.2.1, Loss-of-Coolant (LOCA) Long-Term Mass and Energy Releases. The cases address break location, and postulated single failure. Only the limiting cases, which address the containment peak pressure case and limiting long-term EQ case, are presented herein.

Calculation of the containment response following a postulated LOCA was analyzed by use of the digital computer code GOTHIC. GOTHIC version 7.2 was used for the LOCA containment response analysis. The GOTHIC Technical Manual (*Reference 54*) provides a description of the governing equations, constitutive models, and solution methods in the solver. The GOTHIC Qualifications Report (*Reference 55*) provides a comparison of the solver results with both analytical solutions and experimental data.

The GOTHIC containment modeling for Ginna is consistent with the recent NRC approved Kewaunee evaluation model (*Reference 56*). The latest code version is used to take advantage of the diffusion layer model (DLM) heat transfer option. This heat transfer option was approved by the NRC (*Reference 57*) for use in Kewaunee containment analyses with the condition that mist be excluded from what was earlier termed as the mist diffusion layer model (MDLM). The GOTHIC containment modeling for GINNA has followed the conditions of acceptance placed on Kewaunee. Kewaunee and Ginna both have large dry containment designs with similar sized containment volumes and active heat removal capabilities. The differences in GOTHIC code versions are documented in Appendix A of the GOTHIC User Manual Release Notes (*Reference 57*). Version 7.2 is used consistent with the restrictions identified in *Reference 56*; none of the user-controlled enhancements added to version 7.2 were implemented in the Ginna containment model. A description of the Ginna GOTHIC model are provided in Section 6.2.1.2.2.6, Description of the LOCA GOTHIC Containment Model.

Accident Description

A break in the primary RCS piping causes a loss-of-coolant, which results in a rapid release of mass and energy to the containment atmosphere. Typically, the blowdown phase for the large LOCA events is over in less than 30 seconds. This large and rapid release of high-energy, two-phase fluid causes a rapid increase in the containment pressure, which results in the actuation of the containment recirculation fan coolers and containment spray systems.

The RCS accumulators begin to refill the lower plenum and downcomer of the reactor vessel with water after the end of blowdown. The reflood phase begins after the vessel fluid level reaches the bottom of the fuel. During this phase, the core is quenched with water from both the accumulators and pumped SI. The quenching process creates a large amount of steam and entrained water that is released to containment through the break. This two-phase mixture would have to pass through the steam generators and also absorb energy from the secondary side coolant if the break were located in the cold leg or pump suction piping.

The LOCA mass and energy release decreases with time as the system cools. Core decay heat is removed by nucleate boiling after the reflood phase is complete. The core fluid level is maintained by pumping water back into the vessel from either the SI or sump recirculation

system. The containment heat removal systems continue to condense steam and slowly reduce the containment pressure and temperature over time.

Input Parameters, Assumptions, and Acceptance Criteria

The major modeling input parameters and assumptions used in the Ginna containment evaluation model for the LOCA event are identified in this section. The assumed initial conditions and input assumptions associated with the fan coolers and containment sprays are listed in Table 6.2-16. The containment recirculation fan cooler (CRFC) heat removal capability data used is presented in Table 6.2-17. The primary function of the residual heat removal system (RHR) is to remove heat from the core by way of the emergency core cooling system (ECCS). The recirculation system alignment is outlined in Table 6.2-18. The containment structural heat sink input is provided in Table 6.2-19, and the corresponding material properties are listed in Table 6.2-20.

The LOCA containment analysis described herein utilized input and assumptions in support the Ginna EPU program, while addressing analytical conservatisms. The following summarized assumptions are areas where known differences exist between the SEP licensing analysis and the EPU Program containment integrity analysis.

1. All exposed concrete and carbon steel surfaces areas are conservatively assumed to have an overcoat and primer coatings (reduces heat transfer through heat sink).
2. To simulate the gap between insulation, steel and concrete, a thin air gap was modeled between these layers.
3. For the GOTHIC LOCA model, all of the sump heat sinks (3, 4, 5, 6 and 7) are considered to be submerged. The UFSAR model assumed the basement floor as submerged with the sump wall split between liquid and vapor. The conductors assumed submerged are essentially insulated from the vapor after the pools develop.
4. The containment initial pressure was assumed to be 15.7 psia for the EPU program.
5. Non-condensable accumulator gas addition is modeled in the EPU Program model.
6. Current Ginna calculations support a containment high-high pressure setpoint for containment spray pump initiation of 32.5 psig. The analysis value modeled was 33.5 psig to provide more margin for instrument uncertainty.
7. A sump recirculation model i.e., modeling coupled residual and component cooling heat exchangers, and service water piping was developed for the EPU Program.

The major assumptions made in the containment response analysis are listed below:

- The LOCA mass and energy release input to the containment model is described in Section 6.2.1.2.2.1, Loss-of-Coolant (LOCA) Mass and Energy Releases.
- Homogeneous mixing is assumed. The steam-air mixture and the water phases each have uniform properties. More specifically, thermal equilibrium between the air and the steam is assumed. However, this does not imply thermal equilibrium between the steam-air mixture and the water phase.

- Air is taken as an ideal gas, while compressed water and steam tables are employed for water and steam thermo-dynamic properties.
- For the blowdown portion of the LOCA analysis, the discharge flow separates into steam and water phases at the breakpoint. The saturated water phase is at the total containment pressure, while the steam phase is at the partial pressure of the steam in the containment. Steam and water releases are input separately for the post-blowdown portion of the LOCA analysis.
- The T_{sat} at the partial pressure of the steam is used for heat transfer to the heat sinks and the fan coolers.

Design Basis Accident

The Ginna LOCA containment response analysis considered a spectrum of cases as discussed in Section 6.2.1.2.2.1, Loss-of-Coolant (LOCA) Long-Term Mass and Energy Releases. The cases address break location, and postulated single failure (minimum and maximum safeguards). Only the limiting cases, which address the containment peak pressure case and limiting long-term EQ case, are presented herein. The LOCA pressure and temperature response analyses were performed assuming a loss of offsite power and a worst single failure (loss-of-one emergency diesel generator (EDG) that is, loss-of-one containment cooling train).

The limiting minimum safeguards case was based on a diesel train failure (loss of one cooling train) i.e., the active heat removal is:

- One containment spray pump, injection-phase only
- Two CRFCs
- One RHR pump and heat exchanger (HX)
- One component cooling water pump and two CCW HXs
- Two service water pumps

The calculation for the double-ended pump suction (DEPS) case was performed for a 30-day transient in support of EQ. The sequence of events for the containment peak pressure case, a double-ended hot leg break (DEHL) and the DEPS (EQ transient) is shown in Tables 6.2-21 and 6.2-22, respectively.

The Ginna GOTHIC containment evaluation model consists of a single-lumped parameter node; the DLM heat and mass transfer option is used.

Acceptance Criteria

The containment response for design basis LOCA containment integrity is an ANS Condition IV event, an infrequent fault. The relevant requirements to satisfy Nuclear Regulatory Commission acceptance criteria are as follows:

- GDC-16 and 50: In order to satisfy the requirement of GDC-16 and 50, the peak calculated containment pressure should be less than the containment design pressure of 60 psig.

- GDC-38: In order to satisfy the requirement of GDC-38, the calculated pressure at 24 hours should be less than 50% of the peak calculated value. (This is related to the criteria for containment leakage assumptions as affecting doses at 24 hours.)

Note that although Ginna is not a SRP plant, for completeness, the SRP long-term cooling criterion is also examined

>The containment design pressure for Ginna is 60 psig. The containment design temperature is 286°F.

6.2.1.2.2.7 *Description of the LOCA GOTHIC Containment Model*

Noding Structure

The Ginna GOTHIC containment evaluation model for the LOCA event consisted of one volume. Additional boundary conditions, volumes, flow paths, and components are used to model accumulator nitrogen release and sump recirculation. Injection of accumulator nitrogen during a LOCA event is modeled by a boundary condition. The recirculation system model uses GOTHIC component models for the RHR and CCW HXs and the CCW pumps. Recirculation flow from the sump is modeled using a boundary condition.

Volume Input

GOTHIC requires the volume, height, diameter, and elevation input values for each node. The containment is modeled as a single control volume in the containment model. The minimum free volume of 1,000,000 ft³ was used. The height, diameter, and elevation input values are not important for this single-volume containment model, so standard values of 100 feet, 10 feet, and 0 feet were used respectively.

A conservatively calculated pool surface area is used to model interfacial heat and mass transfer to liquid pools on the various floor surfaces inside containment. The conductors representing the floors are essentially insulated from the vapor after the pools develop; however, there can still be condensation or evaporation from the surface of the liquid pools. The pool area input value represents the sum of the three-floor conductor surface area (3, 4, 5, 6, and 7) from Table 6.2-19. Using this method to model the interfacial heat and mass transfer between the pools and the atmosphere was previously approved by the NRC for the Kewaunee containment design basis accident (DBA) and equipment qualification analyses (*Reference 54*).

The LOCA containment response model input values for the RHR and CCW system's volume, height, diameter, and elevation are not important for modeling the sump temperature response after recirculation. Values of 50 feet, 35.0 feet, 10.0 feet, and 0 feet, respectively were used; the model is not sensitive to these representative input values.

Initial Conditions

The containment initial conditions are listed below:

- Pressure: 15.7 psia
- Relative Humidity: 20%

- Temperature: 125°F

The LOCA containment response model contains volumes representing the RHR and the CCW system. The RHR system volumes were initially filled with water (125°F) at containment pressure (15.7 psia). The CCW system volumes in the LOCA containment response model were initially filled with water (85°F), but at a higher pressure of 60 psia. The CCW surge tank was modeled as a boundary node at a constant pressure.

Flow Paths

Flow paths connect the boundary conditions to the containment volume. The flow rate is specified by the boundary condition, so most of the flow path input is not important. Standard values are used for the area, hydraulic diameter, friction length, and inertia length of the flow path. Since this is a single volume model, the elevation of the break flow paths is arbitrarily set to 50 feet and the elevation of the spray flow paths is arbitrarily set to 90 feet.

Flow boundary conditions model the LOCA break flow to the containment. The boundary conditions are linked to functions that define the mass and energy of the break flow. The boundary conditions are connected to the containment control volume via flow paths.

The containment spray is also modeled as a boundary condition which is connected to the containment control volume via a flow path.

Heat Sinks

The heat sinks in the containment are modeled as GOTHIC thermal conductors. The heat sink data is based on conservatively low surface areas and is summarized in Table 6.2-19.

A thin air gap is assumed to exist between the steel and concrete for steel-jacketed heat sinks. A gap conductance of 10 Btu/hr/ft²/°F is assumed between steel and concrete. A gap of 20 Btu/hr/ft²/°F is assumed for the minimum gap conductance between layers of insulation and steel or concrete. The gap width is determined by dividing the gap thermal conductivity by the gap conductance.

The volumetric heat capacity and thermal conductivity for the heat sink materials is summarized in Table 6.2-20. The specific heat value was calculated based on the volumetric heat capacity.

Heat and Mass Transfer Correlations

GOTHIC has a number of heat transfer coefficient options that can be used for containment analyses.

The direct heat transfer coefficient set is used, along with the Diffusion Layer Model (DLM) mass transfer correlation, for all of the heat sinks inside containment. This heat transfer methodology was reviewed and approved for use in the Kewaunee containment DBA analyses (*Reference 56*). The DLM correlation does not require the user to specify a revaporization input value, as was done in previous analyses using the Uchida correlation.

The direct heat transfer coefficient set is used for the heat sinks representing floors. The submerged conductors are essentially insulated for the vapor after the pool develops. Insulated surfaces are modeled with a constant (0.0 Btu/hr-ft²/°F)

Modeling Sump Recirculation

The calculated containment peak pressure and temperature occur long before the transfer to sump recirculation. However, a sump recirculation model consisting of simplified RHR and CCW system models, was added to the Ginna containment model to calculate the long-term LOCA containment pressure and temperature response.

The recirculation system is actuated after a low RWST level signal. The RHR heat exchanger cools the water from the containment sump. The RHR system injects the cooled water into the RCS to cool the core. The RHR heat exchanger is cooled with CCW water and service water provides the ultimate heat sink, cooling the CCW heat exchangers.

Boundary Conditions

LOCA Mass and Energy Release

The LOCA mass and energy release methodology generates the releases from both sides of the break (or two flow paths: mass and energy exiting from the vessel side of the break; and mass and energy exiting from the steam generator side of the break). The LOCA transient M&E releases are calculated as separate flow paths (for the first 3,600 seconds) and input to the GOTHIC containment model via boundary conditions. The break mass and enthalpy are input to the containment model through forcing functions on flow boundary conditions. The M&E releases from the boundary conditions are analyzed for Ginna out to 3,600 seconds i.e., time at which all energy in the primary heat structures and steam generator secondary system is assumed to be released/depressurized to atmospheric pressure, (i.e., 14.7 psia and 212°F). Section 6.2.1.2.2.1, Loss-of-Coolant (LOCA) Long-Term Mass and Energy Releases, describes the LOCA long-term M&E release methodology (*Reference 47*). The boundary conditions are linked to functions that define the mass break flow and the enthalpy of the break flow.

The SG secondary metal mass above the water level of the secondary side is assumed to transfer to containment over a 24-hour period. This change was made as part of the containment temperature increase from 120°F to 125°F (*Reference 60*).

The liquid portion of the break flow is release as drops with an assumed diameter of 100 microns (0.00394 in). This is consistent with the methodology approved for Kewaunee (*Reference 56*) and is based on data presented in *Reference 58*.

The long-term post-one-hour mass and release (boil-off from the core at the decay heating rate) calculations are performed through user defined functions by GOTHIC. These input functions are used to incorporate the sump water cooling in the long term and are consistent with the Westinghouse methodology previously approved by the NRC. After primary system and secondary system energy have been released (depressurized to atmospheric pressure, (i.e., 14.7 psia and 212°F), the M&E release to the containment is assumed to be from long-term steaming of decay heat. A flow boundary condition is defined to provide the long-term

boil-off M&E release to containment. The mass flow rate and enthalpy of the flow is calculated using GOTHIC control variables.

The long-term boil-off calculation used the American Nuclear Society (ANS) Standard 5.1 decay heat model (+2 σ uncertainty) for the determination of long-term boil-off from core (*Reference 50*). Table 6.2-4 lists the decay heat curve used. This assumption is consistent with the long-term M&E methodology documented in *Reference 47*.

Containment Recirculation Fan Coolers

The containment fan coolers are modeled with a cooler component. There are two trains of containment safeguards available, with two fan coolers per train. Consistent with the application of single-failure criterion presented in Section 6.2.1.2.2.2. The inherent assumption is that offsite power is lost with the pipe rupture. This results in the actuation of the EDGs, powering the two trains of safeguards equipment. Operation of the EDG delays the operation of the safeguards equipment that is required to mitigate the transient.

Two cases have been analyzed to assess the effects of a single failure. The first case assumes minimum safeguards based on the postulated single failure of an EDG. This assumption results in the loss-of-one train of safeguards equipment. Thus, the remaining equipment is conservatively modeled as: two CRFCs and one containment spray pump. The other case assumes maximum safeguards, which assumes both EDGs are available. With the maximum safeguards case the limiting single failure assumption postulated is the failure of one containment spray pump. The analysis of the cases described provides confidence that the effect of credible single failures is bounded.

The fan coolers in the containment evaluation model are modeled to actuate on the containment high pressure setpoint with a biased high uncertainty, (6 psig), and begin removing heat from containment after a specified 44-second delay. The heat removal rate per containment fan cooler is given as a function of containment steam saturation temperature and is presented in Table 6.2-17. The heat removal rate is read into a GOTHIC function and a multiplier, based on the number of fan coolers running is used to calculate the heat removal rate from containment.

Containment Spray System

The containment spray is modeled with a boundary condition. As previously identified in the fan cooler modeling discussion, Ginna has two trains of containment safeguards available, with one spray pump per train. Consistent with the application of single-failure criterion presented in Section 6.2.1.2.2.2. The inherent assumption is that offsite power is lost with the pipe rupture. This results in the actuation of the EDG, powering the two trains of safeguards equipment. Operation of the EDG delays the operation of the safeguards equipment that is required to mitigate the transient.

Relative to single failure criterion with respect to a LOCA event, one spray pump is considered inoperable, whether due to an EDG failure (minimum safeguards case) or as the limiting single failure in the maximum safeguards case.

The containment spray is modeled to actuate on the containment high-high pressure setpoint with a biased high uncertainty (33.5 psig) and to begin injecting 104°F water after a specified 28.5 second delay. The containment spray flow versus containment pressure is listed in Table 6.2-16 per spray pump in the injection

phase. The spray flowrate is modeled in GOTHIC as a function of time and containment pressure. The containment spray is credited only during the injection phase of the transient and is terminated during the transition to sump recirculation (i.e., at 2652 seconds).

Accumulator Nitrogen Gas Modeling

The accumulator nitrogen gas release is modeled with a flow boundary condition in the LOCA containment model. The nitrogen release rate was conservatively calculated by maximizing the mass available to be injected. The nitrogen gas release rate was used as input for the GOTHIC function, as a specified rate over a fixed time period. Nitrogen gas was released at a rate of 212.3 lbm/seconds, beginning at 44.17 seconds (average accumulator tank water volume empty time) and ending at 64.17 seconds.

6.2.1.2.2.8 *Results*

The containment pressure, steam temperature, and water (sump) temperature profiles for the DEHL case (peak pressure case) are shown in Figures 6.2-1 through 6.2-3. Table 6.2-21 provides the transient sequence of events for the DEHL transient. The results of the DEPS (long-term EQ transient) are shown in Figures 6.2-4 through 6.2-6. Table 6.2-22 presents sequence of events for the DEPS transient. Table 6.2-23 provides the containment pressure and temperature results relative to peak containment conditions and also at 24 hours for EQ support and the acceptance limits for these parameters.

A review of the results presented in Table 6.2-23, shows that the analysis margin (analysis margin is the difference between the calculated peak pressure and temperature and the acceptance limits) is maintained. From the containment response analysis, performed in support of the Ginna EPU program, the containment peak pressure and temperature is 55.42 psig and 285.0°F.

As indicated in Table 6.2-23, the peak temperature @ 24 hours of 161.0°F exceeds the acceptance limit of 152°F for a short time. The impact of the 24-hour peak temperature exceeding the acceptance criteria on equipment qualification is evaluated in Section 6.1.2.1.2.

LOCA Containment Response Transient Description Double Ended Pump Suction Break with Minimum Safeguards

This analysis assumes a loss of offsite power coincidence with a double-ended rupture of the RCS piping between the steam generator outlet and the RCS pump inlet (suction). The associated single-failure assumption is the failure of a diesel generator to start resulting in one train of ECCS and containment safeguards equipment being available. The containment heat removal systems that are assumed available are one RHR heat exchanger, two CCW heat exchangers, one containment spray pump (injection phase), and two containment recirculation fan coolers. Further loss of offsite power delays the actuation times of the safeguards equipment due to the time required for diesel startup after receipt of the safety injection signal.

The postulated RCS break results in a rapid release of mass and energy to the containment with a resultant rapid increase in both the containment pressure and temperature. This rapid rise in containment pressure actuates the containment HIGH pressure signal at 0.39 seconds

and a containment HIGH-HIGH pressure signal at 4.21 seconds. The containment pressure continues to rise rapidly in response to the release of mass and energy, reaching the blow-down peak pressure of 50.57 psig at 13.01 seconds. The end of blowdown marks a time when the initial inventory in the RCS has been exhausted and a slow process of filling the RCS downcomer in preparation for reflood has begun. Since the mass and energy release during this period is low, pressure continues to decrease slightly. At approximately 44.98 seconds the accumulators have emptied, and the pressure increases in response to the loss of steam condensation in the RCS loops and the introduction of the accumulator nitrogen gas to containment out to a second peak which occurred at 70.06 seconds.

During this period the containment spray (32.79 seconds) and containment recirculation fan coolers (44.4 seconds) have also started and are removing heat. Reflood continues at a reduced flooding rate due to the buildup of mass in the RCS core, which offsets the downcomer head. This reduction in flooding rate and the continued action of the containment recirculation fan coolers and the containment injection spray leads to a slowly decreasing containment pressure out to the end of reflood, which occurs at 226.08 seconds.

At this juncture, by design of (*Reference 47*) mass and energy release evaluation model, energy removal from the steam generator secondary side begins at a very high rate, resulting in a rise in containment pressure from 226.08 seconds out to 1,220 seconds when the ultimate peak pressure of 55.42 psig is reached. Energy continues to be removed from the secondary side of the broken loop and intact loop steam generators until the secondary temperature is the saturation temperature (T_{sat}) at the containment design pressure. This point is reached at 1,131.10 and 1,228.63 seconds for the broken loop and intact loop steam generators, respectively. Energy removal from the secondary side of the steam generators continues by way of intermediate pressure equilibration stages until the final depressurization, when the secondary reaches the mandatory reference temperature of T_{sat} at 14.7 psia, and 212°F, at 3,600 seconds. The heat removal of the broken loop and intact loop steam generators are calculated separately. The intermediate equilibration stages are met at 1,226.51 seconds for the broken loop steam generator and 1,331.55 seconds for the intact loop steam generator. After the peak containment pressure is reached and during the steam generator depressurization period, the mass and energy release is reduced since the large energy removal has been accomplished. Containment pressure slowly decreases until the initiation of sump recirculation at 2,652 seconds. At this time, the emergency core cooling system (ECCS) is realigned for sump recirculation resulting in an increase in safety injection temperature (due to the delivery from the hot sump and a reduction in steam condensation). Also at 2,652 seconds the containment injection phase spray is terminated from the refueling water storage tank. Without crediting recirculation spray, the containment pressure and temperature will begin to increase out to approximately 3,600 seconds. At this time, the energy removal from the two operating containment recirculation fan coolers exceeds the energy release and the pressure and temperature turnaround. This trend continues to the end of the transient at 2.592E+6 seconds.

The LOCA containment response analysis has been performed as part of the EPU program for Ginna. As illustrated in the Section 6.2.1.2.2.8, Results, all cases were well below the

containment acceptance limits of 60 psig and 286°F. In addition, the long-term DEPS case was well below 50% of the peak containment pressure value within 24 hours. Based on the results, all applicable SRP criteria have been met.

6.2.1.2.3 Secondary System Pipe Break Analysis

The limiting steam line break and containment response is documented in *Reference 44*. Several combinations of initial power level, single failures, and availability of offsite power were analyzed to determine the limiting combination of initial conditions and limiting steam line break is for a 1.4 sq. ft. break from 70% power with offsite power available and a failure of a vital bus.

The analysis of a steam line break inside containment is performed to demonstrate that peak containment pressure is less than the design pressure of 60.0 psig. The analysis performed here shows that the design pressure is not violated for any steam line break inside containment.

6.2.1.2.3.1 Event Analysis

A comprehensive set of break sizes, initial power levels, single failure assumptions, and offsite power availability was considered so that there is reasonable assurance that the limiting cases have been covered. Historically, 130 cases were evaluated (*Ref. 3, 21, and 42*).

The single failures considered in the analysis have been selected based upon their potential for increasing the amount of mass and energy released into containment and for reducing the amount of heat removed from containment (containment recirculation fan coolers (CRFCs) and containment spray). Postulated failures were:

- Feedwater control system failure
- Auxiliary feedwater throttle valve failure
- Vital bus failure
- One diesel generator failure to start

6.2.1.2.3.2 Protective Features

The primary design features which provide protection for steam ruptures are:

- A. Safety injection system actuation from any of the following:
 1. Two-out-of-three pressurizer low-pressure signals.
 2. Two-out-of-three low-pressure signals in any steam line.
 3. Two-out-of-three high containment pressure signals.
- B. If the reactor trip breakers are closed, reactor trip may be actuated from a power range neutron flux high trip, overpower delta T signal, or upon actuation of the safety injection system.

Redundant isolation of the main feedwater lines. Following a safety injection signal all main feedwater regulating valves, MFRV valves, main feedwater bypass valves and main feedwater isolation valves (MFIVs) are rapidly closed, and the main feedwater (MFW)

pumps tripped. Opening of the main feedwater (MFW) pump breakers also generates a signal to close the main feedwater pump discharge valves (MFPDV).

- C. Trip of the fast-acting, main steam isolation valves (MSIVs) which are designed to close in less than 5 seconds with no flow on the following signals:
1. Two-out-of-three high-high containment pressure signals.
 2. One out of the two high-high steam flow signals in a steam line in coincidence with any safety injection signal.
 3. One out of the two high steam flow signals in a steam line in coincidence with two out of four indications of low reactor coolant average temperature and any safety injection signal.

Each steam line has a fast-closing main steam isolation valve (MSIV) and a non-return check valve. These four valves prevent blowdown of more than one steam generator for any break location.

The protective functions credited in the limiting case steam line break are SI caused by low steamline pressure and reactor trip caused by SI actuation.

6.2.1.2.3.3 *Single Failures Assumed*

The worst single failure for the limiting steam line break (1.4 sq. ft. break, 70% power, no loss of offsite power) that pressurizes containment is the failure of a vital bus that powers one safeguard train. The active heat removal from the containment is reduced by the loss of 1 train of fan coolers and one containment spray pump. One train of safety injection is also lost.

6.2.1.2.3.4 *Operator Actions Assumed*

The operator action assumed in the analysis is that the operator isolates auxiliary feedwater (AFW) to the faulted steam generator within 10 minutes from the initiation of the break.

6.2.1.2.3.5 *Chronological Description of Event*

The event starts with a steam line break inside containment. For the limiting steam line break, containment pressure rapidly increases. The non-return check valve on the faulted steam generator closes preventing blowdown of the intact steam generator. The low steam-line pressure setpoint is reached starting safety injection (SI) and causing a reactor trip. The main feedwater (MFW) pumps trip from the safety injection (SI) signal and start to coast down. The main feedwater regulating valve (MFRV) on the faulted steam generator closes. The containment recirculation fan coolers (CRFCs) start to remove heat from containment and the auxiliary feedwater (AFW) pumps start. The reduction in steam generator pressure causes the feedline to flash adding additional mass and energy into containment. The high-high containment pressure setpoint is reached causing containment spray to start. At ten minutes, auxiliary feedwater (AFW) is isolated.

6.2.1.2.3.6 *Impact on Fission Product Barriers*

This analysis demonstrates that a steam line break inside containment does not exceed the containment design limits. Therefore, the containment remains available as a fission product barrier for breaks inside containment.

6.2.1.2.3.7 *Reactor Core and Plant System Evaluation*

The analysis for a steam line break inside containment is performed in two parts. The first part of the analysis involves the calculation of mass and energy releases. The second part of the analysis uses the mass and energy releases to calculate the containment pressure response.

The RETRAN code is used to calculate the mass and energy releases. Any assumption that controls the amount of mass which is contained in or added to the faulted steam generator during the transient is an important assumption. This includes the initial steam generator mass assumption and the main and auxiliary feedwater flow rate assumption. The initial steam generator masses are modeled conservatively high for the faulted and intact SG. Since initial steam generator was in the intact SG has essentially no impact on peak containment pressure.

Main feedwater (MFW) flow rates are modeled as a function of steam generator pressure as well as a function of the main feedwater regulating valve (MFRV) position. Each case is initiated with the MFRVs in both loops at the nominal position which corresponds to the initial power level. As soon as the break occurs, a mismatch will be present between steam flow and feed flow. This may cause the MFRV to open. In the analyses, the faulted loop MFRV is assumed to ramp to full open in two seconds and remain full open. The intact loop MFRV remains in its nominal position until reactor trip. At the time of reactor trip, the intact loop MFRV is assumed to step closed. The flow rates are input as a function of pressure within the constraints of the valve positions discussed above.

The auxiliary feedwater (AFW) flow rates assumed for each of the cases in which offsite power is assumed to be available are conservatively high flow rates assuming that both of the motor driven auxiliary feedwater (MDAFW) pumps are operating at full capacity for all cases initiated at less than 50% power. The assumed flowrate is 235 gpm to each SG. The faulted loop flow rates assumed for the cases without offsite power available and for cases initiated above 50% power with offsite power available include 630 gpm flow from the turbine driven auxiliary feedwater (TDAFW) pump in addition to both motor driven auxiliary feedwater (MDAFW) pumps. With initial power above 50%, previous experience has shown that low-low level will be reached in both steam generators following a reactor trip, due to shrink and swell phenomenon. Auxiliary feedwater (AFW) is assumed to be initiated 25 seconds after the SI setpoint is reached (the minimum time for the system to start) and is assumed to be terminated via operator action at 10 minutes.

The moderator density coefficients assumed in the calculations are conservative end-of-life coefficients. The assumed shutdown margin is 1.3% Δ K/K.

The containment recirculation fan coolers (CRFCs) are actuated on the high containment pressure signal, while the containment spray pumps are actuated on the high-high

containment pressure signal. The high containment pressure set point is 4 psig, but a value of 6 psig is applied in the model in consideration of instrument uncertainty. A 34 second activation delay is applied for the case with offsite power available, while a 44 second delay is applied for the loss of offsite power condition. The heat removal is based on 85°F lake water temperature (highest service water (SW) value). Values assumed in the analysis are listed on Table 6.2-25.

The high-high containment pressure setpoint is 28 psig. A value of 33.5 psig is applied in the model in consideration of instrument uncertainty. Spray activation delay is based on the number of pumps operating and offsite power condition availability.

At the initiation of the accident, the containment atmosphere is assumed to be at a temperature of 125°F, with a containment pressure of 15.7 psia, and a relative humidity of 20%. The relative humidity is conservative for containment pressurization purposes (versus the nominal 50% relative humidity shown in Table 3.11-1). The temperature of the refueling water storage tank (RWST), and consequently the containment spray water, has been assumed to be 104°F. The net free volume applied in the containment integrity analysis was 1,000,000 ft³.

6.2.1.2.3.8 *Input Parameters and Initial Conditions*

The major parameters associated with the mass and energy release calculations are illustrated on Table 6.2-24.

The major parameters associated with the containment pressure response calculation are illustrated on Tables 6.2-19, 6.2-20 and 6.2-25.

6.2.1.2.3.9 *Methodology*

RETRAN is used to calculate the time dependent mass and energy out the break. The time dependent mass and energy is input into GOTHIC to calculate the time dependent containment pressure.

6.2.1.2.3.10 *Acceptance Criteria*

The containment pressure should not exceed the design pressure of 60 psig.

6.2.1.2.3.11 *Results*

Figure 6.2-7 illustrates containment pressure for the limiting case. The containment temperature transient for this case is illustrated in Figure 6.2-8. The sequence of events is listed on Table 6.2-26.

The limiting case results in a peak containment pressure of 59.6 psig.

6.2.1.2.3.12 *Radiological Consequences*

The steam line break discussed in this section is associated with peak containment pressure. As such, the steam from the break is contained within containment. Since no steam is released to the environment, doses are negligible and bound by a steam line break outside containment. See Section 15.1.5.5.

6.2.1.2.3.13 Conclusion

The peak containment pressure for the limiting steam line break inside containment is less than containment design pressure.

6.2.1.3 Evaluation of Containment Internal Structures

6.2.1.3.1 Introduction

The containment internal structures, such as the reactor coolant loop compartments and the reactor shield wall, are designed for the pressure buildup that could occur following a LOCA. If a LOCA were to occur in these relatively small volumes, the pressure would build up at a rate faster than the overall containment, thus imposing a differential pressure across the walls of the compartments.

6.2.1.3.2 Reactor Coolant Loop Compartment Pressure

A digital computer code, COMCO, was developed to analyze the pressure buildup in the reactor coolant loop compartments. The COMCO code is largely an extension of the COCO code in that a separation of the two-phase blowdown into steam and water is calculated and the pressure buildup of the steam-air mixture in the compartment is determined. Each compartment has a vent opening to the free volume of containment and this is included in the calculation. The calculations were made assuming that the reactor coolant loop compartment access blocks were in place. Two of the three blocks for each compartment are normally not in place, which increases the vent area to containment above that included in the calculations.

The main calculation performed is a mass and energy balance within the control volume of a compartment. The pressure builds up in the compartment until a mass and energy relief through the vent exceeds the mass and energy entering the compartment from the break. The reactor coolant loop compartments are designed for the maximum calculated differential pressures resulting from an instantaneous double-ended rupture of the reactor coolant pipe. The following results were calculated for the two reactor coolant loop compartments:

	Volume (ft ³)	Vent Area (ft ²)	Pressure (psi)
Compartment A	28,800	608	12.1
Compartment B	47,700	553	14.0

6.2.1.3.3 Thermal Gradients

Thermal gradients through the internal structures have been considered in the design of these structures and found to be negligible. No significant temperature differential exists under operating conditions. The maximum thermal gradient that can exist within the containment is approximately 30°F, since normal operating temperature is approximately 120°F and concrete structural components are maintained below 150°F. These gradients are well within the scope of ACI-318-63 design requirements. Under accident conditions, the temperature gradient occurring does not penetrate into the structure sufficiently to affect stress distributions in the structure during the approximate 1-sec duration of the unbalanced pressure differential.

6.2.1.3.4 Reactor Vessel and Steam Generator Annulus Pressure

Concrete around and under the reactor has been designed to limit stress in the concrete reinforcement to below its ultimate strength value under the pressure differential resulting from a postulated instantaneous double-ended rupture of a reactor coolant pipe (i.e., LOCA) in the concrete annulus surrounding the reactor vessel. Calculations for this limiting case show the maximum pressure in the annulus surrounding the reactor vessel is 175 psi and for the concrete under the reactor is 38 psi.

Concrete structures around the steam generators are also designed for the maximum calculated differential pressures resulting from an instantaneous double-ended rupture. Stresses in the reinforced concrete under these pressure differentials are limited to those values presented in Part IV-B of the ACI-318-63 Code as reduced by suitable capacity reduction factors as defined in the ACI-318 Code.

Pressure differential values given are based on the following assumptions:

- A. A longitudinal split of area equivalent to the cross-sectional area of a reactor coolant pipe, i.e., 4.5 ft². A circumferential failure at the reactor vessel nozzle would result in a much smaller flow discharge area because the vessel and pipe arrangement is such that no significant relative movement can take place.
- B. An initial flow from the break of 98,000 lb/sec.
- C. The buildup of pressure in the annulus around the failed pipe causes the reactor vessel nozzle inspection plug to blow free. This plug does not present any problem from a missile generation standpoint because it is made of sand bags contained in a steel wire mesh container.
- D. A total flow area out to the pipe annulus of approximately 14.8 ft².
- E. The flow entering the cavity is 37% of the initial flow at the break. The reason for this is that the discharge area from the pipe annulus into the cavity is 37% of the total flow area out of the pipe annulus.
- F. The cavity outflow area during the initial blowdown period is approximately 42.9 ft².

In addition, analyses related to subcompartment pressure loading were performed within the scope of the resolution of Unresolved Safety Issue A-2, Asymmetric LOCA loads. The results of those analyses are documented in *Reference 4*.

6.2.1.3.5 Seismic Evaluation

The structural integrity of the containment internal structures under seismic condition and loadings is discussed in Section 3.8.3. It is concluded in Section 3.8.3 that the estimated seismic stresses of interior structures, including concrete shield walls, steel and concrete columns, and crane support structures are low and therefore the structures are capable of withstanding a safe shutdown earthquake.

6.2.1.3.6 Technical Evaluation for Extended Power Uprate (EPU) Conditions

An evaluation was conducted to determine the effect of the Ginna EPU program on the short-term LOCA-related M&E releases that support the analysis of subcompartments discussed in the Ginna Section 6.2.1.3.

6.2.1.3.6.1 Introduction

The containment internal structures are designed for a pressure buildup that could occur following a postulated LOCA. If a LOCA were to occur in these relatively small volumes, the pressure would build up at a faster rate than the overall containment, thus imposing a differential pressure across the walls of the compartments.

Short-term LOCA M&E release calculations are performed to support analysis of the RCL compartments, the concrete around and under the reactor vessel, and the concrete structures around the steam generator. The breaks assumed in the current licensing basis for these structures are large primary loop pipe breaks (a longitudinal split of area equivalent to the cross-sectional area of a reactor coolant pipe, i.e., 4.5 ft²). These analyses are performed to ensure that the walls in the immediate proximity of the break location can maintain their structural integrity during the short-pressure pulse (generally less than 3 sec) that accompanies a LOCA within the region.

Ginna has been approved for LBB methods as part of the NRC staff's resolution of unresolved safety issue A-2 (*Reference 59*). With the elimination of the large RCS breaks, the only break locations that need to be considered are the largest branch lines off of the primary loop piping. These branch lines are the pressurizer surge line, the pressurizer spray line, and the accumulator lines. The releases from these smaller breaks are considerably lower than would result from the large RCS breaks.

6.2.1.3.6.2 Input Parameters and Assumptions

The short-term LOCA M&E release analysis is sensitive to the assumed characteristics of various plant systems, in addition to other key modeling assumptions. Where appropriate, bounding inputs are utilized and instrumentation uncertainties are included. For example, the RCS operating temperatures were chosen to bound the temperature range of all operating cases, and a temperature uncertainty allowance (-4°F) was then included. Nominal parameters are used in certain instances. For example, the RCS pressure in this analysis is based on

a nominal value of 2250 psia plus an uncertainty allowance (+60 psi). All input parameters are chosen consistent with accepted analysis methodology. The blowdown M&E release rates are affected by the initial RCS temperature conditions. Since short-term releases are linked directly to the critical mass flux, which increases with increasing pressures and decreasing temperatures, the short-term LOCA releases are expected to increase due to changes associated with the current RCS conditions.

Increased power has no impact on the short-term releases because of the duration of the event (i.e., ~1.0 sec). Only changes in the initial RCS pressure and temperature conditions would affect the results.

For the M&E releases, the core-stored energy and flow behavior through the core have the potential of changing as a result of a fuel change. However, any changes to the flow characteristics past the fuel are assumed small, and as such, would have an insignificant impact on the short-term LOCA M&E releases. Any possible change in the core-stored energy does not adversely affect the normal plant operating parameters, system actuations, accident mitigating capabilities or assumptions important to the short-term LOCA M&E releases. This change does not create conditions more limiting than those assumed in the analyses. Any change in core-stored energy would have no effect on the releases because of the short duration of the postulated accident.

Therefore, the only effect that needs to be addressed is the change in RCS coolant temperatures and RCS coolant pressure.

In summary, the following assumptions were employed to ensure that the M&E releases were conservatively calculated, thereby maximizing mass release to containment subcompartments:

- RCS vessel outlet temperature goes from 609.8°F to 600.3°F
- RCS vessel/core inlet temperature goes from 552.5°F to 528.3°F
- Allowance for RCS temperature uncertainty is $\pm 4.0^\circ\text{F}$
- Allowance for RCS pressure uncertainty is ± 60 psi

6.2.1.3.6.3 *Acceptance Criteria*

Although Ginna is not a Standard Review Plan (SRP) plant, for completeness the SRP criterion is also examined. A LBLOCA is classified as an ANS Condition IV event -- an infrequent fault. To satisfy the NRC acceptance criteria presented in the SRP Section 6.2.1.3, the relevant requirements are as follows:

- The NRCs NUREG-0800, Section 6.2.1.3, "M&E Release Analysis for Postulated Loss-of-Coolant Accidents," subsection II, Part 3a, provides guidance on NRC's expectations for what must be included in a LOCA M&E release calculation, if that calculation is to be acceptable. The Westinghouse M&E models described in WCAP-8264-P-A, Rev. 1 (*Reference 48*), have been found by the NRC to satisfy those expectations.

6.2.1.3.6.4 *Description of Analysis*

Short-term releases are linked directly to the critical mass flux, which increases with increasing pressures and decreasing temperatures. The short-term LOCA releases are expected to increase due to changes associated with the current RCS conditions. Short-term blowdown transients are characterized by a peak M&E release rate that occurs during a subcooled condition; thus, the Zaloudek correlation, which models this condition, is currently used in the short-term LOCA M&E release analyses (*Reference 48*). This correlation was used to conservatively evaluate the impact of the deviations in the RCS inlet and outlet temperature for the EPU program. Therefore, using lower temperatures maximizes the short-term LOCA M&E releases.

As previously stated, Ginna has been approved for LBB methods as part of the NRC staff's resolution of Unresolved Safety Issue A-2 (*Reference 59*). With the elimination of the large RCS breaks, the only break locations that need to be considered are the largest branch lines off of the primary loop piping. These branch lines include the pressurizer surge line, the pressurizer spray line, the accumulator line, and the RHR line from the hot leg to the first isolation valve. The releases associated with these smaller breaks would be considerably lower than the large RCS breaks.

The reduction in break area comparing a double-ended break in the largest branch line to the current single-ended split in the RCS, the ratio is 4.1. A reduction of this magnitude in pipe break size has been shown to have a significant beneficial impact on the subcompartment loadings. For example, based upon available sensitivities, it is estimated that the peak break compartment pressure was shown to be reduced by a factor of 2.76, and the peak differential across an adjacent wall was reduced by a factor of 3.86.

Using the EPU pressure and temperature (P/T) conditions (i.e., increased pressure and decreased temperature), the lower temperature increases the short-term LOCA M&E release. For conservatism, uncertainties were applied only to the EPU data to maximize the possible increase. Using the critical mass flux equation, it was determined that the release could increase by 9.7%. Therefore, from a very conservative perspective, the current design basis releases could increase by a factor of 1.10 due to RCS initial condition (P/T) effects.

6.2.1.3.6.5 *Short-Term LOCA M&E Releases Results*

In summary, the effect of eliminating the large RCS breaks, and instead considering the branch nozzles, is a factor of 4 (400%) reduction in the break area, whereas the penalty associated with the uprate is a release increase of only a factor of 1.10. So, the benefit of LBB more than offsets any penalty to the current licensing basis short-term mass and energy and subcompartment analysis associated with uprating.

The current licensing basis LOCA M&E releases used for the subcompartment results for the RCL compartments, the reactor vessel compartment, and the steam generator compartment remain bounding for the short-term subcompartment analysis by virtue of applying LBB methods.

6.2.1.4 Minimum Operating Conditions

Containment operability is defined in the Technical Specifications. The minimum operating conditions related to containment are specified in Section 3.6 of the Technical Specifications for MODES 1, 2, 3, and 4, and Section 3.9.3 for MODE 6 (Refueling).

6.2.1.5 Instrumentation Requirements

Instrumentation is provided to actuate the engineered safety features to ensure the containment design limit is not exceeded, as discussed in Section 7.3, and to monitor the containment pressure, temperature, sump level, radiation level, and hydrogen concentration (See Section 6.2.5).

6.2.1.5.1 Pressure

Containment pressure is a variable required for both normal operating conditions and post-accident monitoring. Seven transmitters are installed outside the containment with sensing lines which monitor containment pressure. The pressure is indicated for all seven channels on the main control board and inputs are provided for six of the channels to the plant process computer system (PPCS). Three pressure transmitters have a calibrated range of 0 to 60 psig (PT-945, 947, 949) and have operability requirements per Technical Specification LCO 3.3.2. Three have a calibrated range of -5 to +185 psig (PT-946, 948, 950) and have operability requirements per Technical Specification LCO 3.3.2 and LCO 3.3.3. PT-946 and 948 are designed in accordance with Regulatory Guide 1.97 and are dedicated to continuously record (on the PPCS) the containment pressure as required by NUREG 0737, Item II.F.1.4. The seventh channel is a narrow-range transmitter with a calibrated range of -3 to +3 psig (PT-944) that is used to ensure compliance with Technical Specification LCO 3.6.4. PT-950 is indicated on the main control board but provides no input to PPCS.

6.2.1.5.2 Sump Level

Containment sump A level transmitters (LT-2039 and LT-2044) are provided to monitor sump level during MODES 1, 2, 3, and 4. These redundant level transmitters indicate water level inside the containment from the bottom of sump A (elevation 205 ft) to the basement floor (elevation 235 ft) with an accuracy of $\pm 5\%$. This provides a narrow-range containment water level indication which will provide useful information only for leaks or small fluid line breaks. It will not provide useful information for most loss-of-coolant accident (LOCA) break sizes, but is used to ensure that assumptions with respect to resolution of Unresolved Safety Issue 2, "Asymmetric LOCA Loads", are being met. Operability of these level transmitters is addressed by Technical Specifications.

The wide-range containment water level monitor is located in sump B and is designed to monitor water depths corresponding to at least 500,000 gallons. Redundant safety-related containment sump B level indicators (LI-942 and LI-943) show that water has been delivered to the containment following an accident, when to begin the recirculation procedure, and whether to terminate safety injection, if still in progress. The wide-range water level monitoring system is composed of a series of level switches rather than a continuous level monitor.

The switches are located at specific levels of sump B determined to be of importance for operator action and equipment protection. The level switches are located at 8 in., 78 in., 113

in., 180 in., and 214 in. from the bottom of the sump. The monitor system includes a level indicator light display in the control room and is recorded by the plant process computer system (PPCS). The indicator lights are labeled in inches of water and each light is actuated by one level switch. At the moment an indicator light is illuminated the containment water level is plus or minus approximately 1/2 in. of the indicated level. The PPCS records level switch actuation only; therefore, at the moment a level switch is actuated, the containment water level is plus or minus approximately 1/2 in. of the recorded level. The level switches have been qualified in accordance with IEEE 323-1974 and IEEE 344-1975. The two functionally independent channels are physically separated, seismically supported, and fed from separate Class 1E power supplies as required by NUREG 0737. Operability requirements of these level switches are addressed by Technical Specifications.

6.2.1.5.3 Radiation

Two Victoreen Model 875 high-range containment area radiation monitors, R-29 and R-30, physically separated, are provided according to the requirements of NUREG 0737, Item II.F.1.2.C. The monitors have a maximum range of 10^7 rem/hr. The monitors are designed and qualified to function in an accident environment. The system is qualified to IEEE 323 and Regulatory Guides 1.97 and 1.89. Operability requirements of these radiation monitors are addressed by Technical Specifications.

6.2.1.5.4 Containment Temperature and Dewpoint

The temperature and dewpoint instrumentation monitors containment atmosphere conditions during normal and shutdown operations and can be used during containment leak rate testing. The temperature instrumentation is also qualified to function during post-accident conditions. The instrumentation receives inputs from six temperature sensors (TE-6031, 6035, 6036, 6037, 6038, and 6045) and six dewpoint sensors located inside containment. The computer multiplexer (MUX) instrumentation converts each temperature input to allow the information to be displayed on the plant process computer system (PPCS) and the safety parameter display system (SPDS). The temperature display range is 0-300°F with an accuracy of ± 0.9 . The dewpoint display range is 0-170°F with an accuracy of ± 2.0 °F. The instrumentation is significant to safety but not required nor designed to function during or after a safe shutdown earthquake. The temperature sensors serve as partial backup for containment fire detection instruments, and are used to verify compliance with a Technical Specification for containment temperature. There are an additional 18 local terminal boxes to permit temporary installation of resistance temperature detectors during containment integrated leak rate testing, thus providing 24 resistance temperature detectors located throughout the containment at several elevations if required for the integrated leak rate tests.

6.2.2 CONTAINMENT HEAT REMOVAL SYSTEMS

The following design criteria 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 (see Section 3.1.1). Conformance with 1972 General Design Criteria of 10 CFR 50, Appendix A, is discussed in Section 3.1.2. The criteria discussed in Section 3.1.2 as they apply to the containment heat removal systems include 38, 39, and 40.

CRITERION: Where an active heat removal system is needed under accident conditions to prevent exceeding containment design pressure, this system shall perform its required function, assuming failure of any single active component (AIF-GDC 52).

Two means of removing heat from the containment atmosphere are provided: the containment recirculation fan cooler (CRFC) units and the containment spray system. As shown in Sections 6.2.1.2.3 through 6.2.1.2.3.3 for containment integrity, at least one train of each of these systems is required to provide sufficient steam-condensing capacity to ensure against containment overstress and to remove that portion of the residual heat and chemical reaction heat released to the containment.

The service water (SW) system is the heat sink for the containment recirculation fan coolers (CRFC). Operation of two of the installed service water (SW) pumps will provide sufficient cooling water for the minimum required number of containment recirculation fan cooler (CRFC) units.

Electrical power for the CRFC motors and the service water (SW) pumps is provided from the 480-V station auxiliary supplies. If auxiliary power is not available, the onsite diesel-generator power units supply power as described in Section 8.3.

Testing of Operational Sequence of Containment Pressure Reducing Systems

CRITERION: A capability shall be provided to test initially under conditions as close as practical to the design and the full operational sequence that would bring the containment pressure reducing systems into action, including the transfer to alternate power sources (AIF-GDC 61).

Capability is provided to test initially to the extent practical the operational startup sequence beginning with transfer to alternative power sources and ending with near design conditions for the containment spray and containment recirculation fan cooling (CRFC) systems.

6.2.2.1 Containment Recirculation Fan Cooler (CRFC) System

6.2.2.1.1 Design Bases

6.2.2.1.1.1 Capacity

The cooling capacity of the CRFC system is sized to remove the normal heat loss from equipment and piping in the reactor containment during plant operation and to remove sufficient heat from the reactor containment following the design-basis accident and subsequent containment pressure transient to keep the containment pressure from exceeding design pressure. The fans and cooling units continue to remove heat after the loss-of-coolant accident (LOCA) and in conjunction with the containment spray system to reduce the pressure close to atmospheric within the first 24 hours.

The fission product control function of the containment post-accident charcoal system which utilizes the CRFC units is discussed in Section 6.5.

The original CRFC units were replaced during the 1993 MODE 6 (Refueling) outage.

6.2.2.1.1.2 *Design Objectives*

The following objectives are met to provide the engineered safety features cooling function.

- a. The containment heat removal requirements are bounded by the assumptions of two evaluations. They are the containment integrity evaluation that is discussed for the steam line break in Section 6.2.1.2 and the LOCA analysis under 10 CFR 50, Appendix K, that is discussed in Section 15.6.4.2. The upper bound heat removal capability is established by the LOCA analysis. The lower bound heat removal capability is established by containment design pressure limits following the postulated main steam line break accident. The establishment of basic heat transfer parameters for the cooling coils of the containment recirculation fan cooler (CRFC) units, and the calculation for design air flow are discussed in Section 6.2.2.1.2.2. Each of the containment recirculation fan coolers (CRFC) installed during the 1993 refueling outage provides a minimum heat removal capability of 54.6×10^6 Btu/hr with containment conditions of 74.7 psia, 286°F saturated steam and air mixture, with a minimum service water (SW) system flow of 915 gpm at 80°F maximum inlet temperature and a fouling factor of $0.001 \text{ hr-ft}^2\text{-°F/Btu}$. This heat removal capability is achieved at an air flow of 33,000 cfm and a coil face velocity of 440 fpm.

The impact on CRFC heat removal capability for an increase in maximum lake temperature from 80°F to 85°F during the injection phase of a design basis LOCA has been evaluated in combination with reduced SW flow to the CRFC coolers due to degraded SW pump performance and bounding SW piping hydraulic resistance. The resulting decrease in CRFC heat removal calculated by *Reference 43* has been used to establish a bounding minimum CRFC heat removal capability that is used for bounding design basis containment pressure and temperature transient analyses.

The maximum heat removal capacity of 149.1×10^6 Btu/hr was evaluated in *Reference 26*. This capacity was determined at a maximum service water (SW) flow of 1750 gpm at 35°F and no heat exchanger fouling. The air-side conditions were at the LOCA conditions of 286°F, 74.7 psia, and 0.202 lb/ft^3 . The maximum heat removal capacity was reevaluated in *Reference 35* for a service water temperature of 30°F and determined to be 175.32×10^6 Btu/hr. This reevaluation was performed to support a reduction of the Technical Specifications screen house lower temperature limit.

In the discussion above, the stated service water (SW) flow is through the containment recirculation fan cooler (CRFC) coils only and does not include service water (SW) flow through the fan motor coolers.

- b. In removing heat at the design-basis rate, the coils are capable of discharging the resulting condensate without impairing the flow capacity of the unit and without raising the exit temperature of the service water (SW) to the boiling point. Since condensation of water from the air-steam mixture is the principal mechanism for removal of heat from the post-accident containment atmosphere by the cooling coils, the coil fins will operate as wetted surfaces under these conditions. Entrained water droplets added to the air-steam mixture, such as by operation of the containment spray system, will therefore have essentially no effect on the heat removal capability of the coils.

6.2.2.1.1.3 *Special Features*

The equipment in the containment recirculation fan cooler (CRFC) system is qualified to operate in a post-accident environment as defined in Figures 6.1-1 and 6.1-2.

Portions of other systems that share functions and become part of the systems credited with containment cooling are designed to meet the criteria of this section. Any single active component failure in such systems will not degrade the heat removal capability of the CRFC units.

Where portions of the systems are located outside of the containment, the following features are incorporated in the design for operation under post-accident conditions:

- a. Means for isolation of any section under anticipated malfunction or failure conditions (expected fault conditions).
- b. Means to detect and control radioactivity leakage into the environs, to the limits consistent with guidelines set forth in 10 CFR 50.67.

6.2.2.1.2 System Design

6.2.2.1.2.1 *System Description*

The containment recirculation fan cooling system is designed as Seismic Category I (see Drawing 33013-1863).

The containment recirculation fan cooler (CRFC) system consists of four CRFC units, each including motor, fan, cooling coils, moisture separators and high efficiency particulate air filters, duct distribution system, and instrumentation and controls. The units are located on the intermediate floor between the containment wall and the primary compartment shield walls. Two of the four CRFC units are equipped with activated charcoal filter units, normally isolated from the main air recirculation stream, through which the air-steam mixture is bypassed to remove volatile iodine following an accident. The filter units are located on a platform above the operating floor. The fans are direct-driven, centrifugal type, and the coils are enhanced heat transfer plate fin-tube type. Air-operated, tight-closing, 125-lb USAS butterfly valves isolate any inactive air handling system from the duct distribution system. Duct work distributes the cooled air to the various containment compartments and areas. During normal operation the flow sequence through the CRFC units is as follows: cooling coils, moisture eliminator, high efficiency particulate air filters, fan, and discharge header.

In the event of an accident, the flow sequence would be the same except that the fan discharge of two fans would be directed through an alternative bypass line to the post-accident charcoal filters before being discharged onto the operating floor area of containment.

The design of the post-accident charcoal filter units is discussed in Section 6.5.1.2.

Two of the air handling assemblies are required during the post-accident period for depressurization of the containment. Coils are fabricated of copper Turbex plate fins vertically oriented on austenitic stainless steel (AL-6XN) tubes. The Turbex fins are a corrugated plate design that induces air turbulence, which enhances heat transfer. The coils and coil inlet

louvers are provided with adequate drain pans and drain piping to prevent flooding. Condensate drain water is directed to the reactor cavity sump A.

The coil internal design pressure is 150 psig. Local flow and temperature indication of service water (SW) at each CRFC unit and alarms indicating abnormal service water (SW) flow, temperature, and radioactivity are provided in the control room.

The location of the CRFC units and the distribution system ensures that air will not enter the suction side of the units until after it has circulated through the spaces requiring cooling and ventilation. Air is never drawn directly from inside the primary compartment to the unit suction.

The duct distribution system provides a distribution header, with branch ducts that discharge inside the primary compartment walls. The system also provides two branch ducts at opposite extremes of the containment wall that are routed upward along the edge of the containment wall to the top center of the containment. Air is supplied to the containment from outlets at intervals along the complete length of the duct riser. After being discharged along the walls of the primary compartments, ventilation air circulates upward through the steam generator compartments to the operating floor level. Other air, after being discharged from the duct riser above the operating floor, combines with the air circulating up from the primary compartment and the resultant mixture is drawn through the annulus between the operating floor and the containment wall to the suction side of the CRFC units.

Since all four CRFC units during MODES 1 and 2 and two CRFC units during accident conditions discharge into a common distribution header, no space in the containment is dependent on a single unit for cooling and ventilation.

6.2.2.1.2.2 *Design Analysis*

In order to set the containment recirculation fan coolers (CRFC) normal air flow rates such that the accident heat removal rate is between the bounds of the LOCA analysis (Section 15.6.4.2) and containment integrity analysis (Section 6.2.1.2), the following analysis was performed:

- a. The heat removal capacity of the CRFC unit coils installed during the 1993 refueling outage was evaluated as a function of steam-air flow for cooling water temperatures of 35°F and 80°F at coolant flow rates of 1750 and 915 gpm. This thermal analysis was performed using the Marlo Coil Water Coil Rating Computer Program, Version 2.0. The program predicts heat transfer of the cooling coils for condensing, partial condensing, and sensible modes of operation. Verification of the Marlo Coil Water Coil Rating Computer Program, Version 2.0, is documented in *Reference 26*. The maximum heat removal capacity was later evaluated at a cooling water temperature of 30°F as documented in *Reference 35* to support an increase of maximum service water flow and a reduction of the minimum service water temperature. Additionally, the minimum heat removal capability was later evaluated by *Reference 43* at a cooling water temperature of 85°F.
- b. The CRFC system air-side pressure drops for the enhanced coil fin design were evaluated for normal conditions. A fan performance curve was developed from the original Westinghouse curve but was based on the original onsite test data.

The fan laws were used to convert the original fan performance curve to accident conditions. The pressure drops for accident conditions are summed and the system resistance curve intersection with the fan performance curve determines the steam-air flow rate. The correlation between MODES 1 and 2 air flow and accident generation steam-air flow is plotted on Figure 6.2-10.

Assumptions

Significant assumptions that were made for the analysis are:

1. Accident operation atmosphere is a saturated steam-air mixture at 286°F with a density of 0.202 lb/ft³ and pressure of 74.7 psia.
2. Normal operation atmosphere conditions are 125°F, density of 0.0717 lb/ft³, and pressure of 14.7 psia.
3. The cooling coil tubesheet fouling factor is assumed to be 0.001, which is a typical specification for the tube material and cooling water supply at Ginna Station. The TEMA Standard for this application is 0.001 fouling factor.
4. All components are assumed to be clean, i.e., there is no additional air flow resistance due to dust or dirt loading.
5. Air flow resistance was considered for high efficiency particulate air filters and moisture separators, typical for the equipment used at Ginna Station.

6.2.2.1.2.3 *Redundancy Provisions*

The cooling water requirements for the four containment recirculation fan cooler (CRFC) units can be supplied by any one of the four service water (SW) pumps. Each of the two emergency diesels powers a service water (SW) pump which would be automatically started as part of the emergency bus-loading sequence on loss of normal ac power, coincident with a requirement for engineered safety features operation (safety injection signal).

The CRFC units are supplied by individual lines from the containment service water (SW) loop header. Each inlet line is provided with a shutoff valve and drain valve. Similarly, each discharge line from the CRFC unit is provided with a shutoff valve and drain valve. This allows each CRFC unit to be isolated for draining or maintenance.

The CRFC unit cooling water discharge is monitored for radioactivity using a radiation monitor (R-16) in the common discharge line. Upon indication of radioactivity in the effluent, each CRFC unit discharge line can be valved individually to locate the defective cooling coil. However, since the cooling coils and service water (SW) lines are completely closed inside the containment, no contaminated leakage is expected into these units. In the unlikely event of a leak occurring in a containment recirculation fan cooler (CRFC) unit following a LOCA, the direction of leakage would depend upon the location of the break, since the containment post-accident pressure is in excess of or equal to the pressure at the service water (SW) discharge of the containment recirculation fan cooler (CRFC). The containment post-accident pressure is only higher than the containment recirculation fan cooler (CRFC) supply line pressure for the first hour of the design-basis pressure transient.

During normal operation, cooling water flow through the units is throttled for containment temperature control purposes by a valve on the common discharge header from the CRFC units. An independent full-flow valve opens automatically in the event of a safety injection signal to bypass the control valve. Both valves fail in the open position upon loss of air pressure and either valve is capable of passing the full flow required for all four containment recirculation fan cooler (CRFC) units.

6.2.2.1.2.4 *Actuation Provisions*

When the high-containment-pressure or automatic safety injection signal is received, the butterfly valves for the CRFC units are tripped to the accident position. Accident position is also the fail-safe position.

Butterfly valves for two CRFC units are used to route the discharge air flow through the post-accident charcoal filters; these valves have only two positions, full open or full closed. These valves are air-operated and spring loaded. Upon loss of control signal or control air, the spring actuates the valve to the open accident position (fail-safe operation).

Redundant electrically operated three-way solenoid valves are used at each butterfly valve to control the instrument air supply (control air). These valves are arranged so that failure of a single solenoid valve to respond to the accident signal will not prevent actuation of the butterfly valve to the accident position (fail-safe operation).

The containment pressure is sensed through six separate pressure transducers located outside the containment. Containment pressure is communicated to the transducers by three 3/8-in. stainless steel lines penetrating the containment vessel. The high-containment-pressure signal from these sensors trips the containment isolation dampers and valves and sends a signal to start the fan motors.

Overload protection for the fan motors is provided at the switchgear by over-current trip devices in the motor feeder breakers. The fan motor feeder breakers can be operated from the control room and can be reclosed from the control room following a motor overload trip.

6.2.2.1.2.5 *Environmental Protection*

Safety Related CRFC components have been Environmentally Qualified for operation under normal and post-accident containment ambient conditions. CRFC dampers controlled by instrument air will fail to their accident position upon loss of air. Redundant flow switches in the system indicate whether air is circulating in accordance with the design arrangement. The flow switches are not environmentally qualified nor required to operate post-accident. Abnormal flow alarms are provided in the control room.

All fan parts, louver dampers, valve shaft and disk seating surfaces, and ducts in contact with the containment fluid are protected against corrosion. The fan motor enclosures, electrical insulation, and bearings are designed for operation during accident conditions, as described in Section 6.2.2.1.1.

No materials known to react adversely with wet sodium hydroxide (present in the post-accident containment atmosphere from spray operation) are used in the construction of the

CRFC units, except for the HEPA filters, which were upgraded. Refer to Section 6.5.1.2.2.3 for additional details. In particular, original equipment specifications explicitly forbid the use of any aluminum.

Considerable protection of the CRFC equipment against the effects of the sodium hydroxide is afforded by the components arrangement. Following the accident, the entering air steam mixture, with entrained moisture in the form of spray droplets, flows first through the cooling coils. The coil fins operate as wetted surfaces while condensing the steam; the entrained moisture combines with the condensate to greatly reduce the sodium hydroxide concentration of the mixture. At design accident conditions, the average dilution factor exceeds 300.

The cooling coils are constructed of corrosion resistant materials throughout with galvanized steel casings, stainless steel tubes, and copper fins. Condensate collection pans are made of stainless steel.

Following the cooling coils is the moisture separator (demister) section, designed to remove entrained moisture originating in the cooling coils. Two separate moisture removal processes are employed; the first by direct impingement on vertical hooked louvers, and the second by trapping on separator pads. Runoff from both stages flows into collection pans from which it is piped to the containment sump. The moisture separator casings, hooked louvers, and collection pans are fabricated of galvanized steel. Glass fiber media, with resorcinol-formaldehyde binder, is used for the separator pads and the pad retaining grids are of stainless steel.

The resorcinol-formaldehyde binder used in the manufacturing of the separator pads is formed by reaction of resorcinol and formaldehyde in the presence of a basic catalyst. Subsequent exposure to the slightly basic solution at high temperature is not expected to cause significant deterioration of the resin.

6.2.2.1.3 Design Evaluation

The four CRFC units operate in conjunction with the containment spray system as a means of limiting containment overpressure following an accident. Only two of the CRFC units are required in the post-accident condition. Provisions have been included in these units to balance and adjust air and water flow to satisfy both normal and post-accident conditions. Operability requirements of the CRFC units are addressed by Technical Specification LCO 3.6.6.

The containment recirculation fan cooler (CRFC) units are dependent on the operation of the electrical and service water (SW) systems. Cooling water to the coils is supplied from the service water (SW) system. Four service water (SW) pumps are provided, two of which are required to operate immediately during the post-accident period.

During MODES 1 and 2, each of the containment recirculation fan cooler (CRFC) units is in continuous or intermittent operation. Collection and measurement of condensate from the cooling coils is one method used to determine leakage from fluid systems within the containment as described in Section 5.2.5.4. Since the pressure of the service water (SW) system is in excess of that of the containment during MODES 1 and 2, any leakage occurring in a cooling coil would result in leakage of service water (SW) into the containment. This

leakage would flow into the collecting pan at a rate greatly in excess of the normal flow rate from condensation, and would indicate the presence of a leak, which could be confirmed by isolation of service water (SW) flow to the affected CRFC unit. Individual flow and temperature indicators located on the discharge from each CRFC unit with alarms on the control board provide additional means of detecting a leak in a containment recirculation fan cooler (CRFC) unit.

In the case of leakage into the service water (SW) system following a LOCA (i.e., a passive failure), the radiation monitor located in the discharge header from the CRFC units would alarm on the control board and by isolating each CRFC unit in turn, the faulty unit could be identified. Operation of the containment recirculation fan cooling (CRFC) system would then be continued using the remaining three CRFC units.

In the case of service water (SW) leakage into containment, the flow monitor on the affected CRFC unit would alarm if a gross leak occurred, signifying a decrease in exit flow. The largest leak that could escape detection by the flow monitor would be approximately 230 gpm.

Smaller leaks would be detected by means of successive samples of recirculating core coolant, indicating progressively lower boron concentration. It is calculated that if sufficient boron is added via the Emergency Core Cooling System (ECCS), dilution by one-half will not cause the shutdown reactivity to be lost following a major loss-of-coolant accident. At a rate of 230 gpm, about 15 hours of continuous service water (SW) addition would not reduce the boron concentration by this amount. Boron concentration would be measured with sufficient frequency by sampling the recirculating coolant to permit corrective action to be taken before a critical concentration is reached.

6.2.2.1.4 Tests and Inspections

The fan motors and the fan rotating assemblies are statically and dynamically tested for proper balance.

The cooling coils can be leak tested by stopping the fans and continuing the cooling water flow. Any significant cooling water leakage would be seen as flow into the collecting pan. Specific CRFC unit testing requirements are provided in Technical Specification LCO 3.6.6.

6.2.2.1.5 Instrumentation

The cooling water discharge flow and exit temperature of each of the four coolers are alarmed in the control room if the flow is low or if the temperature is high. The transmitters are outside the reactor containment. In addition, the exit flow is monitored for radiation (R-16) and alarmed in the control room if high radiation should occur. This is a common monitor and the faulty cooler can be located locally by manually valving each one out in turn. Operability of R-16 is addressed in the Offsite Dose Calculation Manual.

6.2.2.2 Containment Spray System

6.2.2.2.1 Design Bases

6.2.2.2.1.1 Design Criteria

The following design criteria 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 (see Section 3.1.1). Compliance with 1972 General Design Criteria of 10 CFR 50, Appendix A, is discussed in Section 3.1.2. The criteria discussed in Section 3.1.2 as they apply to the containment spray system include 38, 39, and 40.

Inspection of Containment Pressure Reducing Systems

CRITERION: Design provisions shall be made to the extent practical to facilitate the periodic physical inspection of all important components of the containment pressure reducing systems, such as pumps, valves, spray nozzles, and sumps (AIF-GDC 58).

Where practicable, all active components and passive components of the containment spray system are inspected periodically to demonstrate system readiness. The pressure containing systems are inspected for leaks from pump seals, valve packing, flanged joints, and safety valves. During operational testing of the containment spray pumps, the portions of the systems subjected to pump pressure are inspected for leaks.

Testing of Containment Pressure Reducing Systems Components

CRITERION: The containment pressure reducing systems shall be designed, to the extent practical so that active components, such as pumps and valves, can be tested periodically for operability and required functional performance (AIF-GDC 59).

All active components in the containment spray system are tested both in preoperational performance tests in the manufacturer's shop and in-place testing after installation. Thereafter, periodic tests are also performed, which include tests performed after any component maintenance. Testing of the components of the safety injection system that are used for containment spray purposes are described in Section 6.3.5.

Testing of Containment Spray Systems

CRITERION: A capability shall be provided to the extent practical to test periodically the delivery capability of the containment spray system at a position as close to the spray nozzles as is practical (AIF-GDC 60).

Permanent test lines for all the containment spray loops are located so that all components up to the isolation valves at the containment may be tested. These isolation valves are checked separately.

The air test lines for checking that spray nozzles are not obstructed connect downstream of the isolation valves. This verification can be performed by injecting hot air flow through the nozzles and monitoring the nozzles by the use of thermography. Other means of verification

of air movement through the nozzles may include low pressure air, smoke, or floatation devices.

6.2.2.2.1.2 *Performance Objectives*

The containment spray system, in conjunction with the containment cooling system and the Emergency Core Cooling System (ECCS), is designed to remove sufficient heat from the containment atmosphere following an accident condition to maintain the containment pressure below design limits. The containment spray system in conjunction with the sodium hydroxide (NaOH) system is also capable of reducing the iodine and particulate fission product inventories in the containment atmosphere such that the offsite radiation exposure resulting from a LOCA is within the guidelines established by 10 CFR 50.67.

The containment spray system is designed to spray borated water into the containment building whenever the coincidence of two sets of two-out-of-three high-containment-pressure signals occurs. Either of two subsystems containing a pump and associated valving and spray headers is independently capable of delivering one-half of the assumed flow. Refer to Table 15.6-21 for the flow rate assumed in the accident analysis.

The spray system is designed to operate over an extended time period following a LOCA as required to restore and maintain containment conditions at near atmospheric pressure. Together with the containment recirculation fan coolers (CRFC), it has the capability of reducing the containment post-accident pressure as required, taking into account any reduction due to single failures of active components.

Portions of other systems that share functions and become part of the containment cooling system when required are designed to meet the criteria of this section. Any single failure of active components in such systems does not degrade the heat removal capability of containment cooling.

These portions of the spray system located outside of the containment which are designed to circulate, under post-accident conditions, radioactively contaminated water collected in the containment meet the following requirements:

- a. Shielding within the guidelines of 10 CFR 100.
- b. Collection of discharges from pressure relieving devices into closed systems.
- c. Means to detect and control radioactivity leakage into the environs to the limits consistent with guidelines set forth in 10 CFR 50.67.

The active system components are redundant. System piping located within the containment is redundant and separate in arrangement unless fully protected from damage that may follow any primary coolant system failure.

System isolation valves relied upon to operate for containment cooling are redundant, with automatic actuation or manual actuation.

6.2.2.2.1.3 *Service Life*

All portions of the system located within containment are designed to withstand, without loss of functional performance, the post-accident containment environment and operate without benefit of maintenance for the duration of time to restore and maintain containment conditions at near atmospheric pressure.

6.2.2.2.1.4 *Codes and Classifications*

The codes and classifications to which the system was originally designed and the evaluation with respect to current standards are discussed in Section 3.2 and Table 3.2-1.

6.2.2.2.2 *System Design*

6.2.2.2.2.1 *Operational Requirements*

The containment spray process flow arrangement is shown in Drawing 33013-1261. The containment spray system consists of two pumps, two spray headers, spray nozzles, and the necessary piping and valves. The system initially takes suction from the refueling water storage tank (RWST). When a low level is reached in the refueling water storage tank (RWST), the spray pump suction is fed from the discharge of the residual heat removal pumps if continued spray is required. Containment spray is only required for containment pressure control for beyond-design-basis events during the sump recirculation phase (see Section 6.3.3.9)

During the period of time that the spray pumps draw from the refueling water storage tank (RWST), approximately 20 gpm of spray additive will be added to the refueling water in each train by using a liquid eductor motivated by the spray pump discharge. The fluid passing from the tank will then mix with the fluid entering the pump suction. The results will be a solution suitable for the removal of iodine. The iodine removal function of the containment spray and NaOH systems is discussed in Section 6.5.2.2.

The spray system will be actuated by the coincidence of two sets of two-out-of-three high-containment-pressure signals. This starting signal will start the pumps and open the discharge valves to the spray header (860A, B, C, and D). The valves associated with the spray additive tank will be opened automatically after the containment spray signal is actuated. Sodium hydroxide will flow to the suction of the spray pumps and mix with refueling water prior to being discharged through the spray nozzle into the containment.

The system design conditions were selected to be compatible with the design conditions for the low-pressure injection system, since both of these systems share the same suction line from the RWST.

The physical arrangement of spray nozzles is such that a maximum portion of the free volume of the containment is contacted by the sprays. Exception is made of the region below the operating deck because (1) the proximity of large components and supporting structure restricts the available free trajectory of the spray, and (2) the possibility of missile damage places difficult constraints on the routing of spray piping. The fraction of the free volume that is not washed by sprays is about 22%.

Another factor that tends to induce mixing in the containment atmosphere during the early phases of spray operation is the bulk transport of steam from the general regions of the containment (especially the region near the reactor coolant system rupture point) to the zone just below the spray nozzles where most of the condensation occurs.

6.2.2.2.2.2 *Refueling Water Storage Tank (RWST)*

The containment spray system shares the refueling water storage tank (RWST) liquid capacity with the safety injection system. The description of the refueling water storage tank (RWST) is provided in Section 6.3.2.2.3.

6.2.2.2.2.3 *Containment Spray Pumps*

The two containment spray pumps are of the horizontal centrifugal type driven by electric motors. These motors can be powered from both normal and emergency power sources.

The design head of the pumps is sufficient to continue at rated capacity with a minimum level in the refueling water storage tank (RWST) against a head equivalent to the sum of the design pressure of the containment, the head to the upper-most nozzles, and the line and the nozzle pressure losses. Pump motors are direct-coupled and large enough for the maximum power requirements of the pump. The materials of construction are suitable for use in mild boric acid solutions, such as stainless steel or equivalent corrosion-resistant material.

Design parameters are presented in Table 6.2-27.

The pump motors are direct-coupled and non-overloading to the end of the pump curve. The containment spray pump characteristic curves of total dynamic head and net positive suction head as a function of flow are shown in Figure 6.2-19.

6.2.2.2.2.4 *Liquid Jet Eductor*

The system incorporates the use of a liquid jet eductor in each train to entrain the sodium hydroxide solution and mix it with the borated water from the refueling water storage tank (RWST). Approximately 5% of the spray pump discharge is diverted through the eductor. The kinetic energy of the spray pump discharge passing through the eductor causes a low-pressure area to exist at the throat of the eductor. This low-pressure area allows the sodium hydroxide to flow (20 gpm) into and mix with the borated water passing through the eductor. The now caustic solution passes out of the eductor where it returns to the suction side of the spray pump.

6.2.2.2.2.5 *Spray Ring Headers*

Once the caustic solution leaves the spray pump discharge, it passes through the respective motor-operated discharge valves (860 A, B, C, and D). The solution passes into the containment structure and through two 6-in. spray headers which climb the interior walls of the containment building. At the top of the containment building, the two spray headers terminate in two spray rings. Each ring has many hollow cone nozzles at varying angle orientations and relative header positions to ensure a minimum of 90% area coverage and uniform heat and fission product removal.

6.2.2.2.2.6 *Spray Nozzles*

The spray nozzles, of the ramp bottom design, are not subject to clogging by particles less than 0.25 in. in diameter and are capable of producing a mean drop size of approximately 1000 microns in diameter with the spray pump operating at design conditions and the containment at design pressure. The 179 nozzles are mounted on two concentric ring headers each comprised of 3-in. and 4-in. pipes.

During the injection phase of an accident, the spray pumps draw suction from the refueling water storage tank (RWST), which is not expected to have any large particles. During the recirculation phase of an accident, the strainer system associated with containment sump B prevents particles greater than 1/16 in. in diameter from entering the spray nozzles (see Section 6.3.2.2.6).

6.2.2.2.2.7 *Environmental Qualification*

All of the active components of the containment spray system are located outside the containment and therefore are not required to operate in the steam-air environment produced by the accident.

Parts of the system in contact with borated water, the sodium hydroxide spray additive, or mixtures of the two are stainless steel or an equivalent corrosion-resistant material. The piping is welded except for flanged connections at the pumps, post-accident charcoal filters fire protection, various instruments, and blind flanges.

6.2.2.2.2.8 *System Tests*

The proper functioning of the spray pumps and the isolation valves can be tested by first manually valving off the spray header at the containment and opening the spray test line, Drawing 33013-1261. On actuation of the valves and pumps, flow will pass back through the test line to the refueling water storage tank (RWST).

Nozzle availability can be tested by blowing hot air (approximately 200°F) through the nozzles and observing the flow by use of thermography or by other means which may include verifying air movement through the nozzles by low pressure air, smoke, or floatation devices. Specific system testing requirements are specified in Technical Specification LCO 3.6.6.

6.2.2.2.3 *Design Evaluation*

6.2.2.2.3.1 *Design Basis*

The system design is based on the spray water being raised to the temperature of the containment in falling through the steam-air mixture within the building. The minimum fall path of the droplets is approximately 60 ft from the spray ring headers to the operating floor. The actual fall path is longer due to the trajectory of the droplets sprayed out from the ring header. Heat transfer calculations, based upon 1000 micron droplets, show that thermal equilibrium will be reached in a distance of approximately 5 ft. Thus, the spray water reaches essentially the saturation temperature. The containment spray system is designed to provide sufficient

heat removal capability to maintain the post-accident containment pressure below the design value in conjunction with the minimum number of CRFC units.

6.2.2.3.2 *Heat Transfer Calculations*

When a spray drop enters the hot saturated steam-air environment, the vapor pressure of the water at its surface is much less than the partial pressure of the steam in the atmosphere. Hence, there will be diffusion of steam to the drop surface and condensation on the drop. This mass flow will carry energy to the drop. Simultaneously, the temperature difference between the atmosphere and the drop will cause a heat flow to the drop. Both of these mechanisms will cause the spray drop temperature and vapor pressure to rise. The vapor pressure of the drop will eventually become equal to the partial pressure of the steam and the condensation will cease. The temperature of the drop will be essentially equal to the temperature of the steam-air mixture.

The terminal velocity of the drop can be calculated using the formula given by Weinberg (*Reference 5*), where the drag coefficient C_D is a function of the Reynolds number:^a

$$V^2 = \frac{4Dg(\rho - \rho_m)}{3C_D\rho_m}$$

(Equation 6.2-1)

For the 700 micron drop size expected from the nozzles, the terminal velocity is less than 7 ft/sec. For a 1000 micron drop, the velocity would be less than 10 ft/sec. The Nusselt number for heat transfer, Nu , and the Nusselt number for mass transfer, Nu' (Sherwood number), can be calculated from the empirical relations given by Ranz and Marshall (*Reference 6*).

$$Nu = 2 + 0.6 (Re)^{1/2} (Pr)^{1/3}$$

$$Nu' = 2 + 0.6 (Re)^{1/2} (Sc)^{1/3}$$

The Prandtl number and the Schmidt number for the conditions assumed are approximately 1.7 and 0.6, respectively. Both of these are sufficiently independent of pressure, temperature, and composition to be assumed constant under containment conditions (*References 7 and 8*). The coefficients of heat transfer, h_c , and mass transfer, k_G are calculated from Nu and Nu' , respectively. The equations describing the temperature rise of a falling drop are as follows:

$$\begin{aligned} (d/dt) (Mu) &= mh_g + q \\ (d/dt) (M) &= m \end{aligned}$$

a. Nomenclature used is given at the end of this discussion

where:

$$q = h_c A (T_s - T)$$

$$m = k_G A (P_s - P_v)$$

These equations can be integrated numerically to find the internal energy and mass of the drop as a function of time as it falls through the atmosphere. Analysis shows that the liquid drop temperature rises to the steam-air mixture temperature in less than 0.5 sec, which occurs before the drop has fallen 5 ft. These results demonstrate that the spray will be 100% effective in removing heat from the atmosphere.

The nomenclature is as follows:

A = area

C_D = drag coefficient

D = droplet diameter

g = acceleration of gravity

h_c = coefficient of heat transfer

h_s = steam enthalpy

k_G = coefficient of mass transfer

M = droplet mass

m = diffusion rate

Nu = Nusselt number for heat transfer

Nu' = Nusselt number for mass transfer

P_s = steam partial pressure

P_v = droplet vapor pressure

Pr = Prandtl number

q = heat flow rate

Re = Reynolds number

Sc = Schmidt number

T = droplet temperature

T_s = steam temperature

t = time

u = droplet internal energy

V = velocity

ρ = droplet density

ρ_m = steam-air mixture density

The design evaluation on the iodine removal function of the containment spray and NaOH systems is discussed in Section 6.5.2.2.

6.2.2.2.3.3 *Reliance on Interconnected Systems*

The containment spray system operates in conjunction with the containment recirculation fan cooling system for containment integrity. For operation in the recirculation mode, water is supplied through the residual heat removal pumps. Spray pump cooling is supplied from the component cooling loop when required during the recirculation mode.

During the recirculation phase some of the flow leaving the residual heat exchangers may be bled off and sent to the suction of either the containment spray pumps or the high-head safety injection pumps. Minimum flow requirements are set for the flow being sent to the core via the residual heat removal and safety injection systems and for the flow being sent to the containment spray pump suction.

Normal and emergency power supply requirements are discussed in Section 8.3.

6.2.2.2.3.4 *Reliability Considerations*

A failure analysis has been made on all active components of the system to show that the failure of any single active component will not prevent fulfilling the design function. This analysis is summarized in Table 6.2-28.

6.2.2.2.3.5 *Containment Spray Pump Net Positive Suction Head Requirements*

The net positive suction head for the containment spray pump is evaluated for both the injection and recirculation phase operation of the design-basis accident. The end of the injection phase operation gives the limiting net positive suction head requirement and the net positive suction head available is determined from the elevation head and vapor pressure of the water in the refueling water storage tank (RWST) and the pressure drop in the suction piping from the tank to the pumps.

6.2.2.2.3.6 *Equipment Protection*

The containment spray pumps are protected against overcurrent. Additionally, the safety injection pump fans start when either containment spray pump starts. These fans are not required or credited in the accident analysis.

The containment spray pump discharge valves are protected against overcurrent and excessive torque. Also, each valve is equipped with a mechanical interlock feature, such that the valve must stroke fully open before it can be closed and vice versa.

The containment spray system pumps are powered from safeguards buses 14 and 16. The containment spray pumps will start on a high-high-containment-pressure signal. The pumps can be supplied by an emergency diesel generator and do not trip on under-voltage conditions on the respective bus. The containment spray discharge valves are supplied from engineered safety features motor control centers supplied by safeguards buses 14 and 16. The motor control centers do not shed vital equipment on loss of voltage. The pumps and valves would continue to operate when power was restored.

6.2.2.2.4 Minimum Operating Conditions

The limiting conditions for operation are specified in Technical Specification LCO 3.6.6. The basis of the limits is to ensure that prior to raising the reactor coolant system to elevated temperature and pressure, adequate containment cooling engineered safety features are operable.

6.2.2.2.5 Tests and Inspections

The active components of the containment spray system are tested on a regular schedule as follows:

- A. Spray pumps. These pumps are tested individually by opening the valves in the full flow test line back to the refueling water storage tank (RWST). Each pump in turn is started by operator action and checked for flow establishment. The spray injection valves (860A, B, C, and D) are also stroked open and closed during this test.
- B. Spray nozzles. With the motor-operated valves closed, the lines are drained and low-pressure air, hot air, or smoke is blown through the test connections.

During these tests the equipment is visually inspected for leaks. Leaking seals, packing, or flanges are tightened or replaced to eliminate the leak. Valves and pumps are operated and inspected after any maintenance to ensure proper operation.

The Ginna Station Gas Intrusion Management Program was established in response to Generic Letter (GL) 2008-01. The GL requires all plants to evaluate the Emergency Core Cooling, Decay Heat Removal and Containment Spray systems for gas intrusion. The program provides guidance for the identification, evaluation, and elimination of gas voids if found. Additional references can be found in Section 6.3.5.5.

6.2.2.2.6 Instrumentation

6.2.2.2.6.1 Interlock and Control Features

The containment spray pumps are controlled remotely by two (STOP/AUTO/CLOSE) selector switches on the main control board. Both pumps will start automatically when two of the three instruments on two-of-two channels sense 28 psig containment pressure. Manual actuation can be accomplished by depressing two buttons simultaneously on the main control board. Additionally, the four containment spray header isolation valves (860A, B, C, and D) will open automatically on a high-high containment pressure signal. Manual operation of the spray header isolation valves can be accomplished from the main control board.

6.2.2.2.6.2 Control Room and Local Indication

Control Room Alarms

The following is a list of conditions that would sound an alarm or annunciator in the control room:

- Containment spray, two channels, two-of-three instruments, 28 psig.
- Containment spray pump cooling water outlet low flow.

- Containment pressure channel alert.
- Safeguards breaker trip - if any safeguards pump trips.
- Breaker lockout - if any safeguards pump is locked out.

Control Room Indication

The following is a list of parameters that are measured and provide indication in the control room:

- Containment pressure, 0-60 psig front control board.
- Containment pressure, 10-200 psia back control board.
- Refueling water storage tank (RWST) level 0-100%.
- Containment spray pump discharge valves AUTO, OPEN, or CLOSED.
- Containment spray pump start switch manual.
- Containment spray actuation two-of-two buttons.

Local Indication

The following is a list of containment spray system parameters that provide local indication:

- Containment spray system test line flow.
- Refueling water storage tank (RWST) level 0-100%.
- Containment spray pump motor fan cooler switch at pumps.

6.2.3 SECONDARY CONTAINMENT

The Ginna Station design does not require the use of a secondary containment to reduce MODES 1 and 2 radiological releases or to mitigate the consequence of postulated accidents. Therefore, this section is not applicable to Ginna Station.

6.2.4 CONTAINMENT ISOLATION SYSTEM

6.2.4.1 Design Criteria

The following design criteria were used during the licensing of Ginna Station. They represent the AIF version of proposed criteria issued by the AEC for comment on July 10, 1967 (see Section 3.1.1). Conformance with 1972 General Design Criteria (GDC) of 10 CFR 50, Appendix A, is discussed in Section 3.1.2. The criteria discussed in Section 3.1.2 as they apply to the containment isolation system include 54, 55, 56, and 57. The more recent SEP evaluation of the design against these criteria is discussed in Section 6.2.4.4.1.

CRITERION: Penetrations that require closure for the containment function shall be protected by redundant valving and associated apparatus (AIF-GDC 53).

Isolation valves for all fluid system lines penetrating the containment provide at least two barriers for redundancy against leakage of radioactive fluids to the environment in the event of a loss-of-coolant accident (LOCA). These barriers, in the form of isolation valves or closed

systems (i.e., containment isolation boundaries), are defined on an individual line basis. In addition to satisfying containment isolation criteria, the barriers are designed to facilitate normal operation and maintenance of the systems and to ensure reliable operation of other engineered safety features.

With respect to numbers and locations of isolation boundaries, the criteria applied are generally those outlined by the five classes described in Section 6.2.4.4.

CRITERION: Capability shall be provided to the extent practical for testing functional operability of valves and associated apparatus essential to the containment function for establishing that no failure has occurred and for determining that valve leakage does not exceed acceptable limits (AIF-GDC 57).

Capability is provided to the extent practical for testing the functional operability of valves and associated apparatus during periods of reactor shutdown. The isolation valves are also subject to periodic type C leak rate tests as required by the Containment Leakage Rate Testing Program (Technical Specification 5.5.15).

6.2.4.2 Design Basis

6.2.4.2.1 Functional Requirements

The function of the containment isolation system is to isolate the non-essential process lines that penetrate the containment to ensure that the total leakage of activity will be within design limits in the event of an accident. The system consists of many valves and the logic and circuitry necessary to automatically close these valves on a containment isolation signal. All major nonessential lines that penetrate the containment, except the auxiliary feedwater lines (preferred and standby), component cooling water for the reactor coolant pumps, and the main steam lines, have either automatic isolation valves or are normally maintained closed when the reactor is above MODE 5 (Cold Shutdown).

Each system whose piping penetrates the containment leakage limiting boundary is designed to maintain or establish isolation of the containment from the outside environment under any accident for which isolation is required, and assuming a coincident independent single failure or malfunction occurring in any active system component within the isolated bounds.

Piping penetrating the containment is designed for pressures at least equal to the containment design pressure. Containment isolation boundaries are provided as necessary in lines penetrating the containment to ensure that no unrestricted release of radioactivity can occur. Such releases might be due to rupture of a line within the containment concurrent with a LOCA or due to rupture of a line outside the containment which connects to a source of radioactive fluid within the containment or communicates with the containment environment.

Isolation of a line inside the containment prevents flow from the reactor coolant system or any other large source of radioactive fluid in the event that a piping rupture outside the containment occurs, even though a piping rupture outside the containment at the same time as a LOCA is not considered credible, since the penetrating lines are Seismic Category I design up to and including the second isolation boundary and are assumed to be an extension of containment.

Closure times for isolation valves are provided in an administrative procedure. Containment isolation becomes necessary under the same conditions that require operation of the other engineered safety features. The containment isolation signal is derived from the same signals which automatically activate safety injection.

6.2.4.2.2 Seismic Design

The isolation valves included within the containment isolation system are designed and qualified to ensure that they are capable of withstanding the maximum potential seismic loads.

To facilitate their adequacy in this respect:

- A. Valves are located, when possible, in a manner to reduce the accelerations on the valves. Valves suspended on piping spans are reviewed for adequacy for the loads to which the span would be subjected. Valves are mounted in the position recommended by the manufacturer.
- B. Valve yokes are reviewed for adequacy and strengthened as required for the response of the valve operator to seismic loads.
- C. Where valves are required to operate during seismic loading, the operating forces are reviewed to ensure that system function is preserved. Seismic forces on the operating parts of the valve are expected to be small compared to the other forces present.
- D. Control wires and piping to the valve operators are designed and installed so that the flexure of the line does not endanger the control system. Appendages to the valve, such as position indicators and operators, are checked for structural adequacy.
- E. As part of the RG&E seismic piping upgrade program, all containment isolation valves were reviewed and reanalyzed to ensure that current seismic criteria are met.

6.2.4.3 System Design

The five classes described in Section 6.2.4.4 are the general categories into which lines penetrating containment are classified. Also described in Section 6.2.4.4 are the basic isolation boundary arrangements used to provide two barriers between the reactor coolant system or containment atmosphere and the environment. System design is such that failure of one boundary to close does not prevent isolation, and no manual operation is required for immediate isolation. Automatic isolation is initiated by the containment isolation signal.

A containment isolation signal is generated by any automatic safety injection signal or manually by depressing one of two containment isolation pushbutton switches on the main control board left panel. A manual safety injection signal will not, by itself, generate a containment isolation signal. Operability requirements of the containment isolation signal are addressed by Technical Specification LCO 3.3.2. The safety injection signals which initiate containment isolation are low pressurizer pressure, low steam line pressure, and high containment pressure.

Main steam line isolation will occur manually or on a high-high steam flow coincident with a safety injection signal, high steam flow and low TAVG coincident with a safety injection signal, and high-high containment pressure. Operability requirements of the main steam

isolation instrumentation are addressed by Technical Specification LCO 3.3.2. Automatic containment isolation valves are designed to isolate the process stream in a maximum of 60 seconds depending on the particular isolation valve. Operability requirements of the containment isolation valves are addressed by Technical Specification LCO 3.6.3.

Containment isolation valves and associated fans which directly communicate with the containment atmosphere and which can provide a potential release path to the outside environment are also isolated by a containment ventilation isolation signal. The signal is generated by gaseous (R-12) and particulate (R-11) radiation monitors, a containment isolation signal or manual containment spray signal. The containment ventilation isolation signal serves as a backup to the containment isolation signal and is not specifically credited in the accident analysis. Operability requirements of the containment ventilation isolation instrumentation are addressed by Technical Specification LCO 3.3.5.

6.2.4.3.1 Isolation Valve Parameters Tabulation

A summary of the fluid system lines penetrating containment and the boundaries employed for containment isolation is presented in Table 6.2-29 and Figures 6.2-13 through 6.2-78. The containment isolation valves (CIV) and boundaries (CIB) are also indicated in the figures. However, the intent of Figures 6.2-13 through 6.2-78 is to provide only schematic representations of the penetrations for leak testing purposes and not to depict complete as-built piping configurations. Each boundary is listed in Table 6.2-29 as to type, position indication in the control room, and open or closed status during normal operation, and immediate post-accident conditions.

Containment isolation boundaries are provided with actuation and control equipment appropriate to the valve type. For example, air-operated and diaphragm (Saunders patent) valves are generally equipped with air diaphragm operators, with fail-safe operation ensured by redundant control devices in the instrument air supply to the valve. Solenoid valves are also designed for fail safe operation. Motor-operated valves are capable of being supplied from reliable onsite emergency power as well as their normal power source. Closed systems, manual valves, and check valves, of course, do not require actuation or control systems. These non-automatic isolation boundaries are used in lines that must remain in service, at least for a time, following an accident. These are closed manually if and when the lines are taken out of service.

6.2.4.3.2 Isolation Valve Operability

All containment isolation valves, actuators, and controls are located so as to be protected against missiles which could be generated as the result of a LOCA. Only valves so protected are considered to qualify as containment isolation valves.

Only isolation valves located inside containment are subject to the high-pressure, high-temperature, steam-laden atmosphere resulting from an accident. Operability of these valves in the accident environment is ensured by proper design, construction, and installation, as reflected by the following considerations.

- A. All components in the valve installation, including valve bodies, trim and moving parts, actuators, instrument air and control, and power wiring, are qualified to perform their safety function in a post-accident environment if they need to change position. Provisions for valves locked in their safe post-accident conditions ensure no adverse changes of position due to accident conditions.
- B. In addition to normal pressures, the valves are designed to withstand maximum pressure differentials in the reverse direction imposed by the accident conditions.

All air-operated valves that receive a containment isolation signal or are considered a containment isolation valve are listed in Table 6.2-32. The effects of loss of air to these valves were considered in the safety analysis of all systems in the plant. Throughout the overall design of the plant, it has been acknowledged that the air supply is not a Class 1 safety-related system. All systems have been designed accordingly with careful attention to the manner of operating equipment to ensure that each component will assume the safe position upon loss of air pressure.

6.2.4.4 Design Evaluation

6.2.4.4.1 Current Safety Criteria

The containment isolation system conforms with the requirements of diversity in the parameters used for containment isolation, i.e., automatic isolation of normally open nonessential systems by the containment isolation signal. The design is such that resetting the isolation signal will not result in the automatic reopening of containment isolation valves but reopening will require deliberate operator action. The essential versus nonessential system containment penetrations are given in Table 6.2-33.

The containment isolation system at Ginna Station was evaluated (*Refs. 9 and 10*) by the NRC under the Systematic Evaluation Program (SEP) Topic VI-4. The safety criteria used in the evaluation were as follows.

- A. 10 CFR Part 50, Appendix A, General Design Criteria (GDC) 54, 55, 56, and 57.
- B. NUREG 75/087, Standard Review Plan (SRP) for the Review of Safety Analysis Reports for Nuclear Power Plants (SRP 6.2.4, Containment Isolation System, where applicable).
- C. Regulatory Guide 1.11, Instrument Lines Penetrating Primary Reactor Containment.
- D. Regulatory Guide 1.141, Revision 1, Containment Isolation Provisions for Fluid Systems.

There are five classes of penetrations for categorizing the lines penetrating the containment. The following discussion addresses the conformance of each penetration to applicable safety criteria. The penetration numbers (or line numbers) correspond to those provided in Table 6.2-29.

6.2.4.4.2 Class 1 Penetrations (Outgoing Lines, Reactor Coolant System)

6.2.4.4.2.1 Applicable Lines

Normally operating outgoing lines connected to the reactor coolant system were originally designed with at least one automatically operated isolation valve and one manual isolation valve in series located outside the containment. GDC 55, which applies to Class 1 lines, specifies that one valve should be located inside the containment and one valve should be located outside the containment, with the valves being either locked closed or being automatic isolation valves. Furthermore, a simple check valve outside containment may not be used as an automatic isolation valve. The following lines are included in this class: 108, 110b, 112, 140, 205, 206a, and 207a.

6.2.4.4.2.2 Class 1 Penetration Evaluation

The containment isolation provisions for lines 108 and 110b differ from the explicit requirements of GDC 55 from the standpoint of the number of isolation valves. There is no containment isolation valve in these lines inside the containment. Therefore, an automatic isolation valve for line 108 and a locked-closed manual valve for line 110b would be required inside the containment to meet current criteria. For line 108 (reactor coolant pump seal water return and excess letdown line) it was concluded that installing a second automatic isolation valve was not required since it is a 3-in. line which has an automatic, leak-tested, motor-operated valve outside containment. The line also terminates in the volume control tank, which has a design pressure higher than the containment accident pressure. The NRC agreed that an additional valve was not necessary and the current configuration was found acceptable (*Reference 12*).

Backfitting was not recommended by the NRC for line 110b (safety injection test line) because:

- a. The penetration is isolated further upstream from the cold leg by check valves which are periodically leak tested.
- b. The penetration is isolated from the hot legs by two check valves and a closed motor-operated valve.
- c. During safety injection the line is pressurized by the safety injection system and because the line is connected vertically to the reactor coolant system, a water seal would exist after injection.
- d. The line is of small diameter (3/4-in.). Additionally, from a risk standpoint, the probabilistic risk assessment conducted in support of the Systematic Evaluation Program ranked this issue of low importance and estimated the impact of the resolution to be low (*Reference 10*).

Penetration 112 (letdown to nonregenerative heat exchanger) differed from the explicit requirements of GDC 55 from the standpoint of valve location at the time of the SEP review. The penetration was subsequently modified to provide automatic isolation valves both inside and outside containment. The penetration now meets the explicit requirements of GDC 55 (*Reference 11*).

The containment isolation provisions for line 140 (residual heat removal system letdown line) differ from the explicit requirements of GDC 55 from the stand-point of valve location. A single motor-operated valve is provided inside the containment; there is no containment isolation valve in the line outside the containment. However, the closed, safety-grade system outside the containment (residual heat removal system) is a suitable isolation barrier in lieu of a valve adjacent to the containment and GDC 55 permits isolation provisions that differ from the explicit requirements, provided the basis for acceptability is defined. Therefore, the isolation provisions for line 140 satisfy GDC 55 on some other basis.

The isolation provisions for lines 205, 206a, and 207a differ from the explicit requirements of GDC 55 from the standpoint of valve location and actuation. Each of these lines is provided with an air-operated valve and one manual valve in series located outside the containment. Locating both containment isolation valves outside containment was considered acceptable based on the design of the piping between the containment and the first valve being sufficiently conservative to provide adequate assurance of integrity. Each of these lines also contains an air-operated valve located inside the missile barrier in containment. The controls for these air-operated valves inside the missile barrier were modified to fail closed when the instrument air to the containment is automatically isolated. These valves also receive a containment isolation signal and remain closed on reset of containment isolation. However, due to the location of these valves, they are not containment isolation valves. This configuration was found acceptable by the NRC (*Reference 10*).

6.2.4.4.3 Class 2 (Outgoing Lines)

6.2.4.4.3.1 Applicable Lines

Normally operating outgoing lines not connected to the reactor coolant system and not protected against missiles throughout their length inside the containment were originally designed with at least one automatically operated isolation valve or one remotely operated stop valve located outside the containment. Manual isolation valves in series with the isolation or remote operated valves were also provided outside containment.

GDC 56 applies to Class 2 penetration lines and specifies that one valve should be located inside the containment and one valve should be located outside the containment with the valves being either locked closed or being automatic isolation valves. Furthermore, a simple check valve outside the containment may not be used as an automatic isolation valve. The following lines are included in this class: 107, 120b, 121c, 123a, 143, 203a, 305e, and 332c.

6.2.4.4.3.2 Class 2 Evaluation

The containment isolation provisions for lines 107, 121c, 143, 203a, and 332c differ from the explicit requirements of GDC 56 from the standpoint of valve location. All of these lines have isolation valves located outside the containment.

It is not practical for lines 107 (sump discharge line) and 143 (reactor coolant drain tank discharge line) to have an isolation valve located inside the containment because the valve may be submerged as a result of a LOCA. Standard Review Plan 6.2.4, Item II.3, has provided guidance in this concern. Therefore, the valving arrangement for these lines with both valves located outside the containment was found acceptable (*Reference 9*).

Lines 121c, 203a, and 332c are small sensing lines for the containment pressure transmitters and are open to the containment atmosphere; they were installed as a requirement of the TMI Lessons Learned. The pressure transmitters form a closed boundary outside the containment. A manual valve is also provided in each line for double-barrier isolation capability. In light of the post-accident monitoring function of these lines, the isolation barriers satisfy GDC 56 on some other defined basis (Reference ANSI/ANS-56.2, 1984).

The containment isolation provisions for lines 120b, 123a, and 305e differ from the explicit requirements of GDC 56 from the standpoint of valve location and actuation. Each of these lines is provided with an air-operated valve and a manual valve in series outside the containment.

Backfitting to meet explicit valve location requirements was not recommended by the NRC for these lines for the following reasons (*Reference 10*):

- a. As a plant design basis, the piping between the containment and the containment isolation valves is at least equal to containment design pressure. Isolation valves are similarly rated.
- b. Piping runs between the containment penetrations and the containment isolation valves have been kept as short as possible and are Seismic Category I.
- c. All piping penetrations are solidly anchored to the containment wall. External guides, stops, increased pipe thickness, or other means are provided, where required, to limit motion and moments to prevent ruptures by making the penetration the strongest part of the system. In addition, all penetrations and anchorages are designed for forces and moments that might result from postulated pipe ruptures.
- d. These piping penetrations are located in areas that are protected from tornado missiles.

Backfitting to meet explicit valve actuation requirements was not recommended by the NRC for the following reasons (*Reference 10*):

- aa. These lines are small (3/8 in. and 1 in.).
- bb. The valves are located near the containment wall.
- cc. The piping and valves are designed as Seismic Category I.
- dd. These valves are small, air-operated, fail-closed valves and have had no previous history of failure to close at Ginna Station.

Additionally, from a risk standpoint, the SEP pipe rupture analysis ranked this issue of low importance and estimated the impact of the resolution to be low.

Consequently, these lines are acceptable.

6.2.4.4.4 Class 3 (Incoming Lines)

Two subclasses are identified for Class 3 penetration lines. GDC 55 or 56 apply to Class 3 lines, depending on the line function:

6.2.4.4.1 *Class 3A Penetrations*

Incoming lines connected to open systems outside the containment are provided with one of the following arrangements: (1) a check valve located inside containment and a remote-operated valve or closed manual valve located outside the containment, (2) both a check valve and a remote-operated valve or closed manual valve located outside the containment, or (3) two remote-operated valves located outside containment. The following penetration lines are included in this subclass: 120a, 121a, 121b, 129, 305b, 310a, and 310b.

The containment isolation provisions for lines 121b and 129 (branch a) differ from the explicit requirements of GDC 56 from the standpoint of valve type. A check valve and a locked-closed manual valve, both of which are leak tested, provide for containment isolation. This provides two containment isolation barriers, which does not meet the explicit requirements of GDC 56, but provides equivalent protection. This configuration was found acceptable by the NRC since the probabilistic risk assessment found this issue to be of low significance (*Reference 10*).

The containment isolation provisions for lines 129 (branch b) and 305b differ from the explicit requirements of GDC 56 from the standpoint of valve location. These lines have two automatic air-operated isolation valves located outside containment. These were found acceptable based on the location discussion for Class 2 lines 120b, 123a, and 305e (*Reference 10*).

The containment isolation provisions for lines 120a, 121a, 310a, and 310b satisfy the explicit requirements of GDC 56.

6.2.4.4.2 *Class 3B Penetrations*

Incoming lines connected to closed systems outside containment are provided with at least one check valve or normally closed isolation valve located inside containment, a check valve located outside containment, or two remote-operated valves inside containment. The following lines are included in this subclass: 100, 101, 102, 105, 106, 109, 110a, 111, and 113.

The containment isolation provisions for lines 100, 102, 106, and 110a (the charging lines) differ from the explicit requirements of GDC 55 from the standpoint of valve number. A simple check valve in each line, inside containment, is identified as the containment isolation valve. These lines are connected to the chemical and volume control system outside the containment. Since the charging system does not have a required post-accident safety function, these lines would have to be automatically isolated to meet GDC 55 requirements.

Backfitting was not recommended by the NRC for these lines (*Reference 10*) because:

- a. The piping system is designed to operate at 2250 psi, significantly above the containment design pressure.
- b. The piping is Seismic Category I.
- a. The charging pumps are positive displacement pumps and, therefore, leakage back through the pumps is expected to be minimal.

The containment isolation provisions for lines 101 and 113, safety injection system, differ from the explicit requirements of GDC 55 from the standpoint of valve location and actuation. Each safety injection line is provided with a check valve outside containment. Additionally, each safety injection line is provided with two parallel motor-operated valves inside the containment that are remotely controlled from the control room. Since the safety injection system outside the containment is a closed, safety-grade system, two isolation boundaries are provided. Consequently, GDC 55 is met on some other defined basis (*Reference 9*).

ECP-15-000655 added relief valve 890 to provide over pressure protection to line 101 ("B" Safety Injection Discharge Header), as well as valves 891 and 892. Valve 891 provides a pressure boundary and safety related function to isolate flow to avoid flow diversion from line 113 ("A" SI Pump Discharge Header) and from line 101. Valve 892 provides a pressure boundary function and a flow path to relief valve 890. This modification does not degrade Ginna's containment isolation capabilities. The additional relief valve 890 will serve as a containment isolation valve inside containment. This isolation valve is relatively close to the corresponding penetration 101 and is equivalent to relief valve 887 with respect to containment isolation satisfying containment environmental conditions. This modification does not change this line's closed loop outside containment design. It is noted that containment isolation to P110 and P113 are enhanced by having an additional valve (891) that is locked closed.

ECP-16-000114 added relief valve 893 for overpressure protection of line 113 ("A" Safety Injection Discharge Header). The additional relief valve 893 serves as a containment isolation valve inside containment. This isolation valve is relatively close to the corresponding penetration 113, and the modification does not change this line's closed loop outside containment design.

The containment isolation provisions for lines 105 and 109 (containment spray pump discharge lines) differ from the explicit requirements of GDC 56 from the standpoint of valve number and type. A simple check valve is provided in each line outside containment, which is not an acceptable automatic isolation valve per current criteria. Although these lines have a post-accident safety function, they are open to the containment atmosphere and, therefore, the isolation provisions should satisfy GDC 56 on some other defined basis. Backfitting for these lines was not recommended based on the same criteria discussed for Class 2 lines 120b, 123a, and 305e (*Reference 10*).

The containment isolation provisions for line 111, the residual heat removal supply line, differ from the explicit requirements of GDC 55 from the standpoint of valve location and actuation. The valving arrangement for this line is the same as for line 140, with a single motor-operated valve located inside the containment that is remotely operated from the control room. There is no isolation valve outside the containment except for valve 959, which isolates a branch flow path. Since the system outside the containment is a closed, safety-grade system, it constitutes an appropriate isolation boundary in lieu of a valve in the line outside the containment. Also, the line has a post-accident safety function and automatic isolation of the line is not appropriate. Therefore, the valve location and provisions for line 111 satisfy GDC 55 on some other defined basis (*Reference 9*).

6.2.4.4.5 Class 4 Penetrations (Closed System, Missile Protected)

6.2.4.4.5.1 Applicable Lines

Normally operating incoming and outgoing lines, which are connected to a closed system inside the containment and protected against missiles throughout their length, were originally designed with at least one manual isolation valve outside containment.

GDC 57 applies to Class 4 lines. This criterion specifies the isolation provisions for closed systems inside the containment that are neither part of the reactor coolant pressure boundary nor connected directly to the containment atmosphere. For these closed systems to qualify as containment isolation boundaries, they must be safety-grade design since the containment isolation system is an engineered safety feature. Further guidance is provided by Standard Review Plan 6.2.4 in this regard. Closed systems must, in part, be protected against missiles and pipe whip, designated Seismic Category I, and classified Safety Class 2. Furthermore, GDC 57 specifies that a locked closed, remote manual, or automatic isolation valve must be provided outside the containment and that a simple check valve may not be used as the automatic isolation valve. The following lines are included in this class: 119, 123b, 124a, 125, 126, 127, 128, 130, 131, 201a, 201b, 206b, 207b, 209a, 209b, 308, 311, 312, 315, 316, 319, 320, 321, 322, 323, 401, 402, 403, and 404.

6.2.4.4.5.2 *Class 4 Evaluation*

The containment isolation provisions for lines 201a, 201b, 209a, 209b, 308, 311, 312, 315, 316, 319, 320, and 323 differ from the explicit requirements of GDC 57 from the standpoint of valve actuation. All of these service water (SW) lines are equipped with local manual isolation valves outside the containment and a closed system inside containment. GDC 57 requires that these penetrations be provided with an automatic isolation valve. However, these valves would only be needed for containment isolation if there was a significant breach of the closed system inside containment. Since the manual isolation valves are accessible following a design-basis LOCA, RG&E agreed to upgrade the existing valves. The allowable leakage rate for these valves is based on manufacturers' recommendations and not on 10 CFR 50, Appendix J, limits. The NRC accepted this configuration and testing methodology (*Reference 10*).

Penetration line 124a is the component cooling water supply to the excess letdown heat exchanger. This line is provided with a check valve and a closed system inside containment. However, the check valve is located less than 3 ft from containment. The NRC accepted similar penetration configurations during the SEP (see discussion for Class 2 lines 120b, 123a, and 305e). Therefore, GDC 57 is met on some other defined basis.

Penetration lines 401 and 402 are the main steam lines. Each line is provided with a main steam stop valve that is air-operated and capable of remote manual operation. Branch lines are normally isolated or can be closed by operators when required. Additionally, the main steam blowdown and sample lines (206b, 207b, 321, and 322) are provided with air-operated isolation valves. These valves are available to automatically or remote manually isolate the main steam blowdown lines. All six penetrations credit the steam generator tubes as a closed system inside containment. Therefore, the isolation provisions satisfy GDC 57 (*Reference 9*). The main steam lines can resist the postulated tornado missiles without loss of required safety function.

Penetration lines 403 and 404 are the feedwater lines. Each line is provided with a safety related remotely activated valve and a check valve in series. The safety-related valve was originally a manual valve that was up-graded by the plant Extended Power Uprate (EPU) program. Additionally, each feedwater line has two 3-in. auxiliary feedwater lines joining it

downstream of these two valves. Under accident conditions, the preferred auxiliary feedwater system is automatically initiated and providing water at a higher pressure than containment pressure. The isolation provisions of lines 403 and 404 differ from the explicit requirements of GDC 57 from the standpoint of valve type. Since the check valve in the feedwater lines outside the containment is not an appropriate automatic isolation valve, the original manual valve outside the containment would have to be upgraded to a remote manual isolation valve. Backfitting for lines 403 and 404 was not recommended for the same reasons discussed under valve location for Class 2 lines 120b, 123a, and 305e (*Reference 10*). Additionally, the feedwater lines can resist postulated tornado missiles without loss of required safety function.

As a result of the Extended Plant Update (EPU), it was required to provide a second automatic main feedwater isolation valve (MFIV) in each main feedwater line. The purpose of the second MFIV is to automatically isolate the main feedwater line following a steam line break in containment. To support this uprate requirement, the existing 14" manual valves (V3994 and V3995) downstream of the main feedwater check valves in penetration lines 403 and 404 were modified to include a remote acting actuator. Although these valves are now automatic valves and are located in the Class 2 portion of the main feedwater lines, they are not considered containment isolation valves (*Reference 44*).

Containment isolation provisions for penetration lines 119, 123b, 124c, 125, 126, 127, 128, 130, and 131, satisfy the requirements of GDC 57.

6.2.4.4.6 Class 5 Penetrations (Special Service)

6.2.4.4.6.1 Applicable Lines

Lines that penetrate the containment and which may be opened to the containment atmosphere, but which are normally closed during reactor operation, are provided with either two isolation valves in series, one isolation valve and one blind flange, one isolation valve and a closed system or two blind flanges. Normally one of these devices is located inside and the other is located outside the containment.

GDC 56 applies to Class 5 penetration lines. The following lines are included in this class: 2, 29, 103, 124b, 124d, 132, 141, 142, 202a, 202b, 203b, 203c, 204, 210, 300, 304a, 304b, 305a, 305c, 305d, 307, 309, 313, 317, 324, 332a, 332b, and 332d.

6.2.4.4.6.2 Class 5 Evaluation

The following penetration lines differ from the explicit requirements of GDC 56 from the standpoint of isolation boundary type: 2, 29, 204, 300, and 313. A blind flange inside or outside the containment is an acceptable isolation barrier in lieu of an isolation valve, if it is leak testable. Line 2, the steam generator inspection and maintenance cabling access penetration, is isolated by a double-gasketed, leak-testable flange both inside and outside containment.

Line 29, the fuel transfer tube, is isolated by a double-gasketed resilient seal flange located inside the containment that is leak testable. Line 313, leak test depressurization, is equipped with a leak testable blind flange inside the containment and a normally closed valve outside the containment. Lines 204 and 300, purge supply and purge exhaust respectively, are equipped with leak testable blind flanges with double O-rings inside containment.

Containment isolation provisions for lines 124b, 124d, 202a, 202b, 203b, 203c, 210, 304a,

304b, 305a, 305c, and 305d differ from the explicit requirements of GDC 56 from the standpoint of valve location. All of these lines have two normally closed isolation valves in series located outside the containment. This is acceptable, however, based on the discussion under Class 1 for lines 205, 206a, and 207a (*Reference 10*).

The containment isolation provisions for lines 141 and 142, containment sump recirculation system, differ from the explicit requirements of GDC 56 from the standpoint of valve actuation. Due to their post-accident safety function, a single motor-operated valve is used for each line. The valve actuation provisions are acceptable and meet GDC 56 on some other defined basis (*Reference 9*).

Containment isolation provisions for lines 332a, 332b, and 332d differ from the explicit requirements of GDC 56 from the standpoint of isolation boundary type. The isolation provisions for these hydrogen monitor instrumentation lines consist of a normally closed solenoid-operated valve and a closed system outside containment that is Seismic Category I and designed to withstand maximum containment accident pressure. The lines are 3/8-in. (O.D.) stainless steel tubing. GDC 56 is met on some other defined basis (Standard Review Plan 6.2.4, Item II.6) for these lines (*Reference 10*).

Containment isolation provisions for lines 103, 132, 307, 309, 317, and 324 satisfy the explicit requirements for GDC 56.

6.2.4.4.7 Special Cases

Containment penetrations 1000, personnel hatch, and 2000, equipment hatch, are not covered by the penetration classes discussed above, but are evaluated under containment isolation provisions. These penetrations are described in Section 3.8.1.5 and shown in Figures 3.8-30 and 3.8-31. These openings are provided with redundant closures and/or seals, and are closed during MODES 1 and 2. They are also leak testable.

6.2.4.4.8 Instrumentation and Controls Evaluation

The instrumentation and control aspects of the override of the containment ventilation isolation and other engineered safety feature actuation signals were also reviewed by the NRC against current review guidelines. The evaluation concluded (*Reference 13*) that the electrical, instrumentation, and control aspects of the override of the containment ventilation isolation and Engineered Safety Features Actuation System (ESFAS) signals are acceptable, except for a lack of adequate physical protection for some of the engineered safety features reset push buttons.

The review resulted in two design modifications. One was to remove the blocking mechanisms that existed in the actuation and reset logic of the containment ventilation isolation, containment isolation, and containment spray systems so that any actuating signal, manual or automatic, would cause the related system to perform its safety function at any time. The other modified the containment isolation and containment ventilation isolation valve control circuits to provide individual resetting of each isolation valve. Resetting a valve after automatic closure requires operation of a containment isolation or containment ventilation isolation reset switch on the main control board and a valve control reset switch located at the containment isolation valve reset panel.

6.2.4.4.9 Containment Purging During Normal Plant Operation

In response to NRC concerns relative to containment shutdown purge valve operability in the event of a LOCA, the purge valves at Ginna Station were analyzed and evaluated for stress. It was concluded that the valve assembly is able to meet ASME Code Service Label B allowables at a valve opening angle of 30 degrees. This determination was based on torque data reflecting a worst-case upstream elbow oriented 90 degrees out of plane with respect to the valve shaft, with the leading edge of the valve disk closing toward the outer wall of the elbow. The travel stops of the valves were adjusted to a maximum opening of 30 degrees.

Subsequently, the inboard containment isolation valves in the purge air supply and purge air exhaust lines were removed and replaced with blind flanges with double O-ring seals. The volume between the inboard flange and outboard isolation valve is leak testable. See Figures 6.2-51 and 6.2-58. The blind flange is securely bolted in place during MODES 1, 2, 3, and 4 and is only removed when required during MODE 5 (Cold Shutdown) and MODE 6 (Refueling) shutdowns. Since the blank flanges have a double seal, it is no longer necessary to rely on the outer 48-in. valves for containment isolation during power operation. Therefore, the 30-degree travel stops have been removed.

A 1500-cfm mini-purge system capable of purging containment during all modes of operation has been installed. The supply and exhaust systems include a 6-in. line penetrating containment at penetration 309 and 132, respectively, with air-operated butterfly valves inside and outside containment. The valves are capable of closing against 60 psig in 2 seconds or less measured from the time the actuator receives the isolation signal to the time the valves are fully closed. The outlet of the mini-purge supply line and the inlet to the mini-purge exhaust line, inside containment, are equipped with debris screens to prevent fouling of the isolation valves. See Figures 6.2-66 and 6.2-41. The 6-in. mini-purge exhaust valves are opened periodically, for about one-half hour at a time, to reduce containment pressure when it reaches approximately 0.5 psig.

The exhaust valves are administratively controlled so that the total time that they are opened is as low as reasonably achievable. The air-operated mini-purge containment isolation valves are manually controlled from the control room. They fail in the closed position on loss of instrument air or electrical power. They also close automatically on a containment ventilation isolation signal. The shutdown purge valves also receive an automatic closure signal from the containment ventilation isolation instrumentation. (See Sections 9.4.1.2.8 and 9.4.1.2.9 for descriptions of the containment shutdown purge and the containment mini-purge systems.)

The trip setpoints for containment ventilation isolation while purging are established to correspond to the limits of 10 CFR Part 20 for unrestricted areas. The setpoints are determined procedurally by calculating effluent monitor count rate limits, while taking into account appropriate factors for detector calibration, ventilation flow rate, and average site meteorology.

6.2.5 COMBUSTIBLE GAS CONTROL IN THE CONTAINMENT

The requirement for Ginna and other Plants to analyze postaccident hydrogen concentration was removed from 10CFR50.44. Subsequently, TSTF-447 allowed removal of hydrogen recombiners and monitors from Technical Specifications (see Section 6.2.5.1 and 6.2.5.2). Ginna implemented the change in amendment 90 to the Technical Specifications. Because the hydrogen analysis is no longer required, Ginna did not update the calculations for EPU,

therefore reference to specific quantities in this section are retained for historical purposes.

Following a loss-of-coolant accident (LOCA), hydrogen may be produced inside the reactor containment by reaction of the zircaloy/ZIRLO®/Optimized ZIRLO™ fuel cladding with water or steam, by radiolysis of water, and by corrosion of materials. In addition, hydrogen dissolved in the reactor coolant and contained in the pressurizer vapor space may be released to the containment atmosphere. To ensure that the containment hydrogen concentration is maintained at a level below the lower hydrogen flammability limit, a combustible gas control system is provided. This subsection describes the sources of post-LOCA hydrogen and the systems that are provided to control the buildup of hydrogen within the containment.

6.2.5.1 Introduction

The design for combustible gas control within containment provides the capability to (1) measure the hydrogen concentration in the containment, (2) ensure a mixed atmosphere, and (3) control the combustible gas concentration, thus satisfying the criteria of 10 CFR 50.44 and the GDC 41, 42, and 43 of Appendix A to 10 CFR 50 (see Section 3.1.2).

Two redundant hydrogen concentration monitoring devices can measure and record hydrogen concentration in containment continuously after a design-basis event. Each train includes a sample line from the containment, a local analyzer/control panel in the intermediate building discharging to a common sample return line to the containment, a remote monitoring/control panel in the relay room, and a recorder in the control room. The instrumentation is in compliance with the requirements of NUREG 0737, Item II.F.1.6, and Regulatory Guide 1.97. However, the hydrogen monitors were removed from the Ginna Technical Specifications by Amendment 90 (*Reference 44*). With the elimination of the DBLOCA hydrogen release, hydrogen monitors are no longer required to mitigate Design Basis Accidents and, therefore, the hydrogen monitors do not meet the definition of a safety-related component as defined in 10 CFR 50.2. RG 1.97 recommends classifying the hydrogen monitors as Category 1. RG 1.97, Category 1, is intended for key variables that most directly indicate the accomplishment of a safety function for DBA events and, therefore, are items usually addressed within Technical Specifications. As part of the rulemaking to revise 10 CFR 50.44, the NRC found that the hydrogen monitors no longer meet the definition of Category 1 in RG 1.97. The NRC further concluded that Category 3, as defined in RG 1.97, is an appropriate categorization for the hydrogen monitors because the monitors are required to diagnose the course of beyond DBAs. Hydrogen monitoring is not the primary means of indicating a significant abnormal degradation of the reactor coolant pressure boundary. Section 4 of Attachment 2 to SECY-00-0198, "Status Report on Study of Risk-Informed Changes to the Technical Requirements of 10 CFR Part 50 (Option 3) and Recommendations on Risk-Informed Changes to 10 CFR 50.44 (Combustible Gas Control)", found that the hydrogen monitors were not risk-significant. Therefore, the NRC found that hydrogen monitoring equipment requirements no longer meet any of the four criteria in 10 CFR 50.36(c)(2)(ii) for retention in Technical Specifications and, therefore, may be relocated to other licensee-controlled documents. Because the monitors are required to diagnose the course of beyond DBAs, Ginna was required to verify that it has, and make a regulatory commitment to maintain, a hydrogen monitoring system capable of diagnosing beyond DBAs.

The analyzers are capable of monitoring hydrogen content by volume from 0 to 10%, over a containment atmosphere range of -2 to 60 psig, 40° to 290°F, and relative humidity of 0 to 100%. Control power of 125-V dc is supplied by vital batteries. Control and instrument power of 120-V ac is supplied from dedicated constant voltage, 480-V/120-V, transformers

fed from the 480-V ac supplying the analyzer motors.

The hydrogen monitoring system is capable of operation during all post-accident conditions. During MODES 1 and 2, it is maintained in an isolated standby mode. On-line calibration capability is provided by two hydrogen tanks, one for each train, which are capable of delivering a known concentration of hydrogen gas to the monitoring system. Operability of these hydrogen monitors is addressed by Technical Requirements Manual (TRM) Section 3.3.6.

Containment hydrogen concentration can also be determined by the postaccident sampling system (see Section 9.3.2.3.2).

A mixed containment atmosphere is provided by the containment recirculation fan coolers (CRFC), each unit having the capability of recirculating approximately 40,000 ft³/min. The entire containment air volume recirculates four to five times each hour if at least two CRFC units are operating.

Combustible gas control is provided by the hydrogen recombiner system.

6.2.5.2 Hydrogen Recombiner System

6.2.5.2.1 Description

The hydrogen recombiner system is shown in Drawing 33013-1275, Sheets 1 and 2. The recombiner system consists of two full-rated subsystems, each capable of maintaining the post-LOCA containment hydrogen concentration below 4 volume percent. Each subsystem contains a combustor, fired by an externally supplied fuel gas, employing containment air as the oxidant. Hydrogen in the containment air is oxidized in passing through the combustion chamber. Hydrogen gas is also used as the externally supplied fuel so that the noncondensable combustion products that would cause a progressive rise in containment pressure are avoided. Oxygen gas is made up through a separate containment feed to prevent depletion of oxygen below the concentration required for stable operation of the combustor.

Each recombiner is equipped with an air supply blower to deliver primary combustion air and quench air which reduces the unit exhaust temperature, an ignition system, and associated monitoring and control instrumentation. The system is designed to operate at ambient steam overpressures corresponding to 0 to 5 psig in the containment and to withstand the design basis transient environment prior to operation.

The basis for the system design is the analysis presented in *Reference 14*. *Reference 14* contains the analysis of performance and sensitivity to changes in operating parameters based on proof tests performed on the prototype combustor.

In the event that hydrogen fuel is required following an accident, it would be brought to the site. Operation of the hydrogen recombiners would prevent hydrogen from reaching 4 volume percent in the containment when calculated by the more conservative (homogeneous) energy deposition model. Subsequent operation will require an average fuel consumption of 8100 scf per day over the next 60 days, and about half as much oxygen, to maintain the ambient hydrogen at 4 volume percent. Bulk gas could be delivered in trailer-mounted tubes at 60-80,000 scf per load, requiring about eight such deliveries of hydrogen during that 60-day period, and four similar deliveries of oxygen. Consumption would be more rapid than the average rate during the early phase and less rapid later, due to the decay of the radiolysis

source.

The hydrogen recombiners were removed from the Ginna Technical Specifications by Amendment 90 (Reference 44). The revised 10 CFR 50.44 no longer defines a DBLOCA hydrogen release, and eliminates requirements for hydrogen control systems to mitigate such a release. The installation of hydrogen recombiners and/or vent and purge systems required by 10 CFR 50.44(b)(3) was intended to address the limited quantity and rate of hydrogen generation that was postulated from a DBLOCA. The NRC found that this hydrogen release is not risk-significant because the DBLOCA hydrogen release does not contribute to the conditional probability of a large release up to approximately 24 hours after the onset of core damage. In addition, these systems were ineffective at mitigating hydrogen releases from risk-significant beyond design-basis accidents (DBAs). Therefore, the NRC eliminated the hydrogen release associated with a DBLOCA from 10 CFR 50.44 and the associated requirements that necessitated the need for the hydrogen recombiners and the backup hydrogen vent and purge systems. As a result, the NRC staff found that requirements related to hydrogen recombiners no longer meet any of the four criteria in 10 CFR 50.36(c)(2)(ii) for retention in Technical Specifications and the existing Technical Specifications requirements may, therefore, be eliminated.

6.2.5.2.2 Containment Venting

Venting of the containment atmosphere prior to accumulation of an explosive mixture of hydrogen in the containment would obviate dependence on the recombiner, provided the radioactive constituents of the atmosphere can be trapped or safely dispersed in the environment. An assessment of this procedure for use at the Ginna Station site leads to the following conclusions.

- A. Venting must be accomplished before 6% by volume hydrogen is accumulated. A higher concentration, if accidentally ignited, could result in dynamic overpressures capable of damaging the containment.
- B. Favorable meteorological conditions and/or protective action on behalf of the nearby population would reduce the possible dose from venting the airborne activity at 21 days to an acceptable level.

It is concluded that proper protection of the health and safety of the public is served by providing the recombiner system, thus avoiding the necessity of venting at any specific time. However, the alternative means exists of avoiding a serious hazard by controlled venting if for any reason the recombiners were not operable (*Reference 30*).

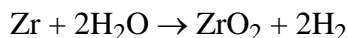
6.2.5.3 Design Evaluation

6.2.5.3.1 Hydrogen Production and Accumulation

Following a LOCA, hydrogen will be produced inside the reactor containment by reaction of the zirconium in fuel cladding with water and steam, by radiolysis of the core and sump solutions, by corrosion of aluminum and zinc, and by release of the hydrogen contained in the reactor coolant system. The design evaluation of these sources of hydrogen are described below. Major parameters and assumptions considered in the evaluation are summarized in Table 6.2-34.

6.2.5.3.1.1 *Zirconium-Water Reaction*

One of the major sources of hydrogen immediately following a LOCA is associated with the reaction of the zircaloy/ZIRLO®/Optimized ZIRLO™ fuel cladding with water. Zirconium reacts with steam according to the reaction:



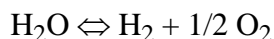
For each mole of zirconium that reacts, 2 moles of free hydrogen are produced.

The extent of the zirconium-water reaction depends on the effectiveness of the emergency core cooling system (ECCS). A value of 5% of the total zircaloy/ZIRLO®/Optimized ZIRLO™ cladding mass was considered as the amount of cladding that undergoes the zirconium-water reaction. This is consistent with the 10CFR50, paragraph 44 requirement that the fraction of the cladding considered for hydrogen generation be 5 times the extent of the maximum calculated reaction under 10CFR50, paragraph 46 (ECCS performance criteria assessment). Since this calculated amount cannot exceed 1% of the cladding in the active core region, the assumed value of 5% is a conservative and bounding value.

The zirconium cladding weight considered in the evaluation is included in Table 6.2-34, along with other plant parameters and assumptions that were considered.

6.2.5.3.1.2 *Radiolytic Hydrogen Generation*

Water radiolysis is a complex process involving reactions of numerous intermediates. However, the overall radiolytic process can be described by the reaction:



Water is decomposed into free hydrogen and oxygen by the absorption of energy emitted by fission products contained in the fuel and fission products intimately mixed with the LOCA water. The rate of production of gases from radiolysis of coolant solutions depends on (1) the amount and quality of radiation energy absorbed in the specific coolant solutions and (2) the net yield of hydrogen generated from the solution due to the absorbed radiation energy. Factors such as coolant flow rates and turbulence, chemical additives in the coolant, impurities, and coolant temperature can all exert an influence on the gas yields from radiolysis.

Post-accident conditions in the containment create two distinct radiolytic environments. One environment exists inside the reactor vessel, where radiolysis can occur when energy emitted by decaying fission products in the fuel is absorbed by the solution pumped through the reactor to cool the core. The other environment exists outside the reactor vessel, in the containment sump solution, where radiolysis can also occur when decay energy emitted by dissolved fission products is absorbed by the sump solution. The two basic differences between the core environment and the sump environment that affect the rate of hydrogen production are the rate of energy absorption and the type of flow regime.

The assumptions given in Regulatory Guide 1.7 were used to determine the fission product distribution after the accident. This distribution is assumed to occur instantaneously after the accident, and the hydrogen production is assumed to begin immediately. All noble gas activity is released from the fuel and is present in the containment atmosphere. Fifty percent of the halogens and one percent of the other fission products in the core are assumed to be released from the fuel and intimately mixed with the water in the reactor sump. The total decay energy ($\beta + \gamma$) in the sump is tabulated in Table 6.2-35 and the total decay energy ($\beta + \gamma$) available for deposition in the reactor core is included in Table 6.2-36. The values are based on operation with extended fuel cycles prior to the accident. All of the energy listed is considered to be absorbed by the sump solution, but only a fraction of the core energy is considered to be absorbed by the solution. In the core, all of the beta energy and 90% of the gamma energy is absorbed in the fuel/fuel cladding matrix, rather than the core solution.

Radiolytic decomposition of water is a reversible reaction. In the core, where the products of radiolysis are continuously flushed away by the circulation of cooling solutions, there is little chance for hydrogen and oxygen to accumulate. Consequently, recombination of hydrogen and oxygen is assumed not to occur because significant quantities of the two reactants are not available. The sump, however, is a relatively deep and static environment, where the products of radiolysis are removed by molecular diffusion. Experimental tests simulating sump conditions demonstrate that there is significant reverse reaction in the sump. Hence, there is an apparent reduction in the quantity of hydrogen produced per unit energy absorbed.

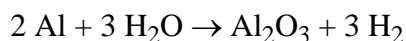
The results of Westinghouse and Oak Ridge National Laboratory studies indicated maximum hydrogen yields of 0.44 molecules per 100 eV for core radiolysis and 0.3 molecules per 100 eV for sump radiolysis. The results of these studies are published in *References 32, 33, and 34*. However, the design basis hydrogen formation rate of hydrogen is based on the conservative recommendations of Regulatory Guide 1.7; that is, a hydrogen yield of 0.5 molecules per 100 eV of energy absorbed for both core and sump radiolysis.

The major assumptions considered in the analysis of radiolysis from core and sump solutions are included in Table 6.2-34.

6.2.5.3.1.3 *Corrosion of Materials*

Following a LOCA, hydrogen may be produced inside the containment by corrosion of aluminum, which is found in the containment as aluminum metal components. A tabulation of the masses and areas of such components is included in Table 6.1-3.

Aluminum corrosion may be described by the overall reaction:



Three moles of hydrogen gas are produced for every two moles of aluminum that is oxidized. Approximately 20 scf of hydrogen gas are produced for each pound of aluminum corroded.

The time-temperature cycle (Table 6.1-5) considered in the calculation of aluminum corrosion is a representation of the postulated post-accident containment temperature transient. The corrosion rates at various temperatures are shown in Figure 6.1-9. The contribution of

corrosion to the hydrogen accumulation in the containment following the DBA was evaluated using these corrosion rates and the baseline aluminum inventory given in Table 6.1-3. No credit was taken for the protective shielding effects of insulation or enclosures; i.e., complete and continuous immersion in spray was assumed. Further, it was conservatively assumed that the long term corrosion rate of aluminum is maintained at 200 mils/yr; consistent with Regulatory Guide 1.7.

6.2.5.3.1.4 *Initial Inventory in the RCS and Pressurizer*

During normal operation of the plant, hydrogen is dissolved in the reactor coolant and is also contained in the pressurizer vapor space. Following a LOCA, this hydrogen is assumed to be immediately released to the containment atmosphere.

The initial inventory of hydrogen is based on the following:

- a. Reactor coolant hydrogen concentration of 50 cm³ (STP)/kg of coolant.
- b. Control banks of pressurizer heaters will modulate to control pressurizer heat losses to maintain constant pressurizer temperature and pressure.
- c. Bypass spray rate of 1 gal/min. (This value is conservative with respect to the actual plant flow which is higher because the valves are normally full open.)
- d. Normal liquid level in the pressurizer less the uncertainty or 45 percent.
- e. Pressurizer power operated relief valves (PORVs) closed.

6.2.5.3.2 Effect of Recombiners

The hydrogen production rates, accumulated hydrogen inside the containment, and the effect of recombiner initiation are presented in Figures 6.1-10, 6.1-12, and 6.2-80. Figure 6.1-10 presents the production rates from the time-dependent sources of hydrogen as a function of time after LOCA. Figure 6.1-12 shows the accumulated hydrogen inventory in the containment from the various sources with no recombiners in operation. Figure 6.2-80 shows the hydrogen concentration buildup for recombiner actuation at 24 hours, at 3.5 volume percent, and at 5.5 volume percent; along with the concentration buildup with no recombiner actuation. Figure 6.2-80 assumes the hydrogen recombiner runs continuously once started at a capacity of 100 cfm. The recombiner installed has a capacity of 350 cfm and is expected to be operated intermittently to maintain the containment hydrogen concentration less than 4 volume percent (*Reference 40*).

During the 2003 Refueling Outage it was discovered that the post-LOCA containment hydrogen generation analysis of record (which is the basis for Figures 6.1-10, 6.1-12, and 6.2-80) did not include the aluminum inventory associated with the reactor vessel insulation. A sensitivity analysis (*Reference 46*) was performed to determine the effect on the hydrogen generation rate. The analysis determined that the time to reach a hydrogen concentration of 3.5 volume percent was reduced from approximately 8.8 days to 6.2 days and the time to reach a concentration of 5.5 volume percent (without recombiner actuation) was reduced from approximately 26 days to 21 days. The figures have not been revised to include the effects of

the addition hydrogen generation and remain valid for showing the generation trend and effect of the recombiners.

The capability of a hydrogen recombiner to maintain the containment hydrogen concentration to well below the Regulatory Guide 1.7 flammability limit of 4 volume percent is illustrated in Figure 6.2-80. The figure shows that the recombiner is effective at immediately limiting the hydrogen concentration buildup in containment to the value at which it is initiated and reducing the concentration thereafter.

Values greater than 4 volume percent of hydrogen are presented for illustrative purposes only. There is no intent to operate at levels greater than 4 volume percent hydrogen. The design limit is 4 volume percent which is consistent with Regulatory Guide 1.7.

6.2.6 *CONTAINMENT LEAKAGE TESTING*

6.2.6.1 Containment Design Leakage

The containment leakage limiting boundary is provided in the form of a single steel liner in the vessel. Each system whose piping penetrates this boundary is designed to maintain isolation of the containment from the outside environment. Provision is made to periodically monitor leakage by pressurizing the penetrations and containment.

The leakage rate acceptance limits are specified in the Containment Leakage Rate Testing Program as required by Technical Specification 5.5.15. This program contains leakage limits for containment, isolation boundaries, containment air locks, and mini-purge valves with leakage limits applicable in MODES 1, 2, 3, and 4. These leakage rate limits, under hypothetical accident conditions and with minimum containment engineered safeguards operating, will maintain public exposure below 10 CFR 100 guidelines.

6.2.6.2 Tests and Inspections

6.2.6.2.1 Design Criteria

The following design criteria related to containment and containment penetration leakage testing were used during the licensing of Ginna Station. They represent the AIF revision of proposed criteria issued by the AEC for comment on July 10, 1967 (see Section 3.1.1). Conformance with 1972 General Design Criteria (GDC) of 10 CFR 50, Appendix A, as they apply to containment and penetration leakage testing is discussed in Section 3.1.2. These include GDC 16, 50, 52, 53, and 54. Leakage test compliance with 10 CFR 50, Appendix J, is discussed in Section 6.2.6.3.

6.2.6.2.2 Initial Containment Leakage Rate Testing

CRITERION: Containment shall be designed so that integrated leakage rate testing can be conducted at the peak pressure calculated to result from the design-basis accident after completion and installation of all penetrations and the leakage rate shall be measured over a sufficient period of time to verify its conformance with required performance (AIF-GDC 54).

After completion of the containment structure and installation of all penetrations and weld channels, an initial integrated leakage rate test was conducted at the peak calculated accident pressure (60 psig), maintained for a minimum of 24 hours, to verify that the leakage rate was no greater than 0.1 wt % of the containment volume per day.

The absolute method was used, and a leakage test was also performed at a reduced pressure of 35 psig. Containment recirculation fan cooler (CRFC) units operated continuously throughout the test to ensure good air mixing and temperature control. See Section 14.6.1.6.9 for additional details of these tests.

6.2.6.2.3 Periodic Containment Leakage Rate Testing (Type A Tests)

CRITERION: The containment shall be designed so that an integrated leakage rate can be periodically determined by test during plant lifetime (AIF-GDC 55).

A leak rate test at the peak calculated accident pressure using the same method as the initial leak rate test can be performed at any time during the operational life of the plant, provided the plant is not in operation and precautions are taken to protect instruments and equipment from damage.

Subsequent to the initial containment leakage test, periodic containment integrated leak rate tests (i.e. Type A) were conducted at reduced pressure (35 psig), with appropriate compensatory modifications to the leakage acceptance criteria. This practice is no longer allowed following implementation of 10 CFR 50, Appendix J, Option B with all future integrated leakage rate tests required to be performed at a pressure of 60 psig as specified in Technical Specifications 5.5.15 (*Reference 31*).

The containment liner has channels attached to each seam weld, which are sealed with a threaded plug. The channels were used during construction for acceptance testing of the liner welds. The NRC position is that all containment liner weld channels will be vented to the containment atmosphere during the containment integrated leak rate test unless it is demonstrated that the weld channels will maintain their integrity when subject to the loading conditions of a design-basis LOCA.

The weld channels were installed in conformance with ASTM A-36 and were attached with 1/4-in. continuous fillet welds that were tested by liquid penetrant and radiography. The channel weld connections were tested to 69 psig. Thus, there is reasonable assurance that the weld channels will maintain their integrity during the design-basis LOCA and leak rate tests are conducted without venting the weld channels.

6.2.6.2.4 Provisions for Testing of Type B Penetrations

CRITERION: Provisions shall be made to the extent practical for periodically testing penetrations which have resilient seals or expansion bellows to permit leaktightness to be demonstrated at the peak pressure calculated to result from occurrence of the design-basis accident (AIF-GDC 56).

A permanently piped monitoring system was provided such that all penetrations can be checked for leaktight integrity at any time throughout the operating life of the plant. Leakage

tests of Type B penetrations can be performed utilizing this monitoring system. Periodic leakage tests are performed in accordance with the Containment Leakage Rate Testing Program (Technical Specification 5.5.15).

Penetrations are designed with double seals so as to permit pressurization of the interior of the penetration whenever a leak test is required. The containment air locks (i.e., the equipment hatch and personnel hatch) are equipped with double-tongue, single-gasket seals with the space between the tongues connected to the pressurization system. The system utilizes a supply of clean, dry, compressed air, which will place the penetrations under an internal pressure as required for the test. Leakage from the monitoring system is checked by measurement of the integrated makeup air flow or by pressure decay. In the event excessive leakage is discovered, each penetration can then be checked separately at any time.

6.2.6.2.5 Provisions for Testing of Isolation Valves (Type C)

CRITERION: Capability shall be provided to the extent practical for testing functional operability of valves and associated apparatus essential to the containment function for establishing that no failure has occurred and for determining that valve leakage does not exceed acceptable limits (AIF-GDC 57).

Capability is provided to the extent practical for testing the functional operability of valves and associated apparatus during periods of reactor shutdown. The Type C tests for containment isolation valves are performed in accordance with Technical Specification 5.5.15, Containment Leakage Rate Testing Program.

6.2.6.3 Leakage Test Compliance with 10 CFR 50, Appendix J

In March 1978, the NRC staff issued Amendment No. 17 to the Provisional Operating License for Ginna Station (*Reference 16*). The amendment modified the Ginna Technical Specifications regarding containment testing for the purposes of clarification to satisfy the requirements of Appendix J to 10 CFR 50, and granted certain exceptions to the appendix. In May 1981, the NRC staff issued a safety evaluation report (*Reference 17*) that concluded that no additional exemptions from the requirements of Appendix J were necessary as a result of the evaluation, and with the amended Technical Specifications and the completion of certain modifications, Ginna Station was in compliance with Appendix J to 10 CFR 50 with the only exceptions being the exemptions granted in 1978.

As a result of the 1981 NRC evaluation, certain test connection configurations for lines inside containment were modified to meet the requirements of Appendix J for type C leakage tests. The modification involved installation of block valves and test connections for penetrations 107, 143, 202, 210, and 304. (See Table 6.2-29.)

Another modification resulting from the evaluation was to provide the capability to drain fluid away from certain containment isolation valves and ensure exposure of the valves to containment air test pressure. The following penetrations were thus modified:

- Makeup to pressurizer relief tank (penetration 121a).
- Reactor coolant system charging (penetration 100).

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- Reactor coolant system alternate charging (penetration 102).
- Reactor coolant pump A seal injection (penetration 106).
- Reactor coolant pump B seal injection (penetration 110a).
- Demineralized water (penetration 324).
- Pressurizer liquid and gas sample (penetrations 206a and 207a).
- Containment sump A (penetration 107).

On April 26, 1995, the NRC staff issued Amendment No. 59 to the Ginna Technical Specifications (*Reference 18*). The amendment supported an RG&E request for a one-time exemption from the requirements of 10 CFR 50, Appendix J, Section III.D.3. The NRC staff approved the exemption (*Reference 19*) that allowed type C leakage tests for 129 containment isolation valves to be waived during the 1995 MODE 6 (Refueling) outage and extended the required two-year testing interval by up to one month to allow postponement of the testing of these valves until the 1996 refueling) outage.

Subsequent to this, RG&E implemented 10 CFR 50, Appendix J, Option B (*Reference 31*) which specifies frequencies of Type A, B, and C tests based on previous test performance. Implementation of Option B voided all previous exemptions to Appendix J. The testing program used to meet Option B is contained in Technical Specification 5.5.15, Containment Leakage Rate Testing Program.

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**Table 6.2-1
SYSTEM PARAMETERS INITIAL CONDITIONS**

<u>Parameters</u>	<u>Value</u>
Core Thermal Power (MWt)	1811.0
RCS Total Flow Rate (lbm/sec)	18,000.0
Vessel Outlet Temperature ^a (°F)	615.8
Core Inlet Temperature ^a (°F)	544.2
Vessel Average Temperature ^a (°F)	580
Initial Steam Generator Steam Pressure (psia)	855.0
Steam Generator Design	BWI
SGTP (%)	0
Initial Steam Generator Secondary Side Mass (lbm)	108,548.0
Assumed Maximum Containment Backpressure (psia)	74.7
Accumulator	
Water volume (ft ³ per accumulator (minimum) ^b	1090
N ₂ cover gas pressure (psia) (minimum)	714.7
Temperature (°F)	125
SI Start Time. (sec) [total time from beginning of event, which includes the maximum delay from reaching the setpoint]	35.1
Auxiliary Feedwater Flow (gpm/steam generator) (Minimum Safeguards)	0
Auxiliary Feedwater Flow (gpm/steam generator) (Maximum Safeguards)	170
NOTES: Core thermal power, RCS total flow rate, RCS coolant temperatures, and steam generator secondary side mass include appropriate uncertainty and/or allowance.	

- a. RCS coolant temperatures include +4.0°F allowance for instrument error and deadband.
- b. Does not include accumulator line volume

Table 6.2-2
SAFETY INJECTION FLOW - MINIMUM SAFEGUARDS

<u>RCS Pressure (psia)</u>	<u>Total Flow (gpm)</u>
<u>Injection Mode (Reflood Phase)</u>	
14.7	1800
20	1776
40	1683
60	1580
80	<1466
100	1335
120	1170
140	820
214.7	600
314.7	600
414.7	600
514.7	600
<u>Recirculation Mode</u>	
<u>RCS Pressure (psia)</u>	<u>Total Flow (gpm)</u>
14.7	1000

Table 6.2-3
SAFETY INJECTION FLOW - MAXIMUM SAFEGUARDS

<u>RCS Pressure (psia)</u>	<u>Total Flow (gpm)</u>
<u>Injection Mode (Reflood Phase)</u>	
14.7	4452
20	4449.8
40	4441.6
60	4433.4
80	4425.2
100	4417
114.7	4411
120	4408.7
140	4400.1
175	4385
176	994.5
214.7	978
314.7	933<
414.7	885.5
514.7	838
Recirculation Mode	
RCS Pressure (psia)	Total Flow (gpm)
0	3000

Table 6.2-4
LOCA M&E RELEASE ANALYSIS - CORE DECAY HEAT FRACTION

<u>Time (sec)</u>	<u>Decay Heat Generation Rate (Btu/Btu)</u>
10	0.053876
15	0.050401
20	0.048018
40	0.042401
60	0.039244
80	0.037065
100	0.035466
150	0.032724
200	0.030936
400	0.027078
600	0.024931
800	0.023389
1000	0.022156
1500	0.019921
2000	0.018315
4000	0.014781
6000	0.013040
8000	0.012000
10,000	0.011262
15,000	0.010097
20,000	0.009350
40,000	0.007778
60,000	0.006958
80,000	0.006424
100,000	0.006021
150,000	0.005323
200,000	0.004847
400,000	0.003770
600,000	0.003201

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<u>Time (sec)</u>	<u>Decay Heat Generation Rate (Btu/Btu)</u>
800,000	0.002834
1,000,000	0.002580
2,000,000	0.001909
4,000,000	0.001355

**Table 6.2-5
DOUBLE-ENDED HOT LEG BREAK BLOWDOWN M&E RELEASE**

	<u>Break Path No. 1*</u>		<u>Break Path No. 2**</u>	
<u>Time</u>	<u>Flow</u>	<u>Energy</u>	<u>Flow</u>	<u>Energy</u>
<u>Seconds</u>	<u>Lbm/sec</u>	<u>Thousand Btu/ sec</u>	<u>Lbm/sec</u>	<u>Thousand Btu/ sec</u>
0.00	0.00	0.00	0.00	0.00
0.001	45,065.95	28,537.79	45,064.06	28,535.37
0.002	45,254.64	28,656.36	45,003.49	28,491.46
0.003	44,802.27	28,369.59	44,258.93	28,014.05
0.10	36,007.58	23,129.05	25,762.22	16,272.75
0.20	33,879.18	21,695.19	22,306.78	13,977.66
0.30	32,871.20	21,005.54	19,831.00	12,228.22
0.40	31,688.51	20,248.83	18,641.74	11,288.26
0.50	31,383.05	20,060.14	17,887.68	10,651.04
0.60	30,740.72	19,702.77	17,389.10	10,206.29
0.70	30,419.88	19,596.76	17,028.02	9876.33
0.80	29,887.60	19,399.52	16,719.93	9602.59
0.90	28,937.73	18,921.36	16,527.38	9413.51
1.00	27,857.05	18,362.40	16,333.08	9236.01
1.10	26,841.71	17,840.58	16,192.69	9099.69
1.20	25,842.44	17,325.35	16,092.64	8993.09
1.30	24,770.61	16,745.20	16,033.65	8915.39
1.40	23,626.93	16,100.72	16,013.76	8864.12
1.50	22,480.79	15,446.54	16,025.68	8834.91
1.60	21,319.25	14,727.98	16,054.92	8818.92
1.70	20,519.18	14,209.65	16,097.73	8814.06
1.80	19,864.61	13,952.94	16,143.57	8814.28
1.90	18,896.99	13,557.88	16,184.57	8815.36
2.00	17,968.39	13,007.59	16,212.26	8812.47
2.10	17,501.39	12,616.12	16,226.14	8805.34
2.20	17,338.50	12,341.13	16,229.06	8794.99
2.30	17,269.50	12,130.22	16,218.10	8779.66
2.40	17,170.36	11,938.65	16,190.57	8757.66
2.50	17,017.52	11,750.35	16,143.99	8727.32

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	<u>Break Path No. 1*</u>		<u>Break Path No. 2**</u>	
<u>Time</u>	<u>Flow</u>	<u>Energy</u>	<u>Flow</u>	<u>Energy</u>
<u>Seconds</u>	<u>Lbm/sec</u>	<u>Thousand Btu/ sec</u>	<u>Lbm/sec</u>	<u>Thousand Btu/ sec</u>
2.60	16,828.55	11,565.86	16,076.61	8687.53
2.70	16,642.76	11,395.89	15,984.10	8635.69
2.80	16,496.42	11,257.72	15,862.01	8569.17
2.90	16,396.55	11,159.79	15,687.98	8475.76
3.00	16,335.54	11,079.09	15,448.90	8348.41
3.10	16,328.66	11,016.89	15,197.54	8215.79
3.20	16,376.88	10,972.11	14,929.65	8075.65
3.30	16,453.94	10,939.35	14,645.51	7927.78
3.40	16,540.89	10,916.00	14,335.07	7766.43
3.50	16,618.81	10,891.66	13,982.75	7583.28
3.60	16,672.13	10,864.45	13,596.10	7382.25
3.70	16,661.20	10,818.05	13,189.52	7171.26
3.80	16,636.42	10,768.96	12,758.16	6948.23
3.90	16,621.33	10,725.79	12,339.57	6732.82
4.00	16,614.19	10,685.05	11,916.58	6516.41
4.20	16,599.60	10,603.37	11,096.60	6099.71
4.40	16,658.75	10,546.93	10,325.27	5709.52
4.60	13,159.46	8914.19	9586.29	5335.15
4.80	13,252.49	8968.85	8921.34	4999.69
5.00	13,038.55	8783.27	8340.47	4708.44
5.20	12,872.33	8653.67	7840.70	4459.64
5.40	12,784.91	8541.17	7410.77	4246.97
5.60	12,586.65	8384.69	7033.31	4060.76
5.80	12,432.88	8193.30	6698.89	3896.28
6.00	11,947.50	7918.23	6397.06	3748.33
6.20	11,733.43	7752.72	6121.30	3613.87
6.40	11,473.15	7565.67	5867.65	3490.95
6.60	11,175.75	7360.29	5627.36	3375.27
6.80	10,830.57	7133.20	5399.71	3266.67

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	<u>Break Path No. 1*</u>		<u>Break Path No. 2**</u>	
<u>Time</u>	<u>Flow</u>	<u>Energy</u>	<u>Flow</u>	<u>Energy</u>
<u>Seconds</u>	<u>Lbm/sec</u>	<u>Thousand Btu/ sec</u>	<u>Lbm/sec</u>	<u>Thousand Btu/ sec</u>
7.00	10,420.26	6880.95	5176.94	3161.20
7.20	9951.66	6609.98	4953.14	3055.69
7.40	9444.85	6329.28	4715.50	2944.70
7.60	8925.93	6036.97	4463.91	2831.15
7.80	8511.40	5804.66	4208.15	2721.00
8.00	7985.42	5542.55	3945.89	2612.73
8.20	7524.41	5299.57	3684.20	2507.79
8.40	7080.88	5064.44	3427.37	2405.78
8.60	6588.57	4813.83	3184.13	2308.48
8.80	5930.63	4517.78	2970.47	2222.37
9.00	5286.46	4250.25	2763.54	2125.98
9.20	4712.65	3985.92	2563.50	2034.82
9.40	4190.85	3741.39	2369.15	1956.14
9.60	3680.06	3510.26	2177.80	1880.52
9.80	3163.36	3217.52	2002.93	1805.41
10.00	2849.00	2911.10	1856.17	1738.34
10.20	2655.22	2704.79	1730.61	1676.33
10.40	2419.32	2526.89	1619.47	1623.95
10.60	2150.65	2342.90	1505.33	1581.30
10.80	1874.87	2133.68	1389.56	1536.35
11.00	1682.71	1960.73	1279.29	1484.78
11.20	1510.21	1754.79	1147.70	1385.59
11.40	1119.86	1323.91	1027.99	1260.71
11.60	723.34	881.31	949.33	1169.69
11.80	428.85	528.49	899.40	1110.52
12.00	243.21	299.47	818.88	1013.35
12.20	167.17	205.14	696.94	864.48
12.40	128.12	157.62	631.67	786.09
12.60	206.60	259.19	592.82	738.55

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	<u>Break Path No. 1*</u>		<u>Break Path No. 2**</u>	
<u>Time</u>	<u>Flow</u>	<u>Energy</u>	<u>Flow</u>	<u>Energy</u>
<u>Seconds</u>	<u>Lbm/sec</u>	<u>Thousand Btu/ sec</u>	<u>Lbm/sec</u>	<u>Thousand Btu/ sec</u>
12.80	316.54	401.96	525.76	655.78
13.00	315.80	400.24	454.88	568.30
13.20	0.00	0.00	396.05	495.59
13.40	0.00	0.00	346.78	434.54
13.60	0.00	0.00	292.99	367.73
13.80	359.79	459.35	245.98	309.23
14.00	447.50	567.97	200.75	253.03
14.20	451.17	570.49	198.97	251.27
14.40	488.74	591.81	207.43	262.04
14.60	454.11	563.16	202.38	255.54
14.80	419.04	524.48	197.48	249.51
15.00	472.75	572.98	188.11	237.89
15.20	462.52	570.70	178.68	226.04
15.40	448.68	551.14	174.09	220.46
15.60	444.78	546.51	170.35	215.86
15.80	162.53	208.06	171.75	217.66
16.00	71.22	92.48	142.85	181.14
16.20	0.00	0.00	0.00	0.00
* - Mass and Energy exiting the Reactor Vessel (RV) side of the break				
** - Mass and Energy exiting from the SG side of the break				

**Table 6.2-6
DOUBLE-ENDED HOT LEG BREAK - MASS BALANCE**

		<u>Time (Seconds)</u>		
		<u>0.00</u>	<u>16.20</u>	<u>16.20</u>
		<u>Mass (Thousand lbm)</u>		
Initial	In RCS and Accumulators	403.55	403.55	403.55
Added Mass	Pumped Injection	0.00	0.00	0.00
	Total Added	0.00	0.00	0.00
Total Available		403.55	403.55	403.55
Distribution	Reactor Coolant	265.54	45.90	52.05
	Accumulator	138.01	103.14	96.98
	Total Contents	403.55	149.04	149.04
Effluent	Break Flow	0.00	254.51	254.51
	ECCS Spill	0.00	0.00	0.00
	Total Effluent	0.00	254.51	254.51
Total Accountable		403.55	403.55	403.55

**Table 6.2-7
DOUBLE-ENDED HOT LEG BREAK - ENERGY BALANCE**

		<u>Time (Seconds)</u>		
		<u>0.00</u>	<u>16.20</u>	<u>16.20</u>
		<u>Energy (Million Btu)</u>		
Initial Energy	In RCS, Accumulators, SG	423.03	423.03	423.03
Added Energy	Pumped Injection	0.00	0.00	0.00
	Decay Heat	0.00	2.89	2.89
	Heat from Secondary	0.00	-1.04	-1.04
	Total Added	0.00	1.85	1.85
Total Available		423.03	424.88	424.88
Distribution	Reactor Coolant	153.86	11.71	12.26
	Accumulator	13.08	9.77	9.22
	Core Stored	15.53	6.61	6.61
	Primary Metal	84.70	79.61	79.61
	Secondary Metal	36.76	35.71	35.71
	Steam Generator	119.10	114.97	114.97
	Total Contents	423.03	258.38	258.38
Effluent	Break Flow	0.00	166.17	166.17
	ECCS spill	0.00	0.00	0.00
	Total Effluent	0.00	166.17	166.17
Total Accountable		423.03	424.55	424.55

Table 6.2-8
DOUBLE-ENDED PUMP SUCTION BREAK MIN SI
BLOWDOWN M&E RELEASE

	<u>Break Path No. 1*</u>		<u>Break Path No. 2**</u>	
<u>Time</u>	<u>Flow</u>	<u>Energy</u>	<u>Flow</u>	<u>Energy</u>
<u>Seconds</u>	<u>Mass lbm/sec</u>	<u>Thousand Btu/ sec</u>	<u>Mass lbm/sec</u>	<u>Thousand Btu/ sec</u>
0.00	0.00	0.00	0.00	0.00
0.0010	83,175.48	44,548.02	39,062.43	20,881.94
0.0020	39,933.04	21,348.04	39,667.09	21,204.27
0.0030	39,924.60	21,344.34	39,452.84	21,088.90
0.0041	39,918.34	21,341.56	39,222.19	20,964.59
0.10	39,695.23	21,291.82	19,431.21	10,377.68
0.20	40,333.13	21,778.56	21,147.26	11,302.95
0.30	41,066.08	22,365.74	22,326.07	11,940.55
0.40	41,839.35	23,023.09	22,700.47	12,146.71
0.50	42,591.52	23,711.47	22,462.94	12,025.49
0.60	43,047.34	24,251.68	22,065.12	11,818.89
0.70	43,043.49	24,525.09	21,665.72	11,610.46
0.80	42,387.09	24,400.93	21,347.73	11,444.85
0.90	41,364.85	24,042.74	21,117.38	11,324.86
1.00	40,229.43	23,599.14	20,914.73	11,218.28
1.10	39,016.10	23,089.96	20,691.79	11,099.55
1.20	37,652.75	22,475.61	20,422.02	10,954.55
1.30	36,070.01	21,709.81	20,108.43	10,785.69
1.40	34,297.03	20,815.93	19,875.46	10,660.12
1.50	32,899.07	20,160.55	19,660.73	10,544.56
1.60	31,826.62	19,732.84	19,386.24	10,396.52
1.70	30,676.21	19,278.53	19,050.08	10,214.82
1.80	29,305.13	18,701.94	18,692.09	10,021.20
1.90	27,569.23	17,903.00	18,325.42	9822.98
2.00	22,470.06	14,819.33	17,939.10	9614.21
2.10	18,720.51	12,575.23	17,539.53	9398.39

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CHAPTER 6 ENGINEERED SAFETY FEATURES

	<u>Break Path No. 1*</u>		<u>Break Path No. 2**</u>	
<u>Time</u>	<u>Flow</u>	<u>Energy</u>	<u>Flow</u>	<u>Energy</u>
<u>Seconds</u>	<u>Mass lbm/sec</u>	<u>Thousand Btu/ sec</u>	<u>Mass lbm/sec</u>	<u>Thousand Btu/ sec</u>
2.20	16,421.07	11,199.91	17,136.14	9181.22
2.30	14,597.80	10,052.62	16,841.65	9023.64
2.40	13,354.36	9263.90	16,631.92	8912.53
2.50	12,583.00	8780.69	16,457.27	8820.81
2.60	12,077.35	8467.88	16,063.86	8610.10
2.70	11,659.20	8209.13	15,604.64	8364.76
2.80	11,258.86	7965.20	15,272.09	8188.58
2.90	10,891.16	7754.20	15,015.15	8053.67
3.00	10,545.44	7569.29	14,786.97	7934.36
3.10	10,198.25	7390.20	14,571.60	7821.84
3.20	9859.84	7220.50	14,369.44	7716.53
3.30	9543.44	7065.19	14,185.70	7621.42
3.40	9260.64	6927.12	14,004.69	7527.71
3.50	9007.64	6801.49	13,828.29	7436.60
3.60	8790.78	6690.71	13,660.59	7350.40
3.70	8606.45	6593.97	13,761.05	7411.40
3.80	8449.48	6510.12	13,959.38	7523.67
3.90	8311.89	6434.84	13,947.46	7521.18
4.00	8186.32	6362.67	13,992.60	7550.45
4.20	7979.39	6238.35	13,988.55	7557.31
4.40	7808.32	6121.55	13,860.21	7495.85
4.60	7641.20	5994.20	13,569.64	7345.41
4.80	7470.29	5843.97	13,223.82	7165.37
5.00	7522.02	5833.76	12,835.33	6962.83
5.20	7723.41	6007.37	12,530.48	6806.97
5.40	7184.58	5961.29	11,981.65	6517.23
5.60	6511.91	5668.97	11,505.52	6266.97
5.80	6177.34	5440.99	11,061.16	6032.76
6.00	5960.94	5231.86	10,638.86	5810.50

GINNA/UFSAR
CHAPTER 6 ENGINEERED SAFETY FEATURES

	<u>Break Path No. 1*</u>		<u>Break Path No. 2**</u>	
<u>Time</u>	<u>Flow</u>	<u>Energy</u>	<u>Flow</u>	<u>Energy</u>
<u>Seconds</u>	<u>Mass lbm/sec</u>	<u>Thousand Btu/ sec</u>	<u>Mass lbm/sec</u>	<u>Thousand Btu/ sec</u>
6.20	5773.74	5022.08	10,210.77	5584.98
6.40	5581.24	4802.46	9775.14	5354.76
6.60	5386.10	4577.87	9349.66	5128.97
6.80	5196.85	4356.51	8953.32	4902.02
7.00	5000.82	4138.39	8602.72	4657.15
7.20	4797.66	3934.58	8319.49	4416.01
7.40	4643.68	3766.83	8405.85	4358.91
7.60	4528.48	3625.12	8216.58	4173.87
7.80	4429.52	3513.67	8201.29	4092.83
8.00	4321.62	3416.58	7780.23	3822.38
8.20	4206.07	3329.71	7719.67	3737.15
8.40	4084.48	3251.51	7129.47	3406.90
8.60	3956.94	3178.28	7117.78	3355.38
8.80	3822.59	3111.51	6684.48	3115.25
9.00	3679.26	3051.49	6471.72	2980.10
9.20	3529.91	2996.46	6249.86	2843.01
9.40	3371.36	2948.36	5870.35	2639.61
9.60	3201.45	2902.32	5567.23	2473.39
9.80	3017.05	2863.57	5349.51	2346.27
10.00	2822.39	2832.26	5095.43	2205.60
10.20	2606.62	2801.01	4796.79	2048.84
10.40	2305.51	2682.72	4445.58	1871.51
10.60	1968.32	2412.85	4048.16	1672.42
10.601	1967.69	2412.17	4047.53	1672.07
10.6015	1966.92	2411.32	4046.78	1671.68
10.602	1966.32	2410.68	4046.21	1671.37
10.80	1682.96	2081.74	3796.05	1525.70
11.00	1459.87	1812.59	3663.18	1420.33
11.20	1275.64	1588.04	3555.16	1327.50
11.40	1077.75	1345.70	3380.62	1221.25
11.60	907.04	1134.65	3055.35	1074.61
11.80	768.47	962.66	2752.42	949.44

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CHAPTER 6 ENGINEERED SAFETY FEATURES

	<u>Break Path No. 1*</u>		<u>Break Path No. 2**</u>	
<u>Time</u>	<u>Flow</u>	<u>Energy</u>	<u>Flow</u>	<u>Energy</u>
<u>Seconds</u>	<u>Mass lbm/sec</u>	<u>Thousand Btu/ sec</u>	<u>Mass lbm/sec</u>	<u>Thousand Btu/ sec</u>
12.00	644.61	808.35	2439.09	827.89
12.20	538.26	675.62	2112.21	704.41
12.40	438.53	550.87	1758.46	574.92
12.60	340.62	428.18	1377.99	441.62
12.80	238.97	300.67	969.51	305.42
13.00	136.63	172.17	542.90	169.11
13.20	28.42	35.95	135.48	42.09
13.40	0.00	0.00	0.00	0.00

* - Mass and Energy exiting the SG side of the break

** - Mass and Energy exiting the pump side of the break

Table 6.2-9
DOUBLE-ENDED PUMP SUCTION BREAK MIN SI
Reflood M&E Releases

	<u>Break Path No. 1*</u>		<u>Break Path No. 2**</u>	
<u>Time</u> <u>Seconds</u>	<u>Flow</u> <u>lbm/sec</u>	<u>Energy</u> <u>Thousand Btu/</u> <u>sec</u>	<u>Flow</u> <u>lbm/sec</u>	<u>Energy</u> <u>Thousand Btu/</u> <u>sec</u>
13.88	0.00	0.00	0.00	0.00
14.08	0.00	0.00	0.00	0.00
14.18	0.00	0.00	0.00	0.00
14.28	0.00	0.00	0.00	0.00
14.33	0.00	0.00	0.00	0.00
14.43	65.71	77.65	0.00	0.00
14.53	16.22	19.16	0.00	0.00
14.65	8.99	10.62	0.00	0.00
14.75	13.77	16.27	0.00	0.00
14.85	18.89	22.32	0.00	0.00
14.95	24.05	28.42	0.00	0.00
15.05	28.06	33.15	0.00	0.00
15.15	32.53	38.44	0.00	0.00
15.25	36.75	43.43	0.00	0.00
15.35	40.81	48.23	0.00	0.00
15.45	44.72	52.84	0.00	0.00
15.55	48.48	57.29	0.00	0.00
15.65	52.11	61.58	0.00	0.00
15.68	53.00	62.63	0.00	0.00
15.75	55.61	65.72	0.00	0.00
15.85	58.98	69.71	0.00	0.00
15.95	61.80	73.04	0.00	0.00
16.05	64.47	76.19	0.00	0.00
16.15	67.04	79.23	0.00	0.00
16.25	69.55	82.19	0.00	0.00
16.35	71.98	85.07	0.00	0.00
16.45	74.35	87.87	0.00	0.00

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CHAPTER 6 ENGINEERED SAFETY FEATURES

<u>Time Seconds</u>	<u>Break Path No. 1*</u>		<u>Break Path No. 2**</u>	
	<u>Flow lbm/sec</u>	<u>Energy Thousand Btu/ sec</u>	<u>Flow lbm/sec</u>	<u>Energy Thousand Btu/ sec</u>
17.45	95.22	112.55	0.00	0.00
18.46	112.74	133.28	0.00	0.00
19.48	127.55	150.81	0.00	0.00
20.14	136.83	161.80	0.00	0.00
20.49	141.38	167.18	0.00	0.00
21.49	168.49	199.29	861.25	109.54
22.58	266.13	315.17	2642.93	339.88
23.58	269.59	319.29	2684.11	347.20
24.58	266.24	315.30	2642.94	343.06
25.08	264.47	313.20	2621.05	340.79
25.58	262.71	311.11	2599.22	338.51
26.58	259.28	307.04	2556.10	334.01
27.58	255.98	303.11	2513.95	329.59
28.58	252.80	299.34	2472.88	325.29
29.58	249.76	295.71	2432.95	321.10
30.58	246.83	292.24	2394.16	317.02
31.58	244.03	288.91	2356.50	313.07
32.58	241.34	285.71	2319.94	309.23
33.58	238.75	282.65	2284.43	305.50
34.58	236.27	279.70	2249.95	301.87
35.58	244.96	290.02	2393.29	311.20
36.58	242.64	287.26	2361.07	307.81
36.68	242.41	286.99	2357.90	307.48
37.58	240.41	284.61	2329.74	304.52
38.58	238.25	282.05	2299.24	301.32
39.58	236.17	279.58	2269.55	298.20
40.58	234.17	277.20	2240.63	295.16
41.58	232.23	274.89	2212.44	292.20
42.58	230.35	272.66	2184.96	289.31

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CHAPTER 6 ENGINEERED SAFETY FEATURES

	<u>Break Path No. 1</u>		<u>Break Path No. 2</u>	
<u>Time Seconds</u>	<u>Mass lbm/sec</u>	<u>Energy Thousand Btu/ sec</u>	<u>Mass lbm/sec</u>	<u>Energy Thousand Btu/ sec</u>
43.18	229.25	271.36	2168.79	287.60
43.58	228.52	270.50	2158.15	286.48
44.58	170.14	201.25	1094.21	175.76
45.58	203.38	240.65	177.01	96.19
46.58	201.34	238.24	176.17	95.18
47.58	199.03	235.51	175.25	94.04
48.58	196.73	232.77	174.32	92.91
49.58	194.41	230.03	173.40	91.78
50.58	192.09	227.27	172.47	90.65
51.58	189.73	224.47	171.53	89.50
52.58	187.44	221.76	170.62	88.39
53.58	185.15	219.05	169.72	87.29
54.58	182.86	216.33	168.81	86.18
55.58	180.56	213.60	167.90	85.07
56.58	178.25	210.87	167.00	83.97
57.58	175.94	208.13	166.09	82.86
58.28	174.32	206.21	165.46	82.09
58.58	173.63	205.38	165.18	81.76
59.58	171.30	202.63	164.28	80.65
60.58	168.97	199.87	163.37	79.55
61.58	166.63	197.09	162.47	78.45
62.58	164.28	194.31	161.56	77.34
63.58	161.93	191.52	160.66	76.24
64.58	159.57	188.73	159.75	75.14
65.58	157.21	185.93	158.85	74.04
66.58	154.84	183.13	157.95	72.95
67.58	152.48	180.32	157.05	71.86
68.58	150.10	177.51	156.51	70.77
69.58	147.73	174.70	155.26	69.68

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	<u>Break Path No. 1</u>		<u>Break Path No. 2</u>	
<u>Time Seconds</u>	<u>Mass lbm/sec</u>	<u>Energy Thousand Btu/ sec</u>	<u>Mass lbm/sec</u>	<u>Energy Thousand Btu/ sec</u>
70.58	145.36	171.89	154.37	68.60
71.58	142.99	169.08	153.48	67.52
72.58	140.61	166.28	152.60	66.45
73.58	138.25	163.47	151.72	65.38
74.58	135.88	160.67	150.84	64.32
76.18	132.11	156.20	149.46	62.63
76.58	131.17	155.09	149.11	62.21
78.58	126.48	149.54	147.40	60.14
80.58	121.83	144.03	145.72	58.11
82.58	117.21	138.57	144.07	56.11
84.58	112.65	133.18	142.45	54.15
86.58	108.15	127.85	140.88	52.24
88.58	103.73	122.62	139.34	50.38
90.58	99.38	117.48	137.85	48.58
92.58	95.12	112.44	136.41	46.84
94.58	90.96	107.52	135.01	45.16
96.58	86.91	102.72	133.68	43.54
98.58	82.97	98.06	132.39	41.99
99.58	81.04	95.78	131.77	41.25
100.58	79.14	93.54	131.17	40.52
102.58	75.43	89.16	130.00	39.11
104.58	71.86	84.93	128.89	37.77
106.58	68.42	80.87	127.85	36.51
108.58	65.13	76.98	126.86	35.33
110.58	63.22	74.72	125.86	34.14
112.58	61.78	73.01	124.92	33.03
114.58	60.40	71.37	124.01	31.95
116.58	59.07	69.81	123.14	30.92
118.58	57.80	68.31	122.30	29.92

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CHAPTER 6 ENGINEERED SAFETY FEATURES

	<u>Break Path No. 1</u>		<u>Break Path No. 2</u>	
<u>Time Seconds</u>	<u>Mass lbm/sec</u>	<u>Energy Thousand Btu/ sec</u>	<u>Mass lbm/sec</u>	<u>Energy Thousand Btu/ sec</u>
120.58	56.59	66.87	121.49	28.97
122.58	55.43	65.51	120.71	28.05
124.58	54.33	64.20	119.96	27.17
126.58	53.28	62.97	119.25	26.32
128.58	52.29	61.80	118.56	25.51
130.58	51.35	60.69	117.91	24.74
132.58	50.47	59.64	117.29	24.01
133.18	50.21	59.33	117.11	23.80
134.58	49.63	58.65	116.70	23.31
136.58	48.84	57.71	116.14	22.65
138.58	48.10	56.84	115.60	22.02
140.58	47.40	56.02	115.10	21.42
142.58	46.75	55.25	114.62	20.85
144.58	46.14	54.53	114.17	20.32
146.58	45.58	53.86	113.74	19.81
148.58	45.05	53.23	113.34	19.34
150.58	44.56	52.65	112.96	18.89
152.58	44.10	52.11	112.60	18.47
154.58	43.67	51.61	112.26	18.07
156.58	43.28	51.14	111.95	17.70
158.58	42.92	50.71	111.65	17.35
160.58	42.58	50.32	111.38	17.02
162.58	42.28	49.96	111.12	16.71
164.58	42.00	49.63	110.88	16.43
166.58	41.75	49.33	110.65	16.16
168.58	41.51	49.06	110.44	15.91
170.58	41.30	48.81	110.24	15.68
172.58	41.11	48.58	110.06	15.47

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CHAPTER 6 ENGINEERED SAFETY FEATURES

	<u>Break Path No. 1</u>		<u>Break Path No. 2</u>	
<u>Time Seconds</u>	<u>Mass lbm/sec</u>	<u>Energy Thousand Btu/ sec</u>	<u>Mass lbm/sec</u>	<u>Energy Thousand Btu/ sec</u>
174.58	40.94	48.38	109.89	15.27
176.58	40.79	48.20	109.74	15.08
176.78	40.78	48.19	109.72	15.07
178.58	40.66	48.04	109.59	14.91
180.58	40.53	47.90	109.46	14.75
182.58	40.43	47.77	109.33	14.61
184.58	40.34	47.66	109.22	14.47
186.58	40.26	47.57	109.11	14.34
188.58	40.19	47.49	109.01	14.23
190.58	40.13	47.42	108.92	14.12
192.58	40.08	47.36	108.84	14.02
194.58	40.04	47.32	108.76	13.93
196.58	40.01	47.28	108.69	13.85
198.58	39.99	47.26	108.62	13.77
200.58	39.98	47.24	108.56	13.70
202.58	39.97	47.23	108.51	13.63
204.58	39.96	47.22	108.46	13.57
206.58	39.96	47.22	108.41	13.51
208.58	39.97	47.23	108.36	13.46
210.58	39.98	47.24	108.32	13.41
212.58	40.00	47.26	108.29	13.37
214.58	40.02	47.29	108.26	13.33
216.58	40.04	47.32	108.23	13.30
218.58	40.07	47.35	180.20	13.27
220.58	40.10	47.39	108.18	13.24
222.58	40.14	47.43	108.16	13.21
224.58	40.18	47.48	108.14	13.19
226.08	40.21	47.52	108.13	13.18

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CHAPTER 6 ENGINEERED SAFETY FEATURES

Table 6.2-10
DOUBLE-ENDED PUMP SUCTION BREAK
Min SI Principle Parameters During Reflood

<u>Time (sec)</u>	<u>Flooding Temp (°F)</u>	<u>Flooding Rate (in/sec)</u>	<u>Carryover Fraction</u>	<u>Core Height (ft.)</u>	<u>Downcomer Height (ft.)</u>	<u>Flow Fraction</u>	<u>Injection Total</u>	<u>Injection Accumulator</u>	<u>Injection Spill</u>	<u>Enthalpy Btu/lbm</u>
							<u>(Pounds mass per second)</u>			
13.4	157.0	0.000	0.000	0.00	0.00	0.500	0.0	0.0	0.0	0.0
14.1	156.3	22.870	0.000	0.57	1.07	0.000	3972.6	3972.6	0.0	94.77
14.28	155.9	25.183	0.000	0.98	1.15	0.000	3951.5	3951.5	0.0	94.77
14.33	155.8	24.975	0.000	1.08	1.16	0.000	3941.0	3941.0	0.0	94.77
15.7	156.1	2.590	0.292	1.50	3.88	0.518	3797.5	3797.5	0.0	94.77
16.5	156.3	2.533	0.391	1.61	5.51	0.554	3726.4	3726.4	0.0	94.77
20.1	158.3	2.831	0.600	2.00	12.90	0.595	3432.4	3432.4	0.0	94.77
22.6	159.6	4.087	0.667	2.25	15.82	0.722	3202.4	3202.4	0.0	94.77
24.6	160.7	3.931	0.691	2.46	15.83	0.722	3082.5	3082.5	0.0	94.77
25.1	161.0	3.888	0.695	2.51	15.83	0.721	3054.8	3054.8	0.0	94.77
30.6	164.4	3.570	0.716	3.01	15.83	0.712	2785.7	2785.7	0.0	94.77
34.6	166.9	3.422	0.721	3.33	15.83	0.706	2622.0	2622.0	0.0	94.77
35.6	167.6	3.500	0.724	3.41	15.83	0.713	2776.2	2575.0	0.0	93.12
36.7	168.3	3.467	0.725	3.50	15.83	0.712	2736.5	2535.1	0.0	93.10
43.2	172.6	3.304	0.728	4.00	15.83	0.704	2526.1	2323.8	0.0	92.95
44.6	173.5	2.735	0.719	4.10	15.83	0.639	1369.2	1163.7	0.0	91.36
50.6	177.7	2.911	0.725	4.51	14.90	0.672	204.1	0.0	0.0	72.03
58.3	183.5	2.665	0.722	5.00	13.72	0.668	204.6	0.0	0.0	72.03
67.6	191.1	2.367	0.718	5.55	12.52	0.662	205.1	0.0	0.0	72.03

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CHAPTER 6 ENGINEERED SAFETY FEATURES

<u>Time</u> <u>(sec)</u>	<u>Flooding</u> <u>Temp</u> <u>(°F)</u>	<u>Flooding</u> <u>Rate</u> <u>(in/sec)</u>	<u>Carryover</u> <u>Fraction</u>	<u>Core Height</u> <u>(ft.)</u>	<u>Downcomer</u> <u>Height</u> <u>(ft.)</u>	<u>Flow</u> <u>Fraction</u>	<u>Injection</u> <u>Total</u>	<u>Injection</u> <u>Accumulator</u>	<u>Injection</u> <u>Spill</u>	<u>Enthalpy</u> <u>Btu/lbm</u>
							<u>(Pounds mass per second)</u>			
76.2	198.3	2.092	0.713	6.00	11.61	0.653	205.5	0.0	0.0	72.03
88.6	208.6	1.713	0.704	6.57	10.65	0.634	206.0	0.0.0	0.0	72.03
99.6	217.3	1.412	0.695	7.00	10.12	0.609	206.4	0.0.0	0.0	72.03
116.6	228.5	1.105	0.683	7.55	9.76	0.577	206.6	0.0.0	0.0	72.03
133.2	237.2	0.953	0.677	8.00	9.70	0.574	206.6	0.0.0	0.0	72.03
154.6	246.5	0.834	0.673	8.52	9.87	0.574	206.5	0.0.0	0.0	72.03
176.8	254.5	0.775	0.673	9.00	10.21	0.577	206.5	0.0.0	0.0	72.03
202.6	262.5	0.749	0.676	9.53	10.70	0.582	206.5	0.0.0	0.0	72.03
226.1	268.9	0.743	0.681	10.00	11.17	0.587	206.5	0.0.0	0.0	72.03

Table 6.2-11
DOUBLE-ENDED PUMP SUCTION BREAK
Post Reflood M&E Release-Minimum Safeguards

<u>Time</u>	<u>Break Path No. 1*</u>		<u>Break Path No. 2**</u>	
<u>Seconds</u>	<u>Flow</u> <u>Mass lbm/sec</u>	<u>Energy</u> <u>Thousand Btu/</u> <u>sec</u>	<u>Flow</u> <u>Mass lbm/sec</u>	<u>Energy</u> <u>Thousand Btu/</u> <u>sec</u>
226.10	88.84	113.36	140.09	50.95
231.10	88.62	113.07	139.98	50.82
236.10	88.39	112.78	139.87	50.69
241.10	88.16	112.49	139.76	50.56
246.10	87.93	112.20	139.65	50.43
251.10	87.71	111.91	139.54	50.30
256.10	87.48	111.62	139.43	50.17
261.10	88.65	113.11	139.32	50.04
266.10	88.42	112.82	139.21	49.91
271.10	88.18	112.52	139.10	49.78
276.10	87.95	112.22	138.99	49.65
281.10	87.72	111.93	138.88	49.52
286.10	87.49	111.63	138.77	49.39
291.10	87.25	111.33	138.66	49.26
296.10	87.02	111.04	138.55	49.13
301.10	88.17	112.50	138.44	48.99
306.10	87.93	112.19	138.32	48.86
311.10	87.69	111.89	138.21	48.73
316.10	87.45	111.59	138.10	48.60
321.10	87.21	111.28	137.98	48.46
326.10	86.98	110.98	137.87	48.33
331.10	86.74	110.67	137.76	48.19
336.10	87.86	112.11	137.64	48.06
341.10	87.62	111.80	137.53	47.92
346.10	87.37	111.48	137.42	47.79
351.10	87.13	111.17	137.30	47.65

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<u>Time</u>	<u>Break Path No. 1*</u>		<u>Break Path No. 2**</u>	
<u>Seconds</u>	<u>Flow Mass lbm/sec</u>	<u>Energy Thousand Btu/ sec</u>	<u>Flow Mass lbm/sec</u>	<u>Energy Thousand Btu/ sec</u>
356.10	86.88	110.86	137.19	47.52
361.10	86.64	110.55	137.07	47.38
366.10	86.40	110.24	136.96	47.24
371.10	87.49	111.64	136.84	47.11
376.10	85.90	109.61	138.46	49.03
381.10	85.66	109.30	138.34	48.88
386.10	85.41	108.99	138.22	48.74
391.10	85.17	108.67	138.09	48.59
396.10	84.92	108.36	137.97	48.44
401.10	86.02	109.76	137.84	48.30
406.10	85.85	109.54	137.72	48.15
411.10	85.67	109.32	137.60	48.01
416.10	85.50	109.10	137.48	47.87
421.10	85.33	108.87	137.36	47.72
426.10	85.15	108.65	137.24	47.58
431.10	84.98	108.43	137.12	47.44
436.10	84.80	108.21	136.99	47.29
441.10	85.93	109.65	136.87	47.15
446.10	85.75	109.42	136.75	47.00
451.10	85.57	109.19	136.63	46.86
456.10	85.39	108.95	136.50	46.71
461.10	85.21	108.72	136.38	46.56
466.10	85.03	108.49	136.25	46.42
471.10	84.84	108.26	136.13	46.27
476.10	85.95	109.67	136.00	46.12
481.10	85.76	109.42	135.88	45.97
486.10	85.57	109.18	135.75	45.82
491.10	85.38	108.94	135.63	45.67
496.10	85.19	108.70	135.50	45.52

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CHAPTER 6 ENGINEERED SAFETY FEATURES

<u>Time</u>	<u>Break Path No. 1*</u>		<u>Break Path No. 2**</u>	
<u>Seconds</u>	<u>Flow Mass lbm/sec</u>	<u>Energy Thousand Btu/ sec</u>	<u>Flow Mass lbm/sec</u>	<u>Energy Thousand Btu/ sec</u>
501.10	85.00	108.46	135.37	45.37
506.10	86.08	109.83	135.24	45.22
511.10	85.88	109.58	135.12	45.07
516.10	84.43	107.73	136.63	46.86
521.10	84.24	107.48	136.49	46.70
526.10	84.05	107.24	136.35	46.53
531.10	85.11	108.59	136.22	46.37
536.10	84.91	108.34	136.08	46.21
541.10	84.71	108.09	135.94	46.05
546.10	84.51	107.83	135.80	45.88
551.10	84.31	107.58	135.66	45.72
556.10	84.11	107.33	135.53	45.56
561.10	85.15	108.64	135.39	45.39
566.10	84.94	108.38	135.25	45.22
571.10	84.73	108.12	135.11	45.06
576.10	84.53	107.85	134.96	44.89
581.10	84.32	107.59	134.82	44.73
586.10	84.11	107.32	134.68	44.56
591.10	85.11	108.60	134.54	44.39
596.10	84.90	108.33	134.40	44.22
601.10	84.69	108.06	134.25	44.05
606.10	84.50	107.82	134.11	43.88
611.10	84.31	107.58	135.53	45.56
616.10	84.12	107.34	135.38	45.38
621.10	83.93	107.10	135.23	45.20
626.10	83.74	106.85	135.07	45.02
631.10	83.55	106.61	134.92	44.84
636.10	84.54	107.87	134.77	44.66
641.10	84.34	107.61	134.61	44.48

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<u>Time</u>	<u>Break Path No. 1*</u>		<u>Break Path No. 2**</u>	
<u>Seconds</u>	<u>Flow Mass lbm/sec</u>	<u>Energy Thousand Btu/ sec</u>	<u>Flow Mass lbm/sec</u>	<u>Energy Thousand Btu/ sec</u>
646.10	84.14	107.36	134.46	44.30
651.10	83.94	107.10	134.30	44.11
656.10	83.74	106.85	134.15	43.93
661.10	84.70	108.07	133.99	43.74
666.10	84.49	107.80	133.84	43.56
671.10	84.28	107.54	133.68	43.37
676.10	84.07	107.27	133.52	43.19
681.10	83.86	107.00	134.86	44.77
686.10	83.64	106.73	134.69	44.57
691.10	83.43	106.46	134.52	44.37
696.10	83.22	106.19	134.36	44.17
701.10	83.01	105.91	134.19	43.97
706.10	83.91	107.07	134.01	43.77
711.10	83.69	106.79	133.84	43.57
716.10	83.47	106.50	133.67	43.36
721.10	83.24	106.22	133.50	43.16
726.10	84.12	107.34	133.33	42.96
731.10	83.89	107.04	133.15	42.75
736.10	82.56	105.34	134.42	44.25
741.10	83.42	106.45	134.24	44.03
746.10	83.19	106.14	134.05	43.81
751.10	82.95	105.84	133.87	43.59
756.10	82.71	105.54	133.68	43.37
761.10	83.55	106.60	133.49	43.15
766.10	83.30	106.28	133.30	42.93
771.10	83.05	105.97	133.11	42.70
776.10	83.86	107.00	132.92	42.48
781.10	83.60	106.67	132.73	42.25
786.10	82.29	105.00	133.93	43.67

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<u>Time</u>	<u>Break Path No. 1*</u>		<u>Break Path No. 2**</u>	
<u>Seconds</u>	<u>Flow Mass lbm/sec</u>	<u>Energy Thousand Btu/ sec</u>	<u>Flow Mass lbm/sec</u>	<u>Energy Thousand Btu/ sec</u>
791.10	83.08	106.00	133.73	43.43
796.10	82.81	105.67	133.52	43.19
801.10	82.55	105.33	133.32	42.95
806.10	83.33	106.32	133.11	42.70
811.10	83.06	105.99	132.91	42.46
816.10	82.80	105.65	132.70	42.22
821.10	83.55	106.60	132.49	41.97
826.10	83.27	106.25	132.28	41.72
831.10	82.99	105.89	133.42	43.06
836.10	82.71	105.54	133.19	42.80
841.10	82.43	105.18	132.97	42.54
846.10	83.13	106.07	132.75	42.27
851.10	82.84	105.70	132.52	42.01
856.10	83.51	106.56	132.30	41.74
861.10	83.20	106.16	132.07	41.47
866.10	81.93	104.54	133.14	42.74
871.10	82.58	105.37	132.90	42.45
876.10	82.27	104.97	132.66	42.17
881.10	82.89	105.77	132.41	41.88
886.10	82.56	105.35	132.17	41.59
891.10	83.16	106.11	131.92	41.29
896.10	81.89	104.48	132.94	42.50
901.10	82.47	105.23	132.67	42.19
906.10	82.11	104.78	132.41	41.87
911.10	82.66	105.48	132.14	41.56
916.10	82.29	105.00	131.87	41.24
921.10	82.81	105.67	131.61	40.92
926.10	82.42	105.17	132.56	42.05
931.10	82.03	104.67	132.28	41.71

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<u>Time</u>	<u>Break Path No. 1*</u>		<u>Break Path No. 2**</u>	
<u>Seconds</u>	<u>Flow Mass lbm/sec</u>	<u>Energy Thousand Btu/ sec</u>	<u>Flow Mass lbm/sec</u>	<u>Energy Thousand Btu/ sec</u>
936.10	82.50	105.27	131.99	41.37
941.10	82.95	105.84	131.70	41.03
946.10	82.52	105.29	131.41	40.69
951.10	82.08	104.74	132.30	41.75
956.10	82.48	105.24	131.99	41.38
961.10	82.02	104.66	131.68	41.01
966.10	82.38	105.12	131.37	40.64
971.10	81.90	104.50	132.22	41.64
976.10	82.21	104.90	131.88	41.25
981.10	82.50	105.27	131.55	40.85
986.10	82.75	105.59	131.21	40.45
991.10	82.19	104.88	132.00	41.39
996.10	81.63	104.16	131.64	40.96
1001.10	81.82	104.40	131.28	40.54
1006.10	81.99	104.61	132.02	41.41
1011.10	82.11	104.78	131.64	40.96
1016.10	82.20	104.89	131.25	40.50
1021.10	81.53	104.03	131.94	41.32
1026.10	81.56	104.06	131.53	40.83
1031.10	82.23	104.93	131.12	40.34
1036.10	81.46	103.94	131.75	41.10
1041.10	82.01	104.65	131.31	40.57
1046.10	81.82	104.40	130.87	40.05
1051.10	81.56	104.07	131.44	40.73
1056.10	81.88	104.48	130.97	40.17
1061.10	81.47	103.96	131.50	40.80
1066.10	81.60	104.12	130.99	40.20
1071.10	81.60	104.11	131.47	40.76
1076.10	82.02	104.66	130.94	40.13

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<u>Time</u>	<u>Break Path No. 1*</u>		<u>Break Path No. 2**</u>	
<u>Seconds</u>	<u>Flow Mass lbm/sec</u>	<u>Energy Thousand Btu/ sec</u>	<u>Flow Mass lbm/sec</u>	<u>Energy Thousand Btu/ sec</u>
1081.10	81.70	104.24	131.36	40.63
1086.10	81.73	104.28	130.79	39.96
1091.10	81.51	104.01	131.15	40.39
1096.10	81.98	104.60	130.55	39.67
1101.10	81.49	103.99	130.85	40.03
1106.10	81.38	103.84	131.11	40.33
1111.10	81.64	104.17	130.42	39.52
1116.10	81.53	104.03	130.61	39.75
1121.10	81.29	103.72	130.75	39.91
1126.10	81.05	103.42	130.83	40.01
1226.41	81.05	103.42	130.83	40.01
1226.51	48.40	60.47	158.14	47.44
1228.24	48.39	60.46	158.16	47.21
1331.55	48.39	60.46	158.16	47.21
1331.65	47.55	54.71	158.99	13.38
2652.00	41.21	47.42	165.33	14.52
2652.10	41.21	47.42	94.57	17.44
3600.00	38.19	43.94	97.59	17.99
3600.10	26.60	30.61	109.18	14.31
10,000.00	19.34	22.26	116.44	15.26
100,000.00	10.34	11.90	125.44	16.44
1,000,000.00	4.43	5.10	131.35	17.22
10,000,000.00	1.39	1.60	134.39	17.62

Notes:

* - Mass and Energy exiting the SG side of the break

* - Mass and Energy exiting the pump side of the break

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CHAPTER 6 ENGINEERED SAFETY FEATURES

Table 6.2-12
DOUBLE-ENDED PUMP SUCTION BREAK MIN SI
Mass Balance

		Time (sec)						
		0.00	13.40	13.40	226.08	1226.51	1331.55	3600.00
		Mass (Thousand Lbm)						
Initial	In RCS and ACC	403.55	403.55	403.55	403.55	403.55	403.55	403.55
Added Mass	Pumped Injection	0.00	0.00	0.00	39.32	245.95	267.64	669.09
	Total Added	0.00	0.00	0.00	39.32	245.95	267.64	669.09
Total Available		403.55	403.55	403.55	442.87	649.50	671.20	1072.65
Distribution	Reactor Coolant	265.54	17.09	41.74	75.46	75.46	75.46	75.46
	Accumulator	138.01	115.93	91.27	0.00	0.00	0.00	0.00
	Total Contents	403.55	133.02	133.02	75.46	75.46	75.46	75.46
Effluent	Break Flow	0.00	270.53	270.53	367.40	585.06	607.22	1008.67
	ECCS Spill	0.00	0.00	0.00	0.00	0.00	0.00	0.00
	Total Effluent	0.00	270.53	270.53	367.40	585.06	607.22	1008.67
Total Accountable		403.55	403.55	403.55	442.86	660.52	682.68	1084.14

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Table 6.2-13
DOUBLE-ENDED PUMP SUCTION BREAK ENERGY BALANCE - MINIMUM SAFEGUARDS

		Time (sec)						
		0.00	13.40	13.40	226.08	1226.51	1331.55	3600.00
		Energy (Million BTU)						
Initial Energy	In RCS, ACC, Steam Generator	423.03	423.03	423.03	423.03	423.03	423.03	423.03
Added Energy	Pumped Injection	0.00	0.00	0.00	2.83	17.72	19.28	55.80
	Decay Heat	0.00	2.42	2.42	15.53	57.67	61.51	131.13
	Heat from Secondary	0.00	8.61	8.61	8.61	8.61	8.61	8.61
	Total Added	0.00	11.03	11.03	26.98	83.99	89.40	195.54
Total Available		423.03	434.06	434.06	450.01	507.02	512.43	618.57
Distribution	Reactor Coolant	153.86	4.67	7.00	20.13	20.13	20.13	20.13
	Accumulator	13.08	10.99	8.65	0.00	0.00	0.00	0.00
	Core Stored	15.53	10.64	10.64	2.77	2.67	2.63	1.81
	Primary Metal	84.70	81.30	81.30	71.51	42.26	40.81	28.37
	Secondary Metal	36.76	36.31	36.31	34.82	20.09	19.13	13.37
	Steam Generator	119.10	127.95	127.95	121.46	66.36	63.32	43.74
	Total Contents	423.03	271.85	271.85	250.69	151.51	146.02	107.44
Effluent	Break Flow	0.00	161.89	161.89	198.94	358.19	361.27	507.22
	ECCS Spill	0.00	0.00	0.00	0.00	0.00	0.00	0.00
	Total Effluent	0.00	161.89	161.89	198.94	358.19	361.27	507.22
Total Accountable		423.03	433.75	433.75	449.63	509.70	507.29	614.66

Table 6.2-14
TABLE DELETED

Table 6.2-15
TABLE DELETED

Table 6.2-16
CONTAINMENT RESPONSE ANALYSIS PARAMETERS

<u>Parameter</u>	<u>Value</u>
Essential Service Water Temperature (°F)	85
RWST Water Temperature (°F)	104
Initial Containment Temperature (°F)	125
Initial Containment Pressure (psia)	15.7
Initial Relative Humidity (%)	20
Net Free Volume (ft ³)	1,000,000
Reactor Containment Fan Coolers	
Total	4
Analysis Minimum	2
Containment High Setpoint (psig)	6.0
Delay Time (sec) Without Offsite Power	44.0
Containment Spray Pumps	
Total	2
Analysis Minimum safeguards	1
Flowrate (gpm) Injection Phase (per pump)	a.)
Containment High High Setpoint (psig)	33.5
Delay Time (sec) Without Offsite Power (1 spray pump)	28.5
ECCS Recirculation Switchover, sec	2,652
Containment Spray Termination Time, (sec) Minimum Safeguards	2,652
Containment ECCS Sump Recirculation Flow, (gpm) Minimum Safeguards	1,000

a.) Containment Pressure (psig)	Containment Spray Flow Rate (gpm)
60	1179
55	1203
50	1227
45	1250
25	1338

Table 6.2-17
CONTAINMENT RECIRCULATION FAN COOLER HEAT REMOVAL CAPABILITY
AS A FUNCTION OF CONTAINMENT STEAM SATURATION TEMPERATURE

<u>Containment Temperature (°F)</u>	<u>Heat Removal Rate [Btu/sec] Per Reactor Containment Fan Cooler</u>
85	0
120	398
220	8,839
240	10,375
260	11,911
280	13,446
286	13,907

Table 6.2-18
LOCA CONTAINMENT RESPONSE ANALYSIS RECIRCULATION SYSTEM
ALIGNMENT PARAMETERS

Residual Heat Removal System	
RHR Heat Exchangers	
Modeled in analysis	1
Recirculation switchover time, sec	
Minimum safeguards	2,652
Flowrate, gpm	
Tubeside (includes 200 gpm pump recirculation)	1,200
Shellside	1,800
Component Cooling water Exchangers	
Modeled in analysis	1
Flowrate, gpm	
Shellside ^a	1,800
Tubeside ^a (service water)	5,000
Additional heat loads, BTU/hr	0.0

a. Minimum heat removal data representing 1 EDG

Table 6.2-19
CONTAINMENT STRUCTURAL HEAT SINK INPUT

<u>Heat Sink Number</u>	<u>Description</u>	<u>Area (ft²)</u>	<u>Material</u>	<u>Thickness (inches)</u>	<u>Thickness (ft)</u>
1	Insulated Containment Wall	36,285	SS	0.019	0.00158
			Gap	0.010	
			Insulation	1.250	0.1042
			Gap	0.010	
			Steel	0.375	0.03125
			Gap	0.021	
			Concrete	42.000	3.5
2	Uninsulated Containment Wall	12,370	Overcoat	0.008	
			Primer	0.002	
			Steel	0.375	0.03125
			Gap	0.021	
			Concrete	30.000	2.5
3	Basement Floor	6,576	Overcoat	0.005	
			Concrete	24.000	2.0
			Gap	0.021	
			Steel	0.250	0.0208
			Gap	0.021	
			Concrete	24.000	2.0
4	Wet Sump Wall A	8.2	Overcoat	0.004	
			Primer	0.002	
			Steel	0.250	0.0208
			Gap	0.021	
			Concrete	36.00	3.0
5	Dry Sump Wall A	2,052.8	Overcoat	0.004	

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<u>Heat Sink Number</u>	<u>Description</u>	<u>Area (ft²)</u>	<u>Material</u>	<u>Thickness (inches)</u>	<u>Thickness (ft)</u>
			Primer	0.004	
			Steel	0.250	0.0208>
			Gap	0.021	
			Concrete	36.0	3.0
6	Sump Floors	366	Overcoat	0.005	
			Concrete	24.000	2.0
			Gap	0.021	
			Steel	0.025	0.0208
			Gap	0.021	
			Concrete	12.000	1.0
7	Walls of Sump B	189	Overcoat	0.005	<
			Concrete	24.000	2.0
			Gap	0.021	
			Steel	0.250	0.0208
			Gap	0.021	
			Concrete	12.000>	1.0
8	Outer Refueling Cavity Wall	6,132	Overcoat	0.005	
			Concrete	35.280	2.94
9	Inner Refueling Cavity Wall	5,609	SS	0.250	0.0208
			Gap	0.021	
			Concrete	24.000	2.0
10	Bottom Refueling Cavity	1,143	SS	0.250	0.0208
			Gap	0.021	
			Concrete	48.0	4.0
11	Loop Compartments	18.846	Overcoat	0.005	
			Concrete	16.938	1.4115

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<u>Heat Sink Number</u>	<u>Description</u>	<u>Area (ft²)</u>	<u>Material</u>	<u>Thickness (inches)</u>	<u>Thickness (ft)</u>
12	Floor of Intermediate Level	9,672	Overcoat	0.005	
			Concrete	3.000	0.25
13	Operating Deck	15,570	Overcoat	0.005	
			Concrete	12.000	1.0
14	Thick Crane Structure	7,225	Overcoat	0.004	
			Primer	0.002	
			Steel	0.750	0.0625
15	Crane Structure	3,374 >	Overcoat	0.004	
			Primer	0.002	
			Steel	0.415	0.03455
16	I-Beam	7,678	Overcoat>	0.004	
			Primer	0.002	
			Steel	0.260	0.0217
17	Thick I-Beam	5,536	Overcoat	0.004	
			Primer	0.002	
			Steel	0.703	0.0586
18	Crane Support	342	Overcoat	0.004	
			Primer	0.002	
			Steel	2.000	0.16667
19	Crane Beams	236	Overcoat	0.004	
			Primer	0.002	
			Steel	1.440	0.12
20	Grating and Misc	14,000	Overcoat	0.004	
			Primer	0.002	
			Steel	0.062	0.005208

Table 6.2-20
MATERIAL PROPERTIES FOR CONTAINMENT STRUCTURAL HEAT SINKS

<u>Material</u>	<u>Conductivity (Btu/hr-ft-°F)</u>	<u>Specific Heat (Btu/lbm-°F)</u>
Concrete	0.81	31.5
Carbon Steel	28.0	54.4
Insulation	0.0208	1.11
Stainless Steel	8.8	54.6
Organic Coating	0.1	20.0
Inorganic Primer	1.0	20.0
Air (gap)	0.0174	0.241

Table 6.2-21
DOUBLE-ENDED HOT LEG BREAK SEQUENCE OF EVENTS

<u>Time (sec)</u>	<u>Event Description</u>
0.00	Break Occurs, Reactor Trip, SG Throttle valve Closure and Loss of Offsite Power are assumed
2.87	Low-Pressurizer Pressure SI Setpoint (1,715 psia) Reached
6.38	Broken Loop Accumulator Begins Injection Water
6.43	Intact Loop Accumulator Begins Injection Water
16.01	Peak Temperature Occurs (280.42°F)
16.01	Peak Pressure Occurs (54.25 psig)
16.20	End of Blowdown Phase
30.00	Transient Modeling Terminated

Table 6.2-22
DOUBLE-ENDED PUMP SUCTION BREAK SEQUENCE OF EVENTS
(Minimum Safeguards)

<u>Time (sec)</u>	<u>Event Description</u>
0.0	Break Occurs, Reactor Trip, SG Throttle Valve Closure and Loss of Offsite Power are Assumed
0.39	Containment HIGH Pressure Setpoint (20.7 psia; 6.0 psig;) Reached
3.09	Low Pressurizer Pressure SI Setpoint (1,715 psia) Reached (Safety Injection Begins coincident with Low Pressurizer Pressure SI Setpoint)
4.21	Containment HIGH-HIGH Pressure Setpoint (33.5) psig; Analysis Value) Reached
6.47	Broken Loop Accumulator Begins Injecting Water
6.55	Intact Loop Accumulator Begins Injecting Water
13.4	End of Blowdown Phase
13.4	Accumulator Mass Adjustment for Refill Period
15.09	Feedwater Isolation Valves Closed
32.79	Containment Spray Pump (RWST) Begins
35.09	Pumped Safety Injection Begins (Includes 32 Second Diesel Delay)
43.58	Broken Loop Accumulator Water Injection Ends
44.43	Containment Fan Coolers Actuate
44.98	Intact Loop Accumulator Water Injection Ends
226.08	End of Reflood
1,131.10	Containment Peak Pressure and Temperature Occurs (55.42 psig; and 284.95 °F)
1,220.06	M&E Release Assumption: Broken Loop Steam Generator (SG) Equilibration when the Secondary Temperature is the Saturation (T_{SAT}) At Containment Design Pressure 74.7 psia
1226.51	M&E Release Assumption: Broken Leg SG Equilibration at Containment Pressure of 64.0 psia
1228.63	M&E Release Assumption: Intact Loop SG Equilibration When the Secondary Temperature is the Saturation (T_{SAT}) at Containment Design Pressure of 74.7 psia

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<u>Time (sec)</u>	<u>Event Description</u>
1,331.55	M&E Release Assumption: Intact Loop SG Equilibration at Containment Pressure of 54.7 psia
2,652.0	Switchover to Cold Leg Recirculation Begins
2,652.0	Containment Spray Terminated
2.592E+6	Transient Modeling Terminated

**Table 6.2-23
LOCA CONTAINMENT RESPONSE RESULTS**

<u>Case</u>	<u>Peak Press. @ Time</u>	<u>Peak Temp. @ Time</u>	<u>Peak Press. (psig) @ 24 hours</u>	<u>Peak Temp. (°F) @ 24 hours</u>
DEHL	54.25 psig @ 16.01 sec	280.42 °F @ 16.01 sec	--	--
DEPS-Minimum Safeguards	55.42 psig @ 1,220 sec.	284.96°F @ 1,220 sec.	7.52	160.99
Containment Pressure - Acceptance Limits				
	Peak Pressure	Pressure @ 24 hours		
Pressure	60 psig	See UFSAR Figure 6.1-2		
Containment Temperature - Acceptance Limits				
	Peak Temperature	Temperature @ 24 hours		
Temperature	286°F	152°F (UFSAR Figure 6.1-1)		

Table 6.2-24
INITIAL CONDITIONS AND MAJOR ASSUMPTIONS FOR THE STEAMLINE
BREAK MASS AND ENERGY RELEASE MODEL (LIMITING CONTAINMENT
PRESSURE CASE)

<u>Parameter</u>	<u>Value</u>
Core power	70%
T _{avg}	571.3°F
Pressure	2250 psia
Steam generator mass, faulted / intact	104,536lb/104,536 lb
Steam generator pressure	896.6 psia
Auxiliary feedwater	
faulted loop	865 gpm
intact loop	235 gpm
Break size	1.4 ft ²
Low steamline pressure setpoint	372.7 psig
Low-Low SG Level	40 %
TDAFW started when faulted loop reaches low-low level with uncertainty	

**Table 6.2-25
MAJOR CONTAINMENT ANALYSIS ASSUMPTIONS**

<u>Initial Conditions</u>	<u>Value</u>						
Initial pressure	15.7 psia						
Initial temperature	125°F						
Initial humidity	20%						
Containment volume	1,000,000 ft ³						
Containment Recirculation Fan Coolers (CRFC)							
Containment safety injection analysis setpoint	6 psig						
CRFC delay with offsite power	34 seconds						
CRFC delay without offsite	power44 seconds						
Heat removal rates per CRFC as a function of containment steam saturation temperature							
Containment steam saturation Temperature, °F	85	120	220	240	260	280	286
Heat removal rate, Btu/sec	0	398	8,839	10,375	11,911	13,446	13,907
Containment Spray System							
Flow rate per containment spray pump	a.)						
Refueling water storage tank (RWST) water temperature	104°F						
Pressure setpoint (High-High)	33.5 psig						
Spray delay (2 sprays)	26.8 seconds						
Spray delay (1 spray)	28.5 seconds						

a.) Containment Pressure (psig)	Containment Spray Flow Rate (gpm)
60	1179
55	1203
50	1227
45	1250
25	1338

Table 6.2-26
SEQUENCE OF EVENTS
STEAMLINE BREAK, VITAL BUS FAILURE

<u>Event</u>	<u>Time (sec)</u>
Break occurs	0.0
Low steamline pressure SI setpoint reached>	<0.05
High containment pressure reached	2.0
Reactor trip	2.0
Main feedwater pumps trip	2.0
MFRV closes	12.0
Auxiliary feedwater (AFW) starts	25.0
Containment Recirculation Fan Cooling (CRFC) starts	36.0
Hi-Hi containment pressure reached	39.6
Low-Low SG Level Reached in Faulted SG and Turbine Driven Auxiliary feedwater (TDAFW) initiated	50.0
Containment spray starts	68.2
Auxiliary feedwater (AFW) terminated to faulted Steam Generator (SG)	600
Containment pressure peaks	612
Break releases stop	714.0

Table 6.2-27
CONTAINMENT SPRAY PUMP DESIGN PARAMETERS

Quantity	2
Design pressure, psig	300
Design temperature, °F	300
Design flow rate, gpm	1200
Design head, ft	475

Table 6.2-28
SINGLE FAILURE ANALYSIS - CONTAINMENT SPRAY SYSTEM

<u>Component</u>	<u>Malfunction</u>	<u>Comments and Consequences</u>
Spray nozzles	Clogged	Large number of nozzles (90 on each of two pairs of headers) ensures that clogging of significant number of nozzles is not credible.
Spray pump	Isolation valve	Two valves in parallel; one left closed required to open.
Spray pump	Fails to start	Two pumps provided; one required to start.

Valves^a

- a. See Table 6.3-9 for discussion of valve failure modes.

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Table 6.2-29
CONTAINMENT PIPING PENETRATIONS AND ISOLATION BOUNDARIES

Note: *Legend for symbols on Table 6.2-31

<u>System</u>	<u>Penetration No.</u>	<u>Boundary EIN</u> *	<u>UFSAR Figure</u>	<u>Notes (Notes only used to supplement Section 6.2.4.4.) (Notes listed on Table 6.2-30)</u>	<u>Position Indication In Control Room</u> *	<u>Normal position (MODES 1, 2, 3, 4)</u> *	<u>Immediate Postaccident Position (Refers to position immediately following receipt of containment isolation signal and containment ventilation isolation signal.)</u> *
Steam generator - inspection/maintenance	2	SPP03 SPP04	6.2-13 6.2-13	1, 2 1, 2	NA NA	C C	C C
Fuel transfer tube	29	SACO5 8152	6.2-13a 6.2-13a	2, 3 ---	NA No	C C	C C
Charging line to B loop	100	370B CLOC	6.2-14 6.2-14	44 4	NA NA	O C	C C
Safety injection pump 1B discharge	101	870B 889B CLOC 12407 PT-923 885B 2817C 2817J 898C 898E 898I Blind Flange 9082 890	6.2-15 6.2-15 6.2-15 6.2-15 6.2-15 6.2-15 6.2-15 6.2-15 6.2-15 6.2-15 6.2-15 6.2-15 6.2-15	42 42 5 42 8, 42 42 42 38, 42 39, 42 42 42 42 42 42	NA NA NA No NA No NA NA No No No No No	C C C C NA O C C C LC C C LC C	O O C C NA O C C C LC C C LC C
Alternate charging to A cold leg	102	383B CLOC	6.2-16 6.2-16	44 4	NA NA	C C	C C
Construction fire - service water	103	Welded cap 5129	6.2-17 6.2-17	7 ---	NA No	C LC	C LC

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<u>System</u>	<u>Penetration No.</u>	<u>Boundary EIN</u> *	<u>UFSAR Figure</u>	<u>Notes (Notes only used to supplement Section 6.2.4.4.) (Notes listed on Table 6.2-30)</u>	<u>Position Indication In Control Room</u> *	<u>Normal position (MODES 1, 2, 3, 4)</u> *	<u>Immediate Postaccident Position (Refers to position immediately following receipt of containment isolation signal and containment ventilation isolation signal.)</u> *
Containment spray pump 1A	105	862A CLOC 2829 869A 2856 2825 2825A 859A 859B 868C 868E 869E	6.2-18 6.2-18 6.2-18 6.2-18 6.2-18 6.2-18 6.2-18 6.2-18 6.2-18 6.2-18 6.2-18 6.2.18	44 8 23 9,44 9,44 44 44 10 10 9,44 9,40,44 41,44	NA NA No No No No No No No No No No	C C LC C C LC C LC LC LC C C	O C LC C C LC C LC LC LC C C
Reactor coolant pump A seal water inlet	106	304A CLOC	6.2-19 6.2-19	44 4	NA NA	O C	C C
Sump A discharge to waste holdup tank	107	1723 1728	6.2-20 6.2-20	43 43	Status Status	O O	C C
Reactor coolant pump seal water return line and excess letdown to VCT	108	313 CLOC	6.2-21 6.2-21	44 11	Both NA	O C	C C

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<u>System</u>	<u>Penetration No.</u>	<u>Boundary EIN</u> *	<u>UFSAR Figure</u>	<u>Notes (Notes only used to supplement Section 6.2.4.4.) (Notes listed on Table 6.2-30)</u>	<u>Position Indication In Control Room</u> *	<u>Normal position (MODES 1, 2, 3, 4)</u> *	<u>Immediate Postaccident Position (Refers to position immediately following receipt of containment isolation signal and containment ventilation isolation signal.)</u> *
Containment spray pump 1B	109	862B CLOC 2830 869B 2858 2826 2826A 859A 859B 868D 868E 869E	6.2-22 6.2-22 6.2-22 6.2-22 6.2-22 6.2-22 6.2-22 6.2-22 6.2-22 6.2-22 6.2-22 6.2-22	44 8 23 9,44 9,44 44 44 10 10 9,44 9,40,44 41,44	NA NA No No No No No No No No No No	C C LC C C LC C LC LC LC C C	O C LC C C LC C LC LC C C C
Reactor coolant pump B seal water inlet	110a	304B CLOC	6.2-23 6.2-23	44 4	NA NA	O C	C C
Safety injection test line	110b	879	6.2-15	12, 42	No	LC	LC
Residual heat removal to B cold leg	111	720 2840 2847 2848 371 959 CLOC 8730	6.2-24 6.2-24 6.2-24 6.2-24 6.2-25 6.2-24 6.2-24 6.2-24	13, 14 --- --- --- 36 37 15, 36 ---	R/G No No No Both Status NA No	C C C C C O C C	C C C C C C C C

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<u>System</u>	<u>Penetration No.</u>	<u>Boundary EIN</u> *	<u>UFSAR Figure</u>	<u>Notes (Notes only used to supplement Section 6.2.4.4.) (Notes listed on Table 6.2-30)</u>	<u>Position Indication In Control Room</u> *	<u>Normal position (MODES 1, 2, 3, 4)</u> *	<u>Immediate Postaccident Position (Refers to position immediately following receipt of containment isolation signal and containment ventilation isolation signal.)</u> *
Letdown to - nonregenerative heat exchanger	112	200A 200B 202 203 CLOC 371 427	6.2-25 6.2-25 6.2-25 6.2-25 6.2-24 6.2-25 6.2-25	16,44 16,44 16,44 44 36 36,44 17	R/G R/G R/G NA NA Both R/G	O/C O/C C C C O O	C C C C C C C
Safety injection pump 1A discharge	113	9081 870A 889A CLOC 12406 PT-922 Cap (PT-922) 885A 898C 898D	6.2-15 6.2-15 6.2-15 6.2-15 6.2-15 6.2-15 6.2-15 6.2-15 6.2-15 6.2-15	42 42 42 5 42 42 6, 42 42 39, 42 42	No NA NA NA No NA NA No No No	LC C C C C NA C O C LC	LC O O C C NA C O C LC
Standby auxiliary feed-water (SAFW) line to steam generator 1A	119	9704A 9723 CLIC	6.2-26 6.2-26 6.2-26	--- --- 19	R/G No NA	O LC C	O LC C
Nitrogen to accumulators	120a	846 8623	6.2-27 6.2-27	--- ---	Both NA	C O/C	C C
Pressurizer relief tank to gas analyzer	120b	539 546	6.2-28 6.2-28	--- ---	Status No	C O	C O
Makeup water to pressurizer relief tank	121a	508 529	6.2-29 6.2-29	--- ---	Both NA	C O/C	C C
Nitrogen to pressurizer relief tank	121b	528 547	6.2-30 6.2-30	--- 20	NA No	C LC	C LC

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<u>System</u>	<u>Penetration No.</u>	<u>Boundary EIN</u> *	<u>UFSAR Figure</u>	<u>Notes (Notes only used to supplement Section 6.2.4.4.) (Notes listed on Table 6.2-30)</u>	<u>Position Indication In Control Room</u> *	<u>Normal position (MODES 1, 2, 3, 4)</u> *	<u>Immediate Postaccident Position (Refers to position immediately following receipt of containment isolation signal and containment ventilation isolation signal.)</u> *
Containment pressure transmitters PT-945 and PT-946	121c	PT-945 1819A PT-946 1819B	6.2-31 6.2-31 6.2-31 6.2-31	6 --- 6 ---	NA No NA No	NA O NA O	NA O NA O
Reactor coolant drain tank to gas analyzer line	123a	1600A 1655 1789	6.2-32 6.2-32 6.2-32	17	No No Status	O O O	C O C
Standby auxiliary feed-water (SAFW) line to steam generator 1B	123b	9704B 9725 9724 CLIC	6.2-26 6.2-26 6.2-26 6.2-26	19	R/G No No NA	O LC C C	O LC C C
Excess letdown heat exchanger cooling water supply	124a	743 CLIC	6.2-33 6.2-33	44 22	NA NA	C C	C C
Postaccident air sample to C fan	124b	Welded Shut	N/A	31	N/A	N/A	N/A
Excess letdown heat exchanger cooling water return	124c	745 CLIC	6.2-33 6.2-33	44 22	R/G NA	C C	C C
Postaccident air sample to common return	124d	1572 1573 1574	6.2-34 6.2-34 6.2-34	--- --- ---	No No No	LC C LC	LC C LC
Component cooling water (CCW) from reactor coolant pump 1B	125	759B CLIC	6.2-35 6.2-35	44 22	R/G NA	O C	O C

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<u>System</u>	<u>Penetration No.</u>	<u>Boundary EIN</u> *	<u>UFSAR Figure</u>	<u>Notes (Notes only used to supplement Section 6.2.4.4.) (Notes listed on Table 6.2-30)</u>	<u>Position Indication In Control Room</u> *	<u>Normal position (MODES 1, 2, 3, 4)</u> *	<u>Immediate Postaccident Position (Refers to position immediately following receipt of containment isolation signal and containment ventilation isolation signal.)</u> *
Component cooling water (CCW) from reactor coolant pump 1A	126	759A CLIC	6.2-36 6.2-36	44 22	R/G NA	O C	O C
Component cooling water (CCW) to reactor coolant pump 1A	127	749A CLIC	6.2-37 6.2-37	44 22	R/G NA	O C	O C
Component cooling water (CCW) to reactor coolant pump 1B	128	749B CLIC	6.2-38 6.2-38	44 22	R/G NA	O C	O C
Reactor coolant drain tank and pressurizer relief tank to containment vent header	129	1713 1793 1786 1787	6.2-39 6.2-39 6.2-39 6.2-39	--- 20 --- ---	NA No Status Status	C LC O O	C LC C C
Component cooling water (CCW) from reactor support cooling	130	814 CLIC	6.2-40 6.2-40	44 22	Both NA	O C	C C
Component cooling water (CCW) to reactor support cooling	131	813 CLIC	6.2-40 6.2-40	44 22	Both NA	O C	C C
Containment mini-purge exhaust	132	7970 7971 Cap	6.2-41 6.2-41 6.2-41	--- --- ---	Both Both NA	O/C C C	C C C

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<u>System</u>	<u>Penetration No.</u>	<u>Boundary EIN</u> *	<u>UFSAR Figure</u>	<u>Notes (Notes only used to supplement Section 6.2.4.4.) (Notes listed on Table 6.2-30)</u>	<u>Position Indication In Control Room</u> *	<u>Normal position (MODES 1, 2, 3, 4)</u> *	<u>Immediate Postaccident Position (Refers to position immediately following receipt of containment isolation signal and containment ventilation isolation signal.)</u> *
Residual heat removal pump suction from A hot leg	140	701 2763 2786 2786B CLOC	6.2-42 6.2-42 6.2-42 6.2-42 6.2-42	13, 14 --- --- --- 15	R/G No No No NA	C C C C C	C C C C C
Residual heat removal pump A suction from sump B	141	850A CLOC 1813A	6.2-43 6.2-43 6.2-43	24 15 14, 25	R/G NA R/G	C C C	O C C
Residual heat removal pump B suction from sump B	142	850B CLOC 1813B	6.2-44 6.2-44 6.2-44	24 15 14, 25	R/G NA R/G	C C C	O C C
Reactor coolant drain tank discharge line	143	1003A 1003B 1709G 1722 1721	6.2-45 6.2-45 6.2-45 6.2-45 6.2-45	--- --- --- --- ---	Status Status No No Status	O O C LC O	C C C LC C
Reactor compartment cooling unit A supply	201a	4757 4775 CLIC	6.2-46 6.2-46 6.2-46	30 --- 27	No No NA	O C C	O C C
Reactor compartment cooling unit B return	201b	4636 4658 4776 PI-2141 Cap (PI-2141) CLIC PI-2000F	6.2-47 6.2-47 6.2-47 6.2-47 6.2-47 6.2-47 6.2-47	26,44 44 --- --- --- 27 ---	No NA No NA NA NA NA	O C C NA C C NA	O C C NA C C NA
B hydrogen recombiner (pilot)	202a	1076B 10211S1	6.2-48 6.2-48	--- 28	No Status	LC C	LC C

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<u>System</u>	<u>Penetration No.</u>	<u>Boundary EIN</u> *	<u>UFSAR Figure</u>	<u>Notes (Notes only used to supplement Section 6.2.4.4.) (Notes listed on Table 6.2-30)</u>	<u>Position Indication In Control Room</u> *	<u>Normal position (MODES 1, 2, 3, 4)</u> *	<u>Immediate Postaccident Position (Refers to position immediately following receipt of containment isolation signal and containment ventilation isolation signal.)</u> *
B hydrogen recombiner (main)	202b	1084B 10213S1	6.2-48 6.2-48	--- 28	No Status	LC C	LC C
Containment pressure transmitter PT-947 and PT-948	203a	PT-947 1819C PT-948 1819D	6.2-49 6.2-49 6.2-49 6.2-49	6 --- 6 ---	NA No NA No	NA O NA O	NA O NA O
Postaccident air sample from D fan	203b	Welded Shut	N/A	31	N/A	N/A	N/A
Postaccident air sample from common header	203c	Welded Shut	N/A	31	N/A	N/A	N/A
Purge supply duct	204	ACD93 5869	6.2-51 6.2-51	2, 29 29	NA Both	C C	C C
Loop B hot leg sample	205	955 956D 966C	6.2-52 6.2-52 6.2-52	17 --- ---	Status No Status	C O C	C O C
Pressurizer liquid space sample	206a	953 956E 966B	6.2-53 6.2-53 6.2-53	17 --- ---	Status No Status	C O C	C O C
Steam generator A sample	206b	CLIC 5735 5749	6.2-54 6.2-54 6.2-54	19 44 ---	NA Status NA	C O C	C C C
Pressurizer steam space sample	207a	951 956F 966A	6.2-55 6.2-55 6.2-55	17 --- ---	Status No Status	C O C	C O C

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<u>System</u>	<u>Penetration No.</u>	<u>Boundary EIN</u> *	<u>UFSAR Figure</u>	<u>Notes (Notes only used to supplement Section 6.2.4.4.) (Notes listed on Table 6.2-30)</u>	<u>Position Indication In Control Room</u> *	<u>Normal position (MODES 1, 2, 3, 4)</u> *	<u>Immediate Postaccident Position (Refers to position immediately following receipt of containment isolation signal and containment ventilation isolation signal.)</u> *
Steam generator B sample	207b	CLIC 5736 5754	6.2-56 6.2-56 6.2-54	19 44 ---	NA Status NA	C O C	C C C
Reactor compartment cooling unit B supply	209a	4635 4637 CLIC	6.2-47 6.2-47 6.2-47	30 --- 27	No No NA	O C C	O C C
Reactor compartment cooling Unit A return	209b	4638 4758 4759 PI-2000AC PI-2232 CLIC	6.2-46 6.2-46 6.2-46 6.2-46 6.2-46 6.2-46	26 44 44 --- --- 27	No No NA NA NA NA	O O C NA NA C	O O C NA NA C
Oxygen makeup to A & B recombiners	210	1080A 10214S1 10214S 10215S1 10215S	6.2-57 6.2-57 6.2-57 6.2-57 6.2-57	--- 28 17, 28 28 17, 28	No Status Status Status Status	LC C C C C	LC C C C C
Purge exhaust duct	300	ACD92 5879	6.2-58 6.2-58	2, 29 29	NA Both	C C	C C
Auxiliary steam supply to containment	301	Welded shut	NA	31	NA	NA	NA
Auxiliary steam condensate return	303	Welded shut	NA	31	NA	NA	NA
A hydrogen recombiner (pilot)	304a	1076A 10205S1	6.2-60 6.2-60	--- 28	No Status	LC C	LC C

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<u>System</u>	<u>Penetration No.</u>	<u>Boundary EIN</u> *	<u>UFSAR Figure</u>	<u>Notes (Notes only used to supplement Section 6.2.4.4.) (Notes listed on Table 6.2-30)</u>	<u>Position Indication In Control Room</u> *	<u>Normal position (MODES 1, 2, 3, 4)</u> *	<u>Immediate Postaccident Position (Refers to position immediately following receipt of containment isolation signal and containment ventilation isolation signal.)</u> *
A hydrogen recombiner (main)	304b	1084A 10209S1	6.2-60 6.2-60	--- 28	No Status	LC C	LC C
Containment air sample postaccident	305a	Welded Shut	N/A	31	N/A	N/A	N/A
Containment air sample inlet	305b	1598 1599	6.2-62 6.2-62	--- ---	Both Both	O O	C C
Containment air sample postaccident	305c	1557 1558 1559	6.2-61 6.2-61 6.2-61	--- --- ---	No No No	LC C LC	LC C LC
Containment air sample postaccident	305d	1560 1561 1562	6.2-61 6.2-61 6.2-61	--- --- ---	No No No	LC C LC	LC C LC
Containment air sample out	305e	1596 1597	6.2-63 6.2-63	--- ---	No Both	O O	O C
Fire service water	307	9227 9229	6.2-64 6.2-64	--- ---	NA NA	C C	C C
Service water (SW) from A fan cooler	308	4629 4633 4655 FIA-2033 TIA-2010 Caps(2) (FIA-2033) PI-2000C CLIC	6.2-65 6.2-65 6.2-65 6.2-65 6.2-65 6.2-65 6.2-65 6.2-65	26,44 --- 44 --- --- --- --- 27	No No NA NA NA NA NA NA	LO C C NA NA C NA C	LO C C NA NA C NA C

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<u>System</u>	<u>Penetration No.</u>	<u>Boundary EIN</u> *	<u>UFSAR Figure</u>	<u>Notes (Notes only used to supplement Section 6.2.4.4.) (Notes listed on Table 6.2-30)</u>	<u>Position Indication In Control Room</u> *	<u>Normal position (MODES 1, 2, 3, 4)</u> *	<u>Immediate Postaccident Position (Refers to position immediately following receipt of containment isolation signal and containment ventilation isolation signal.)</u> *
Mini-purge supply	309	7445 7478	6.2-66 6.2-66	--- ---	Both Both	O/C O/C	C C
Instrument air to containment	310a	5392 5393	6.2-67 6.2-67	--- ---	Both NA	O O	C C
Service air to containment	310b	7141 7226	6.2-68 6.2-68	--- ---	No NA	LC C	LC C
Service water (SW) from B fan cooler	311	4630 4634 4656 FIA-2034 PI-2000D Caps(2) (FIA-2034) TIA-2011 CLIC	6.2-65 6.2-65 6.2-65 6.2-65 6.2-65 6.2-65 6.2-65 6.2-65 6.2-65	26,44 --- 44 --- --- --- --- --- 27	No No NA NA NA NA NA NA NA	LO C C NA NA C NA C	LO C C NA NA C NA C
Service water (SW) to D fan cooler	312	4642 4646 12500K PI-2144 CLIC	6.2-65 6.2-65 6.2-65 6.2-65 6.2-65	30 --- --- --- 27	No No No NA NA	LO C C NA C	LO C C NA C
Leakage test depressurization	313	SAT02 7444 Cap	6.2-69 6.2-69 6.2-69	--- --- ---	NA NA NA	C C C	C C C

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<u>System</u>	<u>Penetration No.</u>	<u>Boundary EIN</u> *	<u>UFSAR Figure</u>	<u>Notes (Notes only used to supplement Section 6.2.4.4.) (Notes listed on Table 6.2-30)</u>	<u>Position Indication In Control Room</u> *	<u>Normal position (MODES 1, 2, 3, 4)</u> *	<u>Immediate Postaccident Position (Refers to position immediately following receipt of containment isolation signal and containment ventilation isolation signal.)</u> *
Service water (SW) from C fan cooler	315	4643 4647 4659 FIA-2035 PI-2000G TIA-2012 Caps(2) (FIA-2035) CLIC	6.2-65 6.2-65 6.2-65 6.2-65 6.2-65 6.2-65 6.2-65 6.2-65 6.2-65	26,44 --- 44 --- --- --- --- --- 27	No No NA NA NA NA NA NA NA	LO C C NA NA NA C C	LO C C NA NA NA C C
Service water (SW) to B fan cooler	316	4628 4632 PI-2138 CLIC	6.2-65 6.2-65 6.2-65 6.2-65	30 --- --- 27	No No NA NA	LO C NA C	LO C NA C
Leakage test supply	317	7443A 7443 Cap	6.2-70 6.2-70 6.2-70	--- --- ---	NA NA NA	C C C	C C C
Deadweight tester	318	Welded shut	NA	31	NA	NA	NA
Service water (SW) to A fan cooler	319	4627 4631 PI-2142 CLIC	6.2-65 6.2-65 6.2-65 6.2-65	30 --- --- 27	No No NA NA	LO C NA C	LO C NA C
Service water (SW) to C fan cooler	320	4641 4645 PI-2136 12500H CLIC	6.2-65 6.2-65 6.2-65 6.2-65 6.2-65	30 --- --- --- 27	No No NA No NA	LO C NA C C	LO C NA C C

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<u>System</u>	<u>Penetration No.</u>	<u>Boundary EIN</u> *	<u>UFSAR Figure</u>	<u>Notes (Notes only used to supplement Section 6.2.4.4.) (Notes listed on Table 6.2-30)</u>	<u>Position Indication In Control Room</u> *	<u>Normal position (MODES 1, 2, 3, 4)</u> *	<u>Immediate Postaccident Position (Refers to position immediately following receipt of containment isolation signal and containment ventilation isolation signal.)</u> *
A steam generator blowdown	321	5738 5752 CLIC	6.2-71 6.2-71 6.2-71	44 --- 19	Status NA NA	O C C	C C C
B steam generator blowdown	322	5737 5756 CLIC	6.2-72 6.2-72 6.2-72	44 --- 19	Status NA NA	O C C	C C C
Service water (SW) from D fan cooler	323	4644 4648 4660 FIA-2036 PI-2000H TIA-2013 Caps(2) (FIA-2036) CLIC	6.2-65 6.2-65 6.2-65 6.2-65 6.2-65 6.2-65 6.2-65 6.2-65 6.2-65	26,44 --- 44 --- --- --- --- --- 27	No No NA NA NA NA NA NA	LO C C NA NA NA C C	LO C C NA NA C C
Demineralized water to containment	324	8418 8419	6.2-73 6.2-73	--- ---	Both NA	C C	C C
Hydrogen monitor instrumentation line	332a	922 924 CLOC 7452 Cap(7452)	6.2-74 6.2-74 6.2-74 6.2-74 6.2-74	--- --- 32 --- ---	Both Both NA No NA	C C C C C	C C C C C
Hydrogen monitor instrumentation line	332b	923 CLOC 7456 Cap(7456)	6.2-74 6.2-74 6.2-74 6.2-74	--- 32 --- ---	Both NA No NA	C C C C	C C C C

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<u>System</u>	<u>Penetration No.</u>	<u>Boundary EIN</u> *	<u>UFSAR Figure</u>	<u>Notes (Notes only used to supplement Section 6.2.4.4.) (Notes listed on Table 6.2-30)</u>	<u>Position Indication In Control Room</u> *	<u>Normal position (MODES 1, 2, 3, 4)</u> *	<u>Immediate Postaccident Position (Refers to position immediately following receipt of containment isolation signal and containment ventilation isolation signal.)</u> *
Containment pressure transmitters PT-944, PT-949, and PT-950	332c	PT-944 1819G PT-949 1819E PT-950 1819F	6.2-75 6.2-75 6.2-75 6.2-75 6.2-75 6.2-75	6 --- 6 --- 6 ---	NA No NA No NA No	NA O NA O NA O	NA O NA O NA O
Hydrogen monitor instrumentation line	332d	921 CLOC 7448 Cap	6.2-74 6.2-74 6.2-74 6.2-74	--- 32 --- ---	Both NA No NA	C C C C	C C C C

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<u>System</u>	<u>Penetration No.</u>	<u>Boundary EIN</u> *	<u>UFSAR Figure</u>	<u>Notes (Notes only used to supplement Section 6.2.4.4.) (Notes listed on Table 6.2-30)</u>	<u>Position Indication In Control Room</u> *	<u>Normal position (MODES 1, 2, 3, 4)</u> *	<u>Immediate Postaccident Position (Refers to position immediately following receipt of containment isolation signal and containment ventilation isolation signal.)</u> *
Main steam from A steam generator	401	3411 3413A 3455 3505A 3505C 3509 3511 3513 3515 3517 3521 3615 3669 11027 11029 11031 PS-2092 PT-468 PT-469 PT-469A PT-482 End caps CLIC	6.2-76 6.2-76	--- 18 --- --- --- --- --- --- --- 18 18 --- 18 --- --- --- 6 6 6 6 6 33 19	R/G No No R/G No NA NA NA NA R/G No No No No No NA NA NA NA NA NA NA NA	C O C C C C C C C O O C C C C NA NA NA NA NA C C	C O/C C O/C C C C C C C O/C C C C C NA NA NA NA NA C C

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<u>System</u>	<u>Penetration No.</u>	<u>Boundary EIN</u> *	<u>UFSAR Figure</u>	<u>Notes (Notes only used to supplement Section 6.2.4.4.) (Notes listed on Table 6.2-30)</u>	<u>Position Indication In Control Room</u> *	<u>Normal position (MODES 1, 2, 3, 4)</u> *	<u>Immediate Postaccident Position (Refers to position immediately following receipt of containment isolation signal and containment ventilation isolation signal.)</u> *
Main steam from B steam generator	402	3410 3412A 3456 3504A 3504C 3508 3510 3512 3514 3516 3520 3614 3668 11021 11023 11025 PS-2093 PT-478 PT-479 PT-483 End caps CLIC	6.2-77 6.2-77	--- 18 --- --- --- --- --- --- --- 18 18 --- 18 --- --- --- 6 6 6 6 33 19	R/G No No R/G No NA NA NA NA R/G No No No No No NA NA NA NA NA NA NA	C O C C C C C C O O C O C C C NA NA NA NA C C	C O/C C O/C C C C C C O C C C C NA NA NA NA C C
Feedwater line to A steam generator	403	3993 3995X 4000C 4003 4011A 4099N 4003A 4099E 8651 CLIC	6.2-78 6.2-78 6.2-78 6.2-78 6.2-78 6.2-78 6.2-78 6.2-78 6.2-78 6.2-78 6.2-78	34 --- 34 34 --- --- --- --- --- --- 19	NA No NA NA No No No No No No NA	O C C C C C C C C C C	C C O/C O/C C C C C C C C

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<u>System</u>	<u>Penetration No.</u>	<u>Boundary EIN</u> *	<u>UFSAR Figure</u>	<u>Notes (Notes only used to supplement Section 6.2.4.4.) (Notes listed on Table 6.2-30)</u>	<u>Position Indication In Control Room</u> *	<u>Normal position (MODES 1, 2, 3, 4)</u> *	<u>Immediate Postaccident Position (Refers to position immediately following receipt of containment isolation signal and containment ventilation isolation signal.)</u> *
Feedwater line to B steam generator	404	3992 3994E 4000D 4004 4012A 3994X 4004A 8650 CLIC	6.2-78 6.2-78 6.2-78 6.2-78 6.2-78 6.2-78 6.2-78 6.2-78 6.2-78	34 --- 34 34 --- --- --- --- 19	NA No NA NA No No No No NA	O C C C C C C C C	C C O/C O/C C C C C C
Personnel hatch	1000	Seal Seal 8403 8404 8405 8406 8407 Caps (2) (PI-2936)	3.8-31 3.8-31 3.8-31 3.8-31 3.8-31 3.8-31 3.8-31 3.8-31 3.8-31	2 2 --- --- --- --- --- --- ---	NA NA No No No No No No NA	C C C C C C C C C	C C C C C C C C C
Equipment hatch	2000	Seal Seal 8060 8412 8413 8414 8415 8416 8417 Cap (8415) Caps (2) (PI-2223)	3.8-30 3.8-30 3.8-30 3.8-30 3.8-30 3.8-30 3.8-30 3.8-30 3.8-30 3.8-30 3.8-30 3.8-30	2 2 --- --- --- --- --- --- --- --- --- ---	NA NA No No No No No No No NA NA NA	C C C C C C C C C C C C	C C C C C C C C C C C C

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Table 6.2-30
CONTAINMENT PIPING PENETRATIONS AND ISOLATION BOUNDARIES - NOTES FOR TABLE 6.2-29

<u>Note</u>	<u>Description</u>
(1)	Penetration number 2 was added as a result of EWR 4998 to facilitate steam generator maintenance activities during reduced inventory operation. This penetration is closed by a double-gasketed blind flange on both ends. The innermost gasket for each flange (i.e., gasket closest to containment wall) provides a containment barrier. Therefore, both flanges are necessary for containment integrity.
(2)	This penetration is provided with redundant seals and is closed during normal operation (MODES 1 - 4). Each seal provides a single containment isolation barrier.
(3)	The end of the fuel transfer tube inside containment is closed by a double-gasketed blind flange to prevent leakage of spent fuel pool water into the containment during plant operation. This flange also serves as protection against leakage from the containment following a loss-of-coolant accident.
(4)	The charging system is a closed system outside containment (CLOC). Verification of this closed system as a containment isolation boundary is accomplished via normal system operation (2235 psig).
(5)	The safety injection system is a closed system outside containment (CLOC). Verification of this closed system as a containment isolation boundary is accomplished via inservice and/or shutdown leakage checks.
(6)	The pressure transmitter assembly, by its design, provides a containment pressure boundary. The integrity of this boundary is verified by annual leakage tests and since the pressure transmitter provides direct indication to the control room, its root valve can be left normally open.
(7)	This penetration was only utilized during initial plant construction and is maintained inactive.
(8)	The containment spray system is a closed system outside containment (CLOC). Verification of this closed system as a containment isolation boundary is accomplished via inservice and/or shutdown leakage checks.
(9)	This valve may be opened during containment spray pump testing since there will always be at least one isolation boundary between the valve and containment for the duration of the test.
(10)	Manual valves 859A and 859B are CIVs for both penetrations 105 and 109.
(11)	A second isolation barrier is provided by the volume control tank and connecting piping per letter from D. D. DiIanni, NRC, to R. W. Kober, RG&E, dated January 30, 1987. This barrier is not required to be tested.
(12)	Only one isolation barrier is provided since there are two Event V check valves in the safety injection cold legs, and two check valves and a normally closed motor-operated valve in the safety injection hot legs. This configuration was accepted by the NRC during the SEP (NUREG 0821, Section 4.22.2).
(13)	10 CFR 50, Appendix J containment leakage testing is not required per D.M. Crutchfield, NRC, letter to J. E. Maier, RG&E, dated May 6, 1981.
(14)	MOVs 1813A, 1813B, 720, and 701 are maintained closed at power with their breakers locked off.
(15)	The residual heat removal system is a closed system outside containment (CLOC). Verification of this closed system as a containment isolation boundary is accomplished via inservice and/or shutdown leakage checks.

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<u>Note</u>	<u>Description</u>
(16)	Containment isolation signals were added to AOVs 200A, 200B, and 202 since AOV 427 fails open on loss of power. The isolation signal for these three valves is relayed from AOV 427.
(17)	This valve receives a containment isolation signal; however, credit is not taken for this function since the valve is inside the missile barrier or outside the necessary class break boundary. Therefore, this valve is not a containment isolation valve, subject to 10 CFR 50, Appendix J leakage testing, nor does it require a maximum isolation time. The containment isolation signal only enhances isolation capability.
(18)	This valve is normally open at power since it is required during power operation or increases the reliability of a standby system. However, this valve can either be closed from the control room or locally when required.
(19)	The main steam, main feedwater, steam generator blowdown, and standby auxiliary feedwater (SAFW) penetrations take credit for the steam generator tubes as a closed system inside containment. Verification of this closed system as a containment isolation boundary is accomplished via normal power operation. The isolation valves outside containment for these penetrations are not required to be Appendix J tested.
(20)	Manual valves 547 and 1793 are locked closed and leak tested to provide equivalent protection for GDC 56 and 57 (see UFSAR Section 6.2.4.4.4.1, Class 3A).
(21)	Deleted
(22)	The component cooling water (CCW) system piping inside containment for this penetration is a closed system (CLIC). Verification of this closed system as a containment isolation boundary is accomplished via inservice and/or shutdown leakage checks.
(23)	This valve is not a containment isolation valve due to the installed downstream welded flange but is maintained normally closed to provide additional assurance of containment integrity.
(24)	Sump lines are in operation and filled with fluid following an accident; therefore, 10 CFR 50, Appendix J leakage testing, is not required for this valve. See D. M. Crutchfield, NRC, letter to J. E. Maier, RG&E, dated May 6, 1981.
(25)	There is no second containment barrier for this branch line. However, MOVs 1813A and 1813B are maintained closed at power and tested to Appendix J. These lines are also filled with water post LOCA, thus providing a barrier to the release of containment atmosphere.
(26)	This manual valve is subject to an annual hydrostatic leakage test and is not subject to 10 CFR 50, Appendix J leakage testing. See NUREG-0821.
(27)	The service water (SW) system piping inside containment for this penetration is a closed system (CLIC). Verification of this closed system as a containment isolation boundary is accomplished via inservice and/or shutdown leakage checks.
(28)	This solenoid valve is maintained inactive in the closed position by removal of its dc control power.
(29)	The flanges and associated double seals provide containment isolation and ensure that containment integrity is maintained for all modes of operation above MODE 5 (Cold Shutdown). During MODE 5 and 6 (Cold Shutdown and Refueling) when the flanges are removed, these valves provide isolation for containment shutdown purge and exhaust. These valves do not require 10 CFR 50, Appendix J leakage testing, nor a maximum isolation time.
(30)	The service water (SW) system is seismically supported and is missile protected inside containment. Therefore, this manual valve is not subject to 10 CFR 50, Appendix J leakage testing. See NUREG-0821. Additionally, except for a brief peak at the beginning of the transient, the SW system pressure at this valve will be above the post-LOCA containment pressure.

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<u>Note</u>	<u>Description</u>
(31)	This penetration is decommissioned and welded shut.
(32)	Acceptable isolation capability is provided for these instrument lines by two isolation boundaries outside containment. One of the boundaries outside containment is a Seismic Category I closed system which is subjected to Type C leakage testing under 10 CFR 50, Appendix J.
(33)	These end caps include those found on the sensing lines for PS-2092, PT-468, PT-469, PT-469A, and PT-482 (Penetration 401) and PS-2093, PT-479, and PT-483 (Penetration 402).
(34)	This check valve can be open when containment isolation is required in order to provide necessary feedwater or auxiliary feedwater to the steam generators. The check valve will close once feedwater is isolated to the affected steam generator.
(35)	Deleted
(36)	AOV 371 and the residual heat removal closed system outside containment (CLOC) are containment boundaries for both penetrations 111 and 112.
(37)	AOV 959 is maintained closed with its control fuses pulled.
(38)	This valve was added in accordance with PCR 2002-0035 and is only used for manual operation of the Safety Injection Accumulator Makeup Pump.
(39)	Manual valve 898C is a CIV for both penetrations 101 and 113.
(40)	Manual valve 868E is a CIV for both penetrations 105 and 109.
(41)	Relief valve 869E is a CIV for both penetrations 105 and 109.
(42)	The Safety Injection pathways do not constitute a potential primary containment atmospheric pathway because they remain either water sealed or water filled for 30-days post-accident. Local leak rate testing of the associated components is not required per the Appendix J Program.
(43)	The Containment Sump A discharge to the Waste Holdup tank pathway does not constitute a potential primary containment atmospheric pathway because this pathway remains water filled for the 30-days post-accident. Local leak rate testing of the associated components is not required per the Appendix J Program.
(44)	These Type C locations (valves) are primary containment boundaries that do not constitute potential primary containment atmospheric pathways during and following a Design Basis Accident (DBA). Therefore, these locations are exempt from Type C leakage testing under 10 CFR 50, Appendix J, per NEI 94-01 and ANSI/ANS 56.8-1994, as well as hydrostatic leak testing being completed in place of 10 CFR 50, Appendix J, testing. Analysis DA-ME-17-007 provides the technical justification for this exemption.

Table 6.2-31
CONTAINMENT PIPING PENETRATIONS AND ISOLATION BOUNDARIES -
LEGEND FOR Table 6.2-29

<u>Symbol</u>	<u>Description</u>
AOV	Air-operated valve
Both	R/G and Status
C	Closed
CIV	Containment isolation valve
CLIC	Closed loop inside containment
CLOC	Closed loop outside containment
LC	Locked closed
LO	Locked open
MOV	Motor-operated valve
NA	Not applicable due to boundary type
O	Open
O/C	Open or closed
R/G	Red/green light on main control board
Status	White status light

Table 6.2-32
EFFECT OF LOSS OF AIR OR POWER SUPPLY TO AIR-OPERATED VALVES

<u>Penetration Number</u>	<u>System</u>	<u>Valve Number</u>	<u>Receive CIS</u>	<u>Position Following Loss of Air or Power</u>
107	Sump A discharge to waste holdup tank	1723 1728	Yes Yes	Fails closed Fails closed
111	Residual heat removal to Loop B cold leg	959	Yes	Fails closed
112	Letdown to non-regenerative heat exchanger	200A 200B 202 5 / 1 427	Yes ^a Yes ^a Yes ^a Yes Yes	Fails closed Fails closed Fails closed Fails closed Fails open ^b
120a	Nitrogen to accumulators	846	Yes	Fails closed
120b	Pressurizer relief tank to gas analyzer	539	Yes	Fails closed
121a	Makeup water to pressurizer relief tank	508	Yes	Fails closed
123a	Reactor coolant drain tank to gas analyzer line	1789	Yes	Fails closed
124c	Excess letdown heat exchanger cooling water supply and return	745	Yes	Fails closed
129	Reactor coolant drain tank and pressurizer relief tank to containment vent header	1786 1787	Yes Yes	Fails closed Fails closed
132	Containment mini-purge exhaust	7970 7971	Yes Yes	Fails closed Fails closed
143	Reactor coolant drain tank discharge line	1003A 1003B 1721	Yes Yes Yes	Fails closed Fails closed Fails closed
204	Purge supply duct	5869	Yes	Fails closed
205	Loop B hot leg sample	955 966C	Yes Yes	Fails closed ^b Fails closed
206a	Pressurizer liquid space sample	953 966B	Yes Yes	Fails closed ^b Fails closed
206b	Steam generator A sample	5735	Yes	Fails closed
207a	Pressurizer steam space sample	951 966A	Yes Yes	Fails closed ^b Fails closed
207b	Steam generator B sample	5736	Yes	Fails closed

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<u><i>Penetration Number</i></u>	<u><i>System</i></u>	<u><i>Valve Number</i></u>	<u><i>Receive CIS</i></u>	<u><i>Position Following Loss of Air or Power</i></u>
300	Purge exhaust duct	5879	Yes	Fails closed ^b
305b	Containment air sample inlet	1598 1599	Yes Yes	Fails closed Fails closed
305e	Containment air sample outlet	1597	Yes	Fails closed
309	Mini-purge supply	7445 7478	Yes Yes	Fails closed Fails closed
310a	Instrument air to containment	5392	Yes	Fails closed
321	Steam generator A blowdown	5738	Yes	Fails closed
322	Steam generator B blowdown	5737	Yes	Fails closed
324	Demineralized water to containment	8418	Yes	Fails closed
401	Main steam from steam generator A	3411 3517	No No	Fails closed ^c Fails closed on loss of air; Fails as-is on loss of power
402	Main steam from steam generator B	3410 3516	No No	Fails closed ^c Fails closed on loss of air; Fails as-is on loss of power

- a. CIS is relayed from control circuit for valve 427.
- b. Credit is not taken for this valve as a containment isolation valve.
- c. Automatic backup N₂ supply provided.

Table 6.2-33
ESSENTIAL AND NONESSENTIAL SYSTEM CONTAINMENT PENETRATIONS

<u>Penetration Number</u>	<u>Identification/Description</u>	<u><i>Essential Versus Nonessential</i></u>
2	Steam generator inspection/maintenance	Nonessential
29	Fuel transfer tube	Nonessential
100	Charging line to loop B	Nonessential
101	Safety injection pump 1B discharge	Essential
102	Alternate charging to loop A cold leg	Nonessential
103	Construction fire service water	Nonessential
105	Containment spray pump 1A	Essential
106	Reactor coolant pump A seal water inlet	Essential
107	Sump A discharge to waste holdup tank	Nonessential
108	Reactor coolant pump seal water return and excess letdown to volume control tank	Nonessential
109	Containment spray pump 1B	Essential
110a	Reactor coolant pump B seal water inlet	Essential
110b	Safety injection test line	Nonessential
111	Residual heat removal to loop B cold leg	Essential
112	Letdown to nonregenerative heat exchanger	Nonessential
113	Safety injection pump 1A discharge	Essential
119	Standby auxiliary feedwater (SAFW) to steam generator 1A	Essential
120a	Nitrogen to accumulators	Nonessential
120b	Pressurizer relief tank to gas analyzer	Nonessential
121a	Makeup water to pressurizer relief tank	Nonessential
121b	Nitrogen to pressurizer relief tank	Nonessential
121c	Containment pressure transmitter PT-945 and PT-946	Essential
123a	Reactor coolant drain tank to gas analyzer	Nonessential
123b	Standby auxiliary feedwater (SAFW) to steam generator 1B	Essential
124a	Excess letdown heat exchanger cooling water supply	Nonessential
124b	Postaccident air sample to containment recirculation fan C	Nonessential ^a
124c	Excess letdown heat exchanger cooling water return	Nonessential

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<u>Penetration Number</u>	<u>Identification/Description</u>	<u>Essential Versus Nonessential</u>
124d	Postaccident air sample to common return	Nonessential
125	Component cooling water from reactor coolant pump 1B	Essential
126	Component cooling water from reactor coolant pump 1A	Essential
127	Component cooling water to reactor coolant pump 1A	Essential
128	Component cooling water to reactor coolant pump 1B	Essential
129	Reactor coolant drain tank and pressurizer relief tank to containment vent header	Nonessential
130	Component cooling water from reactor support cooling	Nonessential
131	Component cooling water to reactor support cooling	Nonessential
132	Containment mini-purge exhaust	Nonessential
140	Residual heat removal pump suction from loop A hot leg	Essential
141	Residual heat removal pump A suction from sump B	Essential
142	Residual heat removal pump B suction from sump B	Essential
143	Reactor coolant drain tank discharge line	Nonessential
201a	Reactor compartment cooling unit A return	Essential
201b	Reactor compartment cooling unit B return	Essential
202a	Hydrogen recombiner B (pilot)	Nonessential
202b	Hydrogen recombiner B (main)	Nonessential
203a	Containment pressure transmitter PT-947 and 948	Essential
203b	Postaccident air sample from containment recirculation fan D	Nonessential ^a
203c	Postaccident air sample from common header	Nonessential ^a
204	Purge supply duct	Nonessential
205	Loop B hot leg sample	Nonessential
206a	Pressurizer liquid space sample	Nonessential
206b	Steam generator A sample	Nonessential
207a	Pressurizer steam space sample	Nonessential
207b	Steam generator B sample	Nonessential
209a	Reactor compartment cooling unit B return	Essential
209b	Reactor compartment cooling unit A supply	Essential

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<u>Penetration Number</u>	<u>Identification/Description</u>	<u><i>Essential Versus Nonessential</i></u>
210	Oxygen makeup to A and B recombiners	Nonessential
300	Purge exhaust duct	Nonessential
301	Auxiliary steam supply to containment	Nonessential ^a
303	Auxiliary steam condensate return	Nonessential ^a
304a	Hydrogen recombiner A (pilot)	Nonessential
304b	Hydrogen recombiner A (main)	Nonessential
305a	Containment air sample postaccident	Nonessential ^a
305b	Containment air sample inlet	Nonessential
305c	Containment air sample postaccident	Nonessential
305d	Containment air sample postaccident	Nonessential
305e	Containment air sample out	Nonessential
307	Fire service water	Nonessential
308	Service water from containment fan cooler A	Essential
309	Mini-purge supply	Nonessential
310a	Instrument air to containment	Nonessential
310b	Service air to containment	Nonessential
311	Service water from containment fan cooler B	Essential
312	Service water to containment fan cooler D	Essential
313	Leakage test depressurization	Nonessential
315	Service water from containment fan cooler C	Essential
316	Service water to containment fan cooler B	Essential
317	Leakage test supply	Nonessential
318	Dead weight tester	Nonessential ^a
319	Service water to containment fan cooler A	Essential
320	Service water to containment fan cooler C	Essential
321	Steam generator A blowdown	Nonessential
322	Steam generator B blowdown	Nonessential
323	Service water from containment fan cooler D	Essential
324	Demineralized water to containment	Nonessential

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<u>Penetration Number</u>	<u>Identification/Description</u>	<u><i>Essential Versus Nonessential</i></u>
332a	Hydrogen monitor instrumentation lines	Nonessential
332b	Hydrogen monitor instrumentation lines	Nonessential
332c	Containment pressure transmitters PT-944, 949, and 950	Essential
332d	Hydrogen monitor instrumentation lines	Nonessential
401	Main steam from steam generator A	Nonessential
402	Main steam from steam generator B	Nonessential
403	Feedwater line to steam generator A	Essential ^b
404	Feedwater line to steam generator B	Essential ^b
1000	Personnel hatch	Nonessential
2000	Equipment hatch	Nonessential

- a. Decommissioned via welded caps or plugs on both sides (Inside & Outside Containment) of penetration.
- b. Used for auxiliary feedwater.

Table 6.2-34
PARAMETERS AND ASSUMPTIONS USED TO DETERMINE HYDROGEN
GENERATION (HISTORICAL)

Core thermal power rating	1,550 MWt
Containment free volume	1,000,000 ft ³
Weight of zirconium cladding	22,309 lbs
Percent of zirconium associated with zirc-water reaction	5%
Hydrogen generated by zirc-water reaction	8,812 scf
Initial inventory	783 scf
Hydrogen recombiner capacity	100 scfm

Core Cooling Solution Radiolysis

Percent of total halogens retained in the core	50
Percent of total noble gases retained in the core	0
Percent of other fission products retained in the core	99
Percent of total decay energy - gamma	50
Percent of total decay energy - beta	50
Percent of gamma energy absorbed by solution	10
Percent of beta energy absorbed by solution	0
Molecules of hydrogen produced per 100 eV of energy absorbed by solution	0.50

Sump Cooling Solution Radiolysis

Percent of total halogens released to the sump solution	50
Percent of total noble gases released to the sump solution	0
Percent of other fission products released to the sump solution	1
Percent of total energy (beta + gamma) absorbed by solution	100
Molecules of hydrogen produced per 100 eV of energy absorbed by solution	0.50

Table 6.2-35
FISSION PRODUCT DECAY ENERGY IN SUMP SOLUTION (HISTORICAL)

<u>Time After</u> <u>LOCA</u>	<u>Sump Fission Product Energy^a</u>	
<u>Days</u>	<u>Energy Release Rate</u> <u>Watts/MWt</u>	<u>Integrated Energy Release</u> <u>Watt-Days/MWt</u>
1	2.32E+02	4.87E+02
5	7.59E+01	9.32E+02
10	4.69E+01	1.23E+03
15	3.35E+01	1.43E+03
20	2.66E+01	1.58E+03
25	2.26E+01	1.70E+03
30	2.00E+01	1.81E+03
40	1.62E+01	1.98E+03
50	1.35E+01	2.13E+03
60	1.14E+01	2.26E+03
70	9.81E+00	2.36E+03
80	8.79E+00	2.45E+03
90	8.29E+00	2.54E+03
100	8.31E+00	2.62E+03

- a. Considers release of 50 percent of core halogens, no noble gases and 1 percent of other fission products to the sump solution.

Table 6.2-36
FISSION PRODUCT DECAY ENERGY IN THE CORE (HISTORICAL)

<u>Time After</u> <u>LOCA</u>	<u>Core Fission Product Energy^a</u>	
<u>Days</u>	<u>Energy Release Rate</u> <u>Watts/MWt</u>	<u>Integrated Energy Release</u> <u>Watt-Days/MWt</u>
1	4.57E+03	6.36E+03
5	3.07E+03	2.08E+04
10	2.44E+03	3.44E+04
15	2.08E+03	4.56E+04
20	1.84E+03	5.54E+04
25	1.67E+03	6.41E+04
30	1.54E+03	7.22E+04
40	1.35E+03	8.65E+04
50	1.20E+03	9.92E+04
60	1.08E+03	1.11E+05
70	9.90E+02	1.21E+05
80	9.14E+02	1.30E+05
90	8.54E+02	1.39E+05
100	8.09E+02	1.48E+05

- a. Considers 50 percent of core halogens, no noble gases and 99 percent of other fission products in the core.

6.3 **EMERGENCY CORE COOLING SYSTEM (ECCS)**

6.3.1 **DESIGN CRITERIA**

The following design criteria 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 (see Section 3.1.1). Conformance with the 1972 General Design Criteria of 10 CFR 50, Appendix A, is discussed in Section 3.1.2. The criteria discussed in Section

3.1.2 as they apply to the Emergency Core Cooling System (ECCS) include 35, 36, and 37.

6.3.1.1 **Emergency Core Cooling System (ECCS) Capability**

CRITERION: An Emergency Core Cooling System (ECCS) with the capability for accomplishing adequate emergency core cooling shall be provided. This core cooling system and the core shall be designed to prevent fuel and clad damage that would interfere with the emergency core cooling function and to limit the clad metal-water reaction to acceptable amounts for all sizes of breaks in the reactor coolant piping up to the equivalent of a double-ended rupture of the largest pipe. The performance of such an Emergency Core Cooling System (ECCS) is evaluated conservatively in each area of uncertainty (AIF-GDC 44).

Adequate emergency core cooling is provided by the safety injection system that constitutes the Emergency Core Cooling System (ECCS) at Ginna Station. The safety injection systems are the passive accumulators, high-pressure safety injection, and low-pressure (residual heat removal) safety injection and recirculation.

The primary purpose of the safety injection system is to automatically deliver cooling water to the reactor core to limit the fuel clad temperature and thereby ensure that the core will remain intact and in place, with its heat transfer geometry preserved. This protection is prescribed for all break sizes up to and including the hypothetical instantaneous double-ended rupture of the reactor coolant pipe, the rod ejection accident, a steam or feedwater line break, a steam generator tube rupture, and other accidents analyzed in Chapter 15.

To ensure effective cooling of the core, limits on peak clad temperature and local metal-water reaction, as prescribed by 10 CFR 50.46, will not be exceeded.

For any rupture of a steam pipe and the associated uncontrolled heat removal from the core, the safety injection system adds shutdown reactivity so that even with a stuck control rod, loss of offsite power, and minimum engineered safety features, the limits of 10 CFR 50.46 and 10 CFR 100 are met. The ability of the safety injection system to meet its capability objectives is presented in Section 6.3.3.

Redundancy and separation of instrumentation and components are incorporated to ensure that postulated malfunctions will not impair the ability of the system to meet the design objectives including the single-failure criterion.

6.3.1.2 Inspection of Emergency Core Cooling System (ECCS)

CRITERION: Design provisions shall, where practical, be made to facilitate access to physical inspection of all critical parts of the Emergency Core Cooling System (ECCS), including reactor vessel internals and water injection nozzles (AIF-GDC 45).

Design provisions are made to the extent practical to facilitate access to the critical parts of the reactor vessel internals, injection nozzles, pipes, valves, and safety injection pumps for visual, boroscopic, and ultrasonic inspection for erosion, corrosion, and vibration wear evidence, and for nondestructive test inspection where such techniques are desirable and appropriate.

6.3.1.3 Testing of Emergency Core Cooling System (ECCS) and Components

CRITERION: Design provisions shall be made so that components of the Emergency Core Cooling System (ECCS) can be tested periodically for operability and functional performance (AIF-GDC 46).

Capability shall be provided to test periodically the operability of the Emergency Core Cooling System (ECCS) up to a location as close to the core as is practical (AIF-GDC 47).

Design provisions are made so that active components of the safety injection system can be tested periodically for operability and functional performance.

Each active component can be individually actuated on the normal power source at any time during plant operation.

The safety injection pumps can be tested periodically during plant operation using the minimum flow recirculation lines in accordance with the inservice pump and valve testing program. The residual heat removal pumps are used every time the residual heat removal loop is put into operation, as well as being periodically tested. All remote operated valves are exercised and actuation circuits are tested during routine plant maintenance.

The accumulators are tested for flow during startup after a MODE 6 (Refueling) shutdown. Accumulator flow is measured when valves in the accumulator test line are opened during the test. This flow is recirculated back to the refueling water storage tank (RWST).

See Section 6.3.5 for a more detailed description of current testing provisions.

6.3.1.4 Testing of Operational Sequence of Emergency Core Cooling System (ECCS)

CRITERION: Capability shall be provided to test initially, under conditions as close as practical to design, the full operational sequence that would bring the Emergency Core Cooling System (ECCS) into action, including the transfer to alternate power sources (AIF-GDC 48).

The design provides for capability to test initially, to the extent practical, the full operational sequence up to the design conditions for the safety injection system to demonstrate the state of readiness and capability of the system. Details of the operational sequence testing are presented in Section 6.3.5, Tests and Inspections. The functional test that was performed

during startup is described in Section 5.4.5.5 and Section 14.6.1. (See also Section 3.1.1.7.12.).

6.3.1.5 Service Life

All portions of the system located within the containment are designed to operate without benefit of maintenance and without loss of functional performance for the duration of time the component is required.

6.3.1.6 Codes and Classifications

The safety injection system and components were designed, fabricated, inspected, and tested in conformance with the applicable codes listed in Table 3.2-1.

As part of the Systematic Evaluation Program (SEP), the codes, standards, and classifications to which the station was built were compared to current code requirements. Details of the comparison which includes the safety injection systems are found in Section 3.2. The seismic qualification of piping and system components is discussed in Section 6.3.3.10. The original quality standards of the safety injection system components are tabulated in Table 6.3-1.

6.3.2 *SYSTEM DESIGN AND OPERATION*

6.3.2.1 System Description

6.3.2.1.1 General

The safety injection system is shown in Drawing 33013-1262, Sheets 1 and 2. The low-head safety injection portion of the system (residual heat removal) is shown in Drawing 33013-1247.

Adequate core cooling following a loss-of-coolant accident is provided by the safety injection (emergency core cooling) system, which operates as follows:

1. Injection of borated water by the passive accumulators.
2. Injection by the high-pressure safety injection pumps drawing borated water from the refueling water storage tank (RWST).
3. Injection by the residual heat removal pumps also drawing borated water from the refueling water storage tank (RWST).
4. Recirculation of reactor coolant and injection water from the containment sump to the reactor coolant system by the residual heat removal pumps.

The principal components of the safety injection system which provide core cooling immediately following a loss-of-coolant accident are the accumulators (one for each loop), the three 50% capacity safety injection (high-head) pumps and the two 100% capacity residual heat removal (low-head) pumps. The safety injection and residual heat removal pumps are located in the auxiliary building.

Sump B has been designed to protect against the entrance of debris through the use of concrete curbing around Sump B, a steel plate sump cover, and multiple strainer modules

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constructed of stainless steel plate with 1/16" perforations (see Section 6.3.2.2.6). The potential for a primary coolant pipe break to damage thermal insulation on the piping as well as that on nearby components, transport it to water sources used for long-term post loss-of-coolant accident (LOCA) recirculation (sump B) and containment spray, and deposit it on debris screens or suction strainers was evaluated and resolved as part of Generic Letter 85-22 and NRC Unresolved Safety Issue A-43. Plant insulation surveys, development of methods for estimating debris generation and transport, debris transport experiments, and information provided as public comments on the findings have shown that debris blockage effects are dependent on the types and quantities of insulation employed, the primary system layout within containment, post-LOCA recirculation patterns and velocities, and the post-LOCA recirculation flow rates. In response to NRC Bulletin 93-02 (*Reference 1*), RG&E stated that the actions currently taken at Ginna during MODE 6 (Refueling) shutdowns and prior to return to power operations include administratively controlling equipment that is stored inside containment during reactor operation and ensuring that the containment recirculation sump (sump B) is inspected and cleaned as necessary following maintenance activities.

Generic Letter 98-04 (*Reference 11*) was issued to request information to evaluate programs for ensuring that Service Level 1 protective coatings inside containment do not detach from their substrate during a design basis loss-of-coolant accident (LOCA). In response to NRC Generic Letter 98-04 (*Reference 12*), RG&E stated that RG&E has implemented controls for the procurement, application, and maintenance of Service Level 1 protective coatings used inside the containment at Ginna. RG&E also described conformance with 10 CFR 50.46(b)(5), "Long-term cooling", and the ability to provide extended decay heat removal.

In its letter dated July 15, 2005, as supplemented on August 31, 2005, R. E. Ginna provided its 90 day response to NRC Generic Letter (GL) 2004-02, "Potential Impact of Debris Blockage on Emergency Recirculation During Design Basis Accidents at Pressurized Water Reactors." In GL 2004-02, the NRC indicated that the primary objective of its technical assessment of Generic Safety Issue (GSI) 191, "Assessment of Debris Accumulation on PWR Sump Performance", was to assess the likelihood that the emergency core cooling system and containment spray system pumps would experience a debris-induced loss of net positive suction head (NPSH) margin during sump recirculation. The NRC also discussed the need to complete corrective actions, including any plant modifications, to address the concerns in GL 2004-02. In addition, the NRC staff stated that all actions should be initiated during the first refueling outage after April 1, 2006, and completed by December 31, 2007. In a letter dated June 29, 2006, Ginna updated its prior response to GL 2004-02 by informing the NRC that it was suspending the activities related to an active sump strainer design and was evaluating the procurement of a replacement passive strainer technology. On July 27, 2006 Ginna submitted a request for extension for completing the corrective actions discussed in GL 2004-02 beyond December 31, 2007. Ginna stated that during its fall 2006 refueling outage, it would install new interim passive strainer modules, which would increase the available screen area by approximately 600 square feet. Ginna stated it would also install a new flow diverter wall in the basement of containment in order to reduce the direct transport path of debris into the recirculation sump from a postulated break in the "B" reactor coolant system compartment, which is the location of the highest postulated generated debris during a loss-of-coolant accident (LOCA). The justification for the extension request was provided in that submittal. The NRC responded in a letter dated October 4, 2006 (*Reference 14*) stating that the

installation of the modifications described above would put mitigation measures in place to adequately reduce the risk for the requested short extension period and, therefore, it is acceptable to extend the completion date for the corrective actions for the issues described in GL 2004-02 until the completion of the Ginna spring 2008 refueling outage.

During the 2008 refueling outage, these interim measures were removed and replaced with a passive sump strainer with approximately 4000 ft² of surface area. Supplementary responses to GL 2004-02, dated February 29, 2008 and July 25, 2008, addressed Requests for Additional Information related to the previous responses. Testing and analyses performed on the Sump "B" strainer resulted in the conclusion that all concerns associated with GL 2004-02 were adequately addressed by the 2008 refueling outage Sump "B" strainer installation.

Analyses (*References 16, 17, and 18*) conforming to the methods developed in WCAP-16530-NP, were performed to determine the quantity of chemical precipitant formation in the containment recirculation pool. Likewise, analyses (*References 19 and 20*) were performed to determine the quantity of post-LOCA debris and its transport to Sump "B". Sump pH calculated in Reference 17 was confirmed to be conservative by a more recent evaluation (See Section 6.1.2.1.4). A scale model of the strainers was tested in a tank, simulating the containment recirculation pool, using scaled quantities of chemical precipitant and other fiber and particulate debris. The test results (*References 21 and 22*) indicate that the 4000 ft² of Sump "B" strainers can withstand the worst case containment post-LOCA debris load, ensuring that the RHR pumps have ample NPSH margin to meet the design flow requirements.

Additional analysis (*Reference 23*) was performed to determine the effect that debris, which bypasses the Sump "B" strainers, has on downstream equipment relied upon during the post-LOCA recovery period. The analysis was performed using the methodology established in topical report WCAP-16406-P. The results of the analysis show that no component is adversely impacted by the debris bypass over the 30 day post-LOCA period. Similarly, the analysis showed that there is an insufficient quantity of debris fiber to block the flow area at the top of the core.

An evaluation of (*Reference 24*) was performed by Westinghouse, using the LOCADM code, to predict the growth of fuel cladding deposits and to determine the clad/oxide interface temperature that results from coolant impurities entering the core following a LOCA. These results show that the calculated fuel cladding deposits and clad/oxide interface temperature do not challenge the acceptance criteria.

The stated acceptance criterion is that the maximum cladding temperature maintained during periods when the core is covered will not exceed a core average clad temperature of 800°F. This acceptance basis is applied after the initial quench of the core and is consistent with long-term cooling requirements stated in 10 CFR 50.46 (b)(4) and 10 CFR 50.46 (b)(5). The maximum fuel clad temperature determined through this analysis was substantially below the acceptance criteria.

An additional acceptance criterion is to demonstrate that the total debris deposition on the fuel rods (oxide + crud + precipitate) is less than 50 mils. This acceptance criterion is based on the maximum acceptable deposition thickness before bridging of adjacent fuel rods by

debris is predicted to occur. Again, the deposition thickness determined through this analysis was substantially below the acceptance criteria.

6.3.2.1.2 Injection Phase

The system passive components (accumulators) function to discharge at least 2550 ppm boric acid water into the cold legs of the reactor coolant piping thus ensuring immediate core cooling. To provide protection for large area ruptures in the reactor coolant system, the safety injection system must respond to rapidly reflood the core following the rapid depressurization and core voiding that is characteristic of large area ruptures. The accumulators act to perform the rapid reflooding function with no dependence on normal or emergency power sources, and also with no dependence on receipt of an actuation signal.

The accumulators are a passive safety feature in that they perform their design function in the total absence of an actuation signal or power source. The only moving parts in the accumulator injection train are the check valves. The path of the check valves is exposed to fluid of relatively low boric acid concentration contained within the reactor coolant loop. Even if some unforeseen deposition accumulated, the differential pressure would be sufficient to allow fluid to be injected.

The check valves operate in the closed position with a nominal differential pressure across the disk of approximately 1535 psi. They remain in this position except for testing or when called upon to function. Since the check valves operate normally in the closed position and are therefore not subject to the abuse of flowing operation or impact loads caused by sudden flow reversal and seating, they do not experience any wear of the moving parts.

Three safety injection pumps (50% capacity each) discharge to loops A and B cold legs.

The function of the high-head safety injection and low-head residual heat removal pumps is to complete the refilling of the vessel and ultimately return the core to a subcooled state. The flow from any two safety injection pumps and one residual heat removal pump is sufficient to provide the required flow to maintain the fuel within required safety limits, as specified in 10 CFR 50.46 and 10 CFR 50, Appendix K. The starting sequence of the safety injection pumps, residual heat removal (RHR) pumps, and the related emergency power equipment is designed so that delivery of the full rated flow is reached at 32 seconds for SI and 30 seconds for RHR.

During the injection phase, the high-head safety injection pumps do not depend on other fluid systems, with the exception of the suction line from the refueling water storage tank (RWST). During the recirculation phase of the accident for small breaks, suction to the high-head safety injection pumps can be provided by the residual heat removal pumps. The residual heat removal (low-head) pumps are normally used during reactor shutdown operations. Whenever the reactor is at power, the pumps are aligned for low-head safety injection.

The safety injection system is designed such that a single active failure of both a residual heat removal pump and a safety injection pump will not prevent the system from fulfilling its design function.

Two independent parallel flow paths from the boric acid storage tanks, each with two motor-operated valves in series (826A, B, C, and D), were originally utilized to permit 12 wt % boric acid to flow to the suction of the safety injection pumps. The system was modified to isolate these flow paths by closing these valves with ac power removed as specified in the Technical Specifications.

6.3.2.1.3 Recirculation Phase

After the injection phase, coolant spilled from the break and water entering containment by operation of the containment spray system and the safety injection system are collected in containment sump B. The collected water is recirculated back to the reactor coolant system by the safety injection system. The residual heat removal pumps take a suction from containment sump B via two independent recirculation sump lines. The residual heat removal pumps provide a net positive suction head for the safety injection pumps during high-head recirculation to cold legs A and B, if needed, by remote manual operator manipulation of valves.

An alternative flow path from the residual heat removal A heat exchanger to safety injection pump C was added during initial construction in advance of proposed changes to AEC regulation. Use of this flow path requires both local manipulation of manual valves (1816A and 1816B) and remote manipulation of motor operated valves (1815A and 1815B). The alternate flow path has been determined not to be required and is not credited within the Ginna current licensing basis. Because of this, use of the alternate flow path is not reflected in emergency operating procedures (EOPs).

6.3.2.2 Component Description

6.3.2.2.1 Accumulators

The accumulators are pressure vessels filled with borated water and pressurized with nitrogen gas. During MODES 1 and 2 and MODE 3 with pressure >1600 psig, each accumulator is isolated from the reactor coolant system by two check valves in series. Should the reactor coolant system pressure fall below the accumulator pressure, the check valves open and borated water is forced into the reactor coolant system. Mechanical operation of the swing-disk check valves is the only action required to open the injection path from the accumulators to the core via the cold leg.

The accumulators are passive engineered safety features because of the gas forced injection; no external source of power or signal transmission is needed to obtain fast-acting, high-flow capability when the need arises. One accumulator is attached to each of the cold legs of the reactor coolant system. The design capacity of the accumulators is based on the assumption that flow from one of the accumulators spills onto the containment floor through the ruptured loop and flow from the remaining accumulator provides sufficient water for the initial core cooling requirements.

The accumulators are carbon steel, clad with stainless steel, and designed to ASME III, Class A. They are located inside the containment but outside the missile barrier; therefore, each is protected against possible missiles. Connections for remotely draining or filling the fluid and gas space during MODES 1 and 2 are provided.

The design parameters of the accumulators are shown in Table 6.3-2.

The accumulators provide redundant level and pressure indications with readouts on the control board. Each indicator is equipped with high and low-level alarms. Specific requirements for accumulator operability, including level, pressure, and boron concentration values, are stated in the Technical Specifications.

6.3.2.2.2 Safety Injection Pumps

6.3.2.2.2.1 Operation

The three high-head safety injection pumps for supplying borated water to the reactor coolant system are horizontal centrifugal pumps driven by electrical motors. Parts of the pumps in contact with borated water are stainless steel or equivalent corrosion resistant material. A minimum flow bypass line with a fixed in-line orifice is provided on each pump discharge to recirculate flow to the refueling water storage tank (RWST) whenever the associated safety injection pump starts. This configuration ensures an acceptable pump minimum flow whenever the main safety injection flow path is passing little or no flow. The bypass line is a 1.50-in. line that includes a 100-gpm flow orifice. The three bypass lines discharge to a 2-in. common header that discharges to the refueling water storage tank (RWST). The pump bypass can be isolated by two motor-operated valves (MOV 897 and 898) in series in the header. These valves are normally open and fail as is on loss of power (see Table 6.3-8). The bypass system is isolated from the refueling water storage tank (RWST) during high-head recirculation when water from the containment sump is recirculated through the residual heat removal pumps to the safety injection pumps and to the reactor coolant system cold legs. Each of the safety injection pumps are sized at 50% of the capacity required to meet the design criteria outlined in Section 6.3.1.

The safety injection pump characteristic curves of total dynamic head as a function of flow are shown in Figure 6.3-2.

The safety injection pumps are provided continuously with service water flow to the outboard (thrust) bearing housing using a water jacket that surrounds the oil bath. During the injection phase post-accident and during line-up for reduced inventory operation, the source of water is the refueling water storage tank (RWST). Since the temperature of the water source is not elevated during these modes, the pump would remain operable in the event that service water was isolated. During the recirculation phase, service water flow is needed to ensure pump reliability, because the temperature of the water source (from containment sump B through the residual heat removal (RHR) pumps and heat exchangers) would be much higher. The pumps are also provided continuously with component cooling water to the inboard and outboard mechanical seal cooling heat exchangers. For the same reasons, component cooling water flow to these heat exchangers is not essential until switchover to the sump recirculation phase. Since emergency procedures do not contain specific instructions concerning these flows, service water and component cooling water is continuously available while the plant is operating in MODES 1-4.

Two residual heat removal pumps used for low-pressure safety injection also supply emergency core cooling water to the reactor coolant system. These centrifugal pumps are driven by electric motors. Details of the residual heat removal pumps are presented in the discussion of the residual heat removal system, Section 5.4.5. Table 6.3-3 gives the design

parameters of the safety injection and residual heat removal pumps. The residual heat removal pump characteristic curves of total dynamic head as a function of flow are shown in Figure 6.3-3.

6.3.2.2.2.2 *Pump Design and Fabrication*

The pressure-containing parts of the safety injection pumps are static castings conforming to ASTM A-351 grade CF8 or CF8M. Stainless steel forgings are procured per ASTM A-182 grade F304 or F316 or ASTM A-336, Class F8 or F8M, and stainless plate is constructed to ASTM A-240, type 304 or 316. All bolting material conforms to ASTM A-193. Materials such as weld-deposited Stellite Colmonoy are used on the pump's shaft sleeve and proven material combinations are used at points of close running clearances in the pumps to prevent galling and to ensure continued performance ability in high velocity areas subject to erosion. All pressure-containing parts of the pumps were chemically and physically analyzed and the results were checked to ensure conformance with the applicable ASTM specification. In addition, all pressure-containing parts of the pump were liquid penetrant examined in accordance with Appendix VIII of Section VIII of the ASME Boiler and Pressure Vessel Code (ASME Code). The acceptance standard for the liquid penetrant test was USAS B31.1, Code for Pressure Piping, Case N-10.

The pump design was reviewed with special attention to the reliability and maintenance aspects of the working components. Specific areas included evaluation of the shaft seal and bearing design to determine that adequate allowances had been made for shaft deflection and clearances between stationary parts.

Where welding of one pressure-containing part to another was necessary, a welding procedure including joint detail was submitted for review and approval by Westinghouse. The procedure included evidence of qualification necessary for compliance with Section IX of the ASME Code Welding Qualifications. This requirement for compliance with Section IX also applies to any repair welding performed on pressure-containing parts. In addition to the above requirements, these welds were radiographed in accordance with Paragraph UW-51 of Section VIII of the ASME Code and subsequently liquid penetrant examined in accordance with Appendix VIII of Section VIII of the ASME Code.

The pressure-containing parts of the pump were assembled and hydrostatically tested to 1.5 times the design pressure for 30 min. Each pump was given a complete shop performance test in accordance with Hydraulic Institute Standards. The pumps were run at design flow and head, shutoff head, and three additional points to verify performance characteristics.

6.3.2.2.3 Refueling Water Storage Tank (RWST)

The refueling water storage tank (RWST) is shown in Drawing 33013-1261. In addition to its usual duty to supply borated water for MODE 6 (Refueling) operations, this tank provides the source of borated water to the safety injection pumps and the residual heat removal pumps for the loss-of-coolant accident.

The capacity of the refueling water storage tank (RWST) is based on the requirement for filling the refueling canal as well as to provide sufficient sump inventory for recirculation. The capacity of the tank is 338,000 gallons. The minimum level is 300,000 gallons with a boron concentration of at least 2750 ppm and no more than 3050 ppm when above MODE 5 (Cold Shutdown). The maximum level, as assumed in the seismic analysis of the tank, is a level which is 6 in. below the top cylindrical edge of the tank (nominally 331,000 gallons).

The water in the tank is borated to a concentration, which ensures reactor shutdown by at least 5% delta k/k when all rod cluster control assemblies are inserted and when the reactor is cooled down for refueling. The maximum boric acid concentration is approximately 1.75 wt % boric acid. At 32°F the solubility limit of boric acid is 2.2%; therefore, the concentration of boric acid in the refueling water storage tank (RWST) is well below the solubility limit at 32°F and heating is not required since the tank is in the auxiliary building.

The refueling water storage tank (RWST) design parameters are given in Table 6.3-4.

6.3.2.2.4 Heat Exchangers

For postaccident operational purposes, the two residual heat exchangers of the residual heat removal loop cool the recirculated sump water. The normal function of these heat exchangers is the cooldown of the reactor coolant system. This is discussed in Section 5.4.5. Table 6.3-5 gives the design parameters of the heat exchangers.

The design of the heat exchangers is in conformance with ASME Code, Section III, and Section VIII, 1965, and the Tubular Exchangers Manufacturers Association (TEMA) standards for Class R heat exchangers. The comparison of these codes to the current requirements was performed under SEP Topic III-1, Codes and Classifications, and is discussed in Section 3.2.

The residual heat exchangers are conventional shell and U-tube type units having one shell and four tube passes. Tubes are seal-welded to the tube sheet. The channel connections are flanged to facilitate tube bundle removal for inspection and cleaning. Each unit has an SA-106 grade A or B carbon steel shell, an SA-234 carbon steel shell end cap, SA-213 type 304 stainless steel tubes, SA-376 type 304 stainless steel channel, SA-240 type 304 stainless steel channel cover, and an SA-240 type 304 stainless steel tubesheet.

6.3.2.2.5 Boric Acid Storage Tanks

The boric acid storage tanks are shown in Drawing 33013-1266. The boric acid storage tank capacity is sized to store sufficient boric acid solution to attain MODE 5 (Cold Shutdown) shortly after full-power operation is achieved, even if the most reactive control rod is not inserted.

In the original plant design the boric acid storage tanks supplied 12%-13% borated water to the safety injection pumps as the initial source of water during the injection phase and were the primary source of compensation for reactivity inserted by the steam line break event. The system was modified as reflected in Technical Specifications requirements to isolate this flow path.

Thus, the boric acid storage tanks no longer perform a safety function related to the Emergency Core Cooling System (ECCS). See also Section 6.3.2.3.5. The concentration of boric acid minimum volume and minimum solution temperature in storage is maintained in accordance with the Technical Requirements Manual (TRM). Periodic manual sampling and corrective action is provided, if necessary, to ensure that these limits are maintained. As a consequence, measured amounts of boric acid solution can be delivered to the reactor coolant to control the chemical poison concentration. The combination overflow and breather vent connections to the boric acid storage tanks have a water loop seal to minimize vapor discharge during storage of the solution.

Redundant tank heaters and line heat tracing are provided to ensure that the solution will be stored at a temperature in excess of the solubility limit. The tank heating elements are located near the bottom of the tank. Minimum solution temperature limits are specified in the Technical Requirements Manual (TRM). (See Section 9.3.4.2.6.)

The two vertical boric acid storage tanks are constructed of austenitic stainless steel. The design free volume in the cylindrical portion of each tank is 3460 gal, the design temperature is 250°F, and the design pressure is atmospheric.

6.3.2.2.6 **Containment Sump B**

Containment sump B collects liquid discharged into the containment following a loss-of-coolant accident and then provides the source of water for long-term recirculation. The sump is located below the basement floor level of containment, surrounded by a 6-inch concrete curb. The entire top surface of the sump is covered with solid metal decking, through which passive strainer modules direct recirculation flow into the sump and the residual heat removal pump suction lines. The sump strainer consists of 3 trains, totaling 16 strainer modules. Modules of each train are connected through a center channel that directs flow to the sump. The total effective strainer surface area is approximately 4000 ft². Each strainer module is constructed of perforated stainless steel plate. The strainer perforated hole size and mechanical fit-up gaps do not exceed 1/16". Multiple penetrations through the sump cover are provided to accommodate various lines, conduits, and valve reach rods. No gap or clearance between the sump cover and these penetrations exceed 1/16". Therefore, all recirculated water entering the pump suction must first pass through the passive sump strainers before entering the pump. The recirculated water will be devoid of any debris greater than 1/16" in diameter.

6.3.2.2.7 **Valves**

6.3.2.2.7.1 **General**

All parts of valves used in the safety injection system in contact with borated water are austenitic stainless steel or equivalent corrosion-resistant material. The carbon steel and ASTM A-193 type 410 stainless steel bolting in borated water systems were replaced with ASTM type 17-4 PH stainless steel to improve corrosion resistance and resistance to cracking. The carbon steel nuts were replaced with type 316 stainless steel. The borated water systems included safety injection and residual heat removal. The motor operators on the low-pressure safety injection deluge line open in the event of a safety injection signal. The injection valves for the high-pressure safety injection are locked in position as needed to direct the safety

injection flow. All valves required for operation of the system have remote position indication in the control room.

Valving is specified for exceptional tightness and, where possible, instrument valves and packless diaphragm valves are used. All valves, except those which perform a control function, are provided with backseats which are capable of limiting leakage to less than 1.0 cm³/ hr/in. of stem diameter, assuming no credit taken for valve packing. Normally closed globe valves are installed with recirculation flow under the seat to prevent leakage of recirculated water through the valve stem packing. Relief valves are totally enclosed. Control and motor-operated valves, 2 in. and above, which are exposed to recirculation flow, were originally provided with double-packed stuffing boxes and stem leakoff connections which are piped to the waste disposal system. Under the Valve Packing Improvement Program, packing arrangements were modified and leakoff connections closed off.

The check valves that isolate the safety injection system (10 in. piping) from the reactor coolant system cold leg piping are installed immediately adjacent to the reactor coolant piping to reduce the probability of an injection line rupture causing a loss-of-coolant accident. The check valves that isolate the safety injection system (2 in. piping) from the reactor coolant system hot leg piping are installed within approximately 8 ft. of the reactor coolant piping to also minimize this probability.

Relief valves are installed in the safety injection pump discharge headers to prevent overpressure in the lines which have a lower design pressure than the reactor coolant system. The relief valves are set at a pressure that is greater than the maximum pump discharge pressure and less than the design capability of the safety injection discharge piping. A relief valve is provided in the accumulator test line for thermal overpressure protection during alignments that isolate the associated piping.

The gas relief valves on the accumulators protect them from pressures in excess of the design value.

A pressure relief valve is installed between each of the two sets of boric acid storage tank isolation valves, 826A and B and 826C and D. These relief valves are provided to avoid an excessive buildup of pressure in the isolated line. Heat tracing remains installed in this line, although the line provides no active function since being isolated during boron reduction plant modifications.

6.3.2.2.7.2 *Motor-Operated Valves*

The pressure-containing parts (body, bonnet, and disks) of the valves employed in the safety injection system were designed per criteria established by the USAS B16.5 or MSS SP66 specifications. The materials of construction for these parts were procured per ASTM A182, F316 or A351, grade CF8M, CF8. All material in contact with the primary fluid, except the packing, is austenitic stainless steel or equivalent corrosion-resistant material. The pressure-containing cast components were radiographically inspected as outlined in ASTM E-71 Class 1 or Class 2. The body, bonnet, and disks were liquid penetrant inspected in accordance with ASME Boiler and Pressure Vessel Code Section VIII, Appendix VIII. The liquid penetrant acceptable standard was as outlined in USAS B31.1 Case N-10.

The body-to-bonnet joint when a gasket is employed was designed per ASME Boiler and Pressure Vessel Code Section VIII or USAS B16.5 with a fully trapped, controlled compression, spiral-wound gasket with provisions for seal welding, or of the pressure seal design with provisions for seal welding. The body-to-bonnet bolting and nut materials were procured per ASTM A193 and A194, respectively.

The entire assembled unit was hydrotested as outlined in MSS SP-61 with the exception that the test was maintained for a minimum period of 30 minutes/in. of wall thickness. Any leakage was cause for rejection. The seating design is of the Darling parallel disk design, the Crane flexible wedge design, or the equivalent. These designs have the feature of releasing the mechanical holding force during the first increment of travel; thus, the motor operator has to work only against the frictional component of the hydraulic imbalance on the disk and the packing box friction. The disks are guided throughout the full disk travel to prevent chattering and provide ease of gate movement. The seating surfaces are hard faced (Stellite No. 6 or equivalent) to prevent galling and reduce wear.

The stem material is ASTM type 316 condition B or precipitation hardened ASTM 17-4 PH stainless procured and heat treated. These materials were selected because of their corrosion resistance, high tensile properties, and resistance to surface scoring by the packing.

The motor operator unit incorporates a hammer blow feature which allows the motor to impact the disks away from the fore or backseat upon opening or closing. The hammer blow feature not only impacts the disk but allows the motor to attain its operational speed.

The valves were assembled, hydrostatically tested, seat-leakage tested (fore and back), operationally tested, cleaned, and packaged per specifications. All manufacturing procedures employed by the valve supplier, such as hard facing, welding, repair welding, and testing, were submitted to Westinghouse for approval.

Valves which must function against system pressure are designed so that they function with a pressure drop equal to full system pressure across the valve disk.

6.3.2.2.7.3 *Manual Valves*

The stainless steel manual globe, gate, and check valves were designed and built in accordance with the requirements outlined in the motor-operated valve description above.

The carbon steel valves were built to conform with USAS B16.5. The construction materials of the body, bonnet, and disk conform to the requirements of ASTM A105 grade II, A181 grade II or A216 grade WCB or WCC. The carbon steel valves pass only nonradioactive fluids and were subjected to the hydrostatic test as outlined in MSS SP-61 except that the test pressure was maintained for at least 30 min/in. of wall thickness. Since the fluid controlled by the carbon steel valves is not radioactive, the double packing and seal-weld provisions are not provided.

6.3.2.2.7.4 *Accumulator Check Valves*

The pressure-containing parts of this valve assembly were designed in accordance with MSS SP-66. All parts in contact with the operating fluid are of austenitic stainless steel or of

equivalent corrosion-resistant materials procured to applicable ASTM or WAPD specifications. The cast pressure-containing parts were radiographed in accordance with ASTM E-94 and the acceptance standard was as outlined in ASTM E-71. The cast pressure-containing parts, machined surfaces, finished hard facings, and gasket-bearing surfaces were liquid penetrant inspected per ASME Code, Section VIII, and the acceptance standard was as outlined in USAS B31.1 Code Case N-10. The final valve was hydrotested per MSS SP-66 except that the test pressure was maintained for at least 30 min. The seat leakage test was conducted in accordance with the manner prescribed in MSS SP-61 except that the acceptable leakage was $2 \text{ cm}^3/\text{hr}/\text{in.}$, nominal pipe diameter.

The valve is designed with a low-pressure drop configuration with all operating parts contained in the working parts within the body, which eliminates those problems associated with packing glands exposed to boric acid. The clapper arm shaft was manufactured from 17-4 PH stainless steel heat treated to Westinghouse specifications. The clapper arm shaft bushings were manufactured from Stellite No. 6 material. The various working parts were selected for their corrosion-resistant, tensile, and bearing properties.

The disk and seat rings were manufactured from a forging. The mating surfaces are hard faced with Stellite No. 6 to improve the valve seating life. The disk is permitted to rotate, providing a new seating surface after each valve opening.

6.3.2.2.7.5 *Leakage Limitations*

Small, normally open valves have backseats which were designed to be capable of limiting leakage to less than $1 \text{ cm}^3/\text{hr}/\text{in.}$ of stem diameter, assuming no credit for packing in the valve. Normally closed globe valves are installed with recirculation flow under the seat to prevent stem leakage from the more radioactive fluid side of the seat.

The specified leakage across the valve disk required to meet the equipment specification and hydrotest requirements is as follows:

1. Conventional globe - $3 \text{ cm}^3/\text{hr}/\text{in.}$ of nominal pipe size.
2. Gate valves - $3 \text{ cm}^3/\text{hr}/\text{in.}$ of nominal pipe size; $10 \text{ cm}^3/\text{hr}/\text{in.}$ for 300-lb and 150-lb USA Standard.
3. Motor-operated gate valves - $3 \text{ cm}^3/\text{hr}/\text{in.}$ of nominal pipe size; $10 \text{ cm}^3/\text{hr}/\text{in.}$ for 300-lb and 500-lb USA Standard.
4. Check valves - $3 \text{ cm}^3/\text{hr}/\text{in.}$ of nominal pipe size; $10 \text{ cm}^3/\text{hr}/\text{in.}$ for 300-lb and 150-lb USA Standard.
5. Accumulator check valves - $2 \text{ cm}^3/\text{hr}/\text{in.}$ of nominal pipe size.

Leakage originally assumed from components of the recirculation loop, including valves, is tabulated in Table 6.3-6. Types of component leakage control were modified from the original design; however, when combined the leakage is within the value assumed in the current analysis (see Section 15.6.4.2.5). Leakage from recirculation systems outside containment is checked by procedure.

6.3.2.2.8 Piping

6.3.2.2.8.1 General

All safety injection system piping in contact with borated water is austenitic stainless steel. Piping joints are welded except for the flanged connections at the safety injection pumps and flow orifices. See Section 5.4.5.2.2.4 for information about the residual heat removal system and Section 6.2.2.2.2.3 for the containment spray system.

The piping beyond the accumulator stop valves is designed for reactor coolant system conditions (2485 psig, 650°F). All other piping connected to the accumulator tanks is designed for 900 psig and 650°F.

The safety injection pump suction piping is designed for low-pressure losses to meet net positive suction head requirements of the pumps. The piping from the refueling water storage tank (RWST) to valves 825A and 825B is designed for 210 psig at 300°F. The piping from these valves to the suction of the safety injection pumps is designed for 495 psig at 300°F.

The safety injection high-pressure branch lines (1500 psig at 300°F) are designed for high-pressure losses to limit the flow rate out of the branch line which may have ruptured at the connection to the reactor coolant loop. Globe valves are provided in the safety injection lines to provide a balanced safety injection flow split. The system design incorporates the ability to isolate the safety injection pumps on separate headers so that full flow from at least one pump is ensured should a branch line break.

6.3.2.2.8.2 Design Criteria

The piping was designed to meet the minimum requirements set forth in (1) USAS B31.1 Code for the Pressure Piping, (2) Nuclear Code Case N-7, (3) USAS Standards B36.10 and B36.19, (4) ASTM Standards, and (5) supplementary standards plus additional quality control measures.

Minimum wall thicknesses were determined by the USAS Code formula found in the power piping Section 1 of the USAS Code for Pressure Piping. This minimum thickness is increased to account for (1) the manufacturer's permissible tolerance of minus 12.5% on the nominal wall and (2) a 10% allowance for wall thinning on the external radius during any pipe bending operations in the shop fabrication of the subassemblies. Purchased pipe and fittings have a specified nominal wall thickness that is no less than the sum of that required for pressure containment, mechanical strength, manufacturing tolerance, and an allowance for wall thinning associated with shop bending.

6.3.2.2.8.3 Design Review

An engineering review of Ginna Station nuclear valves was conducted during the 1974-1975 time period. The review was the first phase of a program to demonstrate acceptable wall thickness on certain valves important to nuclear safety. The measurement program, based on the design and manufacturing requirements of ANSI B16.5 or MSS SP-66, found that the valves either met the requirements or, in the case of one valve, were repaired to meet the requirements. This review is discussed in Section 5.4.9.2.

Thermal and seismic piping flexibility analyses were performed. Special attention was directed to the piping configuration at the pumps with the objective of minimizing pipe imposed loads at the suction and discharge nozzles.

6.3.2.2.8.4 *Materials*

Pipe and fitting materials were procured in conformance with all requirements of the ASTM and USAS specifications. All materials were verified in conformance to specifications and documented by certification of compliance to ASTM material requirements. Specifications imposed additional quality control upon the suppliers of pipes and fittings as listed below.

1. Check analyses were performed on both the purchased pipe and fittings.
2. Pipe branch lines between the reactor coolant pipes and the isolation stop valves conform to ASTM A376 and meet the supplementary requirement S6 ultrasonic testing.
3. Fittings conform to the requirements of ASTM A403. The performance of tension tests as defined by supplementary requirement S4 is required.

6.3.2.2.8.5 *Welding and Fabrication*

Welds for pipes sized 2-1/2 in. and larger were of the full penetration type. Reducing tees were used where the branch size exceeded one-half of the header size. Branch connections of sizes that were equal to or less than one-half of the header size were of a design that conformed to the USAS rules for reinforcement set forth in the USAS B31.1 Code for Pressure Piping. Bosses for branch connections are attached to the header by means of full penetration welds.

All welding was performed in accordance with the ASME Code, Section IX, Welding Qualifications. The shop fabricator was required to submit all welding procedures and evidence of qualifications for review and approval prior to release for fabrication. All welding materials used by the shop fabricator had prior approval.

Butt welds of all high-pressure piping containing radioactive fluid at greater than 600°F temperature and 600 psig pressure or equivalent were radiographed. The remaining piping butt welds were randomly radiographed. The technique and acceptance standards were outlined in UW-S1 of the ASME Code, Section VIII. In addition, butt welds were liquid penetrant examined in accordance with the procedure of ASME Code, Section VIII, Appendix VIII, and the acceptance standard as defined in the USAS Nuclear Code Case N-10. Finished branch welds were liquid penetrant examined on the outside and, where size permitted, on the inside root surfaces.

A postbending solution anneal heat treatment was performed on hot-formed stainless steel pipe bends. Completed bends were then completely cleaned of oxidation from all affected surfaces. The shop fabricator was required to submit the bending, heat treatment, and cleanup procedures for review and approval prior to release for fabrication.

General cleaning of completed piping subassemblies (inside and outside surfaces) was governed by basic ground rules set forth in the specifications. For example, these specifications

prohibit the use of hydrochloric acid and limit the chloride content of service water and demineralized water.

6.3.2.2.8.6 *Packaging*

Packaging of the piping subassemblies for shipment was done so as to preclude damage during transit and storage. Openings were closed and sealed with tight fitting covers to prevent entry of moisture and foreign material. Flange facings and weld end preparations were protected from damage by means of wooden cover plates securely fastened in position. The packing arrangement proposed by the shop fabricator was subject to approval.

6.3.2.2.9 Motors

Motor electrical insulation systems were supplied in accordance with USAS, IEEE, and NEMA standards and were tested as required by such standards.

Temperature rise design selection was such that normal long life is achieved even under accident loading conditions. The RG&E environmental qualification review has determined that the motors are capable of performing their safety function in their postaccident environment.

Criteria for motors of the safety injection and residual heat removal systems require that under any anticipated mode of operation, the motor nameplate rating is not exceeded. The motors have a 1.15 service factor for normal operation.

6.3.2.3 System Operation

6.3.2.3.1 Separation

The residual heat removal pumps deliver through two nozzles that penetrate the core barrel above the level of the top of the core. The three high-head safety injection pumps deliver into two separate headers, with one header injecting into each cold leg of the reactor coolant system; therefore, the ability is provided to isolate the pumps on separate headers and thereby ensure the delivery of the full flow from at least one pump for the special case of a broken injection line.

6.3.2.3.2 System Actuation

A safety injection signal is actuated by low pressurizer pressure on two-of-three detectors, two-of-the-three low steam line pressures, two-of-the-three high containment pressures, or manually. The safety injection signal will open the safety injection system isolation valves and start the high-head safety injection pumps and low-head (residual heat removal) safety injection pumps.

The actuation sequence of the safety injection pumps (1A, 1B, 1C) and the residual heat removal pumps is described in Section 8.3.1.2.4. As described in Section 8.3.1.2.4, the sequencing circuit logic ensures that automatic loading of the safety injection pumps onto their specific power supplies is predicted with certainty. Also, the discharge valves 871A and B are interlocked with safety injection pumps 1A and 1B. If pump 1A fails (no power), valve 871B closes and the discharge from pump 1C is directed to loop B cold leg. If pump 1B fails, valve 871A closes and the discharge from pump 1C is directed to loop A cold leg. Thus,

high-pressure safety injection flow is always directed to the cold legs of both loops if one of the three pumps should fail. See Drawing 33013-1262, Sheets 1 and 2.

6.3.2.3.3 Injection Phase

During MODES 1 and 2 the refueling water storage tank (RWST) is aligned to the suction of the high-head safety injection pumps and residual heat removal pumps. The refueling water storage tank (RWST) outlet line leading to the safety injection pumps contains two series motor-operated valves (896A and B) that are aligned in the open position with direct current control power removed by key switch in accordance with the Technical Specifications. Downstream of these valves the line forms two separate flow paths, each containing a motor-operated valve (825A and 825B), that is open with ac power removed in accordance with the Technical Specifications. These four valves, therefore, have no automatic action. These two flow paths are headered together and lead to the suctions of the three safety injection pumps.

The refueling water storage tank (RWST) outlet line leading to the residual heat removal pumps contains a motor-operated valve (856) and check valve. The motor-operated valve is open with direct current control power removed by key switch in accordance with the Technical Specifications. See Drawing 33013-1247.

The refueling water storage tank (RWST) is equipped with two redundant level indication systems. Each system consists of a level transmitter connected to the refueling water storage tank (RWST) that signals a percent scale level indicator on the main control board and actuates two annunciator alarm windows (low-low and high-low) on the main control board by way of two bistables, each corresponding to an alarm window.

Remote operated valves of the safety injection system which are under manual control (i.e., valves which normally are in their ready position and do not receive a safety injection signal) have their positions indicated on a common portion of the control board. If, during operation, one of these valves is not in the ready position for injection, it is shown visually on the board.

Refer to Table 8.3-1 for the engineered safety features automatic actuation sequence and times after the initiation signal for the cases when the normal power source is available and when only the diesel power source is available.

Because the injection phase of the accident is terminated before the refueling water storage tank (RWST) is emptied, all pipes are kept filled with water before recirculation is initiated. The level indicator and alarms on the refueling water storage tank (RWST) give the operator ample warning to control pump operations and to terminate the injection phase. Level indication (LI-942 and LI-943) is provided for containment sump B which also gives an indication of when recirculation can be initiated.

6.3.2.3.4 Recirculation Phase

After the injection phase, coolant spilled from the break and water injected by the safety injection system and the containment spray is cooled and recirculated to the reactor coolant system by the low-pressure safety injection (residual heat removal) system or, if needed, by the high-pressure safety injection system.

If reactor coolant system depressurization to below the shutoff head of the residual heat removal pumps occurs before the injection mode of the safety injection system is terminated, the residual heat removal pumps will be used in the recirculation mode. The residual heat removal pumps will take suction from the containment sump, circulate the spilled coolant through the residual heat removal heat exchangers, and return the coolant to the reactor via the reactor vessel nozzles. If depressurization of the reactor coolant system proceeds slowly, the high-pressure safety injection pumps are aligned to take suction from the residual heat removal pumps, and inject flow into the reactor coolant system cold legs. In the long term, boron precipitation must be avoided. This is discussed in Section 6.3.3.4.

The recirculation sump lines comprise two independent lines which penetrate the containment. Each line has a remote motor-operated valve located inside and outside the containment. Each line is run independently to the suction of a residual heat removal pump. The system permits long-term recirculation in the event of a single credible passive or active component failure.

In the event of a small break loss-of-coolant accident (4 in. and smaller), supplemental cooling is provided by use of steam dump. Steam dump will be directed to the condenser when outside power is available or directly to the atmosphere when outside power is not available. As discussed in Section 15.6.4, the expected clad temperatures for break sizes 4 in. and smaller are limited to a value below which clad bursting is expected.

Alternative flow paths are provided from the discharge of the residual heat exchangers for both low and high-head recirculation. This is described in Section 6.3.2.1.

The sequence for the changeover from injection to recirculation from the time of the safety injection signal is discussed in Section 6.3.3.3. If recirculation is required to the containment spray or high-head safety injection system, flow instrumentation is provided in the line directing flow from the residual heat removal pumps to the containment spray and safety injection pumps via valves 857A, B, and C. Also, safety injection flow indication is available. The quantity of recirculated fluid being injected into the containment via the spray header can thus be determined. Valves 857A, B, and C are normally closed during safeguards alignment to eliminate the potential for common-mode failure.

A detailed listing of the instrumentation readouts on the control board, which the operator can monitor during recirculation, is given in Table 6.3-7. The safety injection valve operation, interlocks, and associated design features are given in Table 6.3-8.

6.3.2.3.5 Steam Line Break Protection

A large break of a steam system pipe rapidly cools the reactor coolant causing insertion of reactivity into the core. Compensation is provided by injection of boric acid from the refueling water storage tank (RWST). Redundant valves are open with ac power removed (in accordance with the Technical Specifications), providing an injection of boric acid at a concentration between 2750 and 3050ppm boron. This is sufficient to terminate the reactor power transient before any clad damage results. When above MODE 5 (Cold Shutdown), the Technical Specifications require the refueling water storage tank (RWST) contain not less than 300,000 gal of water with a boron concentration not less than 2750 ppm and no more

than 3050 ppm. In addition, at or above a reactor coolant system temperature of 350°F, ac power shall be removed from the safety injection suction valves (MOV 825A and B) with the valves in the open position and from the boric acid storage tank outlet valves (MOV 826A, B, C, and D) with the valves in the closed position.

6.3.2.3.6 Safety Injection System Leakage Outside Containment

Since radioactive fluid can be circulated outside containment during the recirculation phase, provisions are made such that the collection of discharges from pressure relieving devices is routed into closed systems. Also, means exist to detect and control radioactivity leakage into the environs.

Area radiation monitors provide an indication of leakage in the auxiliary building. The discharges from the auxiliary building are monitored and the ventilation path via the plant vent includes charcoal and high efficiency particulate air filters.

Leakage detection exterior to containment is also possible through use of sump level detection. The auxiliary building sump pumps start automatically in the event that liquid accumulates in the sump and the alarm in the control room indicates that water has accumulated in the sump. This leakage is routed to the waste holdup tank. (See Section 5.4.5.3.5.)

Design-basis leakage is considered in the loss-of-coolant accident analysis (Section 15.6.4.2.5). Even with the maximum credible leakage of the residual heat removal pump seals, offsite doses are maintained below the guideline exposure limits of 10 CFR 100.

6.3.3 DESIGN EVALUATION

6.3.3.1 Range of Core Protection

6.3.3.1.1 Safety Injection Requirements Versus Break Size

The measure of effectiveness of the safety injection system is the ability of the pumps and accumulators to keep the core flooded or to reflood the core rapidly when the core has been uncovered for postulated large area ruptures. The result of this performance is to sufficiently limit any increase in clad temperature below a value where emergency core cooling objectives are met (Section 6.3.1). The range of core protection as a function of break diameter provided by the various components of the safety injection system is presented in Figure 6.3-4. The injection from the following combinations of components meets the core cooling objectives.

- Bar A Two safety injection pumps.
- Bar B Two safety injection pumps and one residual heat removal pump.
- Bar C One residual heat removal pump and one accumulator. ^a
- Bar D Two safety injection pumps, one residual heat removal pump, and one accumulator. ^aNo credit is taken for the accumulator which is attached to the ruptured loop.

a. No credit is taken for the accumulator which is attached to the ruptured loop.

With minimum onsite emergency power available (one-out-of-two diesel generators), the emergency core cooling equipment available is represented by Bar D (two-out-of-three safety injection pumps, one-out-of-two residual heat removal pumps, and one-out-of-two accumulators). With these systems, all required Emergency Core Cooling System (ECCS) requirements per 10 CFR 50.46 are met for all break sizes up to and including the double-ended severance of the reactor coolant pipe.

The remaining three combinations (Bars A, B, and C) represent degraded cases with operation of less than the minimum design emergency core cooling equipment. The operation of two safety injection pumps (Bar A) provides core protection for break sizes up to an equivalent break diameter of 3 to 4 in. The operation of two safety injection pumps would allow flow spilling from a broken safety injection line to go uncorrected by operator action. Isolation of the broken line by operator action would increase the range of protection.

The operation of one residual heat removal pump with two safety injection pumps increases the range of core protection to a 10-in. equivalent break diameter (pressurizer surge line break) (Bar B).

The operation of one residual heat removal pump and the accumulators has been specifically analyzed for a range of equivalent break areas between the 10-in. pressurizer surge line break and the double-ended severance of the reactor coolant pipe (Bar C). This analysis shows that the required core protection is provided by these systems for this range. The trend of the results moreover indicates that the range of core protection for this combination actually extends down to an equivalent break diameter of approximately 6 in.

Figure 6.3-4 was developed from the results of the loss-of-coolant accident studies presented in Section 15.6.4. Simulations of a sufficient number of break sizes were performed to demonstrate that the safety injection components meet the emergency core cooling requirements for the loss-of-coolant accident.

A loss-of-coolant accident is defined as a rupture of the reactor coolant system piping or of any line connected to the system up to the first closed valve which would cause loss of coolant at a rate exceeding the flow capability of the charging pumps.

6.3.3.1.2 Makeup System Capacity

The maximum break size for which the normal makeup system can maintain the pressurizer level is obtained by comparing the calculated flow from the reactor coolant system through the postulated break against the charging pump makeup flow at normal reactor coolant system pressure, i.e., 2250 psia. A makeup flow rate from three charging pumps is typically adequate to sustain pressurizer pressure at 2250 psia for a break through a 3/8-in. diameter hole.

6.3.3.1.3 System Evaluation

For the purpose of the evaluation the spectrum of postulated piping breaks in the reactor coolant system is divided into large breaks, defined as a rupture with a total cross sectional area equal to or greater than 1.0 ft², and small breaks, defined as a rupture with total cross

sectional area less than 1.0 ft² but larger than the 3/8-in. diameter hole. Refer to Section 15.6.4 for a detailed description of the accident including acceptance criteria and analytical results.

For small breaks, the analyses presented in Section 15.6.4 show that the high-head and low-head portions of the Emergency Core Cooling System (ECCS), together with the accumulators, provide sufficient core flooding to keep the calculated peak clad temperatures below the required limits of 10 CFR 50.46.

For breaks up to and including the double-ended severance of a reactor coolant pipe, the analyses presented in Section 15.6.4 show that the Emergency Core Cooling System (ECCS) meets the acceptance criteria as presented in 10 CFR 50.46. That is as follows:

- A. The calculated maximum fuel element cladding temperature shall not exceed 2200°F.
- B. The calculated total oxidation of the cladding shall nowhere exceed 17% of the total cladding thickness before oxidation.
- C. The calculated total amount of hydrogen generated from the chemical reaction of the cladding with water or steam shall not exceed 1% of the hypothetical amount that would be generated if all of the metal in the cladding material surrounding the fuel were to react.
- D. Calculated changes in core geometry shall be such that the core remains amenable to cooling.
- E. After any calculated successful initial operation of the ECCS, the calculated core temperature shall be maintained at an acceptably low value and decay heat shall be removed for the extended period of time required by the long-lived radioactivity remaining in the core.

During the injection of emergency cooling water into the reactor coolant system following a loss-of-coolant accident, the concentration of boron will vary depending on the depressurization history of the reactor. If depressurization were slow, the high-head pumps would inject 2750 ppm boric acid which would be diluted by the coolant remaining in the system. Rapid depressurization would bring about early injection of water containing 2550 ppm boric acid from the accumulators. When recirculation begins, the concentration ranges from approximately 2000 ppm to 3000 ppm depending upon the combination of assumptions used (i.e., maximum vs. minimum concentrations and maximum vs. minimum volumes). If chemical additive has been used in the containment spray, the recirculated water will also contain sufficient sodium hydroxide to result in a composition having a pH in excess of 7.0.

6.3.3.2 System Response

The starting sequence of the safety injection pumps (SI), residual heat removal pumps (RHR), and the related emergency power equipment is designed so that delivery of the full-rated flow is reached at 32 seconds for SI and 30 seconds for RHR. The safeguards bus loading and sequencing is given in Table 8.3-1.

6.3.3.3 Safety Injection System Switchover From Injection to Recirculation

The Ginna Station safety injection system design (Drawings 33013-1247, 33013-1261, and 33013-1262) has been reviewed specifically with respect to the post-loss-of-coolant accident switchover to recirculation. The original design intent allowed both low-head safety injection

(residual heat removal) pumps, two high-head safety injection pumps, and one containment spray pump to continue to operate until the refueling water storage tank (RWST) level decreased to a level near the RWST outlet nozzles. All the operating pumps were then stopped and the systems realigned for the recirculation mode before restarting the pumps.

Since the original design resulted in a termination of all injection flow while the system realignment to the recirculation mode was being completed, the reactor coolant would continue to boil away without makeup. The total boiloff time period was determined by the sum of the time increments required for the operator to accomplish several discrete actions to realign the pumps and valves in both the low-head safety injection and if necessary the high-head safety injection parts of the system. It has been shown that sufficient inventory is available above the core that no loss or unacceptable degradation of core cooling would occur during the switchover alignment.

An improvement in the switchover procedure would result if all injection flow is not terminated during the switchover to recirculation by maintaining high-pressure safety injection flow. Since the reactor coolant makeup will be greater than the boiloff rate from the reactor coolant system, a net loss of system inventory will not occur during the switchover of the low-head safety injection pumps.

The Ginna post-loss-of-coolant accident switchover procedure, from the injection mode to the recirculation mode, was modified and improved to avoid the termination of all injection flow by transferring the low-head safety injection and high-head safety injection pumps separately. The emergency procedure for loss-of-coolant accident mitigation was modified to transfer the low-head safety injection (residual heat removal) pumps suction from the RWST to the containment sump when the level reached 28%. Two high-head safety injection pumps and one containment spray pump remain in service until the RWST reaches 15% to provide emergency core coolant and containment cooling during the transfer. Following restart of the residual heat removal pumps, and following the RWST level reaching 15%, the remaining high-head safety injection and containment spray pumps are stopped. The high-head safety injection pump and containment spray pump suctions are then transferred to the recirculation mode.

To provide the required net positive suction head for the low-head safety injection pumps after their suctions are transferred to the containment sump, additional minimum water inventory was provided to the refueling water storage tank (RWST) (from 230,000 to 300,000 gal). The refueling water storage tank (RWST) low-level alarm which signals the proper time to begin the switchover procedure was set at 28%. Assuming pump flow rates in order to minimize the RWST draindown time, this setpoint allows the operator a minimum of 25 minutes in a large loss-of-coolant accident before the RWST reaches 28%. This is conservatively low, because it does not account for the throttling that is performed on the low-head safety injection discharge valve.

A number of procedural actions must take place once the RWST level reaches 28%. These include stopping one high-head safety injection pump, if three were running, stopping both low-head safety injection pumps (residual heat removal), and stopping one containment spray pump, if two were running. The RWST outlet valve to the low-head safety injection pumps

suction is closed, and the suction valves from containment sump "B" to the low-head safety injection pumps are opened. The low-head safety injection pumps can then be restarted. Based upon the maximum flowrates expected, there would be a minimum of 14 minutes and 14 seconds before reaching the RWST low-level alarm of 15%. Under emergency contingency actions, if this could not be accomplished, adequate net positive suction head exists to support continued operation of safety injection and containment spray pumps to an indicated RWST level of 10%.

6.3.3.4 Boron Precipitation During Long-Term Cooling

To prevent boron precipitation, Ginna Station uses simultaneous injection from the residual heat removal and the high-head safety injection systems. The simultaneous injection takes place within 5.5 hours after depletion of RWST, following a loss-of-coolant accident and, requires the primary system to be depressurized to below the shutoff head of the residual heat removal pump. For the large break loss-of-coolant accident the primary system pressure would rapidly drop below the residual heat removal discharge pressure. Emergency operating procedures that provide initial response to LOCA conditions direct operator actions to depressurize the primary system to within the pressure capability of the heat removal system (*Reference 15*).

6.3.3.5 Single Failure Analysis

A single active failure analysis is presented in Table 6.3-9. All credible active system failures are considered. The analysis of the loss-of-coolant accident presented in Section 15.6.4 is consistent with the single-failure criteria.

The effects of a single failure on the boric acid injection from the boric acid storage tanks have been analyzed (*Reference 2*). The analysis considered the failure of (1) a battery, (2) an inverter, and (3) an ac bus on the boric acid storage tank level system. The analysis showed that single-failure criteria was satisfied and boric acid could be injected into the reactor vessel. However, the system was reconfigured in accordance with the Technical Specifications by maintaining valves 826A, B, C, and D closed and securing the valves in position by removing ac power, isolating this flow path, thereby eliminating the need for single active failure considerations.

When the reactor coolant system is being pressurized during the normal plant startup operation, the check valves are tested for leakage as soon as there is about 150 psi differential across the valve. This test confirms the seating of the disk and whether or not there has been an increase in the leakage since the last test. When this test is completed, the discharge line test valves are closed and the reactor coolant system pressure increase continued. There should be no increase in leakage from this point on since increasing reactor coolant pressure increases the seating force and decreases the probability of leakage. As part of the concern regarding separation of high and low-pressure systems, Ginna Station also performs leak rate testing of valves 878G and 878J (safety injection lines) following actuation of the valves, flow through the valves, or maintenance on the valves.

The accumulators can accept leakage back from the reactor coolant system without effect on their availability. However, Technical Specification limits are provided for high and low levels in the accumulators.

The accumulators are located inside the reactor containment and protected from the reactor coolant system piping and components by a missile barrier. Accidental release of the gas charge in the two accumulators would cause an increase in the containment pressure of approximately 0.1 psi.

6.3.3.6 Passive Systems

The accumulators are a passive safety feature in that they perform their design function in the total absence of an actuation signal or power source. The only moving parts in the accumulator injection train are in the two check valves.

During MODES 1 and 2, the flow rate through the reactor coolant piping is approximately five times the maximum flow rate from the accumulator during injection; therefore, fluid impingement on reactor vessel components during injection of the accumulator is not restricting. The high-pressure safety injection piping from the accumulators was analyzed for the effect of forces resulting from the fluid velocity. The calculated maximum stress due to the fluid thrust when combined with other stresses is acceptable. The maximum fluid thrust occurs at the elbow which is located at the third change in the discharge piping direction going back from the reactor coolant pipe.

The critical nature of the accumulator piping to the reactor coolant loop and the high flow rates attainable during operation indicated the need to instrument these lines and their support during preoperational testing. The objective was to measure the actual strains and deflections during high flow testing to demonstrate that operational conditions are within allowable limits. This instrumentation and testing was accomplished concurrent with the full flow accumulator injection test which was part of the preoperational test program.

A vibration test was performed on the reactor coolant system components with the reactor coolant system operating at design operating temperature and pressure. Vibration levels were measured at the reactor coolant pumps, hot-leg piping, cold-leg piping, steam generators, and control rod drive mechanisms. Measurements were made in both the horizontal and vertical directions using a probe type vibrometer.

6.3.3.7 Emergency Flow to the Core

Special attention is given to factors that could adversely affect the accumulator and safety injection flow to the core. These factors are as follows:

1. Steam binding in the core, including flow blockage due to loop sealing.
2. Loss of accumulator water during blowdown.
3. Short circuiting of the accumulator from the core to another part of the reactor coolant system.
4. Loss of accumulator water through the breaks.

All of the above are considered in the analysis and are discussed quantitatively in Section 15.6.4.

6.3.3.8 Recirculation Loop Leakage

Table 6.3-6 summarizes the maximum potential leakage from recirculation loop leak sources. In the analysis, a maximum leakage is assumed from each leak source. For conservatism, three times the maximum expected leak rate from the pump seals was assumed and a leakage of 10 drops/min was assumed from each flange. The total maximum potential leakage resulting from all sources is 2405 cm³/hr to the auxiliary building atmosphere and 31 cm³/hr to the drain tank.

The environmental consequences of the maximum potential leakage is evaluated in Section 15.6.4.2.5.

During recirculation, significant margin exists between the design and operating conditions of the residual heat removal system components. In addition, during normal plant cooldown, operation of the residual heat removal system is initiated when the primary system pressure and temperature have been reduced to 350 psig and 350°F. Since the maximum operating conditions during recirculation are 150 psig and 250°F, significant margin also exists between normal operating and accident conditions. In view of these margins, it is considered that the leakage rates tabulated in Table 6.3-6 are conservative.

During normal reactor cooldown, the tube-side of a residual heat removal heat exchanger is subjected to a reactor coolant system pressure of 325 psig plus the residual heat removal pump discharge pressure, and a temperature of 350°F. During the recirculation phase of a loss-of-coolant accident, these operating conditions are greatly reduced since they are restricted by the conditions existing within the containment; thus, the probability for a leak to develop in the tube side of a residual heat exchanger during recirculation is considered small.

In the unlikely event that a leak occurred, the radiation monitor located in the component cooling system common discharge header from both heat exchangers would alarm if there was any carryover of activity from the recirculated sump water. The same signal would close the vent on the component cooling surge tank, thus confining any activity within the component cooling system. In the event that the leak was not isolated in time, a relief valve at the component cooling water surge tank, which discharges to the waste holdup tank, would be actuated.

By isolating each residual heat exchanger in turn, the faulty heat exchanger could be identified. Recirculation core cooling would then continue using the other heat exchanger.

6.3.3.9 Safety Injection Pump Net Positive Suction Head Requirements

Residual Heat Removal Pumps

The net positive suction head of the residual heat removal pumps is evaluated for normal plant shutdown operation and both the injection and recirculation phase operation of the design-basis loss-of-coolant accident. Recirculation operation gives the limiting net positive suction head requirements and the net positive suction head available is determined from the

containment water level, the temperature and pressure of the sump water, and the pressure drop in the suction piping from the sump to the pumps. Adequate margin between required and available net positive suction head exists under all required operating conditions.

Operation of one residual heat removal pump could support operation of either one or two safety injection pumps in the sump recirculation phase, provided reactor coolant system pressure exceeds containment pressure by 105 psi. However, two safety injection pumps are required to be in operation above this pressure due to safety injection pump runout concerns. Emergency operating procedures require that containment spray be terminated at the end of the injection phase of a loss-of-coolant accident. One containment spray pump may be restarted in the recirculation phase only for containment pressure control for beyond design-basis events if containment pressure is above 43 psig. The containment spray pump would be started and stopped based on criteria that have been established which will ensure that RHR pump runout is not exceeded, RHR pump NPSH margin is adequate, and adequate core cooling is provided. Operation of a high-head safety injection pump in combination with a containment spray pump can be supported by one RHR pump provided established criteria have been met, as covered in the emergency operation procedure for sump recirculation. The limiting set of net positive suction head conditions was determined to exist when both train A and train B core deluge flow paths to the reactor vessel were open and a failure to open one of the motor-operated valves from containment sump B to the suction of the residual heat removal pump occurred just prior to the transfer to the sump recirculation phase. A top mounted hand-wheel was added to the actuators for AOV-624 and AOV-625 to permit local manual throttling of system flow and to act as a travel stop to provide the capability to limit the open position of these valves. The functionality of the hand-wheel operation was tested during the 1999 and 2000 refueling outages. Net positive suction head margin in the sump recirculation phase is improved by placing the valves in a permanently throttled position, (*Reference 13*), thus avoiding the need for operator action post-LOCA, and alleviating operator dose concerns.

Generic Letter 97-04 (*Reference 6*) was issued to request information to confirm the adequacy of the net positive suction head (NPSH) available for emergency core cooling and containment heat removal pumps. The Generic Letter specifically requested information for pumps that took a suction from the containment sump following a design-basis LOCA or secondary line break. RG&E responded to the Generic Letter (*Reference 7*) indicating that the only pumps that take a suction from the containment sump are the residual heat removal (RHR) pumps. The RG&E analysis (*Reference 3*) indicated that the post-accident recirculation phase represented the limiting set of results and that in all cases analyzed, the available NPSH is greater than the required NPSH. After review of RG&E's response, the NRC requested and received additional information (*References 8 and 9*). Upon review of the additional information, the NRC concluded that the requested information had been provided and closed Generic Letter 97-04 for Ginna (*Reference 10*). Subsequently, the design analysis *Reference 3*, has been replaced by the analysis, *Reference 13*, with the same conclusion that NPSH margin is adequate.

Safety Injection Pumps

The net positive suction head for the safety injection pumps is evaluated for both the injection and recirculation phase operations of the design-basis accident. The end of injection phase operation gives the limiting net positive suction head requirement, and the net positive suction head available is determined from the elevation head and vapor pressure of the water in the refueling water storage tank (RWST) and the pressure drop in the suction piping from the tank to the pumps. Adequate margin between required and available net positive suction head exists under all postulated operating conditions.

6.3.3.10 Seismic Analysis

The methodology originally used for the seismic analysis of the safety injection piping and other safety-related mechanical components (heat exchanger pumps, tanks, and valves) is discussed in Section 3.9.2. As a result of the SEP preliminary seismic review of Ginna Station, IE Bulletin 79-14, and other NRC seismic requirements, RG&E initiated a seismic upgrade program after the completion of piping support modifications required by IE Bulletin 79-14. The analysis procedures and criteria used for the piping analysis in the seismic upgrade program conform to current criteria and are discussed in Sections 3.7.3.7.5 and 3.9.2.1.8. The seismic qualification reanalysis for safety-related mechanical equipment under the SEP is discussed in Section 3.9.2.2.4.

6.3.3.11 MODE 4 (Hot Standby) LOCA Evaluation

There are no applicable safety analyses which apply to the emergency core cooling system (ECCS) when in MODE 4 (Hot Standby) due to the stable conditions associated with operation in MODE 4 and the reduced probability of occurrence of a design-basis accident. The MODE 4 loss-of-coolant accident (LOCA) evaluation is bounded by the performance of the ECCS while in MODES 1-3, and Technical Specifications require only a single train of ECCS while in MODE 4. Due to the time available for operators to respond to an accident, single failures are not required to be postulated. An industry concern was promulgated by Westinghouse in *Reference 5*. This was associated with the potential for water hammer in the residual heat removal (RHR) system during plant cooldown operations caused by the transfer of the suction water source to the refueling water storage tank (RWST), after the RHR system had already been lined up and in operation with its suction source provided by the reactor coolant system (RCS). Due to the temperature of the RCS during cooldown (between 200°F and 350°F), it was postulated that, in the event of a LOCA, a transfer of the suction source to the cooler RWST could cause the water to flash to steam resulting in vapor binding, water hammer, and mechanical pump damage to the residual heat removal piping and pumps.

This condition was evaluated and appropriate changes to operating and emergency procedures were implemented to prevent the risk of a water hammer event. The temperature associated with saturated conditions that would exist at the residual heat removal pump suction is 282.7°F. Emergency procedures dictate that, if the residual heat removal water temperature is initially above 280°F, the transfer of the residual heat removal pump suction to the RWST and pump startup is delayed. Safety injection flow would be provided by a high-head safety injection (SI) pump through both the hot and cold leg injection paths, during which time the residual heat removal piping is allowed to cool. Design analysis indicated that flow from a

single high-head SI pump can provide sufficient decay heat removal capability. Analysis also indicated that, when the residual heat removal water temperature has cooled to 250.9°F, transfer of the suction source to the RWST and pump start can be initiated. If the residual heat removal pump start was delayed until the RWST water inventory was depleted to the level for sump switchover (28%), calculations showed that sufficient time would have elapsed such that the residual heat removal piping would have cooled to a temperature below the saturation temperature at the pump suction (233°F) to permit suction transfer to sump B without the risk of adverse consequences associated with water hammer. The analysis was performed assuming one or two high-head safety injection pumps in operation. Each case produced acceptable results.

6.3.3.12 Alternate RCS Injection (BDB)

This system is part of a comprehensive strategy being credited in response to both NEI 12-06 and NFPA-805 for events, where existing plant systems are disabled due to fire, tornado, earthquake or flood. The Alternate RCS Injection System (BDB), supports the ability to inject borated water into the reactor at pressure, and to preserve inventory and reactivity control. This system provides an alternate means for compensating for reactor coolant pump (RCP) seal leakage, normally provided by the Chemical and Volume Control (CVCS) System. Unlike the Chemical and Volume Control (CVCS) System or the Safety Injection System (SI), the Alternate RCS Injection System will be provided with power sources (DDSAFW and NFPA-805 diesel generators) totally independent from existing plant power sources.

The overall operating strategy of the Alternate RCS Injection System is to draw on an existing source of borated water (RWST) that will be available following external events (tornado, earthquake or flood) and pump the borated water, using a high pressure positive displacement pump, into the cold leg of the reactor, via the Safety Injection lines. The Alternate RCS Injection pump suction piping is tied into the RWST recirculation pump suction line that takes suction off the bottom of the tank (el. 237' 8"). The 3" Alternate RCS Injection pump suction piping is run through the Auxiliary Building to buried lines between the Auxiliary Building and the SAFW Building Annex, to the Alternate RCS Injection pump in the SAFW Building. The Alternate RCS Injection pump is mounted on the pad of the former SAFW test tank. The positive displacement pump is capable of providing a flow of 78 gpm at 1585 psig, through a 2" pump discharge line that parallels the suction line into the Auxiliary Building basement, west of the RWST, and into the Safety Injection "A" and "B" header. The entire system is manually operated and controlled, making it impervious to Auxiliary Building fires or floods. A booster pump can be manually aligned to the Alternate RCS Injection pump suction, in the Auxiliary Building basement, in order to utilize the RWST volume below elevation 271'.

Since the Alternate RCS Injection System interfaces with the Safety Injection pump discharge headers, safety related isolation valves are installed to provide a boundary. Each Alternate RCS Injection branch line feeding into the SI "A" and "B" headers are equipped with a safety related normally closed ball valve. These valves are classified as Containment Isolation Valves (CIVs). Additionally, a check valve in the common line is installed to ensure that contaminated water does not migrate through the Auxiliary Bldg. to the SAFW Bldg. or Annex, which are not Radiological Controlled Areas (RCA).

Since the Alternate RCS Injection pump is a positive displacement pump, the flow from the pump is constant. A regulating valve on the pump skid ensures downstream pressure does not exceed the valve setting. Downstream pressure above the setting will cause a portion of the flow to be bypassed back to the pump suction. The regulating valve setting of 1575 psig ensures that the required flow of 75 gpm can be fed into the reactor pressurized to 1500 psig. Additionally, a relief valve on the pump skid is set at 1775 to 1875 psig to ensure that the pressure remains well below the pressure/temperature rating of the downstream piping.

The Alternate RCS Injection System design provides a means for testing the Alternate RCS Injection pump, without injecting into the reactor. Branch test lines off the Alternate RCS Injection pump suction and discharge piping allow for taking suction off the DI Water Storage Tank and recirculating it back to the tank to prove its capability to deliver 75 gpm flow at 1540 psig. A pressure breakdown orifice in the test discharge line has been designed to simulate design conditions for testing. Alternate RCS Injection pump pressure and flow instrumentation provide an indication of the pump's capability to meet design pressure and flow.

6.3.4 MINIMUM OPERATING CONDITIONS

The Technical Specifications establish detailed limiting conditions governing the operation and maintenance of the Emergency Core Cooling System (ECCS) components during plant operation.

6.3.5 TESTS AND INSPECTIONS

6.3.5.1 Inspection

All active and passive components of the safety injection system are inspected periodically to demonstrate system readiness.

The pressure containing systems are inspected for leaks from pump seals, valve packing, flanged joints, and safety valves during system testing.

In addition, to the extent that is practical, the critical parts of the reactor vessel internals, injection nozzles, pipes, valves, and safety injection pumps are inspected visually, by boroscopic examination or by ultrasonic testing for erosion, corrosion, and vibration wear evidence, and for nondestructive test inspection where such techniques are desirable and appropriate.

6.3.5.2 System Testing

Testing is conducted in accordance with Technical Specifications to demonstrate proper automatic operation of the safety injection system. A test signal is applied to initiate automatic action. The tests demonstrate the operation of the pumps, valves, pump circuit breakers, and automatic circuitry.

The accumulator pressure and level are continuously monitored during plant operation. Flow from the tanks through the outlet check valves is checked during periodic testing using test lines. The accumulators and the injection piping up to the final isolation valve are maintained with the proper amount of borated water while the plant is in operation.

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Flow in each of the high-head safety injection headers and in the main flow line for the residual heat removal pumps is monitored by flow indicators. Pressure instrumentation is also provided for the main flow paths of the safety injection and residual heat removal pumps. Accumulator isolation valves are blocked closed for this test. The sequence for recirculation operation is capable of being tested following the above injection phase to demonstrate proper sequencing of valves and pumps. This testing is not performed, since pump and valve testing is performed under the Inservice Test Program.

Testing of the safety injection system is performed prior to startup from each refueling outage, with the reactor coolant system pressure less than or equal to 350 psig and temperature less than or equal to 350°F. A test signal is applied to initiate operation of the system. The safety injection and residual heat removal pump motors undergoing the test are prevented from starting during this test. The system is considered satisfactory if control board indication and visual observations indicate that all valves have received the safety injection signal and have completed their travel.

The safety injection pumps and residual heat removal pumps are tested in accordance with the Inservice Pump and Valve Testing Program. Safety injection pumps “B” or “C” may also be utilized as needed for accumulator filling operations. The acceptable limits of performance applied to the residual heat removal pumps are specified in *Reference 28*. The flow rate required for delivery to the RCS from the residual heat removal pumps post-accident over the full range of RCS pressures was reduced by 12% as compared to the required delivery prior to the Ginna extended power uprate (EPU). The acceptable limit of performance applied to the safety injection pumps is 300 gpm. The flow rate required for delivery to the RCS from a safety injection pump post-accident at RCS pressures less than 500 psig is limited to 300 gpm as compared to the previously assumed delivery capability prior to the Ginna EPU. Thus, the 5% degradation limit in the IST Program provides additional margin as compared to the margin that existed prior to EPU, particularly for RCS pressures less than 500 psig. The ECCS delivery values assumed in the accident analysis from high head safety injection and residual heat removal pumps is shown on Table 15.6-17. The

300 gpm flow is achieved through the test line. The accumulator check valves are checked for operability in accordance with the Inservice Pump and Valve Testing Program.

During periodic testing of the safety injection pumps, the test line is opened to provide 300 gpm. Additionally, approximately 100 gpm is delivered through the pump recirculation line to minimize the potential for long term accelerated wear of pump internals (*Reference 4*). The test line is common to the redundant safety injection trains.

6.3.5.3 Components Testing

Preoperational performance tests of the components were performed in the manufacturer’s shop. An initial system flow test demonstrated proper functioning of the system. Periodic tests conducted in accordance with the inservice pump and valve testing program demonstrate that components are functioning properly.

Each active component of the safety injection system may be individually actuated on the normal power source at any time during plant operation to demonstrate operability. The test of the safety injection pumps employs the full flow test line which connects back to the refueling water storage tank (RWST). Remote operated valves are exercised and actuation circuits tested. The automatic actuation circuitry, valves, and pump breakers also are checked during integrated system tests performed during each MODE 6 (Refueling) outage.

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Flow through the accumulator discharge line is tested by opening the remote test valves in the test line connected just downstream of the stop valves. Flow through the test line is measured and the opening and closing of the discharge line stop valves is verified by the flow instrumentation. Test circuits are provided to periodically examine the leakage back through the check valves and to ascertain that these valves seat whenever the reactor system pressure is raised. This test is routinely performed when the reactor is being returned to power after an outage and the reactor pressure is raised above the accumulator pressure. If leakage through a check valve should become excessive, the isolation valve would be closed and an orderly shutdown initiated to repair the check valve.

The accumulator isolation valves are open with power removed during reactor operation. These valves are closed at any time that the reactor coolant system is depressurized.

The recirculation piping was initially hydrostatically tested at 150% of design pressure of each portion of the loop. The entire loop is also pressurized during periodic testing of the engineered safety features components. The recirculation piping is also leak tested at the time of the periodic retests of the containment.

Since the recirculation flow path is operated at a pressure in excess of the containment pressure, it is pressure tested for leakage during periodic retests at the operating pressures. This is accomplished by running each pump utilized during recirculation (safety injection, spray, and residual heat removal pumps) in turn at near shutoff head conditions and checking the discharge and recirculation test lines. The suction lines are tested by running the residual heat removal pumps and opening the flow path to containment spray and safety injection pumps in the same manner as described above.

During the above tests, all system joints, valve packings, pump seals, leakoff connections, or other potential points of leakage are visually examined. Valve gland packing, pump seals, and flanges are adjusted or replaced as required to reduce the leakage to acceptable proportions. For power-operated valves, final packing adjustments are made and the valves are put through an operating cycle before a final leakage examination is made.

The service water pumps and component cooling water pumps were tested prior to initial operation. For those pumps not running during MODES 1 and 2, it is possible to test their operability by interchanging with the operating pumps.

6.3.5.4 Operational Sequence Testing

The full operational sequence of the safety injection system was tested to the extent practical prior to initial plant operation. This test demonstrated the state of readiness and capacity of this system. The safety injection system valving was set initially to simulate the system alignment for plant power operation.

To initiate the test, the safety injection system block switch was moved to the unblock position to provide control power allowing the automatic actuation of the safety injection signal relays from the pressurizer. Simultaneously, the breakers supplying normal power to

the 480-V buses were tripped and operation of the diesel generator power system commenced automatically. The safety injection pumps and the residual heat removal pumps start automatically following the prescribed loading sequence. The valves operate automatically to align the flow path for injection into the reactor coolant system.

The functioning of the accumulators was checked by closing the stop valves, raising the pressure in the tank, and then opening the stop valve and observing the rising pressurizer level.

The rising water level in the pressurizer provided indication of system delivery. Flow into the reactor coolant system was terminated prior to complete filling of the pressurizer.

The functional test provided information to confirm the proper automatic sequencing of load addition to the diesel generator.

The functional test was repeated for the various modes of operation needed to demonstrate performance at partial effectiveness; that is, to demonstrate the proper loading sequence with loss of one of the diesel generator power sources and to demonstrate the correct automatic starting of a second pump should the first pump fail to respond. These latter cases were performed without delivery of water in the reactor coolant system but included starting of all pumping equipment involved in each test.

6.3.5.5 Gas Intrusion Management Program

The Ginna Station Intrusion Management Program was established in response to Generic Letter (GL) 2008-01 (*Ref. 25*). The GL required all plants to evaluate the Emergency Core Cooling, Decay Heat Removal and Containment Spray systems for gas intrusion.

The piping systems have the potential to develop voids and pockets of entrained gas. Maintaining the pump suction and discharge piping sufficiently full of water is necessary to ensure that the systems will perform properly and will inject the flow assumed in the safety analysis upon demand. This prevents unacceptable pump cavitation, water hammer, and injection of non-condensable gas into the reactor vessel following a Safety Injection signal or during shutdown cooling. Consistent with Ginna Station's response to GL 2008-01 (*Ref. 26 and 27*), the Gas Intrusion Management Program addresses the identification, evaluation, and elimination of unacceptable gas voids to maintain the systems sufficiently full of water.

6.3.6 INSTRUMENTATION

Instrumentation and associated analog and logic channels used for initiation of Emergency Core Cooling System (ECCS) operation is discussed in Section 7.3. This section describes the instrumentation available for monitoring Emergency Core Cooling System (ECCS) components during normal and postaccident operation.

6.3.6.1 Containment Sump Level

Redundant containment sump B level indicators (LI-942 and LI-943) show that water has been delivered to the containment following an accident and, subsequently, shows that the residual heat removal pumps have sufficient net positive suction head to allow operation. These switches are designed to withstand accident conditions.

6.3.6.2 Refueling Water Storage Tank (RWST) Level

The refueling water storage tank (RWST) is equipped with two redundant level indication systems. Each system consists of a level transmitter connected to the refueling water storage tank (RWST) which signals a percent scale level indicator on the main control board and actuates two annunciator alarm windows (low-low and high-low) on the main control board by way of two bistables, one corresponding to each alarm window.

During the injection phase, following a loss-of-coolant accident, the high-low level alarm alerts the operator to evaluate the operating engineered safety features pumps and, if required, remove from service enough equipment to maintain an adequate margin to net positive suction head for those pumps continuing in operation. When the level decreases to the low-low level an alarm alerts the operator to suspend the injection phase and switch over to the recirculation phase.

6.3.6.3 Accumulator Pressure and Level

Redundant pressure and level transmitters are provided for each accumulator to monitor the readiness of the accumulators to provide the required injection flow. These indicators are not required to operate after an accident.

6.3.6.4 Boric Acid Storage Tank Level

The boric acid storage tank liquid level system consists of a bubbler system and differential pressure transmitter. The transmitter output transmits the level signal to the main control board indicator and initiates a high-low level alarm and low-low level alarm to the control room annunciator.

6.3.6.5 Residual Heat Exchanger Flow and Temperature

Combined exit flow is indicated during normal RHR cooldown and post-accident alignment, and combined inlet temperature is recorded on the control board to monitor operation of the residual heat exchangers during normal RHR cooldown operation. In addition, the exit temperature of each heat exchanger is locally indicated. These transmitters are outside reactor containment.

6.3.6.6 Safety Injection Line Flow

Safety injection pump flow indication is provided in the feeder to each reactor coolant loop. These transmitters are located inside the containment and are qualified for postaccident operation. Additional flow instrumentation is also provided in the lines from the residual heat removal pumps to the safety injection and containment spray pumps, used during sump recirculation if necessary to provide high-pressure injection or spray flow.

6.3.6.7 Safety Injection Pumps Discharge Pressure

These channels provide additional indication of safety injection pump operation. These transmitters are outside the containment.

6.3.6.8 Pump Energization

All pump motor power feed breakers have indication that they are closed by energizing indicating lights on the control board.

6.3.6.9 Valve Position

All engineered safety feature valves have position indication on the control board to show proper positioning of the valves. Air-operated and solenoid-operated valves are designed to fail in a preferred direction of greater safety on the loss of air or power. Motor-operated valves remain in the position they are in at the time of loss of power to the motor.

REFERENCES FOR SECTION 6.3

1. Letter from R. C. Mecredy, RG&E, to A. R. Johnson, NRC, Subject: 30 Day Response to NRC Bulletin 93-02, dated June 10, 1993.
2. Rochester Gas and Electric Corporation, Amendment No. 2 to Technical Supplement Accompanying Application to Increase Power, February 1971.
3. Rochester Gas and Electric Corporation, Design Analysis NSL-0000-DA027, Revision 1, Ginna Station, Residual Heat Removal Pump NPSH Calculations During Accident Conditions, dated January 28, 1998.
4. Letter from R. C. Mecredy, RG&E, to A. R. Johnson, NRC, Subject: Review of RG&E Actions Taken in Response to NRC Bulletin 88-04, Potential Safety Related Pump Loss, dated March 27, 1991.
5. Westinghouse Nuclear Safety Advisory Letter NSAL-93-004, RHRS Operation as Part of the ECCS During Plant Startup, dated April 20, 1993.
6. Generic Letter 97-04, Assurance of Sufficient Net Positive Suction Head for Emergency Core Cooling and Containment Heat Removal Pumps, dated October 7, 1997.
7. Letter from R. C. Mecredy, RG&E, to G. S. Vissing, NRC, Subject: Response to Generic Letter 97-04, dated January 6, 1998.
8. Letter from G. S. Vissing, NRC, to R. C. Mecredy, RG&E, Subject: Request for Additional Information (RAI) Relating to Response to Generic Letter 97-04 (TAC M99993), dated June 8, 1998.
9. Letter from R. C. Mecredy, RG&E, to G. S. Vissing, NRC, Subject: Response to Request for Additional Information (RAI) Relating to Generic Letter 97-04 (TAC M99993), dated July 21, 1998.
10. Letter from G. S. Vissing, NRC, to R. C. Mecredy, RG&E, Subject: Completion of Licensing Action for Generic Letter 97-04, "Assurance of Sufficient Net Positive Suction Head for Emergency Core Cooling and Containment Heat Removal Pumps" (TAC M99993), dated August 10, 1998.
11. Generic Letter 98-04, Potential for Degradation of the Emergency Core Cooling System and the Containment Spray System After a Loss-of-Coolant Accident Because of Construction and Protective Coating Deficiencies and Foreign Material in Containment, dated July 14, 1998.
12. Letter from R. C. Mecredy, RG&E, to G. S. Vissing, NRC, Subject: Response to Generic Letter 98-04, dated December 1, 1998.
13. Design Analysis DA-ME-2005-085, Revision 0, NPSH for ECCS Pumps during Injection and Sump Recirculation, dated Jan 10, 2006.

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14. Letter from Patrick D. Milano, NRC, to Mary G. Korsnick, Ginna Nuclear Power Plant, LLC, Subject: Approval of Extension Request for Completion of Corrective Actions in Response to Generic Letter 2004-02 (TAC No. MC4687), dated October 4, 2006.
15. Subject: "R. E. Ginna Nuclear Power Plant - Amendment RE: Revised Loss-of-Coolant Accident Analysis (TAC No. MC6860," dated May 31, 2006.
16. Westinghouse Letter Report, LTR-SEE-I-07-49, "Sump pH versus Time Profile and Chemical Effects Evaluation for R.E. Ginna Nuclear Power Plant."
17. Westinghouse Calculation, CN-SEE-I-07-20, Rev. 0, "R.E. Ginna pH versus Time Evaluation."
18. Westinghouse Calculation, CN-SEE-I-07-16, Rev. 1, "R.E. Ginna GSI-191 Chemical Effects Evaluation."
19. Alion Design Calculation, CAL-CONS-3237-02, Rev. 1, "Ginna Reactor Building GSI-191 Debris Generation Calculation."
20. Alion Design Calculation, CAL-GINNA-4376-03, Rev. 0, "Ginna GSI-191 Debris Transport Calculation."
21. T-2139-1 and T-2139-1Rep, CCI, AG Protocol, "Chemical Head Loss MFT Filter Performance Test."
22. 3SA-096.077, CCI, AG Report, "Head Loss Calculation Including Chemical Effects," Revision 1.
23. 170-1356597, "Downstream Effects Evaluation to Support Resolution of GSI-191 for R.E. Ginna Nuclear Power Plant."
24. CN-SEE-I-08-48, "LOCADM Analysis for R.E. Ginna."
25. Generic Letter 2008-01, "Managing Gas Accumulation in Emergency Core Cooling, Decay Heat Removal, and Containment Spray Systems," dated January 11, 2008.
26. Letter from J. Carlin to NRC Document Control Desk, Subject: Nine-Month Response to NRC Generic Letter 2008-01, "Managing Gas Accumulation in Emergency Core Cooling, Decay Heat Removal, and Containment Spray Systems," dated October 13, 2008.
27. Letter from P. Swift to NRC Document Control Desk, Subject: Response to Request for Additional Information Related to Generic Letter 2008-01, dated March 2, 2011.
28. DA-ME-16-007, Rev. 0, "RHR Compensated Differential Pressure Requirements."

Table 6.3-1
QUALITY STANDARDS OF SAFETY INJECTION SYSTEM COMPONENTS

Residual Heat Exchanger

Test and Inspections

Hydrostatic test
Radiograph of longitudinal and girth welds (tube side only)
Ultrasonic test of tubing or eddy current test
Dye penetrant test of welds
Dye penetrant test of tube-to-tube sheet welds
Gas leak test of tube-to-tube sheet welds before hydro and
expanding of tubes

Special Manufacturing Process
Control

Tube-to-tube sheet weld qualifications procedure
Welding, nondestructive testing, and procedure review
Surveillance of supplier's quality control and product

Component Cooling Heat Exchanger

Tests and Inspections

Hydrostatic test
Dye penetrant test of welds

Special Manufacturing Process
Control

Welding, nondestructive testing, and procedure review
Surveillance of supplier's quality control and product

Safety Injection and Residual Heat Removal Pumps

Tests and Inspections

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Performance test

Dye penetrant of pressure retaining parts

Hydrostatic test

Special Manufacturing Process
Control

Weld, nondestructive testing, and inspection procedures for
review

Surveillance of supplier's quality control system and product

Accumulators

Tests and Inspections

Hydrostatic test

Radiography of longitudinal and girth welds

Dye penetrant/magnetic particle of weld

Special Manufacturing Process
Control

Weld, fabrication, nondestructive testing, and inspection
procedure review

Surveillance of supplier's quality control and product

Valves

Tests and Inspections

200 psi and 200°F or below (cast
or bar stock)

Dye penetrant test

Hydrostatic test

Seat leakage test

Above 200 psi and 200°F

Forged valves

Ultrasonic testing of billet prior to forging
Dye penetrant 100% of accessible areas after forging
Hydrostatic test
Seat leakage test

Cast valves

Radiograph 100%^a
Dye penetrant all accessible areas^a
Hydrostatic test
Seat leakage

Functional tests required for:

Motor-operated valves
Auxiliary relief valves

Special Manufacturing Process
Control

Weld, nondestructive testing, performance testing, assembly,
and inspection procedure review
Surveillance of supplier's quality control and product
Special weld process procedure qualification (e.g., hard
facing)

Piping

Tests and Inspections

Class 1501 and below:

Seamless or welded. If welded, 100% radiography is
required; shop-fabricated and field-fabricated pipe weld
joints are inspected as follows:

2501R-601R - 100% radiographic inspection and penetrant
examination

301R-151R - 10% random radiographic inspection

301R-151R - 100% liquid penetrant examination

**Special Manufacturing Process
Control**

Surveillance of supplier's quality control and product

- a. For valves with radioactive service only.

Table 6.3-2
ACCUMULATOR DESIGN PARAMETERS

Number	2
Type	Stainless steel clad/carbon steel
Design pressure, psig	800
Design temperature, °F	300
Operating temperature, °F	60 to 125°F
Maximum pressure, psig	790
Normal set pressure, psig (nominal)	745
Minimum pressure, psig	700
Total volume, ft ³	1750
Minimum water volume, ft ³	1090
Maximum water volume, ft ³	1140
Relief valve setpoint, psig	800
Minimum boron concentration, ppm	2550
Maximum boron concentration, ppm	3050

Table 6.3-3
SAFETY INJECTION SYSTEM PUMPS DESIGN PARAMETERS

	<u>High-Head Safety Injection</u> <u>Pumps</u>	<u>Residual Heat Removal</u> <u>Pumps</u>
Number	3	2
Motor power, hp	350	200
Design pressure, psig	1750	600
Design temperature, °F	300	400
Design flow rate, gpm	300	1560
Maximum flow rate, gpm	625	2500
Design head, ft	2700	280
Maximum shutoff head, ft	3400	340

Table 6.3-4
REFUELING WATER STORAGE TANK (RWST) DESIGN PARAMETERS

Number	1
Material	Stainless steel
Volume, gal	338,000
Code	API Standard 650 and in accordance with the U.S. AEC Division of Technical Information, TID 7024, August 1963, Nuclear Reactors and Earthquakes.
Design conditions	
Fluid	2750 to 3050 ppm boron
Temperature	200°F
Pressure	Atmospheric
Seismic Category	I

Table 6.3-5
RESIDUAL HEAT REMOVAL HEAT EXCHANGERS DESIGN PARAMETERS

Number	2	
Design heat duty, Btu/hr	^a 24.15 x 10 ⁶	
Design UA ^b , Btu/hr/°F	750,000	
Design cycles (85°F-350°F)	200	
	<u>Primary (tube side)</u> <u>(reactor coolant)</u>	<u>Secondary (shell side)</u> <u>(component cooling water)</u>
Design pressure, psig	600	150
Design flow, lb/h	763,000	1,375,000 ^a
Inlet temperature, °F	160	100
Outlet temperature, °F	128.4	117.3

- a. 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 exchanger. See Section 9.2.2.4.1.6.
- b. UA is the total heat transfer coefficient (U) times the area (A).

Table 6.3-6
RECIRCULATION LOOP LEAKAGE INFORMATION USED IN ORIGINAL ANALYSIS

<u>Items</u>	<u>Number of Units</u>	<u>Original Type of Leakage Control and Unit Leakage Rate Used in the Original Analysis</u>	<u>Leakage to Atmosphere (cm³/hr)^a</u>	<u>Leakage Drain Tank (cm³/hr)^a</u>
Residual heat removal pumps	2	Double seal with leakoff - 1 drop/min	0	6
Spray pumps	2	Double seal - 1 drop/min	6	0
Safety injection pumps	3	Same as spray pumps	9	0
Flanges		Gasket - adjusted to zero leakage following any test - 10 drops/min/ flange used in analysis		
Pump	14		420	0
Valves bonnet to body (larger than 2 in.)	48		1440	0
Control valves	8		240	0
Valves - stem leakoffs	25	Backseated, double packing with leakoff - 1 cm ³ /hr used	0	25
Miscellaneous small valves	50	Packed stems - 1 drop/min used	150	0
Isolation valve at loop penetrations 2 in. and above ^b	5	Leakage maximum allowable across disk	140	0
		Total	2405≐0.635gph	31

a. 1 drop/min≐3 cm³/hr

b. This includes only leakage paths to atmosphere.

Table 6.3-7
INSTRUMENTATION READOUTS ON THE CONTROL BOARD FOR OPERATOR
MONITORING DURING RECIRCULATION

<u><i>System</i></u>	<u><i>Valves</i></u>
Safety injection	MOV 825A and B
Residual heat removal	MOV 826A, B, C, and D
Containment spray	MOV 851A and B
	MOV 850A and B
	MOV 1813A and B
	MOV 856
	MOV 857A, B and C
	MOV 852A and B
	MOV 896A and B
	MOV 1815A and B
	MOV 871A and B
	MOV 878A, B, C and D
	MOV 897
	MOV 898
	MOV 860A, B, C and D
	MOV 836A and B
	MOV 704A and B
	MOV 841
	MOV 865
Component cooling water	MOV 738A and B

<u><i>System</i></u>	<u><i>Instrumentation</i></u>
Residual heat removal temperature	TE-630 (RK-3)
Residual heat removal flow	FT 626, FT 689
Residual heat removal to safety injection/containment spray flow	FT 931A and B

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Sodium hydroxide tank level	LT 931
Sodium hydroxide flow	FT 930
Refueling water storage tank (RWST) level	LT 920, LT 921
Safety injection discharge pressure	PT 922, PT 923
Accumulator pressure	PT 936, PT 937 PT 940, PT 941
Accumulator level	LT 934, LT 935 LT 938, LT 939
Safety injection flow	FT 924, FT 925
Containment sump B level	LT 942, LT 943

Pumps

System

Valve Number

Safety injection
Service water
Component cooling
Containment spray
Residual heat removal

Table 6.3-8
SAFETY INJECTION VALVE OPERATION AND INTERLOCKS

<u><i>Valve Number^a</i></u>	<u><i>Design Features</i></u>
MOV 826A, B, C, D	Valves between the boric acid storage tanks and the suction of the safety injection pumps. There are two parallel paths with two valves in each path. All four valves are maintained closed and secured in position with ac power removed in accordance with the Technical Specifications. Valves originally provided high concentration boric acid for the initial duration of the injection phase. Their function was changed and the valves currently perform an isolation function and perform no active function related to the emergency core cooling system (ECCS).
MOV 825A, B	Safety injection pump suction valves from the refueling water storage tank (RWST). They are locked open with ac power removed in accordance with the Technical Specifications and have no automatic action.
MOV 871A, B	Safety injection pump 1C discharge to B and A loop, respectively. They are normally open. During safety injection, if the 1A pump fails to start, then 871B will close. If the 1B safety injection pump fails to start, then 871A will close.
MOV 865, 841	Accumulator discharge to loop cold legs. They are normally open. Alternating current power is removed during operation.
MOV 878B, D	Safety injection pump discharge valves to the reactor coolant loop cold legs. They are normally open. Alternating current power is removed during operation
MOV 878A, C	Safety injection pump discharge valves to the reactor coolant loop hot legs. They are normally closed. Alternating current power is removed during operation.
MOV 852A, B	Reactor vessel deluge valves. They are normally closed and will open on safety injection. Direct current closing power is removed by key switch; so once the valves open, they stay open.
MOV 851A, B	Residual heat removal suction valves from sump B (containment valves). They are maintained open secured in position by removing ac power (in accordance with Technical Specifications) and have no automatic action.
MOV 850A, B	Residual heat removal suction valves from sump B (auxiliary building valves). They are normally closed and have no automatic action. They provide interlock for MOV 857A, B, C, and MOV 700 and 701.
MOV 856	Residual heat removal pump suction valve from the refueling water storage tank (RWST). It is normally open. Direct current control power is removed by a key switch
MOV 896A, B	Refueling water storage tank (RWST) outlet valves. They are normally open and have no automatic action. They provide interlock for MOV 857A, B, and C. Direct current control power is removed by key switch.
MOV 1813A, B	Reactor coolant drain tank pump suction valves from sump B. They are normally closed and have no automatic action.

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<u><i>Valve Number^a</i></u>	<u><i>Design Features</i></u>
MOV 1815A, B	Safety injection pump 1C suction valves. They are normally open and if closed, will open on safety injection signal.
MOV 897, 898	Safety injection pump recirculation valves to the refueling water storage tank (RWST). They are normally open and fail as is on loss of electric power. There is no automatic action. They provide interlock for MOV 857A, B, and C.
MOV 857A, B, C	Residual heat removal discharge valves to safety injection and containment spray pump suction. They are normally closed. If 850A or B is open, then 857A, B, and C cannot be opened unless either 896A or B is closed and either 897 or 898 is closed. This prevents putting water from sump B into the refueling water storage tank (RWST). MOV 857B is also pressure interlocked such that it cannot be opened unless the residual heat removal discharge pressure is less than 250 psi. This prevents overpressurization of the safety injection pump suction piping.
MOV 860A, B, C, D	Containment spray pump discharge valves. They are normally closed and will open on containment spray actuation signal.
AOV 846	Nitrogen supply valve to the accumulators. It is normally closed and fails closed. If open, it will close on a containment isolation signal.
AOV 839A, B AOV 840A, B	Accumulator test lines stop valves. They are normally closed, fail closed, and have no automatic action.
AOV 834A, B	Accumulator vent, or N ₂ fill valves. They are normally closed, fail closed, and have no automatic action.
HCV 945	Accumulator vent valve. It is normally closed, fails closed, and has no automatic action.
AOV 844A, B	Accumulator drain valves to the reactor coolant drain tank. They are normally closed, fail closed, and have no automatic action.
MOV 875A, B MOV 876A, B	Charcoal filter dousing valves. They are normally closed and have no automatic action.
AOV 836A, B	Sodium hydroxide tank outlet valves. They are normally closed, open on containment spray actuation signal, and fail open on loss of air or power.

a See Drawings 33013-1247, 33013-1261, and 33013-1262.

Table 6.3-9
SINGLE FAILURE ANALYSIS - SAFETY INJECTION SYSTEM

<u>Component</u>	<u>Malfunction</u>	<u>Comments</u>
Accumulator (injection phase)	Delivery to broken loop	Totally passive system with one accumulator per loop. Evaluation based on one accumulator delivering to the core and one spilling from the ruptured loop.
Pumps (injection phase)		
High-head safety injection	Fails to start	Three provided; evaluation based on operation of two.
Residual heat removal	Fails to start	Two provided; evaluation based on operation of one.
Component cooling	Fails to start	Two provided; evaluation based on operation of one.
Service water	Fails to start	Four provided; evaluation based on operation of one.
Automatically operated valves (injection phase)		
High-head injection line discharge valves to reactor coolant system cold legs (878 B, D)	Not applicable	Valves locked open with ac power removed (Technical specifications).
Residual heat removal pump deluge valves at injection line (852A, B)	Fails to open to open.	Two parallel valves; one required
Injection valves at high-head safety injection pump suction header (825A, B)	Not applicable	Valves locked open with ac power removed (Technical Specifications).
Refueling water storage tank (RWST) outlet valve to suction of residual heat removal pumps (856)	Not applicable	Valves locked open with the dc control power removed by key switch (Technical Specification.)
Refueling water storage tank (RWST) outlet valves to suction of safety injection and containment spray pumps (896A,B)	Not applicable	Valves locked open with the dc control power removed by key switch (Technical Specification.)
High-head injection line isolation valves from reactor coolant system hot legs (878A, C)	Not applicable	Valves locked closed with the ac power removed (Technical Specification.)
Accumulator isolation valves to reactor coolant system cold legs (865, 841)	Not applicable	Valves locked open with ac power removed (Technical Specification).

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<u>Component</u>	<u>Malfunction</u>	<u>Comments</u>
Valves operated from control room for recirculation (recirculation phase)		
Containment sump recirculation isolation (850A, B and 851A,B)	Fails to open	Two lines in parallel with two valves in series in each line. Valves 851A and B are maintained open with ac power removed in accordance with Technical Specifications; either 850A or B required to open.
Safety injection pump suction valve at residual heat exchanger discharge (857A, B, and C)	Fails to open	Two parallel paths; one required to open. 857B in parallel with series valve 857A and C.
Isolation valve on the test line returning to the refueling water storage tank (RWST) (897, 898)	Fails to close	Two valves in series; one required to close.
Isolation valve at suction header from refueling water storage tank (RWST) (896A, B)	Fails to close	Two valves in series; one required to close.
Isolation valve to refueling water storage tank (RWST) from containment sump (856)		Fails to close Check valve (854) in series provides backup isolation.

NOTE:—The status of all active components of the safety injection system is indicated on the main control board. See Table 6.3-7.

6.4 HABITABILITY SYSTEMS

The control room habitability systems include missile protection, radiation shielding, radiation and toxic gas monitoring, fire detectors and portable extinguishers, air filtration, heating and cooling, and lighting. These habitability systems are provided to permit access to and occupancy of the control room during normal plant operations and emergency conditions.

6.4.1 DESIGN CRITERION

The general design criteria applicable to the control room design are GDC 11 and GDC 19, presented in Section 3.1.1.3.1 and Section 3.1.2.2.10, respectively. GDC 11 was used during the licensing of Ginna Station. It is the Atomic Industrial Forum (AIF) version of the proposed criterion issued by the AEC for comment on July 10, 1967. Conformance with 1972 General Design Criteria of 10 CFR 50, Appendix A, which included GDC 19, is discussed in Section 3.1.2.

The plant is equipped with a control room which contains the controls and instrumentation necessary for operation of the reactor and turbine generator under normal and accident conditions.

The control room is continuously occupied by the operating personnel under all operating and design basis accident conditions. Sufficient shielding, ventilation, and habitability provisions exist to ensure that control room personnel can perform all required safety functions from the control room under all credible postulated accident conditions.

It is design policy that the functional capacity of the control room shall be maintained at all times, including during all postulated accident conditions. For certain design basis events, such as fires, control room evacuation may be required. If an event requires control room evacuation, then multiple pathways are available to exit the control room to use procedures to place the plant in a safe shutdown condition. Though self-contained breathing apparatus (SCBA) and potassium iodide tablets are both available to control room operators, neither is relied upon for maintaining operators' exposure within the prescribed limits following a radiological release.

The following features are incorporated in the design to ensure that this criterion is met.

The normal HVAC system provides a large percentage of recirculated air, and the Control Room Envelope (CRE) can be isolated to prevent intake of contaminants from outside the CRE.

The Control Room Emergency Air Treatment System (CREATS) is designed to filter the control room atmosphere during periods when the control room is isolated and to maintain radiation levels in the control room at acceptable levels following the design-basis accident.

Structural and finish materials for the control room, relay room, and battery room are selected on the basis of fire-resistant characteristics. Structural floors and interior walls are reinforced concrete. Interior partitions are metal or masonry on metal joints. The control room ceiling covering is fire retardant acoustical tile and luminous ceiling panels. Door frames and doors are metallic. Use of combustible materials is minimized and controlled by way of the Appendix R reviews performed for any modification that increases combustible loading.

In addition to being continuously occupied, the control room has several fire detectors in the room, and inside cabinets within the control room, which will actuate alarms.

The control room is equipped with portable fire extinguishers that are sized and located in accordance with National Fire Code and National Fire Protection Association specifications. Extinguishers carry the Underwriter's Laboratory label of approval and are considered appropriate for the hazards associated with the control room area.

6.4.2 *SYSTEM DESIGN*

6.4.2.1 Definition of Control Room Envelope (CRE)

The CRE includes all areas served by the Control Room Emergency Air Treatment System (CREATS) and is limited to the top floor of the control building and all CREATS ductwork bounded by CRE isolation dampers. The CRE consists of the control room and its adjacent bathroom, kitchen, and Shift Manager's office. The air volume of the CRE has an effect upon post-accident dose calculations to control room operators.

6.4.2.2 Ventilation System Design

The control room ventilation consists of two separate systems: normal HVAC and CREATS. These two systems provide three different modes of operation: NORMAL, EMERGENCY, and PURGE.

6.4.2.2.1 Normal HVAC System - NORMAL and PURGE Modes of Operation

The normal control room HVAC system is located in the basement floor of the three-story control building and is connected to the CRE by supply and return ducts that are located in the stairwell. The normal HVAC system supports the NORMAL and PURGE modes of operation. The system includes a supply and return air fan and in the NORMAL mode provides fresh outside air and exhaust, coarse filtration, and heating or cooling via electric heating or chilled water cooling coils. The NORMAL system includes a separate fan for lavatory exhaust, which is isolated in the EMERGENCY mode of operation. In the PURGE mode of operation the NORMAL system has the same functions described above while also providing the maximum amount of fresh air and exhaust air to purge airborne contaminants from the CRE.

The normal HVAC system's outside air intake duct is equipped with redundant trains of radiation, chlorine, and ammonia monitors. Any one of these six monitors reaching their setpoint will actuate the EMERGENCY mode of operation and provide an alarm in the control room. The normal HVAC system is also equipped with a smoke detector that monitors return air in the duct between the CRE and the return air fan, and provides an alarm in the control room.

6.4.2.2.2 CREATS System - EMERGENCY Mode of Operation

The Control Room Emergency Air Treatment System (CREATS) consists of two seismic Class I, 1E powered, 100% capacity trains that can each filter, cool or heat, and recirculate 6000 CFM of control room air. The CREATS systems' major components are located in the relay room annex, which is on the east side of the relay room and is described in section

3.3.3.3.6. The relay room annex is east of, and one level below, the control room. Ductwork connects CREATS to the CRE via penetrations in the roof of the relay room annex and in the east wall of the control room; this ductwork is a CRE isolation boundary and, since it is located outside, is designed to survive tornado-driven missiles. The relay room annex is a hardened structure having reinforced concrete walls and roof.

The filters in each train of CREATS were designed per ASME AG-1 1997, and include HEPA filters and 4-in. deep bulk carbon adsorbers. CREATS heating capability comes from electric heating coils, and cooling from direct expansion R-22 refrigerant cooling coils that are supported by air-cooled condensers located on the rooftop of the relay room annex. The cooling system is discussed further in Section 6.4.3.2.4 and Section 6.4.3.4.

Both the heating and cooling functions are provided with Class 1E power that is stripped by a Safety injection (SI) signal; these loads can be manually restored after SI injection is reset to provide normal temperature control in the CRE. The CREATS fans are Class 1E powered and start upon a manual, toxic gas, radiation monitor, or SI signal; in order to maintain the filtration function the CREATS fans will start automatically upon an SI signal.

The CREATS system includes six isolation dampers that, when closed, are CRE isolation boundaries. These six dampers isolate three separate flowpaths and are redundant so that failure of a single damper will not prevent isolation of that flowpath. Four of the dampers are located in the stairwell and isolate the normal HVAC system supply and return ducts; the normal HVAC system fans trip with the isolation signal. The other two dampers are located above the control room's suspended ceiling and isolate the lavatory exhaust duct that discharges to the outdoors through the east wall.

With one train of CREATS in service, airborne particulate activity can be cleaned up at an effective rate of approximately nine filtered air changes per hour. Though not required in order to meet GDC 19 criteria, control room personnel have access to portable respiratory equipment. There are adequate numbers of self-contained breathing apparatus (SCBA) located within the control building, and additional SCBA units and spare cylinders located in the fire brigade response room.

The plant has the capability to supply breathing air to 10 people for 6 hours at the rate of two (1.0 hour) bottles per person per hour. A compressor and cascade system are provided onsite to supply the breathing air.

6.4.2.3 Leak Tightness

Tracer gas inleakage testing has been performed to verify the integrity of the CRE boundaries, and to validate the unfiltered air inleakage rate that was assumed in the dose calculations that demonstrate compliance with GDC 19.

6.4.2.4 Interaction with Other Zones and Pressure-containing Equipment.

The CRE has only two adjacent indoor spaces: the turbine building operating level to the north, and the relay room located one floor below the control room. The roof and the other three walls separate the CRE from the outdoor environment. The control room north and east walls include 0.25-in. steel plate.

All of the ductwork located within the CRE is part of an HVAC system serving the control room; there are no intervening ducts that serve adjacent spaces. The outside air intake duct located above the control room's suspended ceiling is a CRE isolation boundary and was modified to ensure seismic qualification and leaktightness. The control room lavatory exhaust duct discharges through the east wall of the control room and is equipped with redundant isolation dampers to meet single failure criteria; only the pressure boundary downstream of these dampers is a CRE isolation boundary.

The normal control room HVAC system is intended to balance fresh outside and exhaust air flows, and CREATS simply isolates and recirculates air within the CRE boundary; thus the design does not maintain a differential pressure between the CRE and any adjacent spaces.

6.4.2.4.1 Interaction with the Turbine Building

The control room's north wall separates it from the turbine building operating level and contains the control room's main entrance/exit door. The door is a side-by-side double door and has no vestibule; it is a security door protecting the control room vital area. The wall and the door are designed to withstand the worst-case turbine building HELB pressure transient. The HELB pressure 'superwall' is a CRE isolation boundary. A short section of the outside air intake duct is located in the turbine building and has also been designed to withstand the worst-case turbine building HELB pressure transient and protect from missiles; thus protecting downstream ductwork within the CRE from these events.

6.4.2.4.2 Interaction with the Relay Room

In the southwest corner of the control room is a doorway to a stairwell which connects the relay room and control room floors of the control building. The stairwell does not extend down into the lowest level of the control building where the normal HVAC system is located. The stairwell contains ductwork which connects the control room normal HVAC system with the CRE. The stairwell is not normally considered part of the CRE but has been evaluated as an acceptable CRE volume (*Reference 5*).

The relay room contains a halon fire suppression system and two separate packaged air conditioning units that employ R-410A refrigerant; all of which have been evaluated and are not considered a threat to CRE habitability (See Section 6.4.3.2).

6.4.2.5 Shielding Design

The control building's roof and the south and west walls face the containment building and are made of 20-in. thick reinforced concrete in order to provide shielding from containment. There is no door facing the containment building and penetrations are minimized in order to prevent radiation streaming into the control room. The post accident dose to control room occupants is presented in section 12.4 and in Chapter 15.

6.4.2.6 System Operational Procedures

The control room HVAC system has three basic modes of operation; NORMAL, EMERGENCY, and PURGE.

The NORMAL mode of operation occurs whenever there is no EMERGENCY signal present and PURGE is not actuated. The normal control room HVAC system provides fresh air and temperature control for the control room. The CREATS fans and emergency cooling/heating are normally secured but can be started as needed for surveillance testing or maintenance.

The EMERGENCY mode of operation is initiated by any signal that actuates CREATS:

- a. Safety Injection signal (SI)
- b. outside air intake radiation monitors
- c. outside air intake toxic gas (chlorine and ammonia) monitors
- d. manual isolation

In the EMERGENCY mode of operation the normal HVAC system fans are tripped, all six isolation dampers close, and the associated train (A or B) of CREATS is actuated to provide filtered recirculation of CRE air. An exception is radiation monitors; either radiation monitor will actuate BOTH trains of CREATS. Unless an SI signal is present, emergency heating or cooling will automatically maintain a normal control room temperature. If an SI signal is present the heating and cooling components are stripped of Class 1E power; after reset of the SI signal these loads can be restored from the control room. Since only one train of CREATS is required the control room operators can secure either train of CREATS at any time after the EMERGENCY mode is actuated. The main control board switch for either CREATS fan can be used to start the associated fan without initiating the EMERGENCY mode of operation.

The PURGE mode of operation is manually actuated from the control room and uses the normal HVAC system to provide the maximum amount of fresh air and exhaust air to the control room. The PURGE mode is overridden by any EMERGENCY actuation signal. With local operator action a purge flow of fresh air can also be established from an alternate outside air source (relay room annex) while in the EMERGENCY mode of operation.

6.4.3 DESIGN EVALUATIONS

Prior to design and installation of the CREATS system the issue of control room habitability was evaluated for operators' exposure to the following:

1. radiological dose
2. toxins
3. smoke and fire
4. temperature extremes

6.4.3.1 Radiological Analysis

As part of the Control Room Emergency Air Treatment System (CREATS) modification, the control room accident dose calculations were updated to reflect the new system configuration. In order to ensure the most limiting accident was analyzed, the decision was made to analyze all credible accidents where a release is postulated. In addition, the new dose calculations were performed using the alternate source term (AST) methodology per *Reference 8*. The atmospheric dispersion values were recalculated using the ARCON96 computer code for

release points specific to the accident (*Reference 9*). The NRC in *Reference 10*, as supplemented by *Reference 11*, approved the AST methodology and analyses. The analysis was updated as part of the Extended Power Uprate (EPU) Project (*Reference 12*) and further updated as part of the Containment Penetration Technical Specification revision (*Reference 15*). The analysis was updated as part of the Containment Temperature increase and Spray flow reduction project (*Reference 4*).

The limiting control room radiological analysis (LOCA) assumes a double-ended rupture of a primary coolant loop. The reactivity release calculation assumes that the core melts; iodine and noble gas activity is released to the containment atmosphere. A portion of the iodine is assumed to plate out on containment surfaces, whereas all of the noble gases are assumed to remain in the containment atmosphere. The iodine and noble gas concentration in the containment atmosphere is reduced by radioactive decay and containment leakage. In addition, iodine is removed by spray and HEPA filters. When recirculation is started, an additional leak path is assumed due to engineered safety features (ESF) equipment leakage during recirculation. The releases are assumed to transport to the control room boundary, diluted only by the atmospheric dispersion coefficient.

The activity then enters the control room through the normal intake and unfiltered inleakage. Activity leaves the control room by radioactive decay, filter absorption, and outleakage. The dose is obtained by combining, as appropriate, control room activity with the dose conversion factors, breathing rate, occupancy factors, and a geometry factor.

Tables 6.4-1 and 6.4-2 present the assumptions used in the analysis. The calculation is performed using the RADTRAD computer package (*Reference 4*). The analysis is documented in the various analyses listed in Tables 6.4-1. The results are presented at the end of Tables 6.4-1. The limits as set in 10 CFR 50.67 are met for total effective dose equivalent (TEDE) following the postulated accidents.

6.4.3.2 Protection from Toxins

Design evaluation of the CREATS system considered the potential for exposure of control room operators to toxins (*Reference 6*), and included evaluation of the following sources:

1. Chlorine from tanks used by the Ontario water treatment plant, located 1.1 miles east of the plant and chlorine from the tanks used by the Webster water treatment plant, located 4.1 miles west of the plant.
2. Ammonia from the ammonium hydroxide tank located onsite, north of the turbine building.
3. Halon from the fire protection system that serves the relay room.
4. Refrigerant from the CREATS cooling system and from several other cooling systems in close proximity to the control room.
5. Sodium hypochlorite from the tank located onsite, east of the screen house.
6. Carbon dioxide generated by control room occupants while isolated in the EMERGENCY mode of operation.

Toxic chemicals were evaluated in a manner similar to that performed for radioactivity, (*Reference 6*). The chemicals identified onsite and offsite are also discussed in Section 2.2.2.6.

6.4.3.2.1 Chlorine

As discussed in Section 2.2.2.6.2, approximately 1.1 miles east of Ginna Station is a water treatment plant that uses chlorine to treat lake water for distribution through the Ontario water system. Additionally, 4.1 miles west of Ginna Station is a water pumping station that also uses chlorine to treat lake water. Exposure to a postulated tank rupture is mitigated by two chlorine detectors located in the outside air intake duct for the normal control room HVAC system. Upon sensing chlorine in the incoming airstream, either detector will automatically isolate the CRE, trip the normal HVAC system, and activate CREATS.

The exposure to control room operators from both the Webster source and the Ontario source is less than the 30mg/m³ limit found in Regulatory Guide 1.78, Rev. 1.

6.4.3.2.2 Ammonia

North of the turbine building is a tank of ammonium hydroxide that is used for secondary side water treatment. Exposure to a postulated rupture of this tank is mitigated by two ammonia detectors located in the outside air intake duct for the normal control room HVAC system. Upon sensing ammonia in the incoming airstream either detector will automatically isolate the CRE, trip the normal HVAC system, and actuate CREATS.

The calculated ammonia exposure to control room operators from this source is less than the 210 mg/m³ limit found in Regulatory Guide 1.78, Rev. 1.

6.4.3.2.3 Halon

The relay room and MUX room located one floor below the control room are protected by a halon 1301 fire suppression system. The halon in the relay room is not a threat to control room habitability because there is a limited volume of halon present, halon is more dense than air and thus will not rise from the relay room into the control room, and there is no HVAC system to circulate the air between these rooms.

6.4.3.2.4 Refrigerant

Installation of the CREATS cooling systems' direct expansion cooling coils created the potential for a postulated rupture of a cooling coil and release of that system's entire refrigerant inventory into the control room. In addition to being an unlikely scenario, the low toxicity and the small volume of the chlorodifluoromethane 'R-22' refrigerant made this accident an insignificant challenge to control room habitability.

6.4.3.2.5 Sodium Hypochlorite

An above ground storage tank located east of the screen house holds sodium hypochlorite used to reduce biological fouling in the circulating and service water systems. This potential toxin source was evaluated and due to its low volatility, low concentration, and distance from the control room's outside air intake, was found not to be a threat to control room habitability.

6.4.3.2.6 Carbon Dioxide

The amount of air inleakage to the control room is minimized in order to limit radiological dose to control room operators, and as a result the levels of carbon dioxide (CO₂) would be expected to rise when the control room remains isolated in the EMERGENCY mode for extended periods of time.

Carbon dioxide is not considered a threat to habitability under accident conditions because the toxicity limit is high and CO₂ levels rise slowly, as a function of the occupants' breathing rate, and a long-term uncontrolled release that requires continuous isolation is unlikely. However, if radioactive or toxic releases require long-term CRE isolation then CO₂ levels can be measured and mitigated.

6.4.3.3 Protection from Smoke and Fire

Design policy is to maintain the control room habitable at all times, and if a single event requires evacuation then multiple pathways shall be available to exit the control room to use procedures to place the plant in a safe shutdown condition.

6.4.3.3.1 Internal Sources of Smoke and Fire

As described in Section 6.4.1, design features have limited the combustibles within the control room, and there are very few ignition sources present. The CRE is approximately 2000 square feet, is all located on the same floor, is continuously occupied, and is equipped with several fire detectors. Therefore it is likely that any fire in the control room would be quickly identified and extinguished by control room operators and/or fire brigade members.

If a fire inside the CRE results in excessive smoke or erratic operations/indications of main control board components, then evacuation to use procedures to place the plant in a safe shutdown condition can occur through the main control room door to the turbine building operating level, or through the back door to the stairwell which leads downstairs to the relay room.

6.4.3.3.2 External Sources of Smoke and Fire

A fire in any space adjacent to the control room is unlikely to spread into the CRE and require a control room evacuation because:

- a. The control room's north wall is protected, on the turbine building side, by a water curtain fire suppression system. Refer to Section 9.5.1.2.3.7 for details.
- b. The relay room, located below the control room, is protected by both a halon and a manually actuated water suppression system. Refer to Sections 9.5.1.2.3.7 and 9.5.1.2.3.8 for details.
- c. The west and south walls of the control room are adjacent to the main, auxiliary, and two offsite transformers. These oil-filled transformers present a significant fire load, but all are equipped with fire detection and water suppression systems. Refer to Section 9.5.1.2.3.7 for details. The west and south walls are also 3-hour fire rated, 20-in. thick reinforced concrete, with a minimum of penetrations.

- d. The roof of the control room is also 20-in. thick reinforced concrete, and the east wall is shielded with 1/4-in. thick armor plate. There is no significant combustible load located outside of the roof or the east wall of the control building.

Smoke from a fire in areas adjacent to the control room is not expected to affect control room habitability because of the limited inleakage across CRE boundaries. If a fire occurred in adjacent spaces, especially the transformer yard, procedures direct the control room operators to actuate the EMERGENCY mode of operation, thus isolating the control room and preventing smoke from outside being brought into the control room by the normal HVAC system.

Redundant dampers make this isolation function single failure proof. If, by some extreme circumstances, excessive smoke entered the control room from fire in an adjacent area, there are three paths available for operators to exit the control building to use procedures to place the plant in a safe shutdown condition:

1. through the main control room door to the turbine building operating level.
2. through the back door to the stairwell, relay room, and then to the mezzanine level of the turbine building.
3. through the back door to the stairwell, relay room, and then through the relay room annex to the outdoors.

It is not credible for a single fire event to simultaneously admit excessive smoke to the control room and obstruct all three of these pathways; thus the design objective is met.

6.4.3.4 Protection from Temperature Extremes

Design of the CREATS system included calculations of the heating and cooling loads for use in sizing the electric heaters and the cooling system. The installed capacity is capable of maintaining the control room temperature approximately 70-74°F under any credible outdoor temperature extremes.

6.4.4 TESTS AND INSPECTIONS

Surveillance testing of filters and isolation dampers is performed in accordance with Technical Specification requirements and assures compliance with the filtration efficiencies assumed in dose calculations. The testing frequency for the Control Room Emergency Air Treatment System (CREATS) is defined in the Technical Specifications and the Ventilation Filter Test Program.

Pressure drop testing at the design flow rate, Freon testing, dioctylphthalate (DOP) testing, charcoal efficiency testing, and compliance with the performance requirements will ensure the operability of the control room Emergency Air Treatment System (CREATS) should it be required. The charcoal efficiency test environment (temperature & humidity) is well above that which is likely to be encountered in the control room under normal or accident conditions.

A successful tracer gas inleakage test of the entire CRE was performed in February 2005 (*Reference 7*). *Reference 13* established surveillance requirements for the CRE, and the

requirement for a control room habitability program to manage and maintain the CRE. The program requirements are consistent with *Reference 14*.

6.4.5 INSTRUMENTATION REQUIREMENT

All instruments and circuits that actuate the EMERGENCY mode are redundant to protect from a single active failure. There are two trains of Safety Injection (SI), two each of the radiation, chlorine, and ammonia monitors, and two pushbuttons for manual isolation. A trip signal from any one of these devices will isolate the CREZ and actuate CREATS. Technical Specifications and the Technical Requirements Manual include required actions to be taken when these instrument channels are inoperable.

REFERENCES FOR SECTION 6.4

1. Letter from D. M. Crutchfield, NRC, to J. E. Maier, RG&E, Subject: NUREG 0737, Item III.D.3.4, Control Room Habitability, R. E. Ginna Nuclear Power Plant, dated April 11, 1983.
2. Deleted
3. Deleted
4. 50.59 Evaluation 5059EVAL-2014-0001, ECP-13-000048 – Containment Air Temperature Increase and Associated Changes, dated September 19, 2014.
5. DBCOR 2006-0010, Control Room Dose Sensitivity for Increase in Volume - Ginna Design Analysis DA-NS-2000-070, Rev 0, titled: Control Room Dose Simulation Removal of MUX Room, Temporary Stairwell Enclosure.
6. Design Analysis, DA-NS-2000-053, Rev 1; "Control Room Toxic Hazards Analysis"
7. Ginna procedure SM-2000-0024-2.3; "Control Room Emergency Zone (CREZ) Tracer Gas In-leakage Test"
8. Regulatory Guide 1.183, Alternate Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors. July 2000.
9. DA-NS-2001-060, Atmospheric Dispersion Factors for the Control Room Intake, Revision 2, dated June 17, 2006.
10. Letter from D. Skay, NRC, to M. G. Korsnick, Ginna NPP, Subject: R.E. Ginna Nuclear Power Plant - Amendment re: Modification of the Control Room Emergency Air Treatment System (CREATS) and Change to Dose Calculation Methodology to Alternate Source Term (TAC No. MB9123), dated February 25, 2005.
11. Letter from D. Skay, NRC, to M. G. Korsnick, Ginna NPP, Subject: R. E. Ginna Nuclear Power Plant - Correction to Amendment No. 87 re: Modification of the Control Room Emergency Air Treatment System (CREATS) (TAC No. MB9123), Dated May 18, 2005.
12. Letter from P. D. Milano, NRC, to M. G. Korsnick, Ginna NPP, Subject: R. E. Ginna Nuclear Power Plant - Amendment re: 16.8 Percent Power Uprate (TAC No. MC7382) dated July 11, 2006.
13. Letter from Douglas Pickett, NRC, to John Carlin, Ginna NPP, Subject: R. E. Ginna Nuclear Power Plant - Amendment re: Control Room Envelope Habitability (TAC No. MD6679), dated August 27, 2008.
14. Regulatory Guide 1.197, Demonstrating Control Room Envelope Integrity at Nuclear Power Reactors.

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15. Letter from D. V. Pickett, NRC, to J. T. Carlin, Ginna NPP, Subject: R. E. Ginna Nuclear Power Plant - Amendment Re: Containment Operability during Refueling Operations (TAC NO ME0203), dated August 12, 2009.

Table 6.4-1
CONTROL ROOM HABITABILITY RADIOLOGICAL EVALUATION -
ASSUMPTIONS AND RESULTS

Parameter

Reactor power	1811 MWt
Control room volume	36,211 (ft ³)
Control room filtered recirculation flow rate (cfm)	6000 cfm (one train)
Control room iodine removal efficiency (%)	
Elemental	94 %
Organic	94 %
Particulate	99 %
Control room unfiltered inleakage (cfm) ^a	300/250 cfm
Time to switch from normal control room HVAC to accident mode (sec)	60 seconds
Control room normal HVAC intake flow rate (cfm)	
< 60 seconds	2200 cfm
≥ 60 seconds	0.0 cfm
Control room emergency HVAC intake flow rate (cfm)	0.0 cfm
Control room breathing rate	3.47 x 10 ⁻⁴ m ³ /sec
Control room occupancy factor (fraction of time)	
0-1 day	1.0
1-4 days	0.6
4-30 days	0.4
Atmospheric Dispersion Factor (sec/m ³)	
Control Room	Accident and release point dependent - see analysis of record

a. 300 cfm assumed for all DBAs with exception of LBLOCA, which assumes 250 cfm.

CONTROL ROOM DOSE ANALYSIS SUMMARY REM TEDE

<u>Accident</u>	<u>Analysis of Record</u>	<u>Control Room Dose (Limit 5 Rem TEDE)</u>
Large Break LOCA	DA-NS-2001-087, Rev.4 and ECN ECP-13-000048-CN-085, also see UFSAR Section 12.4.3.3.15	$3.95 + 0.36 = 4.31$

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CONTROL ROOM DOSE ANALYSIS SUMMARY REM TEDE

<u>Accident</u>	<u>Analysis of Record</u>	<u>Control Room Dose (Limit 5 Rem TEDE)</u>
Fuel Handling Accident (CNMT)	DA-NS-08-050, Rev. 0	4.0416
Fuel Handling Accident (Auxiliary Building)	DA-NS-08-050, Rev. 0	8.0831E-1
Main Steam Line Break (Accident Initiated Iodine Spike)	DA-NS-2002-007, Rev. 4	5.8E-1
Main Steam Line Break (PreAccident Iodine Spike)	DA-NS-2002-007, Rev. 4	1.7E-1
Steam Generator Tube Rupture (Accident Initiated Iodine Spike)	DA-NS-2001-084, Rev. 3	2.4E-1
Steam Generator Tube Rupture (Pre-Accident Iodine Spike)	DA-NS-2001-084, Rev. 3	9.8E-1
Locked RCP Rotor	DA-NS-2002-054, Rev. 2	1.87
Control Rod Ejection	DA-NS-2002-050, Rev. 2	1.83
SFP Tornado Missile ^a	DA-NS-2002-019, Rev. 3	3.38E-1 (Isolation Filtration) 6.3E-1 (No CREATS Actuation) 4.34 (Isolation/No Filtration)
GDT Rupture ^a	DA-NS-2000-057, Rev. 2	1.15E-1

- a. CREATS is not required to function for protection from these events.

For the SFP Tornado Missile event, three cases were evaluated. The first case evaluates the single failure of one train of CREATS to actuate on a high radiation signal (3.38E-1 Rem TEDE). The second case evaluates the case where the radiation monitors do not actuate the CREATS system (6.3E-1 Rem TEDE). The third case is a bounding case which assumes multiple failures beyond the design basis (4.34 Rem TEDE). Because the radiation monitors were not evaluated to actuate during this event, and the third case is beyond design basis, the second case (5.14E-1 Rem TEDE) is considered to be the analysis of record.

NOTE: For the Main Steam Line Break (MSLB) and Steam Generator Tube Rupture (SGTR) accidents two separate cases were analyzed. One assumes a pre-accident iodine spike up to the Technical Specification limit of 60 uCi/gm. The second assumes a post-accident (accident initiated) Iodine spike factor of 335 for the SGTR, and a factor of 500 for the MSLB.

Table 6.4-2
CORE ACTIVITIES^a

<u>Isotope</u>	<u>Activity (Curies)</u>
I-131	4.14×10^7
I-132	6.03×10^7
I-133	8.54×10^7
I-134	9.36×10^7
I-135	7.97×10^7
Kr-85m	1.11×10^7
Kr-85	4.98×10^5
Kr-87	2.13×10^7
Kr-88	3.00×10^7
Xe-131m	4.61×10^5
Xe-133m	2.50×10^6
Xe-133	8.16×10^7
Xe-135m	1.67×10^7
Xe-135	2.16×10^7
Xe-138	7.04×10^7

a. The core activities are based on parameters that are specifically representative of the operation of Cycle 32 (References for *Section 6.4: Reference 7, Table 2*).

6.5 FISSION PRODUCT REMOVAL SYSTEMS

6.5.1 ENGINEERED SAFETY FEATURE FILTER SYSTEMS

6.5.1.1 Introduction

The installation of charcoal filters in an effluent stream provides an effective means of removing iodine from the stream and thereby reduces doses due to the effluent release. There are three areas of Ginna Station where filter systems have been installed and their effectiveness has been assumed to reduce the consequences of design-basis accidents: the post-accident charcoal filtration system in the containment, the charcoal filtration system in the auxiliary building, and the charcoal filtration in the Control Room Emergency Air Treatment System (CREATS) of the control room ventilation. This section discusses only the charcoal filtration system in the containment. The containment spray system iodine removal capability is discussed in Section 6.5.2. The auxiliary building charcoal filtration system is discussed in Section 9.4.2. The control building charcoal filtration system is discussed in Section 6.4.2.

The post-accident charcoal filtration system in the containment is part of the containment recirculation fan cooler (CRFC) system. The post-accident cooling function of the system is discussed in Section 6.2.2. The filtration aspects of the system are discussed in the following sections.

6.5.1.2 Containment Air Filtration System

6.5.1.2.1 Design Basis

The air recirculation filtration capacity of the containment recirculation fan cooler (CRFC) and filtration system is sufficient to reduce the concentration of fission products in the containment atmosphere following a loss-of-coolant accident (LOCA) such that the exposure guidelines of 10 CFR 50.67 are met. Details of the exclusion area boundary (EAB) dose calculations are given in Section 15.6.4.2.5.

The air recirculation filtering capacity used to satisfy the design basis is determined for the following conditions:

- A. Containment leak rate of 0.20 wt % per day.
- B. Conservative meteorology corrected for building wake effects.
- C. Filtration of iodine was assumed. The assumed iodine removal efficiency is 95% for particulate.
- D. Fission product release to containment is per Regulatory Guide 1.183.
- E. Partial effectiveness of the filtration equipment. One train of emergency power is assumed to fail. This results in one train of containment spray and two containment recirculation fan coolers (CRFC) operating, one of which supplies air to a carbon-bed filter unit.

In addition to the design basis specified above, the following objectives are met to provide the engineered safety features functions:

- AA. Each of the four air handling units is connected to a demister and high efficiency particulate air (HEPA) filters rated for full unit flow and capable of 90% removal efficiency for 0.3 micron particles at the post-accident design conditions.
- BB. Two of the four air handling units are capable of supplying air to separate carbon-bed filter units following an accident for fission product iodine removal. The original design flow rate through each carbon filter unit is 38,000 cfm, at a face velocity of approximately 40 fpm. During the 1993 (Refueling) outage, the containment recirculation fan coolers (CRFC) were replaced and as a result, the air flow through each carbon filter unit was reduced to an estimated low of 33,000 cfm. An analysis was performed to verify that an air flow as low as 30,000 cfm, at a face velocity of approximately 30 fpm, through each unit was satisfactory for iodine removal efficiency requirements. Subsequent to that time, Amendment 87 (*References 21 and 22*) to the Ginna operating license implemented the Alternate Source Term method of dose calculations per Regulatory Guide 1.183. This methodology of analysis considers a decreased importance in gaseous iodine and an increased importance on particulate iodine. Consequently, although the charcoal filters still serve to reduce dose, Ginna's dose analysis no longer credits the charcoal filters in the dose analysis to maintain control room and off-site doses within limits. However, because the system has not been physically modified, the system is required to mechanically function to ensure the correct air flows are present for the heat removal and particulate filtering functions to remain valid.

6.5.1.2.2 System Design

6.5.1.2.2.1 General Description

The containment recirculation fan cooler (CRFC) system consists of four air handling systems, each including motor, fan, cooling coils, moisture separators and HEPA filters, duct distribution system, and instrumentation and controls. The units are located on the intermediate floor between the containment wall and the primary compartment shield walls. Two of the four air handling systems are equipped with activated charcoal filter units, normally isolated from the main air recirculation stream, through which the air-steam mixture is bypassed to remove volatile iodine following an accident. The filter units are located on a platform above the operating floor.

6.5.1.2.2.2 Charcoal Filters

Each of the two charcoal filter units consists of an airtight plenum containing two banks of charcoal filter cells. Air flow enters the plenum through two holes in the bottom (one at each end), passes through the charcoal filter banks to the center, and is exhausted from the plenum through a single hole in the top.

The individual filter cells are the flatbed type of construction, with two 2-in. thick horizontal charcoal elements separated by a 2-in. air gap. The sides and back of the cell are enclosed by solid (unperforated) stainless steel sheet metal; the larger (horizontal) surfaces are enclosed by perforated stainless steel sheets. An unperforated stainless steel sheet seals the front edge; this sheet is slightly larger than the basic filter dimensions in order to prevent flanges from clamping in the mounting frame. Several rectangular slots are cut in the front face to permit

air flow. Each filter cell provides approximately 7.9 ft^2 of active surface area for air flow (both elements) and contains approximately 1.32 ft^3 of charcoal.

During operation, air flows vertically downward through the top surface of the filter and upward through the bottom surface, enters the air space between the two charcoal elements, and is discharged through the slots in the front face.

Each filter bank consists of 60 cells in a 4 wide by 15 high array. The mounting racks arrangement permits removal of individual cells from the center of the plenum.

6.5.1.2.2.3 *HEPA Filters*

The original HEPA filter cells are fabricated with cadmium-plated steel frames, glass fiber media containing less than 5% binder (by weight), and waterproof asbestos separators. The binder used in the preparation of the filter media is an acrylic compound that imparts a very high degree of water repellency to the media fibers. The repellency effect is such that the media can support a 30-in. column of water with no penetration. By comparison, normal pressure drop is approximately 1 in. of water at rated flow. Thus, any sodium hydroxide that enters the HEPA filter via entrained moisture is unable to penetrate or react with the binder. The glass fiber media itself is chemically inert, as is the neoprene base adhesive used to seal the media folds to the filter frame.

All of the original HEPA filter cells were replaced during the 1993 refueling outage and subsequent outages, with the last cells being replaced during the 1999 refueling outage. The replacement HEPA filter cells are fabricated with type 409 stainless steel frames and vinyl-coated aluminum separators. An analysis was performed to verify that vinyl-coated aluminum separators are acceptable for loss-of-coolant accident considerations and that the type 409 stainless steel in the cell sides is an acceptable material upgrade.

6.5.1.2.2.4 *Protection From Sodium Hydroxide Attack*

The pleated separators used in the original HEPA filters are made of asbestos and are water proofed during manufacture with sodium silicate solution. Both the asbestos and the sodium silicate are impervious to attack by sodium hydroxide. The aluminum separators used in the replacement HEPA filters are vinyl coated during manufacture. The vinyl coating prevents reaction between the aluminum separator and sodium hydroxide. Note that all of the original pleated separators have been replaced with the vinyl coated type, as the asbestos material is no longer available from the manufacturer.

The gaskets installed between the HEPA filter cells and the supporting frame-work are made of neoprene rubber, which is resistant to attack by sodium hydroxide solution. In addition, the exposed surface area of the gasketing is very small. The larger flat sides of each gasket are compressed between the flat metal surfaces of the joined members; only the narrow edges are exposed to the air-steam mixture. Extensive deterioration of the gasket would have to occur before its inner load-bearing region would be affected. Type 409 stainless steel, which is used in the cell sides of the replacement HEPA filters, is impervious to attack by sodium hydroxide.

Neoprene rubber gaskets are also used to join the stainless steel charcoal filter cells to the supporting framework.

All exposed carbon steel surfaces of the recirculation-filtration equipment are protected against sodium hydroxide reaction by a special painting treatment. These include the containment recirculation fan cooler (CRFC) filter and charcoal filter plenums, fan casing, fan motor and filter, and cooling coil support framework. The painting treatment consists of one coat of Carbonized 11, an inorganic zinc primer applied over sandblasted metal surfaces, followed by a finish coat of either polyamide-cured resin or modified phenolic type paint.

6.5.1.2.2.5 *Fire Protection*

The absolute (HEPA) filters are of the self-extinguishing type. The charcoal filter units are provided with high-temperature detectors (see Section 6.5.1.2.5) and associated alarms in the control room. The possibility of a charcoal filter burning following the maximum credible accident could not be completely excluded. Accordingly, an analysis was performed to determine the potential dose consequences which could be expected if a filter were to burn at some time after the initiation of the maximum credible accident, thereby releasing all of its adsorbed iodine to the containment. If it is assumed that only two of the four filters (i.e., one post-accident charcoal filter unit) have adsorbed 25% of the total core iodine and that at least one continues to remove iodine at an efficiency of 90% (assuming 5% is unfilterable), the additional dose would not exceed 50 rem in 2 hours at the exclusion area boundary (EAB). An additional 2 rem in 30 days could be expected at the low population distance. It was concluded that this improbable circumstance would not result in excessive offsite doses. The containment charcoal filter dousing system can be manually initiated if a fire is detected, but evaluation has shown that the system is not required to mitigate the consequences of any analyzed transients or accidents and therefore, operation of the system is determined to be beyond design basis requirements. Although the above analysis provides informational value, the charcoal filters are not credited in the dose analysis under Alternate Source Term (AST). A loss of either or both filters would have no impact on estimated dose for the design basis accidents provided that adequate air flow continued through the CRFC unit for HEPA and cooler operation.

Each charcoal filter unit is also provided with spray system capability for water dousing upon a signal of high temperature. The borated spray water is provided from the containment spray supply header. Initiation of dousing is carried out manually by the operator. Any significant air temperature rise in the plenum will be readily sensed by the temperature devices because the heat capacity of air is small and convection currents are easily established even in a stagnant environment. As a result, only a small quantity of heat would have to be generated to create sufficient convection to transfer the heat to the nearest detector.

The water dousing system provided with each charcoal filter plenum is designed to drench the absorbers thoroughly in the extremely unlikely event of a charcoal fire during the post-accident recovery. Borated water for this system is obtained from the main headers of the containment spray system through a separate 2-in. stainless steel line to each filter plenum. Two normally closed motor-operated valves in parallel on the 2-in. line can be opened in the event of a coincident beyond design-basis fire in the filters.

The spray water distribution system inside the charcoal filter plenums is of all copper construction. This system provides three individual injection lines, terminating in 3/8-in. brass nozzles, which spray into the air space between each pair of vertically adjacent absorber units. The nozzles discharge horizontally to ensure complete wetting of both upper and lower absorber surfaces. Refer to Figures 6.5-1 and 6.5-2 for details of the filter installation. If ignition should occur, temperature monitors would initiate an alarm and the affected bank could be sprayed.

6.5.1.2.3 Design Evaluation

6.5.1.2.3.1 Decay Heat Generation in the Charcoal Filters

In analyzing decay heat generation in the charcoal filter beds, fission product release to the containment per TID 14844 is assumed, with only one of the two installed charcoal filter units available for fission product trapping at the time of the loss of coolant. The fission product heat source includes beta energy from 25% of core halogens, plus self-absorption in the charcoal of the corresponding gamma energy. Maximum decay heat generation is expected to occur about 1 hour after the loss of coolant, at which time the available core halogens are assumed to have been deposited on the charcoal beds. The resulting beta heat source is conservatively estimated by 2.155×10^{18} MeV/sec, with accompanying gamma energy of approximately 4.3×10^{18} MeV/sec. The corresponding value as updated for the EPU (*Reference 23*) with Alternate Source Term (AST) is estimated to be $5E17$ MeV/sec, based on 2% of the core elemental + organic iodine.

6.5.1.2.3.2 Decay Heat Dissipation With Normal Air Flow

By virtue of their granular, packed-bed construction and assumed uniform air flow distribution across the bed face, the charcoal adsorbers are capable of dissipating the decay heat generated by the entrapped fission products of service conditions (fans operating) with a calculated safety margin greater than 2000. That is, ignition would not occur even if the concentration of beta and gamma heating in the charcoal were more than 2000 times the predicted maximum value. The prediction itself imposes a hot-spot factor of 10, by assuming the distribution of fission products in the filter bed results in deposition of the decay energy in the first 0.2 in. of the 2-in. thick bed.

Additional assumptions upon which these calculations are based are summarized in Table 6.5-1.

Per *Reference 23*, the decay heat generation rate using the Alternate Source Term (AST) and EPU was shown to be less than the previous decay heat based on the TID source term. Therefore, the EPU will not adversely impact decay heat dissipation with normal air-flow.

6.5.1.2.3.3 Decay Heat Dissipation With Loss of Air Flow

It is postulated that all air flow through the charcoal filter unit is lost at the time when maximum heat generation is attained (about 1 hour after the accident). The adiabatic temperature rise of the hot layer of charcoal would be about 5.8°F/sec. If the actual rise were to occur at this rate, the hot layer would approach ignition temperature in about 68 seconds; however, the

temperature switch setpoint (400°F) would be reached in approximately 20 seconds. The resulting control room alarm would alert the operator to the need for actuating the dousing system. Further temperature rise at this rate would bring the hot layer to the ignition temperature of 680°F^a in an additional 48 seconds.

If the loss of cooling air flow were to occur 24 hours later, the calculated adiabatic temperature rise of the hot layer would be only 1.16°F/sec. The alarm setpoint temperature would be reached in about 100 seconds, with an additional 340 seconds being required before the hot layer reached the ignition temperature.

If it is assumed that the trapped fission products are uniformly distributed throughout the 2-in. charcoal bed depth, the calculated adiabatic temperature rise is approximately 0.58°F/sec (1 hour after the accident). The heatup time to the temperature alarm setpoint is thus about 200 seconds and an additional 480 seconds would elapse before the ignition temperature was reached. With the air flow loss occurring 24 hours later, the adiabatic temperature rise decreases to approximately 0.12°F/sec, requiring about 15 minutes to raise the charcoal temperature to the alarm setpoint and an additional 39 minutes to reach the ignition temperature.

No heat losses from the charcoal bed by radiation or conduction through the plenum walls were assumed in calculating the charcoal heating effects discussed above.

Per *Reference 23*, the decay heat generation rate using the alternate source term and EPU was shown to be less than the previous decay heat rate based on the TID source term. Therefore, the EPU will not adversely impact decay heat dissipation with loss of air flow.

Evaluation has shown that the dousing system is not required to mitigate the consequences of any analyzed transients or accidents and therefore, operation of the system is determined to be beyond design basis requirements. Dousing system water flow rate onto the charcoal filter beds at the minimum (design) value is adequate to remove the fission product decay heat generated at the maximum rate (1 hour after the accident) without boiling. The calculated water temperature increase is approximately 90°F for minimum dousing spray flow. If the refueling water storage tank (RWST) at ambient temperature is the source, spray water boiling cannot occur even if the containment is depressurized. With the external recirculation system as the source of water, the spray flow inlet temperature to the charcoal filters is higher than the refueling water temperature, but this condition is accompanied by containment pressure in excess of the corresponding saturation pressure of the heated spray water exiting from the charcoal beds; thus, boiling does not occur.

6.5.1.2.4 Tests and Inspections

6.5.1.2.4.1 HEPA Filter Tests

The HEPA filters used in the containment recirculation fan cooler (CRFC) system are designed and manufactured in accordance with the requirements of ANSI/ASME N509-1980 and Military Specification MIL-F-51068A. They are specified as well for operation in the post-accident containment environment. Materials used in the filter construction are

a. The charcoal ignition temperature of 680°F has been established on the basis of the manufacturer's test.

compatible with the post-accident containment environment. Each filter is subjected to standard manufacturer's efficiency and production tests prior to shipment. These include flow resistance tests, and the standard efficiency-penetration test requiring that penetration does not exceed 0.03% for 0.3 micron diameter homogeneous dioctylphthalate (DOP) particles. Evaluation tests on sample filters constructed from the filter medium to demonstrate retention of strength under wet conditions can be performed as follows:

- a. Expose the filter to a flow that is equivalent to the design flow of wet steam and water spray in a test facility that simulates the actual filter installation. The water would be injected ahead of the filter with a nozzle designed to produce a fine spray. Free (un-entrained) moisture is removed by means of a moisture eliminator upstream of the filter, but no provision is made for removal of entrained moisture entering the filter.
- b. Follow the wet flow test in item 1 above, the filter would be dried and tested to demonstrate that its resistance to flow has not significantly increased.
- c. Following the test in item 2 above, the filter would be subjected to the National Bureau of Standards dust loading test followed by an ultimate strength test with the deposited dust still on the filter.

The installed HEPA filter banks can be tested periodically by injection of locally generated DOP aerosol into the air stream ahead of the filter inlet. The downstream DOP concentration is monitored for indications of abnormal leakage, i.e., defective filter cells, improperly installed gaskets and clamps, and framing cracks, etc.

6.5.1.2.4.2 *Charcoal Filter Tests*

The normally isolated units can be retested periodically. Individual unit cells are removable for periodic sampling.

6.5.1.2.4.3 *System Tests*

The post-accident charcoal filters will be tested at regular intervals to ensure the capability to meet accident analysis assumptions even after prolonged periods without use. Testing will ensure that degradation due to use has not occurred. Mass flow testing of both the charcoal and the HEPA filters at design flow rates will ensure that the filters have not become plugged with foreign matter and that design flows can be achieved. Dioctylphthalate (DOP) testing will ensure that the HEPA filters will adequately remove particulate material

Detailed requirements for in-place testing of the HEPA and charcoal filters are defined in the Technical Specifications.

6.5.1.2.5 Instrumentation Requirements

Capability for detecting and alarming the presence of fires and localized hot spots in the charcoal filters is provided by a system employing both temperature switches and resistance temperature detectors (RTDs). Each charcoal filter plenum (containing two banks of 60 adsorber units each in a 4 wide by 15 high array) is provided with 24 temperature switches. Twelve switches are uniformly distributed for good coverage in each bank, in a 4 wide by 3 high pattern. The temperature switches are set to close at 400°F and are wired parallel in

two redundant circuits (12 switches per circuit) to a common alarm in the control room; thus, closing of a single switch will actuate the alarm to indicate a high temperature condition in the filter plenum.

In addition, four RTDs are mounted in each plenum for temperature monitoring purposes. One RTD is located centrally above each charcoal adsorber bank, and one is located in the air exhaust path from each bank. These devices have a detection range of 0°F to 600°F, and their output is connected to a four-point readout in the control room. This arrangement enables plenum air temperature at each of the four RTD locations to be monitored individually.

6.5.1.3 Generic Letter 96-06 Requirements

Generic Letter (GL) 96-06 (*Reference 12*) and GL 96-06, Supplement 1 (*Reference 13*), alerted licensees to three issues of concern related to equipment operability and containment integrity involving; water hammer in the cooling water systems serving the containment recirculation fan coolers (CRFCs), two-phase flow conditions in the cooling water systems serving the CRFCs, and thermally induced overpressurization of isolated water-filled piping sections penetrating containment. RG&E's evaluation of these issues was provided to the NRC in *Reference 14* and is summarized below:

- A. During loss of offsite power coincident with design-basis accident conditions (loss-of-coolant accident or main steam line break), water hammer in the service water (SW) system associated with the containment recirculation fan coolers (CRFCs) is not a concern. Evaluation showed that the austenitic stainless steel CRFC tubes will be able to withstand peak pressures associated with water hammer and that the U-bend configuration in the service water inlet and outlet piping of each CRFC at the penetrations will assure a heated buffer or transition region to prevent cold service water from trapping steam voids that formed inside the tubes or in their vicinity.
- B. During design-basis accident conditions, two-phase flow in the service water system associated with the CRFCs will not occur for the two service water pump mode of operation. For the one service water pump mode of operation, a low probability exists for voiding at the "C" CRFC cooler discharge and for flashing to steam in the service water piping downstream of the discharge orifice. Analyses have shown that flow readjustment is expected to increase backpressure at the CRFCs to protect the tubes from steam binding and that reduced flow will result in an increase of backpressure at the coils assuring that two-phase flow does not develop at the coils or tubes. In either case, the reduced flow was found to still meet required heat removal rates. With one or more CRFCs out of operation (i.e., fan is off), there is a slight possibility that water hammer can occur at the 14 in. service water discharge header. However, the effects of such water hammer are mitigated by the heat transfer at the tubes which causes an increased exit temperature of the service water as it arrives at the discharge header.

Review of all safety-related systems inside containment that are needed to mitigate design-basis accident conditions showed several that may be subject to thermal overpressurization. The review included all containment penetrations and non-safety-related systems that could interact with safety-related systems. As a result, the containment spray charcoal filter deluge line and the lines at penetrations 121a, 307, and

324 were modified through the installation of thermal relief valves and the lines at penetrations 205, 206a, and 207a were modified through the installation of bypass lines with check valves. These new installations will function to maintain pipe stresses within allowable limits and relieve pressure to below allowable code limits during post-accident containment heatup.

6.5.1.4 Generic Letter 99-02 Requirements

Generic Letter (GL) 99-02 (*Reference 15*) alerted licensees that testing nuclear-grade activated charcoal to standards other than American Society for Testing and Materials (ASTM) D3803-1989, "Standard Test Method for Nuclear-Grade Activated Carbon", does not provide assurance for complying with NRC requirements. NRC requested that all licensees should amend their Technical Specifications (TS) to reference ASTM D3803-1989. RG&E provided a response to this GL (*Reference 16*) in which RG&E committed to submit a proposed TS amendment request to require testing in accordance with the ASTM D3803-1989 protocol.

RG&E submitted an application for amendment to the Ginna Station TS (*Reference 17*). The NRC staff reviewed RG&E's response and stated (*Reference 18*) that the NRC considers GL 99-02 to be closed for Ginna Station. The license amendment request was approved by the NRC (*Reference 19*) and revises Ginna Station TS 5.5.10, "Ventilation Filter Testing Program", to meet the actions requested by GL 99-02. Subsequent to this, Amendment 87 (*References 21 and 22*) negated the need to test the efficiency of the containment charcoal filters.

6.5.2 CONTAINMENT SPRAY AND NAOH SYSTEMS

The containment spray system as a post-accident pressure reducing system is discussed in Section 6.2.2.

In addition to depressurization, the containment spray system (in conjunction with the NaOH system) is effective in scrubbing fission products with the sodium hydroxide additive. The operation and the effectiveness of the containment spray and NaOH systems as a fission product trapping process is discussed below.

6.5.2.1 System Design and Operation

6.5.2.1.1 Spray Additive Tank

The sodium hydroxide solution is stored in the spray additive tank. This austenitic stainless steel tank has a total capacity of 5100 gallons. The minimum volume of 30% to 35% by weight of sodium hydroxide solution to be maintained is 3000 gallons. The design pressure and temperature of the tank are 300 psig and 300°F. An inert nitrogen blanket is maintained over the tank's liquid volume to minimize long-term degradation of the sodium hydroxide.

Since the tank is not vented to atmosphere, the tank design includes two (2) vacuum breakers. The purpose of the vacuum breakers is to allow air to enter the tank when the pumps are running, to avoid excessive vacuum from being developed which could potentially cause the pumps to work harder and potentially implode the tank.

The spray additive tank supports were modified to withstand loads corresponding to the operating-basis earthquake and safe shutdown earthquake in accordance with ASME Code,

Section III, 1980, through Summer 1982 Addenda. The modifications resulted from the SEP review, Topic III-6, Seismic Upgrade.

The two control valves on the sodium hydroxide tank outlet will open automatically upon receipt of a containment spray actuation signal. Also, these control valves may be operated by position switches located on the control board. On a loss of instrument air or electrical power to the controller, these valves fail open.

A spray additive tank low level of 90% will sound an alarm in the control room. Sodium hydroxide tank level (0-100%) and sodium hydroxide flow (0-50 gpm) are NaOH system parameters that are measured and provide indication in the control room. In addition, sodium hydroxide tank level (0-100% at the sodium hydroxide tank) and sodium hydroxide tank flow (0-50 gpm) are NaOH system parameters that provide local indication.

6.5.2.1.2 Effect of Sodium Hydroxide and Boric Acid Mixing

During post-accident operation of the containment spray and NaOH systems, dilution and partial neutralization of the sodium hydroxide additive occurs in two stages: first, as the sodium hydroxide mixes with refueling water in the containment spray pump suction piping, and second as the containment spray solution combines with emergency core cooling water in the containment sump during the recirculation phase.

In the early minutes of the sump mixing stage there is potentially an excess of H_3BO_3 due to the introduction of the accumulator contents and the inventory of the reactor coolant system.

During the injection period, which may last 25 minutes to more than an hour, boric acid and sodium hydroxide are mixed and added to the containment via the containment spray headers. During the injection phase, the pH of the spray estimated for the original plant design was in the range of 8.3 to 9.1, based upon a boron concentration of 2000 to 2300 ppm. Subsequent to the conversion to an 18 month fuel cycle, the Extended Power Uprate (EPU) and the associated higher boron concentration of 2750 to 3050 ppm, the sprayed liquid upper pH value was determined to be 9.90, which is within the limit of 10.5 specified in Standard Review Plan 6.5.2. Sodium hydroxide may be added by blending with the solution recirculated from the sump during the post-LOCA recirculation phase, however, procedures dictate and analysis supports the termination of containment spray prior to entry into the sump recirculation phase.

Assuming instantaneous mixing (maximum acidification of the iodine-bearing spray solution), the pH of the combined solution of the spray and in the sump as a function of time is shown in Figure 6.5-3. This figure is for the original plant design analysis. This analysis assumed that sodium hydroxide was continued to be added during the recirculation phase until the sodium hydroxide tank reached its low-level setpoint. It also assumed the refueling water storage tank (RWST) volume was 230,000 gallons, which was later increased to 300,000 gallons (see Section 6.3.3.3), and a boron concentration less than the current 2300 to 2600 ppm. This figure is retained for historical reference. Post-accident chemistry was reexamined during the Systematic Evaluation Program under SEP Topic VI-1 (*Reference 1*). Figure 6.5-3 also shows the partition coefficient of iodine for both the spray and the sump, calculated for the combination of pH and maximum iodine concentration that could exist with

the original design. The data of Eggleton for 312°F solutions were used as the basis for computing the partition coefficient. The results show that retention of absorbed iodine by the water in the sump is strongly favored from the early moments due to the strong base and weak acid characteristics of the additives present.

Emergency operating procedures require containment spray to be terminated at the end of the injection phase and prior to the recirculation phase. An evaluation was performed to determine the minimum pH in the containment sump if containment spray is terminated after the injection phase (*Reference 2*). It was assumed that only one containment spray pump is operating during the injection phase, which minimizes the volume of sodium hydroxide added following a design-basis LOCA. It was also assumed that two residual heat removal pumps, three safety injection pumps, and the one spray pump operate until the refueling water storage tank (RWST) goes from the 300,000 gallon (88%) level to the 28% level, at which time both residual heat removal pumps and one safety injection pump are stopped, while the remaining pumps continue to operate until the refueling water storage tank (RWST) decreases to the 15% level. At this level the remaining safety injection pumps and the containment spray pump are stopped and transfer to sump recirculation is completed. Containment spray will have operated for 52.4 minutes and, at a sodium hydroxide eductor suction flow of 20 gpm, will have injected 1048 gallons of sodium hydroxide. This sodium hydroxide mixed with the liquid from the refueling water storage tank (RWST), the boric acid storage tanks, accumulators, and the reactor coolant system resulted in a pH level of 7.8 (*Reference 20*) in the containment sump, which is above the minimum pH of 7.0 stated in NRC Branch Technical Position MTEB 6-1. That analysis included the effect of 6100 lb of 12% H_3BO_3 from the boric acid storage tanks added during the injection phase. The safety injection system was subsequently reconfigured to isolate the flow path from the boric acid storage tanks; therefore, less boric acid is available to mix with contents in the containment sump. This, combined with the effect of a higher boron concentration of 2750 to 3050 ppm, resulted in a reevaluation of sump pH for the current range of spray additive tank solution of 30 wt % to 35 wt % (See Section 6.1.2.1.4). A more recent evaluation of sump pH for GSI-191 was performed (See Section 6.3.2.1.1) for the purpose of determining the chemical effects.

The next sections (6.5.2.1.3 through 6.5.2.2.4) have not been updated as a result of the higher boron concentration associated with the conversion to an 18 month fuel cycle EPU or the lower sump pH used in the dose analysis. Containment spray system pH is discussed in Section 6.1.2.1.4. The effect of the higher boron concentration has been evaluated by the Westinghouse (*Reference 10* and *20*) and found to be acceptable. The iodine removal coefficients for containment spray have been re-calculated in *Reference 11* as follows:

Elemental, hr^{-1}	20
Particulate, hr^{-1}	0 (<1.33 minutes) 3.5 (≥ 1.33 , <52 minutes) 0 (≥ 52 minutes)

The values listed above have been used in the LOCA dose calculations as presented in Section 15.6.4.2.5.

6.5.2.1.3 Iodine Retention

Since the current dose calculations presented in Section 15.6.4.2.5 were performed using more current methodology than was utilized in the original licensing of Ginna Station, the following sections represent the original design only, prior to conversion to an 18 month fuel cycle in 1996 or the EPU in 2006. The information is retained for historical reference.

The following combinations of equipment will provide sufficient iodine trapping capability to ensure a post-accident fission product leakage (based on TID 14844 release fractions) that would not exceed the dose limits of 10 CFR 100.

During the injection phase of the design-basis accident, the pH of the containment spray and the resulting partition factor is sufficiently large to prevent any significant amount of iodine from being re-emitted from the sump during the short period of time it takes the sump pH to reach 8.0. Note that at 5 minutes after the start of spray injection, the minimum mixed pH of the sump, i.e., with maximum dilution of spray additive by the boric acid coolant in the sump, is 6.83. The corresponding partition of iodine would result in retention of 74% of the iodine in the liquid phase. Since the theoretical removal rate by spray (Section 6.5.2.2) is approximately 30 hr^{-1} , 5 minutes of spraying would ideally remove about 90% of the available iodine, indicating partial restriction of performance by re-emission if complete and instantaneous mixing were to occur. Realistically, mixing will be less than perfect; hence, the iodine laden water will be at a higher than average pH, and retention will be more favorable. It is concluded that any limitation imposed by the pH history of the sump would be short-lived and would probably influence only the theoretical and not the practical removal rate to any substantial degree.

The fraction of the free volume of the containment that is not washed by containment sprays is about 22%. In the analysis of spray effectiveness as an iodine absorber, (Section 6.5.2.2.2.1, Equation 6.5-6), a conservative allowance is made for this fact by assuming that the only free volume available for contact and dilution between gas and falling droplets is the volume between the operating deck and the spray headers. The containment recirculation fan cooler (CRFC) system minimizes the difference in iodine concentration between the sprayed and unsprayed regions by creating a forced interchange of atmosphere at an average turnover rate of the order of once per minute in the reactor coolant loops compartments.

As discussed in Section 6.5.2.2.2.1, the idealized absorption model for alkaline sodium borate spray predicts a removal rate constant of 29.6 hr^{-1} for elemental iodine in the Ginna Station containment. It is to be expected that some deviations from this model may occur; however, due to the large heat removal duty imposed on the sprays, the design of the system provides a greater flow and dispersal of spray solution than is required to meet the iodine removal objectives. The magnitude of this margin is expressed by the fact that a removal constant of only 3 hr^{-1} is sufficient to reduce the exclusion area boundary (EAB) dose to 300 rem in the first 2 hours.

An order-of-magnitude deviation of spray performance would be required in order to prevent the spray system from meeting this design goal; therefore, from a performance point of view, the spray system design is adequately justified by Containment Systems Experiment and Nuclear Safety Pilot Plant tests which have shown iodine removal to follow very closely the idealized gas-film controlled model prediction.

For evaluation purposes, a conservatism factor of about 2 is adequate to reflect concern for coalescence effects which may occur in the extrapolation from these tests to the full-scale containment.

Additional safety is inherent in the fact that the charcoal filters in the containment recirculation fan cooler (CRFC) system represent an independent means of iodine removal that is also capable of satisfying the regulatory requirements with no help from containment spray. The system capacities are such that the single-failure criterion is met assuming operation of both containment spray trains, both charcoal filters, or one containment spray train and one charcoal filter.

6.5.2.2 Iodine Effectiveness Evaluation of the Containment Spray and NaOH Systems

6.5.2.2.1 Purpose of Chemical Modification

6.5.2.2.1.1 Thermal Capacity

The containment spray system is one of the engineered safety features systems employed following a LOCA inside the containment to reduce the pressure and temperature of the containment atmosphere. The flow rate and inlet subcooling of the spray are sufficient to provide thermal capacity for condensing steam produced by dissipation of heat in the reactor and its associated systems. Minimum operability of this system with onsite power and under a single component failure contingency will prevent pressurization of containment above the design pressure.

6.5.2.2.1.2 Absorption of Iodine in Refueling Water Spray

The containment spray and NaOH systems, by virtue of the large surface area provided between the liquid droplets and the containment atmosphere, afford an excellent means of absorbing the soluble components from the gas phase. If the solubility of the component is sufficiently high, the rate of absorption is limited only by the mass transfer rate of the absorbing species through the gas film. In the case of I₂ vapor, elimination of all but the gas film resistance would permit the absorption by sprays to proceed with a removal half-life of less than 2 minutes, as will be shown later; however, the solubility of I₂ in the refueling water used as spray is limited, in acidic solution, as indicated by the partition coefficient given below (*Reference 3*):

$$K_c = 0.0125 \frac{\text{mole/l gas}}{\text{mole/l liquid}}$$

(Equation 6.5-1)

While this coefficient corresponds to an equilibrium-favoring solution of 60% to 80% of the iodine by the liquid (considering the gas/liquid volume ratios of conventional PWR containments), it is expected that the liquid phase mass transfer resistance would severely limit the removal rate. Assuming a liquid film coefficient of 0.001 cm/sec and a gas film coefficient of the order of 10 cm/sec, the overall mass transfer coefficient, V_T , is obtained as follows:

$$\frac{1}{V_T} = \frac{1}{V_G} + \frac{K_C}{V_L} = \frac{1}{10} + \frac{0.0125}{0.001} = 0.1 + 12.5$$

(Equation 6.5-2)

$$V_T = 0.079 \text{ cm/sec}$$

If the I_2 were infinitely soluble (K_C is approximately 0), the value of V_T would approach 10 cm/sec in this example.

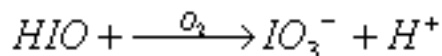
6.5.2.2.1.3 Iodine Absorption with Sodium Hydroxide Addition

To obtain the advantages of an order-of-magnitude improvement of absorption rate and nearly complete removal of I_2 at equilibrium, the chemistry of the spray solution is modified by adding NaOH, raising the pH to 9.5. According to the known behavior of elemental iodine in highly dilute solutions, the hydrolysis reaction,



(Equation 6.5-3)

proceeds nearly to completion (*Reference 4*) at pH greater than 8. The iodide form is highly soluble and HIO readily oxidizes to IO_3^- in the oxygenated medium, this form being likewise soluble:



(Equation 6.5-4)

Griffith (*Reference 3*) suggested that the use of chemical additives which undergo ionic reactions with aqueous I_2 would improve the absorption rate to the point where the gas film mass transfer resistance became limiting, implying that K_C/V_L is much less than 10^{-4} . Griffith's paper called attention to sodium thiosulfate ($Na_2S_2O_3$) as a likely reagent for this purpose and mentioned NaOH as another candidate. Subsequent experiments in a spray medium have shown that both additives bring about absorption rates indicative of gas film control, verifying the desired rate capability.

The selection of NaOH instead of $Na_2S_2O_3$ for this application followed an evaluation program which revealed certain disadvantages for $Na_2S_2O_3$. The results of this evaluation program are included in the Proprietary Westinghouse report, WCAP 7153, March 1968.

By contrast, the same testing program revealed no instability of the solution formed by adding NaOH alone to the borated spray. Corrosion rates of copper and copper-alloy heat exchanger tubing were reduced by more than an order of magnitude compared with high pH $\text{Na}_2\text{S}_2\text{O}_3$ and were acceptably low (less than 0.01 mil/month at 200°F) for the application. These tests showed that pitting or local corrosion did not occur.

6.5.2.2.1.4 *Spray Absorption Process for Iodine Removal*

Therefore, for engineering reasons, further testing was centered on the use of NaOH as the spray additive, leading to the development of a technical basis for its inclusion in the plant engineered safety features as a means of fixing absorbed iodine, enhancing the natural rate of the deposition of I_2 , and thus lowering the calculated offsite thyroid dose resulting from a postulated release of fission products to the containment atmosphere. In summary, this work supports the following conclusions comprising the technical basis for the spray absorption process for iodine removal:

- a. The conversion of absorbed I_2 to I and IO_2^- in pH 9.5 borate solution is quite rapid, such that the absorption process is gas film diffusion controlled.
- b. Mass transfer follows the Ranz-Marshall rate equation (*Reference 5*) for soluble gases, as demonstrated by containment simulation tests performed with a nozzle design, atmospheric conditions, iodine concentration, and spray chemical composition in close approximation to the design-basis accident.
- c. Under a range of conditions bracketing the possible accident modes, the spray experiments and calculations show that the iodine removal process is irreversible (i.e., K_c is not reduced with time if pH is maintained), is compatible with the vital materials and processes of the containment system, and shows high mechanical reliability.

6.5.2.2.2 Technical Basis for Iodine Removal Factor

6.5.2.2.2.1 Analytical Model and Assumptions

The removal of a soluble component by a reactive spray under conditions of a constant mass transfer rate coefficient is exponential:

$$C = C_0 \cdot e^{-\lambda_s t}$$

(Equation 6.5-5)

The removal constant λ_s can be expressed as the product of a mass transfer coefficient, v_G , and the effective absorbing surface area, A .

In addition to the basic assumption that the absorption is gas-film resistance controlled, the following idealizations are made to simplify the physical model:

- a. All droplets behave as spheres or diameter equal to that of the surface-mean diameter droplet, d .

- b. Droplets fall at their terminal velocity, u_t , from the spray nozzle to the operating deck, a distance h .
- c. Iodine concentration in the gas is uniform.

The effective absorbing area is then calculated as follows:

$$A = \frac{6 \cdot F \cdot h}{u_t \cdot V_c \cdot d}$$

(Equation 6.5-6)

where: A = absorbing area per unit volume
 F = volumetric spray flow rate
 V_c = containment free volume

For a given droplet size, the terminal velocity and the mass transfer coefficient are temperature and pressure dependent. In the expression for λ_s , then, these variables can be treated as a dimensionless ratio:

$$\lambda_s = V_G A = \left(\frac{V_G}{u_t} \right) \frac{F \cdot h}{V_c \cdot d}$$

(Equation 6.5-7)

When the remaining parameters are expressed in engineered units, λ_s in reciprocal hours is given by the following equation:

$$\lambda_s = 1470 \cdot \left(\frac{V_G}{u_t} \right) \frac{F \cdot h}{V_c \cdot d}$$

(Equation 6.5-8)

where: F = spray flow gal/minute
 h = fall height, ft
 V_c = volume, ft³
 d = droplet diameter, cm

For the various classes of Westinghouse PWR containments, the following values of the physical parameters are conservatively approximated as follows:

	F	h	V_c	d	Fh/V_cd
Two loop	1,250	70	970,000	0.1	0.90

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	<u>F</u>	<u>h</u>	<u>V_C</u>	<u>d</u>	<u>Fh/V_Cd</u>
Three loop	880	77	2,100,000	0.1	0.32
Four loop	2,340	104	2,600,000	0.1	0.94

The value of V_G/u_t for 0.1 cm droplets in a saturated air-steam atmosphere of pressure P^a is plotted in Figure 6.5-4. These data are obtained from ORNL TM-1911 (*Reference 6*). It is apparent that as pressure decrease during the post-accident period, the value of V_G/u_t , and hence the removal coefficient λ_s , will increase; therefore, the removal rate is underestimated by assuming, for purposes of analysis, the value of this ratio at the design condition of the containment. The results, calculated from Equation 6.5-8, are as follows:

	<u>V_G/u_t</u>	<u>λ_s (hr⁻¹)</u>
Two loop (60 psig, 286°F)	0.0224	29.6
Three loop (42 psig, 264°F)	0.0236	11.1
Four loop (47 psig, 270°F)	0.0231	32.0

-
- a. Defined as the mixture of air and steam produced by adding steam to dry air at an initial temperature of 30°C and one atmosphere pressure, at constant volume.

6.5.2.2.2 *Removal of Elemental Iodine*

The half-life for removal of elemental iodine is obtained from the following expression for exponential decay:

$$T_H = (0.693 \times 3600) / \lambda_s \text{ (in sec)}$$

The dose reduction factor applicable for elemental iodine is the ratio of the average 2-hr inventory of I₂ without removal to the average with removal. It is given by the following expression:

$$DRF_2 = \frac{2\lambda_s}{1 - e^{-2\lambda_s}}$$

(Equation 6.9-1)

The calculated values of λ_s for the three plant types yield the following values of I₂ half-life and I₂ dose reduction factor:

	<u>T_H sec</u>	<u>DRF₂</u>
Two loop	84	59
Three loop	225	22
Four loop	78	64

6.5.2.2.2.3 *Removal of Other Airborne Forms of Iodine*

Concerning other airborne forms of iodine, the removal mechanisms can be characterized in the following ways:

- a. **HI** - Hydrogen iodide may constitute an important fraction of the liberated iodine if oxygen is excluded from the reactor during the melt. The higher diffusivity of HI, compared with I₂, and the fact that favorable partition between vapor and liquid does not require that the absorbed HI molecule undergo chemical reaction, would lead to removal of HI by sprays no less rapid than I₂.
- b. **CH₃I** - A small fraction of the available iodine will exist as organic iodides, of which methyl iodide is the most important. There is preliminary evidence that absorption and chemical decomposition of CH₃I occurs in the reference spray solution. The rate of absorption, which is expected to be liquid film diffusion of liquid-phase reaction inventory of CH₃I vapor is less than the probable error in predicting that inventory. No credit for removal is taken in calculating the 2-hr dose due to organic iodide leakage. (See Section 15.6.3.5.)
- c. **Particles** - Spray may have an important effect on particle removal by increasing the rate of steam condensation. When the bulk flow of steam to the condensing surface is great enough to mask the diffusive motion of particles, as would be the case when cold droplets contact the containment atmosphere during the high-steam period, sub-micron particles are efficiently captured by the spray (*Reference 7*). Larger particles would be removed by HEPA filters or would agglomerate and settle out by gravity, reducing their importance as a potential leakage source, if they could penetrate the leakage path at all. In evaluating the potential benefit of sprays in reducing post-accident iodine leakage, no quantitative consideration is given here to particle removal by condensation because the phenomenon is independent of the chemical modification of the spray solution.

6.5.2.2.2.4 *Experimental Verification*

The droplet size assumed in the spray calculations summarized above was 0.10 cm or 1000 microns. The spray pattern produced by a 3/8-in. aperture ramp bottom nozzle of the type used in these facilities was measured photographically at various operating nozzle pressures. A statistical analysis of the droplet images produced the following results:

<u>Nozzle Pressure (psi)</u>	<u>Flow Rate (gpm)</u>	<u>Number Average (diameter, microns)</u>	<u>Surface Average (diameter, microns)</u>
20	10.9	960	1340
30	12.9	830	1126
40	15.2	735	1012
50	16.9	685	961

The spray system is designed to deliver rated flow with a minimum available nozzle pressure of 40 psi above the containment design pressure. Generally, the nozzle pressure will be more than 40 psi above the actual containment pressure, making the assumption of 1000 microns a realistic one.

A more meaningful demonstration of effective droplet size and verification of the overall mass transfer model is obtained from the spray test program at the Nuclear Safety Pilot Plant. Data from these tests are published through the regular Oak Ridge National Laboratory reporting channels (*References 8 and 9*). The data treatment in this program uses the same basic analytical model as has been presented here and the results are entirely consistent with the premise that I_2 absorption by $N_aOH-HBO_3$ spray is gas-film controlled.

Applying the removal expression (*Equation 6.5-9*) to the Nuclear Safety Pilot Plant system for a typical test condition of 44 psig (266°F), the values corresponding to the plant parameters are as follows:

	f	h	V_C	d	$Fh/V_C d$
NSPP	15	17	1330	0.100	1.92

The ratio V_G/\bar{u}_t for 0.100 cm droplets in a 44 psig steam-air atmosphere (Figure 6.5-4) is 0.0234, giving $\lambda_s = 66 \text{ hr}^{-1}$ and a half-life of 38 seconds. The half-life observed in Nuclear Safety Pilot Plant tests at this condition is in good agreement (*Reference 9*).

To illustrate, the spray absorption rate coefficient, λ_s , has a value of 29.6 hr^{-1} in the Ginna Station calculation when no allowance is made for spectrum perturbation by drop coalescence. Experiments in the Nuclear Safety Pilot Plant and Containment Systems Experiment have shown reasonable agreement with the model where no such allowance is made.

Extrapolation to the full-sized plant allows more opportunity for coalescence, but the effect on absorption rate coefficient was shown to be much less than a factor of 2 in preliminary analyses.

Since the minimum requirements for removal of inorganic iodine in the Ginna Station containment is of the order of 3 hr^{-1} , it is not necessary that this aspect of the analysis be completed in a refined form to show that size distribution effects of the fall height extrapolation will not violate the system design basis. The large margin between removal rate predicted and that required is because the spray system size is governed by the heat removal duty under accident conditions.

REFERENCES FOR SECTION 6.5

1. Letter from John E. Maier, RG&E, to Dennis M. Crutchfield, NRC, Subject: SEP Topic VI-1, Organic Materials and Post-accident Chemistry, dated November 6, 1981.
2. Rochester Gas and Electric Corporation, Termination of Containment Spray After the Injection Phase Post LOCA, 10 CFR 50.59 Safety Evaluation NSL-0000-009, Revision 1, dated October 25, 1989.
3. V. Griffith, "The Removal of Iodine from the Atmosphere by Sprays," AHSB (S) R 45.
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10. Letter from K.C. Hoskins, Westinghouse Electric Corporation, to R.W. Elias, RG&E, Reference No. 95 RGE-G-0037, Subject: 18 Month Fuel Cycle Upgrade Program - Systems Review Summary, dated June 1, 1995.
11. Ginna Station Design Analysis DA-NS-2001-087, Rev 4 Large Break LOCA Offsite and Control Room Doses, October 26, 2006.
12. Generic Letter 96-06, Assurance of Equipment Operability and Containment Integrity During Design-Basis Accident Conditions, dated September 30, 1996.
13. Generic Letter 96-06, Supplement 1, Assurance of Equipment Operability and Containment Integrity During Design-Basis Accident Conditions, dated November 13, 1997.
14. Letter from R. C. Mecredy, RG&E, to G. S. Vissing, NRC, Subject: Response to Generic Letter 96-06, Assurance of Equipment Operability and Containment Integrity During Design-Basis Accident Conditions, dated January 30, 1997.
15. Generic Letter 99-02, Laboratory Testing of Nuclear-Grade Activated Charcoal, dated June 3, 1999.
16. Letter from R. C. Mecredy, RG&E, to G. S. Vissing, NRC, Subject: Response to Generic Letter 99-02, dated November 30, 1999.

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17. Letter from R. C. Mecredy, RG&E, to G. S. Vissing, NRC, Subject: Application for Amendment to Facility Operating License, Ventilation Filter Testing Program Change (5.5.10), dated November 30, 1999.
18. Letter from G. S. Vissing, NRC to R. C. Mecredy, RG&E, Subject: Completion of Licensing Action for Generic Letter 99-02 (TAC No. MA5796), dated February 22, 2000.
19. Letter from G. S. Vissing, NRC to R. C. Mecredy, RG&E, Subject: R. E. Ginna Nuclear Power Plant, Amendment Re: Laboratory Testing of Nuclear-Grade Activated Charcoal (TAC No. MA7266), dated April 12, 2000.
20. Westinghouse Calculation (Proprietary) CN-SEE-05-18, Rev 1, R. E. Ginna Extended Power Uprate (EPU) BORDER Study.
21. Letter from Robert Clark (NRC) to Robert Mecredy (Ginna), Subject: R. E. Ginna Nuclear Power Plant-Amendment Re: Elimination of Post Accident Sampling System (TAC No MB3387), dated January 17, 2002.
22. Letter from D. Skay (NRC) to M. G. Korsnick (Ginna), Subject: R. E. Ginna Nuclear Power Plant - Amendment Eliminating Requirements for Hydrogen Recombiner and Hydrogen Monitors using the Consolidated Line Item Improvement Process (TAC No. MC4195), dated May 5, 2005.
23. Letter from P. Milano (NRC) to M. G. Korsnick (Ginna), R. E. Ginna Nuclear Power Plant - Amendment Re: 16.8 percent power uprate (TAC No. MC7382), dated July 11, 2006.

Table 6.5-1
DATA FOR CHARCOAL FILTER EVALUATION

Conditions of Air-Steam Atmosphere

Pressure	75 psia
Temperature entering	286 °F
Relative humidity	100%
Velocity normal to bed	40 fpm

Charcoal Bed Characteristics

Mean granule diameter	0.06 in.
Bulk density	33 lb/ft ³
Depth	2 in.
Bed dimensions	1.9 ft wide 2.1 ft deep

Heat Source

Fission products	25% of core iodine and bromine (Pre AST) 2% of core elemental + organic iodine (Post AST)
Beta energy absorption	100%
Gamma energy absorption	68%
Charcoal participating	Initial 0.2 in. of bed

6.6 **INSERVICE INSPECTION OF CLASS 2 AND 3 COMPONENTS**

6.6.1 **INTRODUCTION**

The Inservice Inspection (ISI) Program document describes the program for the current inspection interval.

The Inservice Inspection (ISI) Program document for Class 2 and 3 components adheres to the requirements of 10 CFR 50.55a(g) and ASME Boiler and Pressure Vessel Code, Section XI, except where specific written relief has been granted by the NRC pursuant to 10 CFR 50.55a(g)(6)(i).

As indicated in *Reference 1*, an augmented inservice inspection program for high-energy piping outside containment has been established. The inspection program provides for volumetric examination on all circumferential butt welds situated at design break locations or at discontinuity locations where probable failure could occur. Surveillance of these welds can detect material changes in advance of a potential failure, thereby ensuring that the design basis or consequential main steam or feedwater line break will not occur (see Section 3.6.2.1).

6.6.2 **INSERVICE INSPECTION PROGRAM SUMMARY**

6.6.2.1 **Scope**

The specific Class 2 and 3 components to be examined under the program are identified in the Inservice Inspection (ISI) Program document.

6.6.2.2 **Inspection Intervals**

The inservice inspection intervals for Class 2 and 3 components are 10-year intervals. The initial 10-year interval for Class 2 and 3 components commenced May 1, 1973. The second interval was revised to commence on January 1, 1980, to coincide with the interval of the Class 1 program (Section 5.2.4). The third 10-year interval commenced on January 1, 1990. The fourth 10-year interval commenced on January 1, 2000. The fifth 10-year interval commenced on January 1, 2010. The requirements and distribution of examinations within the inspection interval is in accordance with the Inservice Inspection (ISI) Plan.

The inservice inspection interval for the high-energy piping outside containment is also a 10-year interval of service commencing initially on May 1, 1973, and then revised to commence January 1, 1980. The third 10-year interval commenced on January 1, 1990. The fourth 10-year interval commenced on January 1, 2000. The fifth 10-year interval commenced on January 1, 2010. The distribution of examinations within the inspection interval is in accordance with the Inservice Inspection (ISI) Program document.

6.6.2.3 **Extent and Frequency**

Class 2 and 3 components are examined to the extent and frequency as defined in the Inservice Inspection (ISI) Plan document.

6.6.2.4 Examination Methods

Class 2 components are examined by the required visual, surface, or volumetric methods. The examinations include one or a combination of the following: visual, liquid penetrant, magnetic particle, ultrasonic, eddy-current, and radiographic examinations, plus visual leakage examinations. Class 3 components are examined by the required visual or surface examination methods plus visual leakage examinations are performed. Component supports are examined by the visual method. High-energy piping welds outside of containment are volumetrically examined. The required examination methods within this inspection interval are in accordance with the Inservice Inspection (ISI) Plan.

6.6.2.5 Evaluation of Examination Results

The evaluation of nondestructive examination results for Class 2 and Class 3 components and component supports are in accordance with the Inservice Inspection (ISI) Plan. The evaluation of nondestructive examination results for high-energy piping is in accordance with the Inservice Inspection (ISI) Plan document.

6.6.2.6 System Pressure Testing

System pressure testing shall be performed in accordance with the Inservice Inspection (ISI) Plan document.

6.6.2.7 Records and Reports

Records and reports developed from the examinations performed under the Inservice Inspection Plan are maintained in accordance with Article IWA-6000 of the ASME Code, Section XI.

6.6.2.8 Exemptions

Paragraphs IWC-1220, IWD-1220, and IWF-1230 of the ASME Code, Section XI, exempt certain components from examinations where certain conditions are met. These exemptions are applied to the components included in the Inservice Inspection Plan as applicable. Requests for relief are submitted where it is impossible or impractical to examine or test an applicable Class 2 or 3 component or system. Current exemptions in this category are identified in the Inservice Inspection (ISI) Plan document.

REFERENCES FOR SECTION 6.6

1. Rochester Gas and Electric Corporation, Effects of Postulated Pipe Breaks Outside the Containment Building, dated October 29, 1973.