CHAPTER 6

6.0 ENGINEERD SAFETY FEATURES

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6.0 ENGINEERED SAFETY FEATURES

Engineered safety features are included in the design of the HBR 2 facility to mitigate the consequences of a postulated accident in spite of the fact that these accidents are very unlikely. These safety features are:

1. The Safety Injection (SI) System accumulators and pumps, which inject borated water into each coolant loop of the Reactor Coolant System (RCS). This system limits damage to the core and limits the energy released into the containment following a loss-of-coolant accident (LOCA).

2. The Containment Spray System, which is used to reduce containment pressure and to wash down iodine into the containment sump.

3. The air recirculation coolers, which reduce containment pressure following a LOCA.

4. A steel-lined concrete containment structure described herein, with testable penetrations and liner welds, which form a virtually leak-tight barrier to the escape of fission products should a LOCA occur.

5. An Isolation Valve Seal Water System, which creates a leak tight seal in pipes which could communicate with the atmosphere inside the containment following a LOCA.

6. A reactor coolant gas vent system which vents non-condensable gases from the reactor vessel head and the pressurizer steam space during post accident situations.

6.1 ENGINEERED SAFETY FEATURES MATERIALS

6.1.1 METALLIC MATERIALS

6.1.1.1 Materials Selection and Fabrication

6.1.1.1.1 Emergency Core Cooling System Components

Emergency core cooling system (ECCS) components are constructed of austenitic stainless steel or an equivalent corrosion resistant material (except the ECCS coarse screen frames which are carbon steel), and hence are quite compatible with the spray solution over the full range of exposure in the post-accident regime. While this material is subject to crevice corrosion by hot concentrated caustic solution, the NaOH additive cannot enter the containment or the ECCS without first being diluted and partially neutralized with boric acid to a mild solution. Corrosion tests performed with simulated spray showed negligible attack, both generally and locally, in stressed and unstressed stainless steel at containment and ECCS conditions. These tests are discussed in Reference 6.1.1-1. The carbon steel ECCS coarse screen frames structural integrity will not be adversely impacted during the post-accident exposure period.

6.1.1.1.1.1 Pumps

The pressure-containing parts of the pumps were constructed of castings which conformed to American Society for Testing and Materials (ASTM) A-351 Grade CF8 or CF8M specifications. Stainless steel forgings were procured per ASTM A-182 Grade F304 or F316 or ASTM A-336, Class F8 or F8M, and stainless plate was constructed to ASTM A-240 Type 304 or 316 specifications. All bolting material conformed to ASTM A-193. Materials such as weld-deposited Stellite or Colmonoy were used at points of close running clearances in the pumps to prevent galling and to assure continued performance capability in high velocity areas subject to erosion.

All pressure-containing parts of the pumps were chemically and physically analyzed and the results checked to ensure conformance with the applicable ASTM specification. In addition, all pressure-containing parts of the pump were liquid penetrant inspected in accordance with Appendix VIII of Section VIII of the ASME Boiler and Pressure Vessel (B&PV) Code. The acceptance standard for the liquid penetrant test was USAS B31.1, Code for Pressure Piping, Case N-10.

Where welding of pressure containing parts 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 also applied to any repair welding performed on pressure containing parts.

6.1.1.1.1.2 Heat Exchangers

The two residual heat exchangers of the Auxiliary Coolant System conform to the strict rules of the ASME Code regarding the wall thicknesses of all pressure containing parts, material quality assurance provisions, weld joint design, radiographic and liquid penetrant examination of materials and joints, and hydrostatic testing of the unit as well as final inspection and stamping of the vessel by an ASME Code inspector. Each unit has an SA-212-B Carbon Steel shell, an SA-212-B Carbon Steel shell end cap, SA-213 Type-304 Stainless Steel tubes, an SA-240 Type 304 Stainless Steel channel, an SA-240 Type 304 Stainless Steel channel cover and an SA-240 Type 304 Stainless Steel tube sheet.

6.1.1.1.1.3 Valves

All material in accumulator check valves, motor-operated valves, and all other ECCS valves in contact with radioactive fluid were constructed (except the packing) of austenitic stainless steel or materials of equivalent corrosion resistance. Carbon steel was used for manual globe, gate and check valves which pass only non-radioactive fluids.

6.1.1.1.1.3.1 Stainless steel valves (except accumulator check valves)

The pressure-containing parts (body, bonnet and discs) of the valves employed in the Safety Injection (SI) System were designed to meet or exceed criteria established by the USAS B16.5 or MSS SP66 specifications. The materials of construction for these parts were procured to applicable ASME or ASTM specifications for austenitic stainless steel materials.

The pressure containing cast components were radiographically inspected as outlined in ASTM E-71 Class 1 or Class 2. The body, bonnet and discs were liquid penetrant inspected in accordance with the ASME Code Section VIII, Appendix VIII or ASME Code Section III. The liquid penetrant acceptance standard was as outlined in USAS B31.1 Case N-10 or ASME Code Section III.

When a gasket was employed, the body-to-bonnet joint was designed to meet or exceed the ASME 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. RHR-759A and B were evaluated to use Flexpro style gaskets. The body-to-bonnet bolting and nut materials were procured per ASTM A193 and A194, respectively, or equivalent.

The seating surfaces chosen are hard faced (Stellite No. 6, nickel-chrome-boron, or equivalent) to prevent galling and reduce wear.

The stem material chosen was ASTM A276 Type 316 condition B or precipitation hardened 17-4 PH stainless procured and heat treated to Westinghouse or DEP approved Specifications. These materials were selected because of their corrosion resistance, high tensile properties, and their resistance to surface scoring by the packing. With the exception of valves which have had their leak off line removed and various packing arrangements including live loading and standard bolting, the valve stuffing box is designed with a lantern ring leak-off connection with a minimum of a full set of packing below the lantern ring and a maximum of one-half of a set of packing above the lantern ring; a full set of packing is defined as a depth of packing equal to 1-1/2 times the stem diameter. The experience with this stuffing box design and the selection of packing and stem materials has been very favorable in both conventional and nuclear power plants.

6.1.1.1.1.3.2 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 were procured to applicable ASTM or WAPD specifications. The cast pressure containing parts were radiographed in accordance with ASTM E-94 and the acceptance standard 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 the ASME Code, Section VIII, and the acceptance standard was as outlined in USAS B31.1, Code Case N-10.

The valve was designed with a low pressure drop configuration, with all operating parts contained 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 disc and seat ring mating surface and 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. Nickel-chrome-boron may be used as an alternate hard-surfacing material.

6.1.1.1.1.3.3 Carbon steel valves

The carbon steel valves pass only nonradioactive fluids. Since the fluid controlled by the carbon steel valves is not radioactive, the double packing included in the stainless steel valve design described in Section 6.1.1.1.1.3.1 and seal weld provisions were not provided.

The carbon steel valves were built to meet or exceed USAS B16.5. The materials of construction of the body, bonnet and disc conformed to the requirements of ASTM A105 Grade II, A181 Grade II, or A216 Grade WCB or WCC, or equivalent.

6.1.1.1.1.4 Piping

All SI System piping in contact with borated water is austenitic stainless steel. Piping joints are welded except for the flanged connections at the SI and containment spray pumps. The leak off lines for RHR-759A, 759B, 757A, 757B, 757C, and 757D are capped via a threaded joint.

The piping was designed to meet the minimum requirements set forth in:

1. The USAS B31.1 Code for 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 in Section 1, Piping of the USAS Code for Pressure Piping. This minimum thickness was increased to account for the manufacturer's permissible tolerance of minus 12 1/2 percent on the nominal wall. Purchased pipe and fittings had a specified nominal wall/thickness that was no less than the sum of that required for pressure containment, mechanical strength, and manufacturing tolerance.

Pipe and fitting materials were procured in conformance with all requirements of the ASTM and USAS specifications. All materials were verified for conformance to specification and documented by certification of compliance to ASTM material requirements. Specifications imposed additional quality control upon 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. Fittings 3 in. and above have requirements for UT inspection similar to S6 of ASTM A376, except the 6" diameter end caps used in fabricating strainers for the 3/4" diameter piping branching off of the 3" discharge lines of the safety injection pumps.

Welds for pipes sized 2 1/2 in. and larger are butt welded. Reducing tees were used where the branch size exceeds 1/2 of the header size. Branch connections of sizes that are equal to or less than 1/2 of the header size are 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. For new piping installations, it is acceptable to use reinforced branch connections exceeding 1/2 of the header size, as long as the design conforms to the requirements of ANSI/USAS B31.1.

All welding was performed by welders and welding procedures qualified in accordance with the ASME Code, Section IX, Welding Qualifications. The Shop Fabricator was required to submit all welding procedures and evidence of qualification for review and approval prior to release for fabrication. All welding materials used by the Shop Fabricator had prior approval.

All high pressure piping butt welds 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 those outlined in UW-51 of the ASME Code, Section VIII, except for the end cap weld joints used in fabricating the strainers for the 3/4" diameter piping branching off of the 3" discharge lines of the safety injection pumps, in which case USAS B31.1 was used as applicable. In addition, butt welds were either liquid penetrant examined in accordance with the procedure of the ASME Code, Section VIII, Appendix VIII (acceptance standard as defined in USAS Nuclear Code Case N-10) or liquid penetrant examined to the requirements and acceptance criteria of ASME Code Section III. Finished branch welds were liquid penetrant examined on the outside and where size permitted, on the inside root surfaces.

A post-bending 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.

6.1.1.1.1.5 Accumulators

The accumulators are carbon steel, clad with stainless steel, and were designed to ASME Section VIII, Division 2 requirements.

6.1.1.1.1.6 Boron injection tank

The boron injection tank was constructed of solid austenitic stainless steel and was designed to ASME Section VIII, Division 2 requirements.

6.1.1.1.1.7 Refueling water storage tank

The refueling water storage tank was constructed of austenitic stainless steel, and conformed to the requirements of American Water Works Association (AWWA) D100-65. The roof of the tank is stainless steel.

6.1.1.1.2 Containment Spray System Components

Containment Spray System components 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 principal components of the Containment Spray System consist of two pumps, one spray additive tank, spray ring headers and nozzles, and the necessary piping and valves. The containment spray pumps and the spray additive tank are located in the Auxiliary Building and the spray pumps take suction directly from the refueling water storage tank. As all of the active components of the Containment Spray System are located outside the containment, they are not required to operate in the steam-air environment produced by a hypothetical accident.

The Containment Spray System also utilizes the two residual heat removal pumps, two residual heat exchangers and associated valves and piping of the SI System for the long-term recirculation phase of containment cooling and iodine removal (refer to Section 6.1.1.1.1).

The containment spray pumps were designed in accordance with the specifications discussed in Section 6.1.1.1.1 for the pumps in the SI System. The materials of construction are stainless steel or equivalent corrosion-resistant material.

The piping for the Containment Spray System was designed in accordance with the specifications discussed for the piping in the SI System (Section 6.1.1.1.4).

Spray nozzles and piping were built to conform to USAS B31.1. Nozzles are constructed of stainless steel.

The valves for the Containment Spray System were designed in accordance with the specifications discussed for the valves in the SI System, and conformed to the criteria of USAS B16.5. Valving descriptions and valve details are shown in Section 6.5.2.

The spray additive tank was constructed of carbon steel clad with austenitic stainless steel, and conformed to the requirements of the ASME code, Section III, Class C.

The Containment Spray System shares the refueling water storage tank liquid capacity with the SI System. Refer to Section 6.1.1.1.1.7 for a description of this tank.

6.1.1.1.3 Containment Air Recirculating System Components

All fan parts, damper shaft, and blade seating surfaces and ducts in contact with the containment fluid are protected against corrosion. The fan motor enclosures, electrical insulation and bearings were designed for operation during accident conditions.

The coils are fabricated of copper plate fins vertically oriented on stainless steel tubes.

Ducts are constructed of corrosion-resistant material. Where flanged joints use gasket, the material is suitable for temperatures to 300ºF.

6.1.1.1.4 Deleted in Revision 20

6.1.1.1.5 Containment Structural Components

As discussed in Section 3.8.1.6, basically eight materials, of which six are metallic, have been used for construction of the containment structure. Metallic materials and components of the containment are as follows:

a) Reinforcing steel
b) Prestressed Steel System
c) Plate steel penetration frame
d) Liner
e) Equipment hatch and personnel lock, and
f) Pipe piles.

Metallic materials used for pipe piles are discussed in Section 3.8.5.

Metallic materials used for reinforcing steel, the prestressed steel system, the plate steel penetration frames, the liner, the equipment hatch, and personnel lock are discussed in Section 3.8.1.6.

6.1.1.1.6 Isolation Valve Seal Water System Components

The Isolation Valve Seal Water System provides a simple and reliable means for injecting seal water between the seats and stem packing of the globe and double disc types of isolation valves, and into the piping between closed diaphragm-type isolated valves (refer to Section 6.8 for the system design description).

The piping and valves for the system, including the air-operated valves, were designed in accordance with the USAS Code for Pressure Piping (Power Piping System), B31.1.

The isolation valve seal water tank was constructed of ASTM A-240, in accordance with the criteria of the ASME Code, Section VIII. The design data for the tank are given in Table 6.8.2-1.

There are no components of this system located inside containment.

6.1.1.1.7 Containment penetration pressurization system components

The Containment Penetration Pressurization System is capable of providing continuous or intermittent positive pressure gradient into the mechanical and cartridge type electrical containment penetrations and the sealing head assembly of CAPSULE type electrical penetrations (refer to Section 6.9.1 for the System Design Description).

The pressurization air receivers are constructed of ASTM A-285-C in accordance with ASME UPV (Section VIII).

The piping and valves for the system were designed in accordance with the USAS Code for Pressure Piping (Power Piping Systems), B31.1.

For a description of the air compressors, refer to Service Air System, Section 9.3.1.

The nitrogen cylinders used were designed in accordance with Section VIII (Unfired Pressure Vessels) of the ASME Code, for 2000 psig maximum pressure, and contain a total of 17,350 scf of nitrogen.

6.1.1.1.8 Nonmetallic thermal insulation

6.1.1.1.8.1 Piping and equipment insulation

Heat insulation specifications for piping and equipment require the use of low leachable chloride insulation, which has been silicate-inhibited against chloride stress corrosion cracking of austenitic stainless steel.

During the construction phase, the insulation material selected for use in containment was Unibestos block and pipe covering. The insulation was weatherproofed with white duck canvas (instead of an aluminum jacket) to minimize the use of aluminum inside containment.

Each lot and batch of insulation was required to pass a stress corrosion test devised by Knolls Atomic Power Laboratory (Reference 6.1.1-2).

An estimated 6400 ft^3 of Unibestos was originally installed inside containment during construction. Approximately 15 percent of this material has since been replaced by Thermon-12 insulation or equivalent, which was the standard for new or replacement Q-List components and piping.

As a result of the 1984 steam generator replacement project, the steam generators were completely reinsulated with new insulation. This new insulation was a combination of both a metallic reflective and a calcium silicate product consisting of approximately 2600 ft^3.

Removable insulation was installed on areas requiring in-service inspection and access openings such as manways and handholes. Metallic reflective insulation was installed on the lower portion of the steam generators from the channel head to just above the upper set of secondary side handholes.

6.1.1.1.8.2 Containment Liner Insulation

The cylindrical portion of the containment liner was insulated to reduce the design temperature to which it would be exposed.

Containment liner insulation consists of 44 in. x 84 in. x 1 1/4 in. thick, 4 lb/ft^3 density cross-linked polyvinyl chloride (PVC) foam and/or 2 lb/ft^3 density Polyimide foam with an outer covering of 0.019 in. thick stainless steel.

6.1.1.2 Composition, Compatibility and Stability of Containment and Core Spray Coolants

An evaluation program led to the selection of sodium hydroxide (NaOH), as the iodine removal additive to the boric acid containment spray. The results of the evaluation program are detailed in Reference 6.1.1-1. NaOH was found to be chemically stable at post-accident containment temperatures, and resistant to oxidation. The NaOH solution was found to be radiolytically stable, with a relatively low net hydrogen liberation rate.

Corrosion rates of copper and copper-alloy heat exchanger tubing were acceptably low (<0.01 mil/month at 200ºF) for the application. These tests showed that pitting or local corrosion did not occur.

The means of adding NaOH to the spray liquid is provided by a liquid jet eductor, a device which uses the kinetic energy of a pressure liquid to entrain another liquid, mixes the two, and discharges the mixture against a counter pressure. The pressure liquid, in this case, is the spray pump discharge which is used to entrain the NaOH solution and discharge the mixture into the suction of the spray pumps. The two eductors were designed to provide enough NaOH in the mixture so as not to exceed a pH of 10 during the injection phase. The design parameters are presented in Table 6.1.1-2.

Analysis has shown that the minimum expected containment sump pH is slightly above 8.0. This was consistent with the Westinghouse recommendation to maintain sump pH between 8.0 and 10.5 for material compatibility. The minimum expected containment spray pH for the plant is 8.8 which is greater than the Westinghouse minimum acceptance criterion of 8.5 and consistent with the current Nuclear Regulatory Commission (NRC) acceptance criteria, Standard Review Plan 6.5.2, which states that the spray solution must have a pH between 8.5 and 11.0.

Engineered Safety Feature (ESF) coolants are stored in the refueling water storage tank, the boron injection tank, the safety injection accumulators and the containment spray additive tank. The materials selection and fabrication requirements for these vessels are described in Section 6.1.1.1.1.

The refueling water storage tank contains a minimum of 300,000 gal of ≥ 1950 and ≤ 2400 ppm borated water available for delivery. The maximum boric acid concentration is approximately 1.4 weight percent boric acid. This concentration of boric acid in the refueling water storage tank is well below the solubility limit at 32ºF (2.2 percent).

The boron injection tank (BIT) functions as a portion of the safety injection flow path and pressure boundary. Design parameters are 2735 psig and 300ºF. (See FSAR Section 6.3.2.2.7 and Table 6.3.2-3.)

The BIT is vertical with the outlet nozzle on top. A level alarm is provided from a stand pipe/vent arrangement on the outlet pipe at an elevation higher than the top of the tank. This alarm assures that the tank is maintained full at all times.

The three SI accumulators contain 1950 AND 2400 ppm boric acid and are pressurized with nitrogen gas to between 600 and 660 psig, at 70 to 120ºF. Design parameters are 700 psig and 300ºF.

The spray additive tank contains a minimum of 2505 gal of ≥ 30 weight percent sodium hydroxide solution which, upon mixing with the refueling water from the refueling water storage tank, the boric acid from the boric acid tank, the borated water contained within the accumulators, and the primary coolant, will bring the concentration of sodium hydroxide in the containment to approximately 0.6 weight percent solution caustic, and 1.7 weight percent boric acid. This maintains a pH of at least 9.3 and assures the continued iodine removal effectiveness of the containment spray during the recirculation phase of operation after the supply of borated water in the refueling water storage tank has been exhausted. The 300 psig design pressure of the tank is the sum of the refueling water storage tank head and the total developed head of the containment spray pumps at shutoff. Vacuum breaker relief valves on the spray additive tank are designed to actuate prior to achieving a 1.5 psid vacuum to insure adequate system performance. A level indicating alarm is provided to alarm in the Control Room if, at any time, the solution tank contains less than the required amount of sodium hydroxide solution. Periodic sampling confirms that proper sodium hydroxide concentration exists in the tank.

The tank design parameters are given in Table 6.1.1-3.

TABLE 6.1.1-1

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TABLE 6.1.1-2
CONTAINMENT SPRAY EDUCTORS DESIGN PARAMETERS

Quantity	2
Eductor Inlet (motive)	Safety Injection and Post Accident Recirculation Phase
Operating Fluid	Water (with □ 1950 and □ 2400 ppm boron)
Operating Pressure, psig	210 (Injection)/325 (Recirculation)
Operating Temperature	Ambient (Injection)/ 200ºF (Recirculation)
Flow Rate, gpm, max.	80
Discharge Head (including static pressure, friction loss, and discharge elevation) psig	0 - 7
Eductor Suction	
Fluid	30 percent NaOH (solution)
Specific Gravity	1.3
Viscosity (design), cp	10
Suction Pressure, psig	1 to 10
Operating Temperature	Ambient
Suction Capacity (required), gpm	12

TABLE 6.1.1-3

SPRAY ADDITIVE TANK DESIGN PARAMETERS

Number	1
Total Volume (empty), gal	5100
Minimum Volume at Operating Conditions (solution), gal	2505
NaOH Concentration, percent	30
Design Temperature, ºF	300
Design Pressure, psig	300
Design Vacuum, psi	2
Material	Carbon Steel with Austenitic
Stainless Steel Clad	

6.1.2 ORGANIC MATERIALS

Significant quantities of organic material within the Containment Building include containment liner insulation, piping, and equipment insulation, electrical insulation, lubricants, and protective coatings.

The quantity and identity of materials selected for containment liner insulation are described in Section 6.1.1.1.8.2. Materials selected for use as piping and equipment insulation are described in Section 6.1.1.1.8.1.

The ability of electrical equipment in the ESF Systems to withstand radiation exposure would be limited by radiation effects on electrical insulation materials and motor bearing lubrication.

The electrical equipment for the ECCS located in the containment utilizes only inorganic, silicone, and epoxy plastic insulating materials. These materials have a threshold for radiation damage which provides considerable margin above the maximum post-accident radiation dose that would result from the exposure levels and times described in Section 3.11.

The fan cooler motors of the Containment Air Recirculation System contain Class B Thermalastic insulation (NEMA rated total temperature 130ºC). The insulation was impregnated and coated to give a homogeneous insulation system which is highly impervious to moisture.

Where required, because of location in possible high radiation areas, ECCS motor bearings will be lubricated with radiation-rated lubricants.

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

The results of the protective coatings evaluation (Reference 6.1.2-1) showed that several inorganic zincs, modified phenolics and epoxy coatings are resistant to an environment of high temperature (320ºF maximum test temperature) and alkaline sodium borate. Long-term tests included exposure to spray solution at 150 - 175ºF for 60 days, after initially being subjected to the design basis accident (DBA) cycle. Similar tests were conducted at the National Reactor Testing Station at Idaho Falls, Idaho (Reference 6.1.2-2).

The protective coatings, which were found to be resistant to the test conditions, that is, exhibited no significant loss of adhesion to the substrate nor formation of deterioration products, comprise virtually all of the protective coatings exposed in the Duke Energy Progress, LLC containment. Hence, the protective coatings will not add deleterious products to the core cooling solution. Most coatings on carbon steel surfaces exposed to direct impingement by DBA Spray are Carbozinc 11 (an inorganic zinc primer), Phenoline 305 (a modified phenolic epoxy topcoat), or equivalently qualified coatings systems. Other carbon steel surfaces are protected from direct impingement of the spray. For example, the containment vessel liner surface, up to el 367'-10", is protected by the liner insulation and is not exposed to

the DBA spray. The original coating in this area, up to el. 352', is the Keeler & Long 7230 System. It should be noted further, however, that this coating system, while exhibiting blisters in the conservative test environment (Reference 6.1.2-1) did not fail to the extent that significant deterioration products were released from the surface. Coatings systems that are evaluated as equal to or better than the original coatings are utilied for maintenance coating.

The concrete surfaces which have been coated are coated with Phenoline 305 or Carboline 195 Surfacer and Phenoline 305, or equivalently qualified coating systems. Carboline 195 Surfacer is a product generally identical (modified epoxy-polyomide) with protective coating which has been shown to be completely resistant to the DBA environment.

It should be pointed out that several test panels of the types of protective coatings used at HBR 2 were exposed for two DBA cycles and showed no deterioration or loss of adhesion with the substrate.

Some original and replacement components installed in Containment (e.g. light fixtures, gages, fire extinguishers, small pumps, motors, and electrical boxes) have coatings that are not specifically proven to be a DBA resistant type. These coatings have been evaluated for potential debris transport and impact on the ECCS.

Selected equipment and tools stored in Containment which have protective coatings, either have coatings of a DBA resistant type, or they are encapsulated to preclude release of coating debris into the ECCS sump, or they have been evaluated for potential debris transport and impact on the ECCS.

Evaluations of potential coatings debris transport and impact on the ECCS have determined that Net Positive Suction Head (NPSH) requirements for the ECCS are not adversely impacted.

REFERENCES: Section 6.1

6.1.1-1 WCAP-7153, "Investigations of Chemical Additives for Reactor Containment Sprays," M. J. Bell, et al., March 1968 (W Confidential).

6.1.1-2 Karnes, H. F., "The Corrosive Potential of Wetted Thermal Insulation," presented at the AICHE 57th National Meeting, September 25-29, 1965; Conf 650905-2; NSA 19:49949.

6.1.2-1 Picone, L. F., "Evaluation of Protective Coatings for Use in Reactor Containment," WCAP-7198-L, April 1968.

6.1.2-2 Newby, B. J., et al., "Development of Testing Procedures for Protective Coatings to be Used in Nuclear Reactor Contaiment Structures," IN 1253, February 1969.

6.2 CONTAINMENT SYSTEMS

6.2.1 CONTAINMENT FUNCTIONAL DESIGN

6.2.1.1 Containment Structure

6.2.1.1.1 Design Basis

The reactor containment completely encloses the entire reactor and Reactor Coolant System (RCS) and ensures that an acceptable upper limit for leakage of radioactive materials to the environment will not be exceeded even if gross failure of the RCS were to occur.

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

The containment structure was designed to provide biological shielding for both normal and accident situations and limit the amount of radioactivity released to the environment to within applicable limits.

The design pressure and temperature of the containment exceed the peak pressure and temperature occurring as the result of the complete blowdown of the reactor coolant through any rupture of the RCS up to and including the hypothetical severance of a reactor coolant pipe.

The design pressure will not be 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 the engineered safety features (ESF) utilizing only the emergency onsite electric power supply.

The pressure and temperature loadings obtained by analyzing various LOCA, 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.

The containment system was designed to the requirements of 10CFR50, Appendix A, General Design Criteria. A detailed discussion of the implementation of these requirements in the design of H. B. Robinson, Unit 2, is presented in Section 3.1.

6.2.1.1.1.1 Postulated Accident Conditions - LOCA

The design and licensing of nuclear power plants require that the containment be analyzed for pressure and temperature effects. The analyses include pressure and temperature transients to which the containment might be exposed as a result of postulated pipe breaks. Containment integrity analyses are performed on dry containment designs to quantify the margin in the containment design pressure and peak temperature for equipment environmental qualification (EQ), and to

demonstrate the acceptability of the containment safeguards equipment to mitigate the postulated transient. Subcompartment analysis is performed to demonstrate the integrity of containment internal structures when subjected to dynamic, localized pressurization effects that could occur during the very early time period following a design basis event.

This section presents the analysis of the mass & energy releases for postulated Loss-of-Coolant (LOCA) accidents, and the corresponding containment integrity and subcompartment analyses. Also included in this section is a discussion of the input parameters and assumptions, methodology, analyses, acceptance criteria, and the results.

6.2.1.1.1.2 Long-Term LOCA Mass and Energy Releases

6.2.1.1.1.2.1 Introduction

Discussed in this section are the long-term LOCA mass and energy (i.e., M&E) releases for the hypothetical double-ended pump suction (DEPS) and double-ended hot leg (DEHL) break cases. The mass and energy release rates described in this section form the basis of further computations to evaluate the containment response following the postulated LOCA (Section 6.2.1.1.3).

A total of three LOCA mass and energy release cases were analyzed. These cases addressed two different break locations, the double-ended hot leg break and the double-ended pump suction break (see Section 6.2.1.1.1.2.3, "Break Size & Location," for a detailed explanation). The above two break locations were analyzed for both minimum and maximum safeguards (i.e. minimum and maximum pumped ECCS flows). The minimum ECCS cases were performed to address maximum available steam release (minimizing steam condensation) and the maximum ECCS cases were performed to address the effects of maximizing mass flow and subsequent effect on containment response. Reference 6.2.1- 1 has provided justification that these analyses encompass the most limiting assumptions for break location and safeguards operation.

The limiting long-term LOCA mass and energy releases are extended out in time to approximately 1 million seconds and are utilized as input to the containment response analysis, which demonstrates the acceptability of the containment design, EQ limits, and containment safeguards systems to mitigate the consequences of a hypothetical large break LOCA. The containment safeguards systems must be capable of limiting the peak containment pressure to less than the design pressure and to limit the temperature and pressure excursion to below the Environmental Qualification (EQ) limits.

6.2.1.1.1.2.2 Input Parameters and Assumptions

The mass and energy release analysis is sensitive to the assumed characteristics of various plant systems; some of the most-critical items are the RCS initial conditions, core decay heat, accumulators, ECCS flow, and primary and secondary metal mass and steam generator heat

release modeling. Specific assumptions concerning each of these items are discussed in this section. Tables 6.2.1-1 through 6.2.1-3 present key data assumed in the analysis. All input parameters are determined based on accepted methodology (Reference 6.2.1- 1).

Initial Power Level

The initial power level is assumed to be 2346 MWt, which is 100.3% of the rated thermal power (2339 MWt) (adjusted for a calorimetric error of 0.3%) for HBRSEP, Unit No.2. A maximum initial power is conservative for maximizing the mass and energy releases, with respect to reactor coolant system (RCS) temperature, available decay heat energy and initial core stored energy.

Initial RCS Temperature and Pressure

Initial RCS temperatures are chosen to bound the highest average coolant temperature range of all operating cases. The initial T_{HOT} (vessel outlet temperature) of 610.3ºF and initial T_{COLD} (core inlet temperature) of 548.5ºF (which includes +4.0ºF for instrument error and deadband, Reference 6.2.1- 12) were modeled. The use of the higher temperatures is conservative because the initial fluid energy is based on coolant temperatures, which are at the maximum levels attained in steady state operation at 2339 MWt including calorimetric uncertainty. This position on RCS temperatures was originally established in Reference 6.2.1- 7. The RCS pressure is based upon a nominal value of 2250 psia plus an allowance (+30 psi, Reference 6.2.1- 12), which accounts for the measurement uncertainty on pressurizer pressure. This assumption only affects the blowdown phase results. 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. (Note: The RCS initial temperatures were conservatively based upon Steam Generator Tube Plugging (SGTP) level of 0%).

Steam Generator Model

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

Secondary to primary heat transfer is maximized by assuming conservative coefficients of heat transfer (i.e., steam generator primary/secondary heat transfer and reactor coolant system metal heat transfer). Maximum

secondary to primary heat transfer is ensured by maximizing the initial steam generator mass based upon 100% power conditions and then increasing this by 10% to maximize the available energy. The 10% uncertainty addresses uncertainties in SG secondary side volume calculations, and several sources of level measurement errors.

Fuel Design - Core Stored Energy

Core stored energy is the amount of energy in the fuel rods above the local coolant temperature. The selection of the fuel design features for the long-term mass and energy release calculation are based on the need to conservatively maximize the energy stored in the fuel at the beginning of the postulated accident. The following fuel features are considered, 1) Rod Geometry, 2) Rod Power, and 3) Limiting time in life (e.g. Burnup). The Core Stored Energy supplied in Reference 6.2.1- 12 was used in this analysis. Core stored energy is addressed in the analysis as full power seconds.

Core Decay Heat Model

The Nuclear Power Plant Standards Committee (NUPPSCO) of the American Nuclear Society (ANS) approved ANS Standard 5.1 (Reference 6.2.1-2) for the determination of decay heat. This standard was used in the mass and energy release model with the input described below.

Significant assumptions in the generation of the decay heat curve for use in design basis containment integrity LOCA analyses include:

1. Decay heat sources considered are fission product decay and heavy element decay of U-239 and Np-239.

2. Decay heat power from the following fissioning isotopes are included: U-238, U-235, and Pu-239.

3. Fission rate is constant over the operating history of maximum power level.

4. The factor accounting for neutron capture in fission products has been taken from Table 10, of Reference 6.2.1- 3.

5. The fuel has been assumed to be at full power consistent with a cycle average burnup of 42,000 MWD/MTU.

6. The minimum average enrichment is assumed to be 3.7%, and the core fuel loading is assumed to be 68 MTU.

7. The total recoverable energy associated with one fission has been assumed to be 200 MeV/fission.

8. Two-sigma uncertainty (two times the standard deviation) has been applied to the fission product decay.

Based upon NRC staff review, the Safety Evaluation Report (SER) of the March 1979 evaluation model (Reference 6.2.1- 2), the use of the ANS Standard-5.1, August 1979 decay

heat model was approved for the calculation of mass and energy releases to the containment following a loss-of-coolant accident. Table 6.2.1.2-19 provides the Decay Heat Curve.
In 1996, the NRC issued an information notice (Reference 6.2.1- 3) regarding the use of the ANS 5.1 decay heat standard. The following items address that information notice:

1. The comparisons presented in the information notice are for Peak Cladding Temperature only. Even though decay effects are illustrated, there is no mention of LOCA Mass and Energy Releases and Containment Response calculations. However, there is the implied impact on any analysis that has utilized the ANS standard.

2. For LOCA mass and energy, the current methodology (WCAP-10325-P-A) (Reference 6.2.1- 1) utilizes the ANS Standard 5.1 for the determination of the decay heat. The input utilized is called out on page 2-10 of the WCAP. The model, including the decay heat model, has been approved (letter from C. E. Rossi of NRC to W. J. Johnson of Westinghouse, dated 2/17/87, which is included with Reference 6.2.1- 1.)

3. For LOCA mass and energy, the ANS 5.1 standard is used in the selection of inputs. Power history, initial fuel enrichment, and neutron flux level, which are called out in the information notice, are also called out in Reference 6.2.1- 1

Reactor Coolant System Fluid Energy

Margin in RCS fluid volume of 3% (which is composed of 1.6% allowance for thermal expansion and 1.4% for uncertainty) is modeled. These uncertainties were originally introduced into the Reference 6.2.1- 6 methodology that was accepted by the NRC.

Application of Single-Failure Criterion

An analysis of the effects of the single-failure criterion has been performed on the mass and energy release rates for each break analyzed. An inherent assumption in the generation of the mass and energy release is that offsite power is lost. This results in the actuation of the emergency diesel generators, which are required to power the emergency core cooling system (ECCS). Actuation of the Emergency Diesel Generators results in a delay in the time to start both the ECCS and containment safeguards. A delay in the actuation of these accident mitigating components results in a higher containment pressure and temperature for the postulated LOCA. Since the M&E codes (Reference 6.2.1- 1) are uncoupled from the Containment Pressure code (Reference 6.2.1- 7) an assumption on containment pressure is required in the Reference 6.2.1- 1 M&E calculations. Maximum containment backpressure equal to the design pressure is modeled, which reduces the rate of safety injection, condensation of steam by the safety injection, and extends the reflood phase, which maximizes the steam release.

Two single failures have been analyzed: The first postulates the single failure of an emergency diesel generator. This is conservatively assumed to result in the loss of one train of safeguards equipment, which is modeled as: 1 High Head Safety Injection (HHSI) and 1 Low Head Safety Injection (LHSI) pump (Minimum Safeguards). The loss of a diesel generator minimizes ECCS flow and therefore the condensation of steam, increasing the energy release to the containment. The second single failure assumption postulates failure of 1 containment spray pump, resulting in all ECCS equipment operating. This case, referred to as maximum safety injection, maximizes the mass release to containment but also results in more containment heat removal equipment being available. This case considers 2 HHSI and 2 LHSI Pumps (Maximum Safeguards). These two postulated single failures cover the range on possible single failures with regard to the affect on mass and energy releases and containment safeguards availability.

Safety Injection System

Following a Large Break Loss of Coolant Accident (LBLOCA) inside containment, the safety injection system, (SIS) operates to reflood the reactor coolant system. The first phase of the SIS operation is the passive accumulator injection. Three accumulators are assumed available to inject. When the RCS depressurizes to 615 psia (Reference 6.2.1-11) the accumulators begin to inject into the cold legs at the reactor coolant loops. The accumulator injection temperature was modeled at 130ºF (Reference 6.2.1-11). The Sequence of Events tables presented in Section 6.2.1.1.3 provide the actuation times for the accumulators for each case.

The active pumped ECCS operation of the SIS was modeled to address both minimum and maximum safeguards (minimum ECCS and maximum ECCS). The minimum ECCS flow is addressed to calculate the effect on minimizing steam water mixing/steam condensation. The maximum ECCS case addresses the effects of maximizing mass flow out the postulate RCS piping break. The SI signal is assumed to be actuated on the low pressurizer pressure setpoint of 1661.4 psia (Reference 6.2.1-11). For the maximum ECCS case, the SIS was assumed to deliver to the RCS without delay after the generation of this signal where the intent was to maximize mass flow. For the minimum ECCS case, the SIS was assumed to deliver to the RCS 41.7 seconds (Reference 6.2.1-11) after the generation of the SI signal. The ECCS flow is delivered as a function of RCS pressure. The pumped ECCS temperature for the injection phase was assumed to be at 100ºF (Reference 6.2.1-11). In the determination of long term containment pressure and temperature transients, credit is taken for cold leg pumped sump recirculation ECCS flow to the core and sump heat removal via the residual heat exchangers (RHR Heat Exchangers). For the minimum ECCS case during recirculation, (failure of 1 ESF train) 1 HHSI is available. The ECCS configuration for the recirculation phase maximum ECCS case is 2 HHSI. Tables 6.2.1-2 and 6.2.1-3 provide the pumped ECCS flows as a function of RCS pressure for the minimum and maximum safeguards case, respectively. The Sequence of Events tables presented in Section 6.2.1.3 provide the actuation times for the pumped ECCS flow for each case.

6.2.1.1.1.2.3 Description of Analyses

The evaluation model used for the long-term LOCA mass and energy release calculations is the March 1979 model described in Reference 6.2.1- 1. This evaluation model has been reviewed and generically approved. The approval letter is included with Reference 6.2.1- 1. A description of the Reference 6.2.1- 1 methodology is provided below.

Mass and Energy Release Phases

The LOCA mass and energy analysis is typically divided into four phases: blowdown, refill, reflood, and post-reflood. Each of these phases is analyzed by the following codes: blowdown - SATAN-VI; refill/reflood - WREFLOOD; and post-reflood -FROTH and EPITOME

The phases and codes are discussed below. The first phase of a LOCA mass and energy release transient is the blowdown phase, 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 pressure. The blowdown period is typically < 30 seconds. It ends when the RCS active core area is essentially empty, which is within seconds of ECCS injection actuation for the minimum safeguards (Min ECCS) case. For the maximum safeguards case (Max ECCS), ECCS injection is credited after SI signal is reached w/o a delay as noted above in order to maximize the mass flow.

A mass and energy release version of the SATAN-VI code is used for computing the blowdown transient. The code utilizes the control volume (element) approach with the capability for modeling a large variety of 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 6.2.1- 1.

The refill period is the second phase of the LOCA mass and energy release transient. It is the period of time when the lower plenum is being filled by accumulator and pumped 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.

The third phase of a LOCA mass and energy release transient is the core reflooding phase, which begins when the primary coolant system has depressurized (following blowdown) due to the loss of water through the
break. The water from the lower plenum, supplied by the ECCS refills the

reactor vessel and provides cooling to the core. This phase ends when the core is completely quenched. The model conservatively assumes quenching of the core at the 10-foot elevation on the active fuel for containment functional design calculations. During this phase, decay heat generation will produce boiling in the core resulting in a two-phase mixture of steam and water in the core. This two-phase mixture rises above the core and subsequently enters the steam generators. The most-important feature is the steam/water mixing model (described below), which is used during this phase.

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 ECCS and accumulators, reactor coolant pump performance, and steam generator release, 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; i.e., the path through the broken loop and the path through the unbroken loop.

A complete thermal equilibrium mixing condition for the steam and emergency core cooling 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 6.2.1- 1 mass and energy release evaluation model in recent analyses (Reference 6.2.1- 4). Even though the Reference 6.2.1- 1 model credits steam/mixing only in the intact loop and not in the broken loop, justification, applicability, and approval for using the mixing model in the broken loop has been documented (Reference 6.2.1- 4). This assumption is justified and supported by test data, and is summarized as follows.

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 has been reviewed for validation of the containment integrity reflood steam/water mixing model. This data was generated in 1/3-scale tests (Reference 6.2.1- 5), which are the largest scale data available, and thus most-clearly simulates the flow regimes and gravitational effects that would occur in

a PWR. These tests were designed specifically to study the steam/water interaction for PWR reflood conditions.

From the entire series of 1/3-scale tests, a group corresponds almost 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 6.2.1- 1. For all of these tests, the data clearly indicates 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 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, and the other is via reverse flow through the reactor coolant pump. Steam that is not condensed by ECCS injection in the intact RCS loop 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 is contained in References 6.2.1-1 and 6.2.1-5.

Post-reflood describes the period following the reflood transient. For the pump suction break, a two-phase mixture exits the core, passes through the hot legs, is superheated in the steam generators, and exits the break as superheated steam. After the broken loop steam generator cools, the break flow becomes two phase.

The FROTH code (Reference 6.2.1-6) is used for computing the post-reflood transient. The FROTH code calculates the heat release rates resulting from a two-phase mixture level 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, but 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. During the FROTH calculation ECCS injection is addressed for both the injection phase and the recirculation phase.

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 at the saturation

temperature (T_{sat}) at the containment design pressure. After the FROTH calculations, steam generator secondary energy is removed based on first and second stage rates. The first stage rate is applied during the time interval from the broken loop equilibrium at containment design pressure to the estimated intermediate pressure. While stage 2 is the time interval from the estimated intermediate pressure equilibrium out to an SG pressure of 14.7 at 3600 seconds. These rates are applied simultaneously in the transient until the desired depressurization is achieved for each steam generator, which may occur over differing periods of time and rates for each SG. The EPITOME code continues the FROTH calculation for SG cooldown. 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 212ºF. The heat removal of the broken loop and intact loop steam generators are calculated separately.

The Sequence of Events tables 6.2.1-25 through 6.2.1-27 provide the case specific broken and intact loop steam generator equilibration times. By reading the output files from SATAN VI, WREFLOOD, and FROTH, the EPITOME code 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.

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 McAdam's 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 SG 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. 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 212ºF. 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 212ºF. The required depressurization to 14.7 psia at 3600 seconds was arrived at in licensing of the Reference 6.2.1- 1 model.

Metal energy that is considered “inactive” due to a location that prevents cooling by water and thus can only be cooled slowly by steam is assumed to be released slowly. These regions are the upper region of the steam generator secondary side, reactor coolant system (RCS) pressurizer and the reactor vessel head. The steam generator upper region metal and pressurizer metal are assumed to release energy over 24 hours and the reactor head metal has an assumed release period of 7 hours. 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 boiloff/decay heat.

Computer Codes

The Reference 6.2.1- 1 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.

SATAN VI calculates blowdown, the first portion of the thermal-hydraulic transient for the RCS 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 during the core reflood phase. FROTH models the post-reflood portion of the transient. The FROTH code is used for the steam generator heat addition calculation from the broken and intact loop steam generators. 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.

Break Size and Location

Generic studies (Reference 6.2.1- 6) have been performed with respect to 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 post-reflood phases, the break size has little effect on the releases.

Three distinct locations in the reactor coolant system loop can be postulated for pipe rupture:

1. Hot leg (between reactor vessel and steam generator)
2. Cold leg (between Reactor Coolant Pump and the reactor vessel)

3. Pump suction (between steam generator and Reactor Coolant Pump)

The DEHL rupture has been shown in previous studies to result in the highest blowdown mass and energy release rates. 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 bypasses the steam generators venting directly to containment. 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. For the hot leg break, generic studies have confirmed that there is no reflood peak (i.e., from the end of the blowdown period the containment pressure continually decreases). Therefore only the mass and energy releases for the hot leg break blowdown phase are calculated and presented in this section of the report.

The cold leg break location has been found in previous studies to be much less limiting in terms of the overall containment energy releases. 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 (due to the break location the flow will bypass the normal path through the core and go through the path of least resistance to the broken loop) and this results in a considerably lower energy release into containment. Studies have determined that the blowdown transient for the cold leg is less limiting than that for the pump suction and hot leg breaks. During reflood, the flooding rate is greatly reduced because all the core vent paths include the resistance of the reactor coolant pump, in addition to ECCS injection spill, thus the energy release rate into the containment is reduced. Therefore, the cold leg break is not included in the scope of this analysis.

The pump suction break combines the effects of the relatively high core flooding rate, as in the hot leg break, with 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 Reactor Coolant System and secondary side in calculating the releases to containment.

The break locations analyzed for this program are the double-ended pump suction (DEPS) rupture (10.48 ft^2), and the double-ended hot leg (DEHL) rupture (9.18 ft^2). Break mass and energy releases have been calculated for the blowdown, reflood, and post-reflood phases of the LOCA for the DEPS cases.

Sources of Mass and Energy

The sources of mass considered in the LOCA mass and energy release analysis are given in Table 6.2.1-11. These sources are the reactor coolant system, accumulators, and pumped safety injection.

The energy inventories considered in the LOCA mass and energy release analysis are given in Table 6.2.1-12. The energy sources include:

1. Reactor Coolant System Water
2. Accumulator Water (all inject)
3. Pumped Injection Water (RWST/ECCS)
4. Decay Heat
5. Core Stored Energy
6. Reactor Coolant System Metal - Primary Metal (includes SG tubes)
7. Steam Generator Metal (includes transition cone, shell, wrapper, and other internals)
8. Steam Generator Secondary Energy (includes fluid mass and steam mass)
9. Secondary Transfer of Energy (feedwater into and steam out of the steam generator secondary)

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

1. Time zero (initial conditions)
2. End of blowdown time
3. End of refill time
4. End of reflood time
5. Time of broken loop steam generator equilibration to pressure setpoint
6. Time of intact loop steam generator equilibration to pressure setpoint
7. Time of full depressurization (3600 seconds)

Energy Reference Points

Available Energy: 212ºF; 14.7 psia

(The current approved methodology assumes that all energies in the system are taken out to these conditions in the first hour of the event. This is the total available energy.)

Total Energy Content: 32ºF; 14.7 psia

(This is the reference point for the system energy.)

In the mass and energy release data presented, no Zirc-water reaction heat was considered because the clad temperature is assumed not to rise high enough for the rate of the Zirc-water reaction heat to be of any significance. This is a feature of the Reference 6.2.1- 1 methodology based on Peak Cladding Temperature (PCT) analyses using the models of Appendix K to 10CFR50, to meet the criteria specified in 10CFR50.46. These PCT analyses show that less than 1.0% of the total core Zirconium is reacted during the hypothetical LOCA. Thus, the energy release from the Zirconium water reaction would be small and would not significantly affect the mass and energy releases to containment.

6.2.1.1.1.2.4 Acceptance Criteria

A large break loss-of-coolant accident is classified as an ANS Condition IV event, an infrequent fault. To satisfy the Nuclear Regulatory Commission acceptance criteria, the relevant requirements are as follows:

A. HBR2 UFSAR Chapter 3.1 General Design Criteria; as it relates to General Design Criteria 10, 49, and 52, with respect to containment design integrity and containment heat removal.

B. 10 CFR 50, Appendix K, paragraph I.A: as it relates to sources of energy during the LOCA, provides requirements to assure that all energy sources have been considered.

In order to meet these requirements, the following must be addressed.

1. Sources of Energy
2. Break Size and Location
3. Calculation of Each Phase of the Accident
4. Single Failure Criteria

Each of these items is addressed in Section 6.2.1.1.1.2.

6.2.1.1.1.2.5 Results

Using the Reference 6.2.1- 1 methodology, the mass and energy release rates were developed to determine the containment pressure and temperature responses for each of the LOCA cases noted in Section 6.2.1.1.1.2.4. The LOCA mass and energy releases discussed in this section provide the basis for the containment response analysis provided in Section 6.2.1.1.3.

Table 6.2.1-7 present the calculated mass and energy releases for the blowdown phase of the DEPS break for the minimum safeguards case. 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; break path 2 refers to the mass and energy exiting from the pump side of the break.

Table 6.2.1-8 presents the calculated mass and energy release for the reflood phase of the pump suction double-ended rupture, diesel failure (minimum safeguards)

The transients of the principal parameters, such as core flooding rate, core and downcomer level, and safety injection and accumulator injection rates during the core reflooding portion of the LOCA are given in Table 6.2.1-9 for the DEPS case.

Table 6.2.1-10 presents the two-phase post-reflood mass and energy release data for the pump suction double-ended cases.

The sequence of events for the LOCA transient is included in
Table 6.2.1-25.

6.2.1.1.1.2.6 Conclusions

The consideration of the various energy sources in the long-term mass and energy release analysis provides assurance that all available sources of energy have been included in this analysis. Thus, the review guidelines presented in Standard Review Plan Section 6.2.1.3 have been satisfied. Any other conclusions cannot be drawn from the generation of mass and energy releases directly since the releases are inputs to the containment integrity analyses. The containment response must be performed (as documented in Section 6.2.1.1.3).

6.2.1.1.1.2.7 ESF Systems Impact on Energy Removal and Pressure Reduction

Provision was made in the computer analysis for the effects of several engineered safeguards, including internal spray, fan coolers, and recirculation of sump water. The heat removal from containment steam-air phase by internal spray is determined by allowing the spray water temperature to rise to the steam-air temperature.

In the transients one spray pump and two fans starting at 60 sec were assumed. These acted to quickly reduce the pressure after the peak pressures were reached. This is the minimum equipment available considering the single failure criterion in the emergency power system, the spray system, and the fan cooler system. The ability of the fan coolers to limit containment pressure following loss of the component cooling system was examined. If the component cooling loop were lost for any reason during long-term recirculation, core subcooling could be

lost and boiling in the core would begin. Since the fan cooling units are cooled by service water, the energy from the core would be removed from the containment via the fans. The following table summarizes the maximum pressure the containment could reach for assumed times of component cooling system failure.

	3 FANS	2 FANS
C.C. Failure at 12 hr	9.5	27
C.C. Failure at 1 day	7.0	16
C.C. Failure at 1 week	2.0	4.5

The containment heat removal capability started at 60 sec exceeds the energy addition rate and the pressure does not exceed the initial blowdown value. An extended depressurization time results due to the increased heat load on the containment coolers.

The time dependent behavior of containment internal pressure resulting from a LOCA is shown in Figure 3.8.1-29. The loads resulting from the design pressure are shown in Figure 3.8.1-30.

The containment structure is designed to contain the radioactive material that might be released from the core following a LOCA at a leak rate no greater than 0.1 percent of the containment free volume per day at design pressure.

The maximum allowable differential pressure loading from an internal negative pressure is 3.0 psig.

The maximum differential that could occur with 75 percent humidity would be approximately 2.96 psig which is less than the maximum allowable of 3.0 psig. Following an inadvertent initiation of Containment Spray, manual operator action will terminate the event. An alarm will inform the Control Room Operator of the negative Containment pressure.

6.2.1.1.2 Design Features

The design features of the containment and internal structure are described in Sections 3.8.1 and 3.8.3, respectively.

The containment structure, subcompartments, and ESF systems are protected from loss of safety function due to dynamic effects that could occur following postulated accidents. The detailed criteria, locations, and description of protective devices are presented in Sections 3.5 and 3.6.

Codes and standards applied to the design, fabrication, and construction of the containment and internal structure are given in Section 3.8.1.2.

No special design features to mitigate the effects of external pressure loads are required. A Control Room low pressure alarm at 0.4 psi

negative pressure has been incorporated in the containment functional design. However, inadvertent operation of the Containment Heat Removal Systems (CHRS) cannot possibly exceed the negative loading maximum allowable pressure differential. Refer to Section 6.2.1.1.1.2.7 for details.

The equipment and floor drainage system inside containment is described in Section 9.3.3.

Containment cooling and ventilation systems which maintain the containment and subcompartment atmospheres within prescribed pressure, temperature, and humidity during normal operation are described fully in Section 9.4.3.

6.2.1.1.3 Long Term LOCA Containment Response Analysis

With the exception of the Double Ended Pump Suction (DEPS) case with minimum safeguards, the LOCA containment response has been calculated with the COCO code. The containment response for the DEPS case, with minimum safeguards, has been calculated with the GOTHIC code.

6.2.1.1.3.1 Accident Description

The containment system is designed such that for all loss-of-coolant accident (LOCA) break sizes, up to and including the double-ended severance of a reactor coolant pipe, the containment peak pressure remains below the design pressure. This section details the containment response subsequent to a hypothetical LOCA. The containment response analysis uses the long term mass and energy release data from Section 6.2.1.1.1.2

The containment response analysis demonstrates the acceptability of the containment safeguards systems to mitigate the consequences of a LOCA inside containment. The impact of LOCA mass and energy releases on the containment pressure is addressed to assure that the containment pressure remains below its design pressure at the licensed core power conditions. In support of equipment design and licensing criteria (e.g. qualified operating life), with respect to post accident environmental conditions, long term containment pressure and temperature transients are generated to conservatively bound the potential post-LOCA containment conditions.

6.2.1.1.3.2 Input Parameters and Assumptions

An analysis of containment response to the rupture of the RCS must start with knowledge of the initial conditions in the containment. The pressure, temperature, and humidity of the containment atmosphere prior to the postulated accident are specified in the analysis as shown in Table 6.2.1-20. Also, values for the initial temperature of the service water (SW) and refueling water storage tank (RWST) are assumed, along with containment spray (CS) pump flow rate and containment fan cooler (CFC) heat removal performance. All of these values are chosen conservatively, as shown in Table 6.2.1-20. Long term sump recirculation is addressed via Residual Heat Removal System (RHR) heat exchanger performance. The primary function of the RHR system is to remove heat from the core by way of Emergency Core Cooling System (ECCS). Table 6.2.1-20 provides the RHR system parameters assumed in the analysis.

A series of cases was performed for the LOCA containment response. Section 6.2.1.1.1 documented the M&E releases for the minimum and maximum safeguards cases for a DEPS break and the releases from the blowdown of a DEHL break.

For the maximum safeguards DEPS case a failure of a containment spray pump was assumed as the single failure, which leaves available as active heat removal systems, one containment spray pump and four CFCs. Table 6.2.1-22 provides the performance data for one spray pump in operation. (Note: For the Maximum safeguards case a limiting assumption was made concerning the modeling of the recirculation system, i.e., heat exchangers. Minimum safeguards data was conservatively used to model the RHR heat exchangers, i.e., one RHR Heat Exchanger was credited for residual heat removal. Emergency safeguards equipment data is given in Table 6.2.1-20.)

The minimum safeguards case was based upon a diesel train failure (which leaves available as active heat removal systems one containment spray pump and 2 CFCs). Due to the duration of the DEHL transient (i.e. blowdown only), no containment safeguards equipment is modeled.

The calculations for all of the DEPS cases were performed for at least 1.0E5 seconds (approximately 1.16 days). The DEHL cases were terminated soon after the end of the blowdown. The sequence of events for each of these cases is shown in Tables 6.2.1-25 through 6.2.1-27.

The following are assumptions made in the analysis.

(a) The mass and energy released to the containment are described in Section 6.2.1.1.1 for LOCA.

(b) 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.

(c) Air is taken as an ideal gas, while compressed water and steam tables are employed for water and steam thermodynamic properties.

(d) 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. For the post-blowdown portion of the LOCA analysis, steam and water releases are input separately.

(e) The saturation temperature at the partial pressure of the steam is used for heat transfer to the heat sinks and the fan coolers.

6.2.1.1.3.3 Description of COCO Model

With the exception of the Double Ended Pump Suction (DEPS) case with minimum safeguards, the LOCA containment response has been calculated with the COCO code.

Calculation of containment pressure and temperature is accomplished by use of the digital computer code COCO (Reference 6.2.1- 7). COCO is a mathematical model of a generalized containment; the proper selection of various options in the code allows the creation of a specific model for particular containment design. The values used in the specific model for different aspects of the containment are derived from plant-specific input data. The COCO code has been used and found acceptable to calculate containment pressure transients

for many dry containment plants. Transient phenomena within the reactor coolant system affect containment conditions by means of convective mass and energy transport through the pipe break.

For analytical rigor and convenience, the containment air-steam-water mixture is separated into a water (pool) phase and a steam-air phase. Sufficient relationships to describe the transient are provided by the equations of conservation of mass and energy as applied to each system, together with appropriate boundary conditions. As thermodynamic equations of state and conditions may vary during the transient, the equations have been derived for all possible cases of superheated or saturated steam and subcooled or saturated water. Switching between states is handled automatically by the code.

Passive Heat Removal

The significant heat removal source during the early portion of the transient is the containment structural heat sinks. Provision is made in the containment pressure response analysis for heat transfer through, and heat storage in, both interior and exterior walls. Every wall is divided into a large number of nodes. For each node, a conservation of energy equation expressed in finite-difference form accounts for heat conduction into and out of the node and temperature rise of the node. Table 6.2.1-23 is the summary of the containment structural heat sinks used in the analysis. The thermal properties of each heat sink material are shown in Table 6.2.1-24.

The heat transfer coefficient to the containment structure for the early part of the event is calculated based primarily on the work of Tagami (Reference 6.2.1- 8). From this work, it was determined that the value of the heat transfer coefficient can be assumed to increase parabolically to a peak value. In COCO, the value then decreases exponentially to a stagnant heat transfer coefficient that is a function of steam-to-air-weight ratio.

The h for stagnant conditions is based upon Tagami's steady state results. Tagami presents a plot of the maximum value of the heat transfer coefficient, h, as function of "coolant energy transfer speed", defined as follows:

$$h = \frac{\text{total coolant energy transferred into containment}}{\text{(containment volume)(time interval to peak pressure)}}$$

From this, the maximum heat transfer coefficient of steel is calculated:

$$h_{max} = 75 \left(\frac{E}{t_p \ V} \right)^{0.60} \qquad \text{(Equation 1)}$$

Where:

h_{max} = maximum value of h (Btu/hr ft^2 °F).

t_p = time from start of accident to end of blowdown for LOCA and steam line isolation for secondary breaks (sec).

V = containment net free volume (ft^3).

E = total coolant energy discharge from time zero to t_p(Btu).

75 = material coefficient for steel.

(Note: Paint is accounted for by the thermal conductivity of the material (paint) on the heat sink structure, not by an adjustment on the heat transfer coefficient.)

The basis for the equations is a Westinghouse curve fit to the Tagami data.
The parabolic increase to the peak value is calculated by COCO according to the following equation:

$$h_s = h_{max} \left(\frac{t}{t_p}\right)^{0.5}, \; 0 \le t \le t_p \qquad \text{(Equation 2)}$$

Where:

h_s = heat transfer coefficient between steel and air/steam mixture (Btu/hr ft^2 °F).

t = time from start of event (sec).

For concrete, the heat transfer coefficient is taken as 40 percent of the value calculated for steel during the blowdown phase.

The exponential decrease of the heat transfer coefficient to the stagnant heat transfer coefficient is given by:

$$h_s = h_{stag} + (h_{max} - h_{stag})e^{-0.05(t-t_p)} \qquad t > t_p \qquad \text{(Equation 3)}$$

Where:

h_{stag} = $2 + 50X$, $0 < X < 1.4$.

h_{stag} = h for stagnant conditions (Btu/hr ft^2 °F).

X = steam-to-air weight ratio in containment.

Active Heat Removal

For a large break, the engineered safety features are quickly brought into operation. Because of the brief period of time required to depressurize the reactor coolant system or the main steam system, the containment safeguards are not a major influence on the blowdown peak pressure; however, they reduce the containment pressure after the blowdown and maintain a low long-term pressure and a low long-term temperature.

RWST, Injection

During the injection phase of post-accident operation, the emergency core cooling system pumps water from the refueling water storage tank into the reactor vessel. Since this water enters the vessel at refueling water storage tank temperature, which is less than the temperature of the water in the vessel, it is modeled as absorbing heat from the core until the saturation temperature is reached. Safety injection and containment spray can be operated for a limited time, depending on the refueling water storage tank (RWST) capacity.

RHR, Sump Recirculation

After the supply of refueling water is exhausted, the recirculation system is operated to provide long term cooling of the core. In this operation, water is drawn from the sump, cooled in a residual heat removal (RHR) exchanger, and then pumped back into the reactor vessel to remove core residual heat and energy stored in the vessel metal. The heat is removed from the RHR heat exchanger by the component cooling water (CCW). The RHR Heat Exchangers and CCW Heat Exchangers are coupled in a closed loop system, where the ultimate heat sink is the service water cooling to the CCW Heat Exchanger.

Containment Spray

Containment spray (CS) is an active removal mechanism that is used for rapid pressure reduction and for containment iodine removal. During the injection phase of operation, the containment spray pumps draw water from the RWST and spray it into the containment through nozzles mounted high above the operating deck. As the spray droplets fall, they absorb heat from the containment atmosphere. Since the water comes from the RWST, the entire heat capacity of the spray from the RWST temperature to the temperature of the containment atmosphere is available for energy absorption. During the recirculation phase there is a short period of no spray during the switchover of the ECCS pumps and later spray is terminated upon the entry into ECCS hot leg recirculation (11 hours).

When a spray droplet enters the hot, saturated, steam-air containment 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 droplet. This mass flow will carry energy to the droplet. Simultaneously, the temperature difference between the atmosphere and the droplet will cause the droplet temperature and vapor pressure to rise. The vapor pressure of the droplet will eventually become equal to the partial pressure of the steam, and the condensation will cease. The temperature of the droplet will essentially equal the temperature of the steam-air mixture.

The equations describing the temperature rise of a falling droplet are as follows.

$$\frac{d}{dt}(Mu) = mh_g + q \qquad \text{(Equation 4)}$$

Where,

- M = droplet mass
- u = internal energy
- m = diffusion rate
- h_g = steam enthalpy
- q = heat flow rate
- t = time

$$\frac{d}{dt}(M) = m \qquad \text{(Equation 5)}$$

Where,

- q = $h_cA * (T_s - T)$
- m = $k_gA * (P_s - Pv)$
- A = area
- h_c = coefficient of heat transfer
- k_g = coefficient of mass transfer
- T = droplet temperature
- T_s = steam temperature
- P_s = steam partial pressure
- P_v = droplet vapor pressure

The coefficients of heat transfer (h_c) and mass transfer (k_g) are calculated from the Nusselt number for heat transfer, Nu, and the Nusselt number for mass transfer, Nu'.

Both Nu and Nu' may be calculated from the equations of Ranz and Marshall (Reference 6.2.1-9).

$$Nu = 2 + 0.6(Re)^{1/2}(Pr)^{1/3} \qquad \text{(Equation 6)}$$

Where,

- Nu = Nusselt number for heat transfer
- Pr = Prandtl number
- Re = Reynolds number

$$Nu' = 2 + 0.6(Re)^{1/2}(Sc)^{1/3}$$ (Equation 7)

Where,

Nu' = Nusselt number for mass transfer

Sc = Schmidt number

Thus, Equations 4 and 5 can be integrated numerically to find the internal energy and mass of the droplet as a function of time as it falls through the atmosphere. Analysis shows that the temperature of the (mass) mean droplet produced by the spray nozzles rises to a value within 99 percent of the bulk containment temperature in less than 2 seconds. Detailed calculations of the heatup of spray droplets in post-accident containment atmospheres by Parsly (Reference 6.2.1- 10) show that droplets of all sizes encountered in the containment spray reach equilibrium in a fraction of their residence time in a typical pressurized water reactor containment. These results confirm the assumption that the containment spray will be 100 percent effective in removing heat from the atmosphere.

CFC

The reactor containment fan coolers (CFCs) are another means of heat removal. Each CFC has a fan that draws in the containment atmosphere from the upper volume of the containment via a return air riser. Since the CFCs do not use water from the RWST, the mode of operation remains the same both before and after the ECCS change to the recirculation mode. The steam/air mixture is routed through the enclosed CFC unit, past essential service water cooling coils. The fan then discharges the air through ducting containing a check damper. The discharged air is directed at the lower containment volume. See Table 6.2.1-21 for CFCs heat removal capability assumed for the containment response analyses.

6.2.1.1.3.4 Description of GOTHIC Model

Calculation of the containment pressure and temperature response to a DEPS (as the limiting case) is accomplished by use of the digital computer code GOTHIC. GOTHIC (Generation of Thermal-Hydraulic Information for Containments) is an integrated, general purpose thermal-hydraulics code for performing licensing containment analyses for nuclear power plants. The GOTHIC technical manual (Reference 6.2.1-14) provides a description of the governing equations, constitutive models, and solution methods in the solver. The GOTHIC qualifications report (Reference 6.2.1-15) provides a comparison of the solver results with both analytical solutions and experimental data. The GOTHIC containment modeling is consistent with the recent NRC approval of the Dominion evaluation model (Reference 6.2.1-17), taking advantage of the Diffusion Layer Model (DLM) heat transfer option. This heat transfer option was approved by the NRC (Reference 6.2.1-17). The GOTHIC containment modeling has followed the conditions of acceptance presented in Reference 6.2.1-17. The Robinson containment design fulfills the generic qualifications for application of the Dominion methodology; most notably it is a large dry PWR containment. Consistent with the restrictions identified in Reference 6.2.1-17, Version 8.0 of the GOTHIC code is used here as the current and latest release. The differences in GOTHIC code versions are documented in Appendix A “Release Notes” of the GOTHIC User Manual (Reference 6.2.1-16).

The GOTHIC containment evaluation model for the LOCA event consisted of one large (lumped) volume. Additional boundary conditions, volumes, flow paths, and components are used to model accumulator nitrogen release, sump recirculation, containment spray, and fan coolers. The values used in the specific model for different aspects of the containment are derived from plant-specific input data.

Passive Heat Sinks

Structural heat sinks remove significant energy from the containment atmosphere during the early portion of the transient. The containment response accounts for heat transfer to (and heat storage in) both interior and exterior walls. The structural heat sinks in the containment are modeled as GOTHIC thermal conductors. Every thermal conductor is sub-divided into an appropriate number of nodes, depending on the rate of change of temperature. The heat sink geometry data are based on conservatively low surface areas and are summarized in Table 6.2.1-24. The thermal properties for the heat sink materials are summarized in Table 6.2.1-24. The direct heat transfer option with the Diffusion Layer Model (DLM) condensation option is used for the heat sinks representing floors, ceilings, walls, and miscellaneous metal. With the Direct option, all condensate goes directly to the liquid pool at the bottom of the volume. The effects of the condensate film on the heat and mass transfer are incorporated in the formulation of the DLM option. Under the DLM option, the condensation rate is calculated using a heat and mass transfer analogy to account for the presence of noncondensing gases. This heat transfer methodology was reviewed and approved by the NRC for use in the Dominion methodology (Reference 6.2.1-17). The DLM correlation does not require the user to specify a revaporization input value. The following are conservative exceptions to the DLM option for heat transfer coefficient:

1. The submerged conductors are essentially insulated from the vapor after the pool develops. As a simplification, the conductor labeled as “flooded” in Table 6.2.1-24 is assigned a conservatively low constant value for heat transfer coefficient.
2. The exterior surface of the containment dome and cylinder is modeled as an insulated surface with no heat loss (0.0 Btu/hr-ft^2-°F).

Containment Spray

Containment spray is an active means of cooling the containment atmosphere that is used for rapid pressure reduction. During the injection phase of operation, water is drawn from the RWST and sprayed into containment through nozzles mounted high above the operating deck. As the spray droplets fall, they absorb heat from the containment atmosphere. Since the water comes from the RWST, the entire heat capacity of the spray from the RWST temperature to the temperature of the containment atmosphere is available for energy absorption. During the recirculation phase, there is a short period of no spray during the switchover of the ECCS pumps and later spray is terminated upon the entry into ECCS hot leg recirculation at 11 hours.

The lumped parameter approach assumes that conditions are uniform throughout the volume. When sprays are injected into a volume, the drops are assumed to be uniformly distributed throughout the volume. The heat and mass transfer at the spray droplet surface is determined by the drop and atmosphere temperatures, the steam content of the atmosphere, the drop surface area and the heat and mass transfer coefficients. The heat and mass transfer coefficients depend on the fluid properties at the given temperatures, the drop diameter and pressure and the fall velocity of the spray droplets. Appropriate heat and mass transfer coefficients are applied based on the spray drop diameter and fall velocity.

Spray drops typically reach their terminal velocity within a few feet of the nozzle and the fall velocity is assumed equal to the terminal velocity for lumped modeling in GOTHIC. The terminal velocity depends on the drop diameter and the atmosphere properties. GOTHIC calculates appropriate heat and mass transfer coefficients based on the Sauter mean diameter (Reference 6.2.1-17) for spray drops.

Containment spray is modeled with one boundary condition for the injection phase and switching to a second boundary condition for the recirculation phase.

Reactor Containment Fan Coolers

The reactor containment fan coolers (CFCs) are another active means of heat removal. Each CFC has a fan which draws in the containment atmosphere. The steam/air mixture is routed through the enclosed CFC unit, past service water cooling coils. The CFCs are modeled in GOTHIC as a cooler/heater component in the containment volume. See Table 6.2.1-21 for the CFC heat removal capability assumed for the containment response analyses.

6.2.1.1.3.5 Acceptance Criteria

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

A. GDC 10 and GDC 49 from the HBR2 UFSAR Chapter 3.1: In order to satisfy the requirements of GDC 10 and 49, the peak calculated containment pressure should be less than the containment design pressure of 42 psig;

B. HBR2 FSAR Chapter 3.1, GDC 52: In order to satisfy the requirements of GDC 52, the calculated pressure at 24 hours should be less than 50% of the peak calculated value. (This is related to the criteria for doses at 24 hours.)

6.2.1.1.3.6 Analysis Results

The containment pressure, steam temperature and water (sump) temperature profiles from each of the LOCA cases are shown in Figures 6.2.1-1 through 6.2.1-2 for the DEPS break cases.

6.2.1.1.3.6.1 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 to start, resulting in one train of ECCS and containment safeguards equipment being available. This combination results in a minimum set of safeguards being available. Further, loss of offsite power delays the actuation times of the safeguards equipment due to the required diesel startup time 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 resulting rapid rise in both the containment pressure and temperature. This rapid rise in containment pressure results in the generation of a containment HI signal at 0.73 seconds and a containment HI-HI signal at 1.89 seconds. The containment pressure continues to rise rapidly in response to the release of mass and energy until the end of blowdown at 21.6 seconds, with the pressure reaching a value of 38.9 psig. 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 decreases slightly to 37.5 psig and then continues to decrease due to the initiation of the containment spray at 40.09 seconds and fan coolers 46.73 seconds. Reflood continues at a reduced flooding rate due to the buildup of mass in the RCS core that offsets the downcomer head. This reduction in flooding rate and the continued action of the CFCs and Spray leads to a slowly decreasing pressure out to the end of reflood, which occurs at 208.59 seconds. At this juncture, energy removal from the SG secondary begins at a very much increased rate, resulting in a rise in containment pressure out to 989.2 seconds when energy has been removed from the faulted SG bringing the faulted SG secondary pressure down to the containment design pressure of 42 psig. The result of this SG secondary energy release is a containment pressure of 41.8 psig at 989.2 seconds, the ultimate peak pressure for this transient. After this event, the mass and energy released is reduced due to so much energy removal from the SGs having been accomplished and pressure slowly falls out to the cold leg recirculation time of 2442 seconds. At this time, the ECCS is realigned for cold leg recirculation resulting in an increase in the SI temperature due to delivery from the hot sump. At 11 hours, (39600 seconds) containment spray is terminated as a result of aligning the ECCS for hot leg recirculation. The loss of containment spray results in a rapid rise in containment pressure until the steam temperature increases to the level that the fan coolers can remove the decay heat energy at about 60,000 seconds. These changes result in a slower containment pressure reduction rate but containment pressure continues to decrease due to lower decay heat, SG energy release and continued CFC cooling. This trend continues to the end of the transient at 1.0E+07 seconds.

6.2.1.1.3.6.2 Double Ended Pump Suction Break with Maximum Safeguards

The DEPS break with maximum safeguards has a transient history very similar to the minimum safeguards case discussed in Section 6.2.1.1.3.6.1. The results of this event are bounded by the most limiting LOCA event as described in Section 6.2.1.1.3.6.1 Double Ended Pump Suction Break with Minimum Safeguards.

6.2.1.1.3.6.3 Double Ended Hot Leg Break with Minimum Safeguards

This analysis assumes a loss of offsite power coincident with a double-ended rupture of the RCS piping between the reactor vessel outlet nozzle and the steam generator inlet (i.e. A break in the RCS hot leg). The associated single failure assumption is the failure of a diesel to start, resulting in one train of ECCS and containment safeguards equipment being available. This combination results in a minimum set of safeguards being available. Further, loss of offsite power delays the actuation times of the safeguards equipment due to the required diesel startup time after receipt of the Safety Injection signal.

The results of this event are bounded by the most limiting LOCA event as described in Section 6.2.1.1.3.6.1 Double Ended Pump Suction Break with Minimum Safeguards.

6.2.1.1.3.6.4 Double Ended Hot Leg Break with Maximum Safeguards

The DEHL break with maximum safeguards was not analyzed since neither the ECCS pumps or containment safeguards start prior to the end of blowdown. Thus, the maximum ECCS case would be identical to the minimum ECCS case discussed in 6.2.1.1.3.6.3.

6.2.1.1.3.7 Conclusions

As illustrated in the results Section 6.2.1.1.3.5, all cases resulted in a peak containment pressure that was less than 42 psig. In addition, all long-term cases were well below 50% of the peak value within 24 hours. Based on the results, all applicable criteria for HBRSEP, Unit No.2 have been met.

6.2.1.2 Containment Subcompartments

The crane wall has been designed for several pressures as the volume within it is compartmentalized into three compartments each housing one loop of the RCS. The compartments are separated from each other by the refueling canal, missile shield walls, and the in-core instrumentation

room which restricts venting of the steam resulting from the LOCA. The plan locations of the compartments are shown on Figure 6.2.1-13. The pressures for which each compartment is designed are listed below:

COMPARTMENT	DESIGN PRESSURE
Northeast	16 psig
Southeast	13.5 psig
Northwest	22 psig

The primary shield was designed for an internal pressure of 80 psig.

The peak pressures in each compartment were determined by a digital computer code, COMCO, which was developed to analyze the pressure build-up 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 build-up of the steam-air mixture in the compartment is determined. Each compartment has a vent opening to the free volume of the containment.

The main calculation performed is a mass 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 pressure resulting from an instantaneous double ended rupture of the reactor coolant pipe.

Evaluation of Leak-Before-Break Considerations

The current licensing basis pipe break for each reactor coolant loop compartment is the instantaneous double-ended rupture of the reactor coolant pipe. For the reactor cavity region, a longitudinal split of area equivalent to the cross-sectional area of the reactor coolant pipe (i.e. 4.5 ft^2) forms the design basis. HBRSEP, Unit No.2 is approved for leak-before-break (LBB) (Reference 6.2.1-11). LBB allows the dynamic effects of postulated primary loop pipe ruptures to not be considered in the design basis. Since the RCS piping is excluded in the LBB consideration, the connecting large branch nozzles must be considered for design verification. The large branch line nozzles are the pressurizer surge line, accumulator line, and the RHR line. These smaller breaks, which are outside the cavity region, would result in minimal asymmetric pressurization in the reactor cavity region. Additionally, the differential loading is significantly reduced. For example the peak break compartment pressure can be reduced by more than a factor of 2, and the peak differential across an adjacent wall can be reduced by more than a factor of 3, whenever the smaller breaks are considered. Therefore, the decrease in mass and energy releases associated with the smaller RCS nozzle breaks,

as compared to the larger RCS pipe breaks, are within the original mass and energy release calculations. The original licensing basis therefore remains bounding.

6.2.1.3 (Deleted)

6.2.1.4 <u>Containment Analysis for Postulated Secondary System Pipe Ruptures</u>

As stated in the response to General Design Criteria (GDC) 10 in section 3.1.2.10, the design pressure and temperature of the HBRSEP, UNIT NO. 2 containment shall be in excess of the peak pressure and temperature resulting from the complete blowdown of the RCS through any RCS pipe rupture, up to and including a hypothetical LOCA. In addition, the responses to GDCs 49 and 52 (Sections 3.1.2.49 and 3.1.2.52) do not address containment integrity following a Main Steamline Break (MSLB). However, in 1980 the NRC issued IE Bulletin 80-04, which required all licensees to review their analysis of the containment pressure response to an MSLB with runout AFW flow to determine the potential for containment overpressure. In late 1984, IE Notice 84-90 raised the concern that superheated steam released during an MSLB may produce thermal environments more severe than previously analyzed for environmental qualification of safety related equipment. Then, in 1999, a Technical Specification change request for the service water temperature upper limit of 97°F required further analysis of the MSLB for HBRSEP, Unit No 2. The containment pressure and temperature analysis presented in this section addresses these issues and plant changes.

6.2.1.4.1 Mass and Energy Releases

Two power levels have been evaluated for the MSLB: 0% and 102% of 2300 MWt (2346 MWt bounds operation at 2339 MWt including the applicable calorimetric uncertainty). One break area has been analyzed – full double-ended rupture (DER) downstream of the flow restrictor in one steamline. Note that a DER is defined as a rupture in which the steam pipe is completely severed and the ends of the break displace from each other. The break area in the forward-flow direction is 1.4 ft^2, which is the effective blowdown area equivalent to the flow restrictor at the SG outlet nozzle. The reverse flow break area is 1.497 ft^2, which is the cross-sectional area of the flow-restricting venturi.

MSLBs can be postulated to occur with the plant in any operating condition ranging from hot shutdown to full power. Since SG mass decreases with increasing power level, breaks occurring at-lower power level will generally result in a greater total mass release to the containment. However, because of increased stored energy in the primary side of the plant, increased heat transfer in the SGs, and additional energy generation in the fuel, the energy release to the containment from breaks postulated to occur during "at-power" operation may be greater than for breaks postulated to occur with the plant in a hot-shutdown condition. Additionally, steam pressure and the dynamic conditions in the SGs change with increasing power and have a significant influence on the rate of blowdown. Because of the opposing

effects (mass versus energy release) of changing power level on MSLB releases, the two extremes, full-power and zero-power, have been analyzed.

The following five cases were analyzed to determine the most limiting combination of single failure and initial reactor power level with respect to containment pressure and temperature and are presented in the order in which they are discussed in the subsequent text:

- 102% of 2300 MWt with single failure of one Steam Line Check Valve (SLCV).
- Hot Zero Power (HZP) with single failure of one SLCV.
- 102% of 2300 MWt with single failure of one feedwater regulating valve.
- 102% of 2300 MWt with single failure of one E-Bus.
- HZP with single failure of one E-Bus.
- HZP with failure of the Steam Driven AFW pump flow control valve.

The loss of one E-bus results in the loss of one Safety Injection (SI) pump and the loss of one train of containment cooling (one CS pump and two CFC). At HZP, the Feedwater Regulating Valves (FRVs) are closed; therefore, the single failure of a FRV was only considered at 102% power. The failure of a MSIV is bounded by the loss of a Check Valve (CV). Therefore, the loss of one MSIV was not analyzed. With a single failure of the CV in the affected steamline, MSIV closure in the intact steamlines is required to terminate the blowdown of the intact steam generators. A delay time of 4 seconds was assumed (2-second signal processing plus 2-seconds for MSIV closure) with full steam flow assumed through the valve during the valve stroke. An additional case at HZP was analyzed to determine the effect of AFW runout. The single failure for this case is assumed to be the Steam Driven AFW pump flow control valve.

Therefore, a set of cases is defined which encompasses the power range from 102% of 2300 MWt to HZP and different single failures. All the cases also assume the continued availability of offsite power. The largest effect of this assumption is the continued operation of the reactor coolant pumps, which significantly increase the rate of heat transfer to the faulted steam generator. The analyses were performed in two distinct steps. First, the steamline break mass and energy releases are determined. The methodology, modeling of the single failures, and other analysis assumptions are discussed in Section 6.2.4.1.1. Then, the containment response analysis is performed. The methodology, modeling of the single failures, and other analysis assumptions are discussed in Section 6.2.1.4.1.2, 6.2.1.4.1.3, and 6.2.1.4.1.4. The results of both portions of the analysis are presented in Section 6.2.1.4.3, with conclusions summarized in Section 6.2.1.4.4.

6.2.1.4.1.1 Mass and Energy Release Analysis Method

The steamline break mass and energy releases are generated using the NRC-approved LOFTRAN code (Reference 6.2.1.4- 1). LOFTRAN is used for

studies of the transient response of a PWR system to specified perturbations in process parameters. The code simulates a multi-loop system including the reactor vessel, hot and cold leg piping, steam generator (shell and tube sides), and the pressurizer. A neutron point kinetics model is used and the reactivity effects of the moderator, fuel, boron, and rods are included. The secondary side of the steam generator is modeled as a homogeneous saturated mixture. Protection and control systems are simulated, as well as the Emergency Core Cooling System. The calculation of secondary side break flow is based on the Moody critical flow correlation (Reference 6.2.1.4- 2) with fL/D = 0.

The Westinghouse steamline break mass and energy release methodology used for the HBRSEP, Unit No.2 analyses is based on the information in WCAP-8822, “Mass and Energy Releases Following a Steam Line Rupture” (Reference 6.2.1.4- 3). WCAP-8822 forms the basis for the assumptions and models used in the calculation of the mass and energy releases resulting from a steamline rupture. This methodology was approved by the NRC in Reference 6.2.1.4-4.

6.2.1.4.1.2 Single Failure Assumptions

There were four single failures that considered in the steamline break and containment integrity analyses. These were 1) the failure of the main steamline check valve in the faulted Steam Generator (SG), 2) the failure of the Feedwater Regulation Valve (FRV) to close in the faulted SG main feedline, 3) An electrical bus failure which results in losing one train (2 CFCs and 1 Spray pump) of containment pressure reducing equipment and 4) failure of the steam driven auxiliary feedwater pump runout protection system. For the electrical bus failure, the LOFTRAN steamline break mass and energy release analyses model no failure, since the penalty of the single failure is addressed in the containment analysis.

Feedwater Transient When FRV Closes

The set of LOFTRAN cases with either the check valve failure or the E-Bus failure credit the isolation of the main feedwater piping due to the closure of the FRV on the faulted loop. An SI signal generates the FRV closure signal; an electronic time delay of 1.5 seconds is assumed in addition to a valve closure time of 20 seconds. The FRV closure stops the addition of pumped main feedwater into the faulted steam generator. However, when the faulted steam generator depressurizes below the saturation point of the main feedwater, the water in the feedline between the FRV and steam generator may flash into steam. The liquid in the feedline is assumed to exit with the vapor from the flashing, and thus additional mass from the main feedwater system is added to the steam generator even after feedline isolation. The volume of the unisolable feedline piping is assumed to be 355 ft^3 in the cases where the FRV is credited to close.

The safety injection signal that isolates the main feedwater by closing the FRV also trips the main feedwater pumps and closes the Motor Operated Valve (MOV) pump discharge valves. The turbine trip occurs at the same time, terminating the heating source in the main feedwater

heaters. Since the FRV closure is the action that will first terminate main feedwater flow to the faulted steam generator, it is typically the only action modeled in the LOFTRAN analysis. However, when the FRV fails, these other actuations influence the resulting accident sequence of events. The changes in the accident scenario due to the FRV failure are described below.

Feedwater Transient When FRV Fails

The tripping of the main feedwater pumps is credited to terminate the main feedwater addition to the faulted SG. After a 1.5 second electronic delay, the pump coastdown is assumed to occur linearly to zero at the time of the main feedwater block valve closure, which takes 50 seconds.

Motor-operated block valves, located downstream of the feedwater bypass line branch connection and upstream of the FRV, have a closure time of 50 seconds. Their location ultimately defines the extra volume of feedline piping that is unisolated from the faulted steam generators. The water in the unisolable piping is subject to flashing as the faulted steam generator depressurizes to the feedwater saturation pressure. The total unisolable volume for the FRV failure cases is 355 ft^3. However, the volume available to flash prior to closure of the FRV is 1818 ft^3, this additional volume adds to the flashing flow prior to closure of the block valve and was included in the FRV failure case.

Therefore, the net result of the single failure of the FRV on the faulted steam generator is the increased duration of the main feedwater flow and the increased volume for feedwater flashing. The termination of the pumped main feedwater is based on the closure of the block valves.

6.2.1.4.1.3 Analysis Assumptions

This section discusses the key assumptions made in the calculation of the mass and energy releases from the postulated steamline break.

Protection Logic and Setpoints

The timing and results of the steamline break event are largely dependent on the plant's protection logic and the corresponding setpoints and delays. The HBRSEP, Unit No.2 used an older steamline break protection systems that relies of multiple signals from the main steamline header. A reactor trip and SI can occur on either 1) high steam flow in one steam line coincident with low pressure in two of three steam lines, 2) a high pressure differential (110 psi) between any main steamline and the steam header or 3) a containment HI pressure (5.5 psig) signal. All of the analyzed cases discussed in this report have credited the first SI signal, which for the check valve failure and FRV failure cases was the containment HI signal. The E-Bus and steam driven auxiliary feedwater pump controller cases relied upon the high pressure differential signal for SI and reactor trip. Reactor trip may also occur on a non-SI signal such as low pressurizer pressure or overpower

ΔT, but if these signals are generated, it is later than the SI signal that is credited in the analyses.

A non-return check valve is located in each loop's steamline piping. Therefore, for any break location inside of containment, the check valve is a passive device that will prevent reverse flow from the intact steam generator. However, a single failure of a check valve has been assumed in the analyses, and thus the closure of the main steamline isolation valves (MSIVs) upon a steamline isolation signal is necessary to isolate the intact steam generators. The protection logic for steamline isolation is either a containment HI-HI signal at 12.0 psig or low-low level in one of three steam generators. Since LOFTRAN cannot model SG level swell accurately and the containment HI-HI signal will occur first, the HI-HI signal was used for MSIV closure, start of the containment spray pumps with associated delay and start of the steam driven auxiliary feedwater pump. MSIV closure was not needed in the E-Bus and steam driven auxiliary feedwater pump controller failure cases.

Secondary Side Assumptions

This section summarizes the input assumptions associated with the steam generator and the piping attached to it. For the full double-ended rupture steamline break with continued offsite power available, the blowdown is rapid; the containment response is largely controlled by the amount of steam released, as will be discussed in Section 6.2.1.4.3.1. Thus, of key importance are the secondary side assumptions that determine the mass released from the break. The first items that will be discussed are:

- The initial steam generator water inventory,
- The water added from the main feedwater system,
- The water added from the auxiliary feedwater system, and
- The steam in the steamline when the break occurs.

In addition, there are assumptions associated with the quality of the saturated steam that exits the break, and assumptions regarding the time of steam generator tube uncovery, and the containment backpressure that will be discussed at the end of this section.

Initial Steam Generator Inventory

Maximum initial steam generator masses were used in all of the cases. The use of high initial steam generator masses maximizes the steam generator inventory available for release to the containment. The initial masses are provided in Table 6.2.1.4-1.

Main Feedwater System

The main feedwater system has been discussed in Section 6.2.1.4.1.3, since the main feedwater transient is significantly altered by whether the FRV is assumed to close. The MFW flow transient is discussed below for each specified MSLB case. Key assumptions and methods for both single failure scenarios are summarized below
.

- The initial feedwater flow is assumed to be the nominal flow for the power level being analyzed.
- The main feedwater flow increases as the faulted steam generator rapidly depressurizes and the faulted loop FRV opens in response to the increased steam flow. The faulted loop FRV is assumed to be fully open within 0.2 seconds of the event initiation, supplying 116% of rated flow.
- If the FRV closes, the main feedwater flow is terminated by the valve closure, which is assumed to occur linearly for 20 seconds. In these cases credit is taken for the simultaneous coastdown of the main feedwater pumps.
- If the FRV fails to close, the main feedwater flow is terminated due to the trip of the main feedwater pumps. These cases take credit for pump coastdown that is assumed to linearly decrease for over 50 seconds, which is the time required to close the block valve.
- All cases accounted for leakage flow past both the FRV and block valves. The leakage flow was assumed to be 750 gpm when only the FRV was closed and 75 gpm after the block valve closed.
- All cases consider the possibility of the feedline flashing when the feedwater saturation pressure is reached. Only the cases initiated from hot zero power do not experience flashing due to the low temperature of the feedwater.

Auxiliary Feedwater

Generally within the first minute following a steamline break, the auxiliary feedwater system will be initiated due to an SI signal. Addition of auxiliary feedwater to the steam generators will increase the secondary mass available for release to containment. Maximum auxiliary feedwater flow rates are assumed and due to the design of the HBRSEP, Unit No. 2 auxiliary feedwater system all of the auxiliary feedwater was assumed to be delivered to the faulted steam generator. In addition, the pumped auxiliary feedwater flow rate is assumed at the time the SI setpoint is reached, with a 1.5 second electronic delay to start the pumps. The steam driven auxiliary feedwater pump was assumed to start with no delay on either MSIV closure and closure of the main steamline check valve in the faulted main steamline. Operator action is credited to terminate the auxiliary feedwater flow to the faulted steam generator 10 minutes after failure of the main steamline.

Initial Steam in the Steamline

Only one break area has been analyzed – full double-ended rupture (DER) downstream of the flow restrictor in one steamline. Note that a DER is defined as a rupture in which the steam pipe is completely severed and the ends of the break displace from each other. The break area in the forward-flow direction is 1.4 ft^2, which is the effective blowdown area equivalent to the flow restrictor at the SG outlet nozzle. The reverse flow break area is 1.497 ft^2, which is the cross-sectional area of the flow-restricting venturi. Therefore steam in the steamline piping will immediately exit the break in addition to the steam coming from the

faulted steam generator. The steam flow from the piping between the steam generator and its flow-restricting venturi was calculated internally to LOFTRAN based on a volume of 810 ft^3 when credit for check valve closure was taken and 3682 ft^3 when a failure of the check valve is assumed. The rate at which the steam is assumed to exit is based on the initial steam generator pressure and the Moody critical break flow correlation.

Quality of the Break Effluent

Credit for liquid entrainment out the break was not taken in the analyses performed for H. B. Robinson Steam Electric Plant Unit No.2. This is a conservatism in the analyses that if accounted for would reduce the reported peak pressures and steam temperatures.

6.2.1.4.1.4 Reactor Coolant System Assumptions

While the mass and energy released from the break is determined from assumptions that have been discussed in the previous section, the rate at which the release occurs is largely controlled by the conditions in the reactor coolant system. The major features of the primary side analysis model are summarized below and shown in Table 6.2.1.4-1.

- Continued operation of the reactor coolant pumps maintains a high heat transfer rate to the steam generators.

- The model includes consideration of the heat that is stored in the RCS metal.

- Reverse heat transfer from the intact steam generator to the RCS coolant is modeled as the temperature in the RCS falls below the steam generator fluid temperature.

- Minimum flow rates are modeled from ECCS injection, to conservatively minimize the amount of boron that provides negative reactivity feedback.

- Core residual heat generation is assumed based on the 1979 ANS decay heat plus 2σ model (Reference 6.2.1.4- 6).

- Conservative core reactivity coefficients corresponding to end-of-cycle conditions were chosen to maximize the reactivity feedback effects as the RCS cools down as a result of the steamline break.

- Control rod trip reactivity resulting in 1.77%Δk shutdown margin was modeled.

6.2.1.4.1.5 Steamline Break Mass and Energy Releases

There are six tables presented within this section that summarize the mass and energy releases as calculated by LOFTRAN. The results for the 102% of 2300 MWt cases are presented first. The mass release rate is shown for each case in Table 6.2.1.4-2 to Table 6.2.1.4-4. The same information is provided for the HZP cases. The mass release rate is shown for each case in Table 6.2.1.4-5 to Table 6.2.1.4-7. The sequence of events, along with a discussion of the case results, is provided in Sections 6.2.1.4.3.1 and 6.2.1.4.3.4 with the containment results.

6.2.1.4.2 Containment Response Analysis

The following sub-sections describe the analysis method and input assumptions used to determine the containment response to the steamline break mass and energy releases discussed in Section 6.2.1.4.1. The results of the analyses are presented in Section 6.2.1.4.3.

6.2.1.4.2.1 Containment Methods

6.2.1.4.2.1.1 Steamline Break Analyses (MSLB) with COCO

All MSLB containment pressure and temperature responses are performed with COCO with the exception of the HZP MSLB with check valve failure case. The HZP MSLB with check valve failure has been performed with the GOTHIC computer code (6.2.1-17); see section 6.2.1.4.2.1.2 for discussion.

The COCO computer code (Reference 6.2.1.4- 7) is used to analyze the containment pressure and temperature transient response following the postulated steamline break accidents presented in this report. COCO is a mathematical model of a generalized containment; the proper selection of various options in the code allows the creation of a specific model for the particular containment design. The values used for the HBRSEP, Unit No.2 model are summarized in Section 6.2.1.4.2.3.

The COCO computer code consists of time-dependent conservation equations of mass and energy, together with steam tables, equations of state and other auxiliary relationships. Transient conditions are determined for both the containment steam-air mixture and the sump water. The energy equation is applied to the containment shell to obtain transient temperature gradients as well as heat stored in and conducted through the structure. Heat removal by means of energy storage in equipment within the containment, internal sprays, emergency containment coolers, and sump water recirculation cooling system can be included in the model.

The containment air-steam-water mixture is separated into two distinct systems. The first system consists of the air-steam phase, while the second system is the water phase in the containment sump. This division permits more accurate representation of the distinct physical phenomena occurring in each system.

The steam-air mixture and water phase are assumed to have uniform properties. In addition, temperature equilibrium between the air and steam is assumed. However, this does not imply continual thermal equilibrium between the steam-air mixture and water phase. Sufficient relationships to solve the problem independent of this restriction are provided by the equations of conservation of mass and energy as applied to each system, together with appropriate equations of state and heat transfer boundary conditions. As thermodynamic equations of state and conditions may vary during the transient, the equations have been derived for all possible cases of superheated and saturated steam and subcooled or saturated water. Switching between states is handled automatically by the code.

6.2.1.4.2.1.2 Steamline Break (MSLB) Analyses with GOTHIC

As the limiting case, the containment response to a MSLB occurring at HZP with check valve failure was analyzed with the GOTHIC computer code (Reference 6.2.1-17). GOTHIC (Generation of Thermal-Hydraulic Information for Containments) is an integrated, general purpose thermal-hydraulics code for performing licensing containment analyses for nuclear power plants. The GOTHIC technical manual (Reference 6.2.1-14) provides a description of the governing equations, constitutive models, and solution methods in the solver. The GOTHIC qualifications report (Reference 6.2.1-15) provides a comparison of the solver results with both analytical solutions and experimental data. The GOTHIC containment modeling is consistent with the recent NRC approval of the Dominion evaluation model (Reference 6.2.1-17), taking advantage of the Diffusion Layer Model (DLM) heat transfer option. This heat transfer option was approved by the NRC (Reference 6.2.1-17). The GOTHIC containment modeling has followed the conditions of acceptance presented in Reference 6.2.1-17. The Robinson containment design fulfills the generic qualifications for application of the Dominion methodology; most notably the containment is a large dry PWR containment. Consistent with the restrictions identified in Reference 6.2.1-17, Version 8.0 of the GOTHIC code is used here as the current and latest release. The differences in GOTHIC code versions are documented in Appendix A “Release Notes” of the GOTHIC User Manual (Reference 6.2.1-16).

The GOTHIC computer code consists of time-dependent conservation equations of mass and energy, together with steam tables, equations of state and other auxiliary relationships. Transient conditions are determined for both the containment steam-air mixture and the sump water. The energy equation is applied to the containment shell to obtain transient temperature gradients as well as heat transferred to (and stored in) the structure. Heat removal by means of energy storage in equipment and structures within the containment, internal sprays, and emergency containment coolers are included in the model. The GOTHIC containment evaluation model for the LOCA event consisted of one large (lumped) volume. Additional boundary conditions, flow paths, and components are used to model the mass and energy release, containment spray, and fan coolers. The values used for the HBRSEP, Unit No.2 model are summarized in Section 6.2.1.4.2.3.

6.2.1.4.2.2 Single Failure Assumptions

There were four single failures that were considered in the steamline break and containment integrity analyses. The first failure was a failure of the mainsteam CV. The second failure was the failure of the FRV failure. The third failure was in the auxiliary feedwater runout protection system. Each of these failures was accounted for in the LOFTRAN analysis as discussed in Section 6.2.1.4.1. The last failure considered was a failure of an electrical bus resulting in loss of one train of containment pressure reducing equipment. The containment analysis assumes no failure of the containment safeguards for the CV, FRV and auxiliary feedwater runout protection failure cases, since the penalty of the single failure is already accounted for in higher mass and energy releases. The containment for HBRSEP, Unit No.2 has four containment fan coolers and two containment spray pumps. They are all assumed to function in the CV, FRV and auxiliary feedwater pump runout protection failure. The fan coolers and sprays are actuated on a HI containment pressure signal and a HI-HI containment pressure signal, respectively. The delay times, with the assumption of continued offsite power, are:

- A delay of 35.4 seconds from the HI containment pressure setpoint (of 5.5 psig) until the fan coolers start, and

- A delay of 23.5 seconds from the HI-HI containment pressure setpoint (of 12 psig) until the containment sprays start.

The electrical bus failure case assumes the loss of one train of containment safeguards, which eliminates half of the fan coolers and half of the containment sprays. The assumed delay times are not impacted. The following section provides the heat removal and pump flow rates that are assumed for all of the cases.

6.2.1.4.2.3 Analysis Assumptions and Input Values

This section addresses the major input values that are used in the COCO containment response analyses. The assumed initial conditions, the fan cooler heat removal, the containment spray pump flow rate and the containment heat sink input are provided.

At the initiation of the steamline break, the containment is assumed to be at the pressure of 1.0 psig and the temperature of 130°F, with a relative humidity biased low, as listed in Table 6.2.1.4-8. All initial conditions are selected to maximize the containment pressure response. The initial pressure has a direct relationship on the peak containment pressure, and thus is maximized. The initial temperature is maximized because the steady-state temperature of the containment heat sinks is assumed to be the same as the containment air temperature. The higher initial heat sink temperature causes them to be less effective in removing heat. The initial humidity is conservative when it is assumed to be low, since this maximizes the amount of air initially assumed in the containment. The moles of air are non-condensable, and thus will maximize the containment pressure response as the containment temperature increases.

The containment fan coolers each have a fan, which draws in the containment atmosphere, and the steam/air mixture is routed through the enclosed fan cooler unit, past service water cooling coils. The fan then discharges the air back to the containment. The heat removal capability assumed for each fan cooler is summarized in Table 6.2.1.4-9. Note that the electrical bus failure cases credit 2 fan coolers, while the CV, FRV and auxiliary feedwater pump runout protection failure cases credit 4 fan coolers.

The containment spray system flow rate was modeled as a constant flow at all containment pressures, as listed in Table 6.2.1.4-10. During the steamline break blowdown, the containment spray pumps draw water from the refueling water storage tank (RWST) and spray it into the containment through nozzles mounted high above the operating deck.

Finally, the heat transfer through, and heat storage in, interior and exterior walls of the containment structure are considered. Structural heat sinks, consisting of steel and concrete, are modeled as slabs having specific areas and layers of varying thickness. The thermal conductivity, density and specific heat of each layer are specified. The material, heat transfer area and thickness of each component are listed in Table 6.2.1.4-11, while Table 6.2.1.4-12 lists the assumed thermal conductivity and heat capacity.

6.2.1.4.3 MSLB Analysis Results

The mass and energy release analysis described in Section 6.2.1.4.1 and the containment model and assumptions described in Section 6.2.1.4.2 were used to determine the accident progression and containment response to large double-ended rupture steamline breaks. Table 6.2.1.4-13 summarizes the peak containment pressures, steam temperatures and component temperature from the environmental qualification analyses calculated for each of the cases. The results of the cases are discussed in Section 6.2.1.4.3.1 for the check valve failure cases. Section 6.2.1.4.3.2 for the FRV failure case. Section 6.2.1.4.3.3 for the electrical bus failure and Section 6.2.1.4.3.4 for the auxiliary feedwater pump runout protection system failure.

6.2.1.4.3.1 Main Steamline Check Valve Failure Case Results

These cases model the failure of the check valve in the faulted main steamline to close and prevent reverse flow from the intact steam generators. No other failures are taken beyond the check valve, so credit for FRV closure, maximum auxiliary feedwater and all containment safeguards is modeled. The most limiting case with a check valve failure was the case initiated from HZP power. The peak pressure of 41.06 psig occurs at 612.2 seconds. This initial power level results in the highest integrated mass and energy releases, primarily due to the high initial steam generator mass. The 102% of 2300 MWt case calculated a peak pressure of 41.19 psig at 611.78 psig, due primarily to the lower steam generator initial mass.

The turnaround of the containment pressure response is due to the reduction in break energy flow rate as a result of isolating the auxiliary feedwater at 10 minutes (600 seconds).

As discussed above, the containment peak pressure is primarily determined by the high release rates from the break. A lower break flow rate would be a benefit because it would allow the containment heat removal systems to have a larger relative impact. Thus, the large double-ended rupture is the break size for which the most limiting containment pressures are anticipated. Furthermore, the HI containment pressure signal is the first safety injection signal credited in the large double-ended rupture analyses. Any smaller breaks, which would also rely on this same signal for protection, are bounded by the large break size that has been analyzed.

The containment HI-HI signal is credited in the check valve failure cases for generating the signal necessary to isolate the intact steam generators steam lines, as well as starting containment spray.

The sequence of events for the steamline check valve failure cases are summarized in Table 6.2.1.4-14 and Table 6.2.1.4-15. The sequence of events tables presents both the events from the steamline break portion of the analysis and the containment response analysis. It is noted that the break flow continues past the end of the analyzed transient due to the leakage flow around the closed FRV and block valves.

The energy addition coming from the leakage flow is well below the capacity of the fan coolers heat removal capability and therefore, the continued leakage flow does not result in a concern for long-term containment pressure or temperature. The containment pressure and temperature transients for these cases are shown in Figure 6.2.1.4-1 to Figure 6.2.1.4-4.

The containment transient was reanalyzed for environmental qualification conditions by reducing the initial pressure to -1.0 psig (13.7 psia) with COCO and -0.8 psig (13.9 psia) for the limiting HZP calculated with GOTHIC. Two of the conductors (See Table 6.2 conductors, walls 12 & 13) were modeled as components and Figures 6.2.1.4-2 and 6.2.1.4-4 present the component temperature transient.

6.2.1.4.3.2 Feedwater Regulation Valve Failure Case Results

The only case analyzed with an FRV failure was a 102% of 2300 MWt case, since at Hot Zero Power operation the FRVs are closed. The peak pressure of 38.92 psig occurs at 611.73 seconds. With this accident scenario, there is more mass pumped into the faulted steam generator from the main feedwater pumps, but the break mass release is less due to the closure of the main steamline check valve, which limits the release from the intact steam generators. The main feedwater pumped flow rate continues until flow is terminated by closure of the block valve 50 seconds after the SI signal.

The sequence of events for FRV failure case at 102% of 2300 MWt is given in Table 6.2.1.4-16. The sequence of events tables presents both the events from the steamline break portion of the analysis and the containment response analysis. The containment pressure transients for these cases are shown in Figure 6.2.1.4-5 and Figure 6.2.1.4-6.

6.2.1.4.3.3 Electrical Bus Failure Cases

The design of the onsite electric supply system at HBRSEP, Unit No.2 is such that the failure of a single electrical bus (E-Bus) can result in loss of half of the containment pressure reducing equipment. Further, for this assumed single failure case, the associated motor driven auxiliary feedwater pump that would also be lost was assumed to remain available, adding conservatism to this analysis. Other equipment was assumed to operate normally and credit for the main steam line check valves and the main feedwater regulation valves was taken. This single failure was analyzed at the two power levels of 102% of 2300 MWt and HZP. The HZP was slightly more limiting due to the higher initial mass in the faulted steam generator. Table 6.2.1.4-13 shows that for the HZP case the calculated peak pressure was 41.61 psig at 614 seconds and for the 102% of 2300 MWt case the peak pressure was 40.63 psig at 614.1 seconds. These results are close to the limiting result for the check valve failure cases. However, credit for loss of the motor driven auxiliary feedwater pump would substantially reduce the peak pressure for these cases.
Tables 6.2.1.4-17 and 6.2.1.4-18 provides the sequence of events for these two cases while Figures 6.2.1.4-7 through 6.2.1.4-10 provide plots of the containment pressure and temperature responses.

6.2.1.4.3.4 Failure of the Steam Driven Auxiliary Feedwater Runout Protection System

The last single failure to be analyzed was for the steam driven auxiliary feedwater pump runout protection system. This failure will result in increase the total auxiliary feedwater flow from 1209 gpm to 1325 gpm. Since failure of the runout protection system is the single failure, credit for the main steam check valve and the FRVs to close was taken. Further, all containment pressure reducing equipment was assumed to operate. Since the HZP case with a Check Valve failure was the most limiting case, this power level was reanalyzed with the failure of the runout protection system. Table 6.2.1.4-13 shows that the peak pressure was 38.40 psig at 613.23 seconds. Table 6.2.1.4-19 provides the sequence of events and Figure 6.2.1.4-11 & 6.2.1.4-12 provide the containment pressure and temperature transients.

6.1.2.4.4 Conclusions

Containment Integrity Analyses have been performed to provide HBRSEP, Unit No.2 with a current analysis for main steamline break inside containment. The scope of the analysis consists of full double-ended steamline ruptures with two initial power levels of 102% of 2300 MWt and hot zero power, selected to cover the range of power levels at which a steamline break could occur. The failures that have been individually considered are a failure 1) the check valve in the faulted steam generator main steamline, 2) failure of the FRV to close in the main feedline of the faulted steam generator, 3) failure of an electrical bus and 4) failure of the steam driven auxiliary feedwater protection system. The continuation of offsite power is modeled, since this is shown to be more limiting than the loss of offsite power.

The peak pressure occurred at HZP for the check valve failure case and resulted in a peak pressure of 41.06 psig, which is below the design pressure of 42 psig. Analyses performed to establish environmental qualification conditions resulted in a peak component temperature of 322.6°F for the HZP check valve failure case.

6.2.1.5 Minimum Containment Pressure Analysis for Performance Capability Studies of ECCS

Section 6.2.1.1.3, Design Evaluation, presents the results of perturbations in mass and energy release to determine the effectiveness of ECCS for HBR 2.

6.2.1.6 Testing and Inspection

Tests performed on materials and special construction techniques are described in Section 3.8.1.6. Structural integrity tests of the completed Containment Building are described in Section 3.8.1.7. The in-service inspection program for associated ESF components is discussed in Section 3.9.

6.2.1.7 Instrumentation

Instrumentation has been provided to monitor containment atmospheric conditions:

Pressure	-5 to 126 psig
Radiation	10^{-3} - 10^{-9} μCi/cc
Hydrogen Concentration	0 to 10 percent
Water Level	Up to 600,000 gallons

Containment pressure indication will be used to distinguish between various incidents. Pressure taps reflect the effectiveness of the containment and cooling systems and other ESF. High pressure indicates high temperatures and reduced pressure indicates reduced temperatures. Indicators and alarms are provided in the Control Room to inform the operator of system status and to guide actions taken during recovery operations.

Detailed descriptions for all containment instrumentation, including diversity and redundancy considerations, are provided in Section 7.3.

TABLE 6.2.1-1

SYSTEM PARAMETERS INITIAL CONDITIONS

Parameters	Value
Core Thermal Power (MWt)	2300*
102% of Core Thermal Power (MWt)	2346*
Reactor Coolant System Total Flowrate (lbm/sec)	27027.78
Vessel Outlet Temperature (ºF) at 102% Power	610.3
Core Inlet Temperature (ºF) at 102% Power	548.5
Vessel Average Temperature (ºF)	579.4
Initial Steam Generator Steam Pressure (psia)	850
Steam Generator Design	Model 44F
Steam Generator Tube Plugging (%)	0
Initial Steam Generator Secondary Side Mass (lbm)	97,505
Assumed Maximum Containment Backpressure (psia)	56.7
Accumulator	
Water Volume (ft^3) per accumulator	841.
N_2 Cover Gas Pressure (psia)	615
Temperature (ºF)	130.0
Safety Injection Delay, total (sec) (from beginning of event)	
Minimum Safeguards	41.7
Maximum Safeguards	16.4

Note: Core Thermal Power, RCS Total Flowrate, RCS Coolant Temperature, and Steam Generator Secondary Side Mass include appropriate uncertainty and/or allowance.

* Bounds operation at 2339 MWt including the applicable calorimetric uncertainty.

TABLE 6.2.1-2

TOTAL PUMPED ECCS FLOW RATE ASSUMING A DESEL FAILURE (MINIMUM SAFEGUARDS)

INJECTION MODE (REFLOOD PHASE)

RCS Pressure (psia)	Total Flow (lbm/sec)
14.7	568.96
20.0	556.63
40.0	505.35
60.0	451.60
80.0	388.74
100.0	312.04
120.0	205.81
140.0	64.79
160.0	64.17
180.0	63.55
200.0	62.93
220.0	62.35

INJECTION MODE (POST-REFLOOD PHASE)
END OF REFLOOD TO 40.7 MINUTES

RCS Pressure (psia)	Total Flow (lbm/sec)
56.7	460.5

SWITCHOVER FROM ALL INJECTION TO SUMP RECIRCULATION
(40.7 to 50.7 minutes)

RCS Pressure (psia)	Total Flow (lbm/sec)
56.7	HHSI FROM RWST AT 89.75

PRE-PIGGYBACK LONG-TERM RECIRCULATION MODE (50.7 to 77 minutes)

RCS Pressure (psia)	Total Flow (lbm/sec)
14.7	HHSI FROM RWST AT 89.74
14.7	RHR FROM SUMP AT 515.95

PIGGYBACK LONG-TERM RECIRCULATION MODE (after 77 minutes)

RCS Pressure (psia)	Total Flow (lbm/sec)
14.7	57.67

TABLE 6.2.1-3

TOTAL PUMPED ECCS FLOW RATE ASSUMING NO FAILURE (MAXIMUM SAFEGUARDS)

INJECTION MODE (REFLOOD PHASE)

RCS Pressure (psia)	Total Flow (lbm/sec)
14.7	807.92
40.0	717.59
60.0	641.27
80.0	552.01
100.0	443.10
120.0	292.25
140.0	92.01
180.0	90.24
220.0	88.54

INJECTION MODE (POST-REFLOOD PHASE)

RCS Pressure (psia)	Total Flow (lbm/sec)
56.7	653.86

RECIRCULATION MODE

RCS Pressure (psia)	Total Flow (lbm/sec)
14.7	91.45 to 618.91 depending upon configuration prior to hot leg recirculation. 429.0 post hot leg recirculation

TABLE 6.2.1-4

DOUBLE-ENDED HOT LEG BREAK BLOWDOWN
MASS AND ENERGY RELEASES (MINIMUM SAFEGUARDS)

DELETED IN REVISION NO. 21

TABLE 6.2.1-5

DOUBLE-ENDED HOT LEG BREAK MASS BALANCE (MINIMUM SAFEGUARDS)

DELETED IN REVISION NO. 21

TABLE 6.2.1-6

DOUBLE ENDED HOT LEG BREAK ENERGY BALANCE (MINIMUM SAFEGUARDS)

DELETED IN REVISION NO. 21

TABLE 6.2.1-7
DOUBLE-ENDED PUMP SUCTION BREAK BLOWDOWN MASS AND ENERGY RELEASES (MINIMUM SAFEGUARDS)

	BREAK PATH NO.1*		BREAK PATH NO.2**	
TIME (SECONDS)	FLOW (LBM/SEC)	ENERGY THOUSANDS (BTU/SEC)	FLOW (LBM/SEC)	ENERGY THOUSANDS (BTU/SEC)
0.00	0.00	0.00	0.00	0.00
0.001	82418.19	44602.18	39601.35	21390.56
0.002	40639.91	21952.18	40313.38	21774.00
0.003	40627.76	21946.35	40070.56	21641.80
0.004	40616.84	21940.99	39801.01	21494.96
0.10	40058.45	21707.45	19703.19	10630.31
0.20	40530.30	22114.20	22157.46	11966.79
0.30	41147.08	22653.66	23365.60	12626.85
0.40	41830.86	23277.10	23338.43	12616.97
0.50	42371.84	23851.04	22738.54	12298.33
0.60	42480.94	24190.05	22196.49	12011.08
0.70	42010.25	24175.42	21861.97	11836.18
0.80	41046.64	23841.90	21736.62	11773.09
0.90	39956.69	23406.86	21639.99	11724.64
1.00	39049.50	23060.81	21506.62	11655.03
1.10	38293.16	22804.53	21316.59	11553.71
1.20	37512.04	22537.64	21105.34	11440.07
1.30	36586.27	22189.78	20899.53	11328.90
1.40	35583.02	21793.91	20714.23	11228.58
1.50	34590.18	21401.40	20538.48	11133.21
1.60	33596.96	21006.56	20355.95	11033.78
1.70	32542.14	20570.90	20174.73	10934.88
1.80	31461.52	20116.38	20009.16	10844.47
1.90	30341.21	19625.56	19853.25	10759.35
2.00	29280.06	19154.14	19692.26	10671.39
2.10	28160.12	18621.06	19525.72	10580.43
2.20	26943.93	17999.78	19257.96	10433.85
2.30	25207.86	16993.60	18785.16	10175.82
2.40	22566.53	15326.16	18343.53	9935.29
2.50	19852.93	13566.83	18056.31	9779.45
2.60	19131.96	13150.80	17765.74	9621.94
2.70	18308.06	12596.70	17442.83	9446.79
2.80	17146.59	11804.35	17123.23	9273.73
2.90	16290.06	11230.43	16791.18	9094.26
3.00	15354.76	10597.56	16451.63	8910.83

TABLE 6.2.1-7 (Cont'd)
DOUBLE-ENDED PUMP SUCTION BREAK
BLOWDOWN MASS AND ENERGY RELEASES
(MINIMUM SAFEGUARDS)

	BREAK PATH NO.1'		BREAK PATH NO.2"	
TIME (SECONDS)	FLOW (LBM/SEC)	ENERGY THOUSANDS (BTU/SEC)	FLOW (LBM/SEC)	ENERGY THOUSANDS (BTU/SEC)
3.10	14535.12	10050.58	16121.66	8733.22
3.20	13823.04	9581.73	15872.94	8600.22
3.30	13202.95	9179.35	15661.25	8487.49
3.40	12678.65	8848.01	15469.44	8385.54
3.50	12218.28	8564.61	15277.15	8283.21
3.60	11800.92	8313.25	15096.82	8187.44
3.70	11442.41	8104.06	14948.31	8109.08
3.80	11140.58	7934.71	14816.44	8039.75
3.90	10861.64	7778.26	14681.92	7968.78
4.00	10588.51	7623.28	14549.03	7898.60
4.20	10098.43	7354.43	14307.03	7771.15
4.40	9656.72	7117.22	14066.67	7644.53
4.60	9292.52	6921.90	13844.17	7527.96
4.80	9002.72	6759.28	14761.15	8036.42
5.00	8756.17	6614.52	14676.06	7991.10
5.20	8553.43	6483.35	14536.38	7919.85
5.40	8428.43	6393.64	14453.73	7879.38
5.60	8349.49	6326.44	14344.99	7824.65
5.80	8280.52	6272.60	14342.97	7829.22
6.00	8171.55	6197.21	14267.18	7791.76
6.20	8060.03	6121.86	14112.66	7708.49
6.40	7988.75	6064.32	13932.10	7608.94
6.60	7921.59	6004.75	13824.26	7548.52
6.80	7844.57	5935.24	13730.82	7495.32
7.00	7783.12	5861.89	13597.40	7419.38
7.20	7752.15	5793.91	13447.67	7334.47
7.40	7848.88	5811.74	13289.63	7245.32
7.60	8156.86	5998.52	13167.78	7176.87
7.80	7958.53	6177.93	13099.28	7137.28
8.00	7090.17	5880.63	12836.78	6990.76
8.20	6810.48	5686.88	12586.21	6853.01
8.40	6775.42	5621.51	12401.42	6753.27
8.60	6685.31	5558.22	12221.59	6656.20
8.80	6542.16	5456.80	11998.13	6534.16

TABLE 6.2.1-7 (Cont'd)
DOUBLE-ENDED PUMP SUCTION BREAK
BLOWDOWN MASS AND ENERGY RELEASES
(MINIMUM SAFEGUARDS)

	BREAK PATH NO.1*		BREAK PATH NO.2**	
TIME (SECONDS)	FLOW (LBM/SEC)	ENERGY THOUSANDS (BTU/SEC)	FLOW (LBM/SEC)	ENERGY THOUSANDS (BTU/SEC)
9.00	6424.29	5347.95	11785.26	6417.45
9.20	6334.05	5242.14	11586.34	6308.36
9.40	6257.43	5142.34	11379.43	6195.03
9.60	6182.61	5049.48	11177.94	6084.83
9.80	6103.73	4961.32	10984.22	5978.88
10.00	6018.44	4875.46	10788.42	5871.94
10.20	5927.72	4792.78	10594.85	5766.41
10.20	5927.10	4792.24	10593.61	5765.73
10.40	5831.53	4713.26	10404.17	5662.68
10.60	5729.77	4638.31	10219.65	5562.69
10.80	5616.05	4568.74	10040.62	5465.94
11.00	5490.54	4487.18	9844.30	5359.72
11.20	5375.04	4410.04	9666.98	5264.23
11.40	5254.41	4330.11	9482.16	5164.55
11.60	5116.65	4239.09	9267.17	5048.24
11.80	4973.21	4138.58	9071.32	4943.05
12.00	4833.77	4029.91	8878.56	4839.54
12.20	4704.61	3921.09	8673.66	4729.49
12.40	4580.91	3810.72	8482.92	4627.13
12.60	4459.04	3701.62	8280.12	4513.74
12.80	4338.92	3594.37	8066.39	4374.73
13.00	4222.11	3491.08	7865.48	4216.92
13.20	4123.08	3402.85	7860.68	4145.34
13.40	4029.87	3325.93	7513.03	3885.34
13.60	3936.14	3259.77	7689.33	3902.45
13.80	3835.56	3200.74	7090.55	3538.86
14.00	3739.13	3153.20	6997.74	3430.34
14.20	3634.30	3106.24	7435.01	3601.66
14.40	3537.20	3075.09	6407.96	3068.15
14.60	3438.60	3046.80	6523.18	3071.91
14.80	3333.36	3024.43	6639.55	3093.42
15.00	3229.18	3012.64	5925.55	2736.82
15.20	3108.30	2999.55	6239.28	2841.07
15.40	2968.61	2987.50	6287.32	2832.80
15.60	2761.62	2929.62	5391.88	2409.60
15.80	2511.28	2833.78	5030.09	2216.07

TABLE 6.2.1-7 (Cont'd)
DOUBLE-ENDED PUMP SUCTION BREAK
BLOWDOWN MASS AND ENERGY RELEASES
(MINIMUM SAFEGUARDS)

	BREAK PATH NO.1*		BREAK PATH NO.2**	
TIME (SECONDS)	FLOW (LBM/SEC)	ENERGY THOUSANDS (BTU/SEC)	FLOW (LBM/SEC)	ENERGY THOUSANDS (BTU/SEC)
16.00	2289.44	2703.91	4946.16	2134.44
16.20	2105.56	2559.30	4700.27	1989.86
16.40	1915.36	2354.14	4275.47	1782.24
16.60	1758.80	2172.58	3910.60	1607.26
16.80	1624.91	2012.96	3697.54	1496.14
17.00	1497.62	1859.28	3532.34	1402.69
17.20	1380.14	1716.77	3469.59	1344.31
17.40	1265.17	1576.40	3604.57	1354.96
17.60	1157.94	1445.05	3887.98	1415.83
17.80	1053.61	1316.72	4224.28	1495.68
18.00	968.15	1211.42	4432.46	1533.80
18.20	869.10	1089.01	4413.18	1500.10
18.40	772.30	969.03	4186.80	1402.66
18.60	691.41	868.30	3892.05	1288.09
18.80	613.86	771.52	3585.48	1172.84
19.00	541.27	680.76	3282.36	1060.96
19.20	476.05	599.12	2982.81	952.60
19.40	407.42	513.08	2684.82	847.32
19.60	344.04	433.55	2398.80	748.70
19.80	290.70	366.58	2183.27	674.87
20.00	253.18	319.47	1972.44	605.25
20.20	227.00	286.58	1771.92	541.77
20.40	182.08	230.05	1546.69	472.88
20.60	144.22	182.39	1276.97	391.25
20.80	102.84	130.26	870.93	267.86
21.00	63.18	80.22	0.00	0.00
21.20	34.27	43.68	0.00	0.00
21.40	25.92	33.18	0.00	0.00
21.60	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 break

TABLE 6.2.1-8
DOUBLE-ENDED PUMP SUCTION BREAK REFLOOD MASS AND ENERGY RELEASES (MINIMUM SAFEGUARDS)

	BREAK PATH NO.1*		BREAK PATH NO.2**	
TIME (SECONDS)	FLOW (LBM/SEC)	ENERGY THOUSANDS (BTU/SEC)	FLOW (LBM/SEC)	ENERGY THOUSANDS (BTU/SEC)
22.08	0.00	0.00	0.00	0.00
22.28	0.00	0.00	0.00	0.00
22.38	0.00	0.00	0.00	0.00
22.48	0.00	0.00	0.00	0.00
22.58	0.00	0.00	0.00	0.00
22.63	0.00	0.00	0.00	0.00
22.74	39.66	46.66	0.00	0.00
22.84	14.16	16.66	0.00	0.00
22.94	10.45	12.30	0.00	0.00
23.04	13.93	16.39	0.00	0.00
23.14	18.10	21.30	0.00	0.00
23.24	22.68	26.68	0.00	0.00
23.34	27.90	32.83	0.00	0.00
23.44	32.15	37.83	0.00	0.00
23.54	36.04	42.40	0.00	0.00
23.64	39.77	46.80	0.00	0.00
23.74	43.35	51.01	0.00	0.00
23.86	46.30	54.48	0.00	0.00
23.96	49.58	58.34	0.00	0.00
24.06	52.19	61.41	0.00	0.00
24.16	54.69	64.36	0.00	0.00
24.26	57.12	67.21	0.00	0.00
24.36	59.47	69.98	0.00	0.00
24.46	61.75	72.66	0.00	0.00
24.56	63.96	75.27	0.00	0.00
24.66	66.12	77.81	0.00	0.00
25.66	85.12	100.18	0.00	0.00
26.66	100.90	118.79	0.00	0.00
27.66	114.63	134.97	0.00	0.00
28.66	126.82	149.35	0.00	0.00
29.34	134.38	158.27	0.00	0.00
29.66	156.97	184.94	1043.10	152.23
30.74	273.05	322.41	3122.65	474.68
31.74	277.02	327.13	3164.09	487.31
32.74	272.81	322.13	3107.67	481.33

TABLE 6.2.1-8 (Cont'd)
DOUBLE-ENDED PUMP SUCTION BREAK REFLOOD
MASS AND ENERGY RELEASES
(MINIMUM SAFEGUARDS)

	BREAK PATH NO.1*		BREAK PATH NO.2**	
TIME (SECONDS)	FLOW (LBM/SEC)	ENERGY THOUSANDS (BTU/SEC)	FLOW (LBM/SEC)	ENERGY THOUSANDS (BTU/SEC)
33.74	268.32	316.81	3047.59	474.50
34.74	263.94	311.61	2988.45	467.69
34.84	263.51	311.10	2982.62	467.02
35.74	259.71	306.59	2930.80	461.00
36.74	255.63	301.74	2874.81	454.47
37.74	251.70	297.09	2820.52	448.10
38.74	247.93	292.61	2767.93	441.92
39.74	244.31	288.31	2717.00	435.91
40.74	240.82	284.18	2667.68	430.07
41.54	238.13	280.98	2629.33	425.52
41.74	237.47	280.20	2619.90	424.40
42.74	234.24	276.37	2573.59	418.89
43.74	231.13	272.69	2528.68	413.53
44.74	228.13	269.13	2485.11	408.33
45.74	225.24	265.71	2442.80	403.26
46.79	196.31	231.45	1974.98	335.39
47.79	194.71	229.55	1947.26	332.03
48.79	193.01	227.55	1920.82	328.66
49.39	268.91	317.45	262.51	151.23
49.79	277.67	327.90	265.80	156.79
50.79	274.84	324.55	264.48	155.02
51.79	271.15	320.16	262.84	152.71
52.79	267.52	315.85	261.23	150.44
53.79	263.95	311.62	259.65	148.22
54.79	260.44	307.45	258.10	146.04
55.79	256.97	303.33	256.57	143.90
56.69	253.85	299.64	255.21	141.98
56.79	253.50	299.22	255.05	141.77
57.79	250.07	295.15	253.55	139.66
58.79	246.90	291.39	252.16	137.72
59.79	243.78	287.69	250.80	135.82
60.79	240.72	284.06	249.47	133.96
61.79	237.70	280.48	248.16	132.14

TABLE 6.2.1-8 (Cont'd)
DOUBLE-ENDED PUMP SUCTION BREAK REFLOOD MASS AND ENERGY RELEASES (MINIMUM SAFEGUARDS)

	BREAK PATH NO.1'		BREAK PATH NO.2"	
TIME (SECONDS)	FLOW (LBM/SEC)	ENERGY THOUSANDS (BTU/SEC)	FLOW (LBM/SEC)	ENERGY THOUSANDS (BTU/SEC)
62.79	234.73	276.96	246.88	130.35
63.79	231.82	273.50	245.62	128.60
64.79	228.95	270.11	244.38	126.89
65.79	226.13	266.76	243.17	125.21
66.79	223.36	263.48	241.99	123.57
67.79	220.63	260.25	240.82	121.96
68.79	217.96	257.08	239.68	120.39
69.79	215.32	253.96	238.56	118.84
70.79	212.74	250.90	237.47	117.33
71.79	210.20	247.89	236.40	115.86
72.79	207.70	244.94	235.35	114.41
73.69	205.50	242.33	234.42	113.14
73.79	205.26	242.04	234.32	113.00
74.79	202.85	239.19	233.31	111.62
75.79	200.49	236.40	232.33	110.27
76.79	198.18	233.66	231.37	108.95
77.79	195.91	230.97	230.43	107.66
78.79	193.68	228.34	229.51	106.40
79.79	191.50	225.76	228.61	105.17
80.79	189.36	223.22	227.73	103.97
81.79	187.25	220.74	226.87	102.80
82.79	185.20	218.31	226.03	101.65
84.79	181.21	213.59	224.41	99.45
86.79	177.40	209.08	222.87	97.35
88.79	173.75	204.77	221.40	95.36
90.79	170.27	200.66	220.02	93.48
92.79	166.96	196.74	218.70	91.70
94.29	164.58	193.93	217.76	90.43
94.79	163.81	193.02	217.46	90.02
96.79	160.81	189.48	216.28	88.43
98.79	157.97	186.13	215.18	86.94
100.79	155.28	182.95	214.13	85.53

TABLE 6.2.1-8 (Cont'd)
DOUBLE-ENDED PUMP SUCTION BREAK REFLOOD MASS AND ENERGY RELEASES (MINIMUM SAFEGUARDS)

	BREAK PATH NO.1'		BREAK PATH NO.2"	
TIME (SECONDS)	FLOW (LBM/SEC)	ENERGY THOUSANDS (BTU/SEC)	FLOW (LBM/SEC)	ENERGY THOUSANDS (BTU/SEC)
102.79	152.73	179.93	213.15	84.21
104.79	150.32	177.09	212.22	82.96
106.79	148.05	174.41	211.35	81.80
108.79	145.94	171.91	210.52	80.72
110.79	143.94	169.56	209.75	79.70
112.79	142.07	167.35	209.03	78.76
114.79	140.32	165.29	208.36	77.88
116.79	138.69	163.35	207.74	77.05
118.79	137.16	161.55	207.15	76.29
120.79	135.74	159.87	206.61	75.58
122.79	134.41	158.31	206.11	74.92
124.79	133.19	156.86	205.65	74.32
126.79	132.05	155.53	205.22	73.75
128.79	131.01	154.29	204.82	73.24
130.79	130.04	153.15	204.46	72.76
132.79	129.15	152.10	204.12	72.32
134.79	128.34	151.14	203.82	71.92
136.79	127.60	150.27	203.54	71.55
138.79	126.92	149.47	203.28	71.22
140.79	126.31	148.74	203.05	70.91
142.79	125.74	148.07	202.83	70.63
144.79	125.23	147.47	202.64	70.38
146.49	124.83	147.01	202.49	70.18
146.79	124.77	146.93	202.46	70.15
148.79	124.36	146.45	202.30	69.94
150.79	124.00	146.02	202.16	69.76
152.79	123.68	145.64	202.04	69.60
154.79	123.40	145.32	201.93	69.45
156.79	123.16	145.03	201.83	69.33
158.79	122.96	144.79	201.75	69.22
160.79	122.79	144.59	201.68	69.13
162.79	122.65	144.43	201.62	69.05
164.79	122.54	144.30	201.57	68.98
166.79	122.46	144.21	201.53	68.93

TABLE 6.2.1-8 (Cont'd)
DOUBLE-ENDED PUMP SUCTION BREAK REFLOOD
MASS AND ENERGY RELEASES
(MINIMUM SAFEGUARDS)

	BREAK PATH NO.1*		BREAK PATH NO.2**	
TIME (SECONDS)	FLOW (LBM/SEC)	ENERGY THOUSANDS (BTU/SEC)	FLOW (LBM/SEC)	ENERGY THOUSANDS (BTU/SEC)
168.79	122.41	144.14	201.50	68.89
170.79	122.38	144.11	201.48	68.87
172.79	122.37	144.10	201.47	68.85
174.79	122.38	144.11	201.46	68.84
176.69	122.41	144.15	201.46	68.84
176.79	122.41	144.15	201.46	68.84
178.79	122.46	144.21	201.47	68.85
180.79	122.53	144.29	201.48	68.87
182.79	122.61	144.39	201.50	68.90
184.79	122.71	144.50	201.53	68.93
186.79	122.82	144.63	201.56	68.97
188.79	122.95	144.78	201.59	69.01
190.79	123.22	145.10	201.68	69.13
192.79	123.72	145.69	202.26	69.42
194.79	124.17	146.22	203.30	69.78
196.79	124.64	146.78	204.81	70.23
198.79	125.12	147.35	206.71	70.77
200.79	125.59	147.90	208.94	71.35
202.79	126.02	148.41	211.43	71.97
204.79	126.40	148.85	214.12	72.61
206.79	126.71	149.21	216.98	73.26
208.59	126.92	149.47	219.66	73.83

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

TABLE 6.2.1-9

DOUBLE-ENDED PUMP SUCTION BREAK PRINCIPLE PARAMETERS DURING REFLOOD (MINIMUM SAFEGUARDS)

	FLOODING						INJECTION			
Time (Seconds)	Temp (Deg-F)	Rate (in/Sec)	Carryover Fraction	Core Height (Ft)	Downcomer Height (Ft)	Flow Frac	Total (Lb_m/Sec)	Accum (Lb_m/Sec)	Spill (Lb_m/Sec)	Enthalpy (Btu/Lb_m)
21.6	194.2	0.000	0.000	0.00	0.00	0.333	0.0	0.0	0.0	0.00
22.4	192.2	22.234	0.000	0.67	1.11	0.000	5069.7	5069.7	0.0	99.50
22.6	191.2	23.226	0.000	1.05	1.12	0.000	5034.9	5034.9	0.0	99.50
22.9	190.8	2.253	0.096	1.31	1.68	0.232	4957.0	4957.0	0.0	99.50
23.1	190.8	2.427	0.127	1.34	2.11	0.285	4924.3	4924.3	0.0	99.50
24.2	191.0	2.275	0.292	1.50	4.30	0.395	4761.4	4761.4	0.0	99.50
25.7	191.4	2.214	0.443	1.67	7.43	0.417	4551.1	4551.1	0.0	99.50
29.3	192.5	2.481	0.604	2.00	14.59	0.431	4117.2	4117.2	0.0	99.50
30.7	192.9	3.487	0.644	2.13	15.56	0.585	3799.3	3799.3	0.0	99.50
32.7	193.5	3.325	0.673	2.33	15.57	0.583	3612.5	3612.5	0.0	99.50
34.8	194.3	3.167	0.690	2.51	15.57	0.578	3455.5	3455.5	0.0	99.50
41.5	197.1	2.860	0.711	3.00	15.57	0.561	3042.4	3042.4	0.0	99.50
48.8	200.8	2.489	0.717	3.47	15.57	0.519	2259.7	1811.6	0.0	93.26
49.4	201.1	3.045	0.723	3.51	15.53	0.594	427.3	0.0	0.0	68.04
49.8	201.3	3.087	0.724	3.53	15.47	0.597	422.7	0.0	0.0	68.04
56.7	205.8	2.834	0.726	4.00	14.61	0.591	428.9	0.0	0.0	68.04
64.8	211.9	2.583	0.727	4.50	13.82	0.585	435.1	0.0	0.0	68.04
73.7	219.1	2.352	0.727	5.00	13.19	0.576	440.5	0.0	0.0	68.04
84.8	228.3	2.114	0.727	5.56	12.70	0.565	445.6	0.0	0.0	68.04
94.3	235.0	1.954	0.728	6.00	12.49	0.555	448.8	0.0	0.0	68.04

TABLE 6.2.1-9 (Cont'd)

DOUBLE-ENDED PUMP SUCTION BREAK PRINCIPLE PARAMETERS DURING REFLOOD (MINIMUM SAFEGUARDS)

	FLOODING						INJECTION			
Time (Seconds)	Temp (Deg-F)	Rate (in/Sec)	Carryover Fraction	Core Height (Ft)	Dwncomer Height (Ft)	Flow Frac	Total (Lb_m/Sec)	Accum (Lb_m/Sec)	Spill (Lb_m/Sec)	Enthalpy (Btu/Lb_m)
106.8	242.4	1.795	0.728	6.53	12.43	0.543	451.7	0.0	0.0	68.04
118.8	248.4	1.690	0.730	7.00	12.54	0.534	453.2	0.0	0.0	68.04
132.8	254.5	1.609	0.732	7.52	12.82	0.526	454.3	0.0	0.0	68.04
146.5	259.6	1.561	0.736	8.00	13.17	0.521	454.9	0.0	0.0	68.04
162.8	264.9	1.530	0.741	8.55	13.67	0.519	455.2	0.0	0.0	68.04
176.7	268.9	1.518	0.746	9.00	14.12	0.519	455.2	0.0	0.0	68.04
188.8	272.0	1.512	0.751	9.39	14.53	0.520	455.2	0.0	0.0	68.04
192.8	273.0	1.516	0.752	9.51	14.66	0.521	455.1	0.0	0.0	68.04
204.8	275.7	1.523	0.757	9.89	15.02	0.526	454.7	0.0	0.0	68.04
208.6	276.5	1.521	0.759	10.00	15.11	0.528	454.6	0.0	0.0	68.04

TABLE 6.2.1-10
DOUBLE ENDED PUMP SUCTION BREAK POST-REFLOOD MASS AND ENERGY RELEASES (MINIMUM SAFEGUARDS)

	BREAK PATH NO.1*		BREAK PATH NO.2**	
TIME (SECONDS)	FLOW (LBM/SEC)	ENERGY THOUSANDS (BTU/SEC)	FLOW (LBM/SEC)	ENERGY THOUSANDS (BTU/SEC)
208.60	109.12	135.57	351.32	86.21
213.60	108.79	135.15	351.66	86.16
218.60	109.16	135.62	351.28	85.94
223.60	108.82	135.20	351.62	85.90
228.60	109.19	135.65	351.26	85.67
233.60	108.84	135.22	351.61	85.63
238.60	108.49	134.79	351.95	85.59
243.60	108.84	135.23	351.60	85.37
248.60	108.49	134.79	351.95	85.33
253.60	108.14	134.35	352.31	85.29
258.60	108.48	134.77	351.97	85.07
263.60	108.12	134.32	352.33	85.03
268.60	108.45	134.73	352.00	84.81
273.60	108.08	134.28	352.37	84.77
278.60	108.40	134.67	352.05	84.56
283.60	108.03	134.21	352.42	84.52
288.60	107.65	133.75	352.79	84.48
293.60	107.96	134.12	352.49	81.69
298.60	107.58	133.65	352.87	81.66
303.60	107.87	134.01	352.58	81.46
308.60	107.48	133.53	352.96	81.43
313.60	107.76	133.88	352.68	81.23
318.60	107.37	133.39	353.08	81.21
323.60	107.63	133.72	352.81	81.01
328.60	107.23	133.22	353.21	80.99
333.60	107.49	133.54	352.96	80.79
338.60	107.08	133.03	353.37	80.77
343.60	107.32	133.33	353.12	80.58
348.60	106.91	132.82	353.54	80.56
353.60	107.13	133.10	353.31	80.37
358.60	106.71	132.57	353.74	80.35
363.60	106.92	132.84	353.52	80.17
368.60	106.49	132.30	353.96	80.15
373.60	106.69	132.54	353.76	79.97
378.60	106.25	132.00	354.20	79.95

TABLE 6.2.1-10 (Cont'd)
DOUBLE ENDED PUMP SUCTION BREAK POST-REFLOOD MASS AND ENERGY RELEASES (MINIMUM SAFEGUARDS)

	BREAK PATH NO.1*		BREAK PATH NO.2**	
TIME (SECONDS)	FLOW (LBM/SEC)	ENERGY THOUSANDS (BTU/SEC)	FLOW (LBM/SEC)	ENERGY THOUSANDS (BTU/SEC)
383.60	106.43	132.22	354.02	79.77
388.60	105.98	131.66	354.47	79.76
393.60	106.14	131.87	354.30	79.58
398.60	106.30	132.06	354.15	79.41
403.60	105.90	131.56	354.55	79.39
408.60	106.13	131.85	354.32	79.20
413.60	105.74	131.37	354.71	79.17
418.60	105.95	131.63	354.50	78.99
423.60	105.55	131.13	354.89	78.97
428.60	105.74	131.37	354.70	78.79
433.60	105.92	131.60	354.52	78.61
438.60	105.51	131.08	354.94	78.59
443.60	105.67	131.28	354.78	78.42
448.60	105.81	131.46	354.63	78.25
453.60	105.37	130.91	355.07	78.24
458.60	105.50	131.07	354.94	78.07
463.60	105.61	131.21	354.83	77.91
468.60	105.15	130.63	355.30	77.90
473.60	105.24	130.74	355.21	77.75
478.60	105.31	130.84	355.13	77.60
483.60	105.37	130.91	355.07	77.45
488.60	105.41	130.96	355.03	77.31
493.60	104.90	130.32	355.55	77.31
498.60	104.91	130.34	355.53	77.17
503.60	104.91	130.34	355.53	77.04
508.60	104.89	130.32	355.55	76.91
513.60	104.86	130.27	355.59	76.79
518.60	104.80	130.20	355.65	76.67
523.60	104.72	130.10	355.72	76.55
528.60	104.62	129.98	355.82	76.44
533.60	104.50	129.83	355.94	76.34
538.60	104.85	130.27	355.59	76.11
543.60	104.68	130.05	355.77	76.02

TABLE 6.2.1-10 (Cont'd)
DOUBLE ENDED PUMP SUCTION BREAK POST-REFLOOD MASS AND ENERGY RELEASES (MINIMUM SAFEGUARDS)

	BREAK PATH NO.1*		BREAK PATH NO.2**	
TIME (SECONDS)	FLOW (LBM/SEC)	ENERGY THOUSANDS (BTU/SEC)	FLOW (LBM/SEC)	ENERGY THOUSANDS (BTU/SEC)
548.60	104.48	129.80	355.96	75.94
553.60	104.26	129.53	356.19	75.86
558.60	104.48	129.80	355.97	75.67
563.60	104.19	129.44	356.26	75.61
568.60	104.33	129.61	356.12	75.43
573.60	104.42	129.73	356.03	75.27
578.60	104.01	129.22	356.43	75.24
583.60	104.01	129.22	356.43	75.10
588.60	103.95	129.15	356.49	74.98
593.60	103.84	129.00	356.61	74.87
598.60	104.07	129.30	356.37	74.67
603.60	103.83	129.00	356.61	74.59
608.60	103.93	129.12	356.51	74.43
613.60	103.92	129.11	356.52	74.29
618.60	103.80	128.96	356.64	74.19
623.60	103.93	129.12	356.51	74.01
628.60	103.54	128.64	356.90	73.97
633.60	103.71	128.84	356.74	73.79
638.60	103.64	128.76	356.81	73.67
643.60	103.63	128.75	356.82	73.53
648.60	103.60	128.71	356.85	73.40
653.60	103.43	128.50	357.02	73.30
658.60	103.53	128.62	356.92	73.13
663.60	103.28	128.32	357.16	73.06
668.60	103.20	128.22	357.24	72.93
673.60	103.30	128.34	357.15	72.77
678.60	103.10	128.09	357.34	72.67
989.17	103.10	128.09	357.34	72.67
989.27	57.93	71.21	402.51	78.75
993.60	57.87	71.13	402.58	78.60
1529.80	57.87	71.13	402.58	78.60
1529.90	52.21	60.07	408.23	35.80
2442.00	46.81	53.86	413.64	36.77
2442.10	46.81	53.86	42.97	19.36
3042.00	44.38	51.06	45.40	19.80

TABLE 6.2.1-10 (Cont'd)
DOUBLE ENDED PUMP SUCTION BREAK POST-REFLOOD MASS AND ENERGY RELEASES (MINIMUM SAFEGUARDS)

	BREAK PATH NO.1*		BREAK PATH NO.2**	
TIME (SECONDS)	FLOW (LBM/SEC)	ENERGY THOUSANDS (BTU/SEC)	FLOW (LBM/SEC)	ENERGY THOUSANDS (BTU/SEC)
3042.10	44.38	51.06	561.87	130.16
3600.00	42.12	48.47	564.12	130.57
3600.10	34.71	39.94	571.54	117.16
4620.00	31.82	36.61	574.42	117.76
4620.10	29.95	34.46	27.72	4.05
6000.00	27.38	31.50	30.29	4.42
6000.10	27.36	31.49	30.31	4.41
10000.00	23.67	27.24	34.00	4.95
39600.00	17.58	20.23	40.09	5.83
39600.10	17.57	20.22	40.10	5.81
100000.00	13.70	15.76	43.97	6.38
100000.10	13.63	15.68	44.04	6.17
500000.00	7.86	9.04	49.81	6.97
500000.09	7.82	9.00	49.85	6.73
1000000.00	5.76	6.62	51.91	7.01
10000000.00	1.92	2.21	55.75	7.53

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

TABLE 6.2.1.11

DOUBLE-ENDED PUMP SUCTION BREAK
MASS BALANCE (MINIMUM SAFEGUARDS)

MASS BALANCE								
	Time (Seconds)	.00	21.60	21.60	208.59	959.27	1529.80	3600.00
		MASS (THOUSAND LBM)						
Initial	In RCS and Accumulator	557.66	557.66	557.66	557.66	557.66	557.66	557.66
Added Mass	Pumped Injection	.00	.00	.00	72.83	432.28	681.17	1493.34
	Total Added	.00	.00	.00	72.83	432.28	681.17	1493.34
*** Total Available ***		557.66	557.66	557.66	630.49	959.95	1238.83	2051.00
Distribution	Reactor Coolant	401.78	35.30	55.10	108.61	108.61	108.61	108.61
	Accumulator	155.89	115.87	96.07	.00	.00	.00	.00
	Total Contents	557.66	151.17	151.17	108.61	108.61	108.61	108.61
Effluent	Break Flow	.00	406.48	406.48	521.86	881.32	1130.21	1942.37
	ECCS Spill	.00	.00	.00	.00	.00	.00	.00
	Total Effluent	.00	406.48	406.48	521.86	881.32	1130.21	1942.37
*** Total Accountable ***		557.66	557.65	557.65	630.48	989.93	1238.82	2050.98

TABLE 6.2.1-12

DOUBLE -ENDED PUMP SUCTION BREAK ENERGY BALANCE

(MINIMUM SAFEGUARDS)

		ENERGY BALANCE						
	Time (Seconds)	0.00	21.60	21.60	208.59	989.27	1529.80	3600.00
		ENERGY (MILLION BTU)						
Initial Energy	In RCS, Accumulator and Steam Generator	601.60	601.60	601.60	601.60	601.60	601.60	601.60
Added Energy	Pumped Injection	0.00	0.00	0.00	4.96	29.41	46.35	152.62
	Decay Heat	0.00	4.18	4.18	18.60	61.71	86.13	162.34
	Heat From Secondary	0.00	12.01	12.01	12.01	12.02	12.02	12.02
	Total Added	0.00	16.20	16.20	35.56	103.14	144.49	326.98
*** Total Available ***		601.60	617.80	617.80	637.16	704.74	746.10	928.59
Distribution	Reactor Coolant	234.47	8.11	10.08	28.51	28.51	28.51	28.51
	Accumulator	14.47	10.75	8.78	0.00	0.00	0.00	0.00
	Core Stored	20.15	11.04	11.04	3.82	3.81	3.58	2.68
	Primary Metal	135.49	129.16	129.16	97.36	64.45	54.93	40.04
	Secondary Metal	36.71	36.21	36.21	32.79	23.24	18.73	13.42
	Steam Generator	160.31	175.28	175.28	154.07	104.37	84.07	59.76
	Total Contents	601.60	370.56	370.56	316.55	224.39	189.82	144.41
Effluent	Break Flow	0.00	246.77	246.77	303.93	463.67	546.07	774.86
	ECCS Spill	0.00	0.00	0.00	0.00	0.00	0.00	0.00
	Total Effluent	0.00	246.77	246.77	303.93	463.67	546.07	774.86
*** Total Accountable ***		601.60	617.33	617.33	620.48	688.06	735.88	919.27

TABLE 6.2.1-13

DOUBLE-ENDED PUMP SUCTION BREAK BLOWDOWN
MASS AND ENERGY RELEASES (MAXIMUM SAFEGUARDS)

DELETED IN REVISION NO. 21

TABLE 6.2.1-14

DOUBLE-ENDED PUMP SUCTION BREAK REFLOOD
MASS AND ENERGY RELEASES (MAXIMUM SAFEGUARDS)

DELETED IN REVISION NO. 21

TABLE 6.2.1-15

DOUBLE-ENDED PUMP SUCTION BREAK PRINCIPLE
PARAMETERS DURING REFLOOD (MAXIMUM SAFEGUARDS)

DELETED IN REVISION NO. 21

TABLE 6.2.1-16

DOUBLE-ENDED PUMP SUCTION BREAK POST-REFLOOD
MASS AND ENERGY RELEASES (MAXIMUM SAFEGUARDS)

DELETED IN REVISION NO. 21

TABLE 6.2.1-17

DOUBLE-ENDED PUMP SUCTION BREAK MASS BALANCE (MAXIMUM SAFEGUARDS)

DELETED IN REVISION NO. 21

TABLE 6.2.1-18

DOUBLE-ENDED PUMP SUCTION BREAK ENERGY BALANCE (MAXIMUM SAFEGUARDS)

DELETED IN REVISION NO. 21

TABLE 6.2.1-19
DECAY HEAT CURVE 1979 ANS BASED ON PLANT SPECIFIC PARAMETERS PLUS 2 SIGMA UNCERTAINTY

Time (sec)	Decay Heat Generation Rate (P/P_0)
1.00E+01	0.050819
1.40E+01	0.048327
2.00E+01	0.045714
4.00E+01	0.040727
6.00E+01	0.037864
8.00E+01	0.035869
1.00E+02	0.034368
1.40E+02	0.032221
2.00E+02	0.030105
4.00E+02	0.026397
6.00E+02	0.024260
8.00E+02	0.022721
1.00E+03	0.021481
1.40E+03	0.019594
2.00E+03	0.017581
4.00E+03	0.014052
6.00E+03	0.012366
8.00E+03	0.011379
1.00E+04	0.010698
1.40E+04	0.010242
2.00E+04	0.009389
4.00E+04	0.007914
6.00E+04	0.007132
8.00E+04	0.006593
1.00E+05	0.006194
1.40E+05	0.005600
2.00E+05	0.004993
4.00E+05	0.003870
6.00E+05	0.003274
8.00E+05	0.002892
1.00E+06	0.002629
1.40E+06	0.002275
2.00E+06	0.001947
4.00E+06	0.001401
6.00E+06	0.001146
8.00E+06	0.000989
1.00E+07	0.000877

TABLE 6.2.1-20

LOCA CONTAINMENT RESPONSE ANALYSIS PARAMETERS

Service water temperature (°F)	100
RWST water temperature (°F)	100
Initial containment temperature (°F)	130
Initial containment pressure (psia)	15.7
Initial relative humidity (%)	40
Net free volume (ft^3)	2.013x 10^6
Reactor Containment Fan Coolers	
Total	4
Analysis maximum	4
Analysis minimum	2
Containment High setpoint (psig)	5.5
Delay time (sec)	
With Offsite Power	35.4
Without Offsite Power	46.0
Containment Spray Pumps	
Total	2
Analysis maximum	1
Analysis minimum	1
Flowrate (gpm)	
Injection phase (per pump)- See Table 6.2.1-22	932
Recirculation phase (total)	932
Containment High High setpoint (psig)	12
Delay time (sec)	
With Offsite Power (delay after High High setpoint)	23.5
Without Offsite Power (total time from t=0)	38.2
ECCS Recirculation Switchover, sec	
Minimum Safeguards	2442.
Maximum Safeguards	1824.
Containment Spray Termination time, (sec)	
Minimum Safeguards	39600.
Maximum Safeguards	39600.

TABLE 6.2.1-20 (Cont'd) LOCA CONTAINMENT RESPONSE ANALYSIS PARAMETERS	
Emergency Core Cooling System (ECCS) Flows	
Minimum ECCS – (gpm)	
Injection alignment	4119.
Recirculation alignment	3819.
Piggyback alignment	425.
Maximum ECCS – (gpm)	
Injection alignment	5838.
Recirculation alignment	3877.
Piggyback alignment	1052.
Residual Heat Removal System	
RHR Heat Exchangers	
Modeled in analysis *	1
Recirculation switchover time, sec	
Minimum ECCS	2442.
UA, 10^6 * BTU/hr-°F	29.4
Flows – Tube Side and Shell Side – gpm	
PRE-PIGGYBACK LONG-TERM RECIRCULATION MODE (50.7 to 77 min)	
Minimum ECCS Tube Side	3819.0
Qsump shellside *	8970.
PIGGYBACK LONG-TERM RECIRCULATION MODE (77 to 100 min)	
Minimum ECCS Tube Side	425.
Qsump shellside *	8970.
PIGGYBACK LONG-TERM RECIRCULATION MODE PLUS SPRAY (100 min to 11 hr)	
Minimum ECCS Tube Side (425 gpm SI + 932 gpm spray)	1357.
Qsump shellside *	8970.
PIGGYBACK LONG-TERM RECIRCULATION MODE (after 11 hr)	
Minimum ECCS Tube Side	425.
Qsump shellside *	8970.

TABLE 6.2.1-21
CONTAINMENT FAN COOLER PERFORMANCE

Containment Temperature (ºF)	Heat Removal Rate [Btu/sec] Per Reactor Containment Air Recirculation Fan Cooler
130	1820.44
152	3448.69
200	7459.49
263	13112.10
300	16538.24

TABLE 6.2.1-22

CONTAINMENT SPRAY PERFORMANCE

Containment Pressure (psig)	With 1 Pump (gpm)
0	932.
10	932.
20	932.
30	932.
42	932.

TABLE 6.2.1-23

CONTAINMENT HEAT SINKS

No.	Material	Heat Transfer Area ft^2	Thickness ft
1	Containment Cylinder	46,926	
	Stainless Steel		0.00158
	Insulation & Epoxy		0.1045
	Carbon Steel		0.03285
	Concrete		3.5
2	Additional Insulated Portion of the Containment Cylinder	2819.	
	Stainless Steel (foil)		0.001583
	Insulation		0.104167
	Epoxy		0.0005
	Carbon Steel		0.09375
	Concrete		3.5
3	Containment Dome	6,456	
	Stainless Steel		0.00158
	Insulation & Epoxy		0.1045
	Carbon Steel		0.0417
	Concrete		2.5
4	Containment Dome	20,094	
	Epoxy		0.0005
	Carbon Steel		0.0417
	Concrete		2.5
5	Interior Unlined Concrete	59846	
	Epoxy		0.001297
	Concrete		1.97
6	Interior Unlined Concrete (W/internal steel) Flooded	3659	
	Epoxy		0.00292
	Concrete		1.74
	Carbon Steel		0.0221
	Concrete		8.46
7	Interior Unlined Concrete (W/internal Steel) Dry	7318	
	Epoxy		0.00292
	Concrete		1.74
	Carbon Steel		0.0221
	Concrete		8.46

TABLE 6.2.1-23 (CONTINUED)

CONTAINMENT HEAT SINKS

No.	Material	Heat Transfer Area ft^2	Thickness ft
8	Interior Lined Concrete	8847	
	Stainless Steel		0.00198
	Concrete		3.388
9	Structural and Misc Exposed Steel	102261	
	Epoxy coated carbon steel		
	Epoxy		0.000583
	Carbon Steel		0.035065
10	Structural and Misc Exposed Steel	2708	
	Bare Stainless Steel		
	Stainless Steel		0.01425
11	Galvanized Steel	54865	
	Zinc		0.0000833
	Carbon Steel		0.01102
12	Insulted Copper Cable (Used for EQ Calc only)	0.059	
	Hyplon		0.00125
	EPR		0.0025
	Copper		0.005667
13	Carbon Steel Plate (Used for EQ)	0.0872	
	Carbon Steel		0.005208

TABLE 6.2.1-24

THERMOPHYSICAL PROPERTIES OF CONTAINMENT HEAT SINKS

Material	Thermal Conductivity (Btu/hr-ft - °F)	Volumetric Heat Capacity (Btu/ft^3 - °F)
Stainless Steel	9.4	60.1
Carbon Steel	29.53	56.9
Zinc	65.3	40.7
Concrete	1.05	22.5
Insulation & Epoxy	0.0188	0.58
Epoxy	0.23	18.3
Hyplon	0.125	32.537
EPR	0.1445	20.5
Copper	219.0	50.778
Carbon Steel (EQ component)	27.0	48.02

TABLE 6.2.1-25

DOUBLE-ENDED PUMP SUCTION BREAK SEQUENCE OF EVENTS (MINIMUM SAFEGUARDS)

Time (sec)	Event Description
0.0	Break Occurs, Reactor Trip and Loss of Offsite Power are assumed
0.73	Containment HI-1 Pressure Setpoint Reached
1.89	Containment HI-2 Pressure Setpoint Reached
4.80	Low Pressurizer Pressure SI Setpoint = 1661.4 psia Reached
12.30	Broken Loop Accumulator Begins Injecting Water
12.50	Intact Loop Accumulator Begins Injecting Water
21.60	End of Blowdown Phase
24.80	Main Feedwater Fully Isolated
40.09	Containment Spray Pump(s) (RWST) start
46.44	Broken Loop Accumulator Water Injection Ends
46.50	Safety Injection Begins
46.73	Reactor Containment Fan Coolers Actuate
48.89	Intact Loop Accumulator Water Injection Ends
208.59	End of Reflood Phase
989.2	Peak Pressure and Temperature Occur
989.27	Mass and Energy Release Assumption: Broken Loop SG Equilibration to 56.1 psia
1529.80	Mass and Energy Release Assumption: Intact Loop SG Equilibration to 55.5 psia
2442.00	RHR stopped for alignment to cold leg recirculation
3042.00	RHR restarts in cold leg recirculation alignment
4620.00	High Pressure SI stopped in preparation for piggyback operation
6000.00	High Pressure SI restart in piggyback alignment
39600.00	ECCS is aligned for Hot Leg Recirculation
10E+7	Transient Modeling Terminated

TABLE 6.2.1-26

DOUBLE-ENDED PUMP SUCTION BREAK SEQUENCE OF EVENTS (MAXIMUM SAFEGUARDS)

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TABLE 6.2.1-27

DOUBLE-ENDED HOT LEG BREAK SEQUENCE OF EVENTS (MINIMUM SAFEGUARDS)

DELETED IN REVISION NO. 21

TABLE 6.2.1-28

LOCA CONTAINMENT RESPONSE RESULTS (LOSS OF OFFSITE POWER ASSUMED)

CASE	PEAK PRESS. (psig)	PEAK STEAM TEMP. (°F)	PRESSURE (psig) @ 24 hours	LIQUID TEMPERATURE (°F) @ 24 hours
DEPS MINSI	41.8 at 989.2 sec	265.8 at 989.2 sec	13.0 at 86,400 sec	191.7 at 86,400 sec
DEPS MAXSI	NA	NA	NA	NA
DEHL MINSI (30% Relative Humidity Case)	NA	NA	NA	NA
DEHL MAXSI	NA	NA	NA	NA

TABLE 6.2.1.4-1

SYSTEM PARAMETERS INITIAL CONDITIONS

Parameters	Value
Core Thermal Power (MWt)	2300*
102% of Core Thermal Power (MWt)	2346*
Reactor Coolant System Total Flow rate (lbm/sec)	27027.78
Vessel Outlet Temperature (ºF) at 102% Power	610.3
Core Inlet Temperature (ºF) at 102% Power	548.5
Vessel Average Temperature (ºF) at 102% Power	579.4
Vessel No Load Average Temperature (°F)	547.0
Initial Steam Generator Steam Pressure (psia) at 102% Power	850
Steam Generator Design	Model 44F
Steam Generator Tube Plugging (%)	0
Initial Steam Generator Secondary Side Mass (lbm)	
Intact SG at 102% Power	88,641
Faulted SG at 102% Power	94,503
Intact SG at Hot Zero Power	135,000
Faulted SG at Hot Zero Power	137,294
Assumed Maximum Containment Backpressure (psia)	14.7
Accumulator	
Water Volume (ft^3) per accumulator	841.
N_2 Cover Gas Pressure (psia)	615
Temperature (ºF)	130.0
Safety Injection Delay, total (sec) (from beginning of event)	
Minimum Safeguards	41.7

Note: Core Thermal Power, RCS Total Flow rate, RCS Coolant Temperature, and Steam Generator Secondary Side Mass include appropriate uncertainty and/or allowance.

* Bounds operation at 2339 MWt including the applicable calorimetric uncertainty.

TABLE 6.2.1.4-2

102% of 2300 MWt MAIN STEAMLINE BREAK WITH CHECK VALVE FAILURE, MASS AND ENERGY RELEASES

TIME (SEC)	MASS FLOW (LBM/SEC)	ENERGY RELEASE (BTUs/SEC)
.00000	.000000	.000000000
.20000	5184.61	6203624.00
.40000	5126.28	6134872.50
.60000	5087.82	6089828.00
10.000	4175.04	5012488.50
12.000	4037.11	4848898.0
14.000	3223.54	3875781.2
15.600	2202.00	2649272.2
16.000	1946.65	2342660.0
20.000	1339.09	1612887.1
32.000	1010.96	1217071.8
40.000	892.520	1073807.2
50.000	797.060	958228.50
60.000	733.037	880665.62
70.000	687.609	825603.87
80.000	659.426	791439.43
90.000	641.679	769923.31
100.00	629.735	755443.50
120.00	613.819	736148.12
130.00	607.765	728808.56
140.00	602.634	722587.93
150.00	597.918	716869.87
160.00	593.339	711318.87
170.00	588.743	705746.68
182.00	583.091	698895.62
190.00	579.208	694187.43
200.00	563.554	675203.50
202.00	548.320	656736.68
204.00	529.829	634325.81
208.00	485.307	580386.00
212.00	433.846	518043.62
214.00	406.418	484825.34
216.00	378.743	451331.71
218.00	351.635	418559.15
220.00	325.801	387358.46
222.00	301.790	358396.43
224.00	279.999	332141.78
226.00	260.941	309189.12
228.00	244.769	289723.50
232.00	219.664	259541.35
236.00	202.978	239510.95
238.00	197.049	232399.93
240.00	192.419	226851.07
240.19	192.009	226359.71
242.00	188.720	222418.60
244.00	185.814	218937.60
247.60	182.112	214506.00
248.00	181.797	214128.59
252.00	179.407	211268.31
600.00	175.113	206131.18
608.00	176.603	207905.46
612.00	133.545	156476.04
614.00	95.9795	111799.09
618.00	47.2565	54485.789
634.00	21.0466	24210.906
642.00	20.8678	24005.226
644.00	20.9953	24151.945
646.00	21.0308	24192.771
656.00	20.9472	24096.623
658.00	20.9755	24129.173
668.00	20.8779	24016.847
686.00	21.0687	24236.367
696.00	20.9642	24116.203
698.00	20.9876	24143.123
1000.00	20.9663	24118.626

TABLE 6.2.1.4-3

102% OF 2300 MWt MAIN STEAMLINE BREAK WITH FEEDWATER REGULATION VALVE FAILURE, MASS AND ENERGY RELEASES

TIME (SEC)	MASS FLOW (LBM/SEC)	ENERGY RELEASE (BTUs/SEC)
.00	.00	.00
.20	4965.54	5941266.50
.40	4965.54	5941544.00
1.40	2673.00	3202368.75
1.60	2322.76	2783900.25
2.80	2190.18	2627726.75
5.40	1983.54	2383259.00
10.60	1751.04	2106666.50
15.80	1617.28	1946852.75
26.40	1286.70	1549809.50
31.80	1168.04	1406769.00
37.00	1078.44	1298614.12
42.20	1011.52	1217753.00
52.80	913.45	1099138.87
73.60	779.95	937504.94
84.20	735.62	883795.38
105.20	680.46	816937.06
126.20	650.64	780795.13
226.40	607.64	728657.06
228.80	598.27	717285.81
231.60	574.36	688301.81
236.80	512.73	613600.56
247.40	353.36	420641.38
252.60	285.91	339257.66
255.40	257.65	305221.84
258.00	236.94	280312.38
263.20	208.42	246040.06
268.40	192.68	227168.06
273.60	184.37	217213.70
284.20	178.06	209655.48
607.80	176.96	208327.08
608.60	170.80	200946.44
610.40	143.53	168363.09
612.20	104.02	121324.02
613.00	91.40	106381.31
614.60	71.32	82658.25
616.00	55.81	64445.61
618.20	28.02	32236.97
618.40	24.67	28378.60
618.60	20.16	23193.11
618.80	8.75*	9833.00*
1000.00	8.75*	9833.00*

* Time averaged mass and energy release rate during this period.

TABLE 6.2.1.4-4

102% OF 2300 MWt MAIN STEAMLINE BREAK WITH AN ELECTRICAL BUS FAILURE, MASS AND ENERGY RELEASES

TIME (SEC)	MASS FLOW (LBM/SEC)	ENERGY RELEASE (BTUs/SEC)
.00000	.000000	.00000000
.20000	2519.88	3015501.7
.40000	2519.88	3016308.7
.60000	2489.35	2980538.2
.80000	2460.56	2946783.0
1.0000	2432.43	2913754.7
10.000	1778.86	2139991.7
20.000	1383.44	1666019.0
30.000	1130.53	1361232.0
40.000	978.620	1177731.3
50.000	876.396	1054133.0
60.000	799.701	961366.75
70.000	743.433	893264.43
80.000	706.312	848276.00
100.00	660.596	792859.56
110.00	647.123	776523.87
120.00	638.629	766226.56
130.00	630.562	756445.93
140.00	622.980	747254.12
150.00	615.760	738500.81
160.00	608.860	730135.93
170.00	602.211	722074.75
180.00	595.742	714231.87
190.00	550.127	658924.25
200.00	421.546	503144.03
210.00	288.660	342576.21
220.00	214.458	253288.56
230.00	188.148	221733.98
240.00	180.497	212572.75
250.00	178.424	210093.01
260.00	177.800	209345.76
300.00	177.323	208775.17
350.00	177.212	208642.75
400.00	177.156	208574.85
450.00	177.101	208509.15
500.00	177.038	208434.04
525.00	177.002	208391.43
550.00	176.963	208345.04
600.00	176.875	208238.98
607.20	176.958	208355.90
607.40	178.330	209981.26
607.59	178.693	210412.18
607.79	178.671	210383.25
608.00	178.300	209935.95
608.20	177.608	209105.68
608.40	176.623	207924.60
608.59	175.370	206423.28
608.79	173.913	204678.73
609.00	172.268	202710.75
609.20	170.457	200543.28
609.40	168.455	198149.70
609.59	166.319	195595.23
609.79	164.053	192886.25
610.00	161.653	190017.76
611.00	148.226	173984.21
612.00	133.817	156798.92
613.00	114.671	133955.68
614.00	96.2107	112073.40
615.00	82.7049	96106.882
616.00	70.8467	82106.429
617.00	60.1603	69537.796
618.00	49.0184	56533.875
619.00	36.7150	42275.742
620.00	23.6925	27257.626
1000.0	20.9678	24120.263

TABLE 6.2.1.4-5

HZP MAIN STEAMLINE BREAK WITH A CHECK VALVE FAILURE,
MASS AND ENERGY RELEASES

TIME (SEC)	MASS FLOW (LBM/SEC)	ENERGY RELEASE (BTUs/SEC)
.00000	.000000	.00000000
.60000	5798.00	6917185.0
1.0000	5653.13	6749125.0
2.0000	5330.11	6372958.5
4.0000	4805.87	5758514.5
6.0000	4397.65	5277084.5
8.0000	4079.00	4899618.5
10.000	3822.51	4594811.5
12.000	3605.75	4336555.5
14.000	2675.92	3220427.5
16.000	1548.31	1864462.5
18.000	1257.25	1514328.6
20.000	1186.30	1428798.1
30.000	944.712	1136956.0
40.000	799.531	961218.06
50.000	706.380	848353.87
60.000	654.770	785792.18
70.000	625.799	750669.87
80.000	608.768	730023.00
90.000	598.828	717972.81
100.00	593.132	711068.06
110.00	589.455	706611.06
120.00	586.550	703089.18
130.00	583.814	699772.06
140.00	580.961	696313.00
150.00	577.863	692558.00
160.00	574.465	688438.75
170.00	570.746	683929.87
180.00	566.700	679026.25
190.00	562.338	673738.18
200.00	557.666	668076.18
210.00	552.697	662052.37
220.00	547.640	655924.43
230.00	542.512	649708.87
240.00	523.636	626826.18
241.00	516.261	617888.50
242.00	508.060	607951.81
243.00	499.062	597049.81
244.00	489.251	585164.62
245.00	478.563	572218.00
246.00	467.245	558505.75
247.00	455.218	543933.68
248.00	442.515	528544.62
249.00	429.170	512379.15
250.00	415.222	495485.00
251.00	400.725	477930.25
252.00	385.806	459873.43
253.00	370.604	441485.68
254.00	355.200	422862.75
255.00	339.805	404262.78
256.00	324.505	385790.06
257.00	309.540	367735.87
258.00	295.053	350272.43
259.00	281.101	333463.40
260.00	268.020	317706.34
261.00	255.872	303079.78
262.00	244.753	289698.43
263.00	234.717	277628.12
264.00	225.783	266888.40
265.00	217.933	257457.50
266.00	211.121	249278.70
267.00	205.280	242268.40
268.00	200.361	236369.57
269.00	196.266	231459.87
270.00	192.860	227377.04
271.00	190.035	223993.17
272.00	187.703	221199.95
273.00	185.787	218905.07
274.00	184.221	217029.57
276.00	181.920	214275.51
278.00	180.438	212501.76
280.00	179.504	211384.37
300.00	177.687	209210.39
600.00	175.816	206971.93
604.00	175.782	206931.76
608.00	177.771	209306.84
609.00	173.905	204672.84
610.00	165.739	194904.14
611.00	154.458	181425.46
612.00	141.153	165546.00
613.00	128.221	150105.00
614.00	101.087	117829.31
615.00	79.5540	92370.320
616.00	62.0819	71780.781
617.00	43.3850	49989.214
623.00	21.2103	24399.292
1000.00	20.9599	24111.238

TABLE 6.2.1.4-6

HZP MAIN STEAMLINE BREAK WITH AN ELECTRICAL BUS FAILURE, MASS AND ENERGY RELEASES

TIME (SEC)	MASS FLOW (LBM/SEC)	ENERGY RELEASE (BTUs/SEC)
.00000	.000000	.00000000
.20000	2910.41	3469931.5
.40000	2910.41	3471672.5
.60000	2861.52	3415011.0
.80000	2814.78	3360755.7
1.0000	2769.86	3308545.5
10.000	1663.60	2002005.8
15.000	1403.95	1690335.1
20.000	1225.91	1475922.6
30.000	992.324	1194007.5
40.000	844.806	1015741.3
50.000	756.262	908772.18
60.000	702.921	844163.87
70.000	667.369	801068.81
80.000	643.666	772332.37
90.000	627.653	752918.50
100.00	616.609	739529.50
110.00	609.015	730323.68
120.00	603.398	723513.18
130.00	598.755	717884.62
140.00	594.535	712769.18
150.00	590.444	707809.43
160.00	586.324	702814.18
170.00	582.089	697681.06
180.00	577.697	692355.87
190.00	573.122	686810.43
200.00	568.354	681031.00
210.00	563.389	675012.50
220.00	558.228	668756.87
230.00	545.489	653308.50
240.00	436.119	520794.56
250.00	277.169	328724.37
255.00	221.466	261700.26
260.00	193.695	228377.43
265.00	183.052	215631.12
270.00	179.503	211384.00
275.00	178.443	210115.45
280.00	178.080	209681.04
400.00	176.930	208305.01
450.00	176.755	208095.04
500.00	176.546	207845.45
550.00	176.301	207551.90
600.00	176.003	207196.35
608.00	177.891	209449.71
610.00	165.053	194083.42
612.00	139.391	163443.54
613.00	126.040	147423.25
614.00	95.5720	111302.46
615.00	75.6457	87751.750
616.00	58.2949	67343.968
617.00	38.8363	44723.496
617.20	34.5037	39717.214
617.40	29.9293	34439.804
617.59	24.5544	28247.785
1000.0	20.8795	24018.742

TABLE 6.2.1.4-7

HZP MAIN STEAMLINE BREAK WITH FAILURE OF
THE AUXILIARY FEEDWATER RUNOUT PROTECTION SYSTEM,
MASS AND ENERGY RELEASES

TIME (SEC)	MASS FLOW (LBM/SEC)	ENERGY RELEASE (BTUs/SEC)
.00000	.000000	.00000000
.20000	2910.41	3469931.5
.40000	2910.41	3471672.5
.60000	2861.52	3415011.0
.80000	2814.78	3360755.7
1.0000	2769.86	3308545.5
5.0000	2099.26	2520917.7
10.000	1657.95	1995239.6
20.000	1221.94	1471130.7
30.000	988.844	1189798.0
40.000	841.822	1012127.2
50.000	753.338	905229.56
60.000	700.040	840672.18
70.000	664.493	797582.37
80.000	640.803	768860.93
90.000	624.784	749441.00
100.00	613.723	736030.93
110.00	606.102	726792.18
120.00	600.465	719957.56
130.00	595.813	714318.00
140.00	591.595	709204.06
150.00	587.513	704256.00
160.00	583.408	699280.00
170.00	579.195	694172.00
180.00	574.826	688876.31
190.00	570.278	683363.68
200.00	565.540	677619.93
210.00	560.606	671639.43
220.00	555.478	665423.81
230.00	550.160	658978.56
240.00	544.655	652305.93
250.00	475.789	568856.12
255.00	413.002	492797.37
260.00	341.503	406315.28
265.00	276.219	327585.25
270.00	231.580	273860.50
275.00	208.779	246473.15
280.00	199.423	235250.983
300.00	193.900	228668.51
400.00	192.800	227311.10
500.00	192.231	226629.51
600.00	191.255	225459.04
600.20	191.252	225455.95
607.79	193.792	228498.32
608.00	193.781	228483.00
608.20	193.497	228140.40
608.40	192.963	227498.20
608.59	192.197	226577.90
608.79	191.221	225406.29
609.00	190.049	224000.15
609.20	188.696	222377.76
609.40	187.178	220557.25
609.59	185.507	218554.75
609.79	183.696	216384.59
610.00	181.756	214060.48
611.00	170.455	200537.12
612.00	156.951	184397.62
613.00	141.910	166445.87
614.00	125.213	146460.06
615.00	95.7239	111482.05
615.20	91.1601	106087.45
615.40	87.0650	101247.39
615.59	83.1662	96638.031
615.79	79.3150	92087.945
616.00	75.7729	87901.570
616.20	72.0108	83466.445
616.40	68.5844	79427.421
616.59	65.2472	75498.945
616.79	61.8766	71539.960
617.00	58.4064	67474.382
617.20	54.8040	63265.277
617.40	51.0578	58899.500
617.59	47.1678	54377.285
617.79	43.1386	49703.628
618.00	38.9701	44878.253
618.20	34.6433	39878.375
618.40	30.0792	34612.621
618.59	24.7443	28466.480
619.00	22.3836	25748.970
1000.0	20.9793	24133.578

TABLE 6.2.1.4-8

MSLB CONTAINMENT RESPONSE ANALYSIS PARAMETERS

Service water temperature (°F)	100
RWST water temperature (°F)	100
Initial containment temperature (°F)	130
Initial containment pressure (psia)	15.7
Initial relative humidity (%)	30
Initial relative humidity for Check Valve Case	0-100%
Net free volume (ft^3)	2.013×10^6
<u>Containment Fan Coolers</u>	
Total	4
Analysis maximum	4
Analysis minimum	2
Containment High setpoint (psig)	5.5
Delay time (sec) With Offsite Power Without Offsite Power	 35.4 46.0
<u>Containment Spray Pumps</u>	
Total	2
Analysis maximum	2
Analysis minimum	1
Flow rate (gpm) Injection phase (per pump)- See 6.2.1-22 Recirculation phase (total)	 NA
Containment High High setpoint (psig)	12
Delay time (sec) With Offsite Power (delay after High High setpoint) Without Offsite Power (total time from t=0)	 23.5 38.2
ECCS Recirculation Switchover, sec Minimum SG Maximum SG	 NA. NA.
Containment Spray Termination time, (sec) Minimum SG Maximum SG	 NA. NA.

TABLE 6.2.1.4-9

CONTAINMENT FAN COOLER PERFORMANCE

Containment Temperature (ºF)	Heat Removal Rate [Btu/sec] Per Reactor Containment Fan Cooler
130	1820.44
152	3448.69
200	7459.49
263	13112.10
300	16538.24

TABLE 6.2.1.4-10

CONTAINMENT SPRAY PERFORMANCE

Containment Pressure (psig)	(gpm)	
	Train A	Two Trains
0	954.	1933.
10	954.	1933.
20	954.	1933.
30	954.	1933.
42	954.	1933.

TABLE 6.2.1.4-11

CONTAINMENT HEAT SINKS

No.	Material	Heat Transfer Area ft^2	Thickness ft
1	Containment Cylinder	46,926	
	Stainless Steel		0.00158
	Insulation & Epoxy		0.1045
	Carbon Steel		0.03285
	Concrete		3.5
2	Stainless Steel (foil)	2,819	0.001583
	Insulation		0.104167
	Epoxy		0.0005
	Carbon Steel		0.09375
	Concrete		3.5
3	Containment Dome	6,456	
	Stainless Steel		0.00158
	Insulation & Epoxy		0.1045
	Carbon Steel		0.0417
	Concrete		2.5
4	Containment Dome	20,094	
	Epoxy		0.0005
	Carbon Steel		0.0417
	Concrete		2.5
5	Interior Unlined Concrete	59846	
	Epoxy		0.001297
	Concrete		1.97
6	Interior Unlined Concrete (W/internal steel) Flooded	3659	
	Epoxy		0.00292
	Concrete		1.74
	Carbon Steel		0.0221
	Concrete		8.46
7	Interior Unlined Concrete (W/internal Steel) Dry	7318	
	Epoxy		0.00292
	Concrete		1.74
	Carbon Steel		0.0221
	Concrete		8.46
8	Interior Unlined Concrete	8847	
	Stainless Steel		0.00198
	Concrete		3.388

TABLE 6.2.1.4-11 (CONTINUED)

CONTAINMENT HEAT SINKS

No.	Material	Heat Transfer Area ft^2	Thickness ft
9	Structural and Misc Exposed Steel - Epoxy coated carbon steel	102261	
	Epoxy		0.000583
	Carbon Steel		0.035065
10	Structural and Misc Exposed Steel -	2708.	
	Bare Stainless Steel		0.01425
	Stainless Steel		
11	Galvanized Steel	54865	
	Zinc		0.0000833
	Carbon Steel		0.01102
12	Insulted Copper Cable (Used for EQ Calc only)	0.059	
	Hyplon		0.00125
	EPR		0.0025
	Copper		0.005667
13	Carbon Steel Plate (Used for EQ)	0.0872	
	Carbon Steel		0.005208

TABLE 6.2.1.4-12
THERMOPHYSICAL PROPERTIES OF CONTAINMENT HEAT SINKS

Material	Thermal Conductivity (Btu/hr-ft - °F)	Volumetric Heat Capacity (Btu/ft^3 - °F)
Stainless Steel	9.4	60.1
Carbon Steel	29.53	56.9
Zinc	65.3	40.7
Concrete	1.05	22.5
Insulation & Epoxy	0.0188	0.58
Epoxy	0.23	18.3
Hyplon	0.125	32.537
EPR	0.1445	20.5
Copper	219.0	50.778
Carbon Steel (EQ component)	27.0	48.02

TABLE 6.2.1.4-13

PEAK CONTAINMENT PRESSURE, STEAM TEMPERATURE
AND COMPONENT TEMPERATURE RESULTS

CASE	PEAK PRESSURE (PSIG) @ TIME	PEAK STEAM TEMPERATURE (°F) @ TIME	PEAK COMPONENT TEMPERATURE (°F) @ TIME
102% OF 2300 MWt, CHECK VALVE FAILURE	41.19 psig @ 611.78 sec	263.203°F @611.77 sec	264.8°F @ 614.5 sec
102% POWER, FRV FAILURE	40.26 psig @ 610.65 sec	263.51°F @ 29.14 sec	NA
102% POWER, E-BUS FAILURE	40.63 psig @ 614.10 sec	273.50°F @ 34.50 sec	NA
HOT ZERO POWER, CHECK VALVE FAILURE	41.06 psig @ 612.2 sec	322.6°F @ 34 sec	322.6°F @ 34 sec. Note that this represents a component surface temperature for components 12 and 13.
HOT ZERO POWER, E-BUS FAILURE	41.61 psig @ 614.00 sec	267.40°F @ 33.472 sec	NA
HOT ZERO POWER, AUX FEEDWATER RUNOUT PROTECTION	38.40 psig @ 613.23 sec	267.21°F @ 33.579 sec	NA

TABLE 6.2.1.4-14

SEQUENCE OF EVENTS FOR 102% OF 2300 MWt CHECK VALVE FAILURE CASE

Event Description	Time (sec)
Break occurs	0
Containment HI pressure SI setpoint reached(1)	2.96
Auxiliary feedwater starts	4.46
Reactor / turbine trip	4.96
Containment HI-HI pressure setpoint reached(1)	8.35
Main Steamline Isolation on MSIV closure	12.50
Containment sprays start	31.85
Main Feedwater Isolated	34.46
Reactor Containment Air Recirculation Fan Coolers start	38.36
Top of SG tubes uncover	197.6
Auxiliary feedwater to faulted SG terminated	600
Peak containment pressure occurs	611.78
Break flow stops	CONTINUES DUE TO FRV & MFIV LEAKAGE

Notes:

1. The time of containment HI & HI-HI pressure have been credited in the steamline break analysis to the nearest 0.1 second, always rounding up.

TABLE 6.2.1.4-15

SEQUENCE OF EVENTS FOR HOT ZERO POWER CHECK VALVE FAILURE CASE

Event Description	Time (sec)
Break occurs	0
Main Feedwater Isolated	0.00
Containment HI pressure SI setpoint reached[(1)]	3.38
Auxiliary feedwater starts	4.18
Reactor / turbine trip	4.68
Containment HI-HI pressure setpoint reached[(1)]	8.13
Main Steamline Isolation on MSIV closure	12.0
Containment sprays start	31.63
Reactor Containment Air Recirculation Fan Coolers start	38.83
Top of SG tubes uncover	237.0
Auxiliary feedwater to faulted SG terminated	600
Peak containment pressure occurs	612.2
Break flow stops	CONTINUES DUE TO FRV & MFIV LEAKAGE

Notes:

1. The time of containment HI & HI-HI pressure has been credited in the steamline break analysis to the nearest 0.1 second, always rounding up.

TABLE 6.2.1.4-16

SEQUENCE OF EVENTS FOR 102% OF 2300 MWt FRV FAILURE CASE

Event Description	Time (sec)
Break occurs	0
Main Steamline Isolation on SLCV closure	0.1
Containment HI pressure SI setpoint reached	5.41
Reactor Containment Air Recirculation Fan Coolers start	5.51
High Differential Pressure between a Steamline and Steam Header Reached, SI	8.4
Auxiliary feedwater starts	8.6
Reactor / turbine trip	10.4
Containment HI-HI pressure setpoint reached	18.5
Containment sprays start	42.0
Main Feedwater Isolated	59.9
Top of SG tubes uncover	227.0
Auxiliary feedwater to faulted SG terminated	600.0
Peak containment pressure occurs	610.65
Break flow stops	CONTINUES DUE TO FRV AND MFIV LEAKAGE

TABLE 6.2.1.4-17

SEQUENCE OF EVENTS FOR 102% OF 2300 MWt E-BUS FAILURE CASE

Event Description	Time (sec)
Break occurs	0
Main Steamline Isolation on SLCV closure	0.2
High Differential Pressure between a Steamline and Steam Header Reached, SI	3.4
Auxiliary feedwater starts	3.6
Reactor / turbine trip	5.4
Containment HI pressure SI setpoint reached	5.66
Containment HI-HI pressure setpoint reached	19.62
Main Feedwater Isolated	34.7
Reactor Containment Air Recirculation Fan Coolers start	41.06
Containment sprays start	43.12
Top of SG tubes uncover	184.4
Auxiliary feedwater to faulted SG terminated	600
Peak containment pressure occurs	614.10
Break flow stops	CONTINUES DUE TO FRV AND MFIV LEAKAGE

TABLE 6.2.1.4-18

SEQUENCE OF EVENTS FOR HZP E-BUS FAILURE CASE

Event Description	Time (sec)
Break occurs	0
Main Steamline Check Valves Close	0.1
High Differential Pressure between a Steamline and Steam Header Reached, SI	1.6
Containment HI pressure SI setpoint reached	5.86
Auxiliary feedwater starts	3.1
Reactor / turbine trip	3.6
FRVs fully closed	NA
Containment HI-HI pressure setpoint reached	20.33
Containment fan coolers start	41.26
Containment sprays start	43.83
Top of SG tubes uncover	228.2
Peak containment pressure occurs	614
Auxiliary feedwater to faulted SG terminated	600
Break flow stops	CONTINOUS DUE TO FRV AND BLOCK VALVE LEAKAGE

TABLE 6.2.1.4-19

SEQUENCE OF EVENTS FOR HZP RUNOUT PROTECTION FAILURE CASE

Event Description	Time (sec)
Break occurs	0.0
Main Feedwater Isolated	0.0
Main Steamline Isolation on SLCV Closure	0.1
High Differential Pressure between a Steamline and Steam Header Reached, SI	1.6
Auxiliary feedwater starts	1.6
Reactor / turbine trip	3.6
Containment HI pressure SI setpoint reached	5.86
Containment HI-HI pressure setpoint reached	20.5
Reactor Containment Air Recirculation Fan Coolers Start	41.26
Containment sprays start	44.00
Top of SG Tubes Uncover	241.2
Auxiliary Feedwater to faulted SG terminated	600
Peak Containment Pressure Occurs	613.2
Break Flow Stops	Continues - FRV and MFIV Leakage

6.2.2 Containment Heat Removal Systems

6.2.2.1 Design Basis

Adequate post accident heat removal capability for the containment is provided by two separate, full capacity, ESF systems. These are the Containment Spray System (CSS), described in Section 6.2.2.2.1 and the Containment Air Recirculation Cooling System whose components operate as described in Section 6.2.2.2.2. These systems are of different engineering principles and serve as independent backups for each other.

These two ESF systems were designed to remove sufficient heat from the reactor containment, following the initial LOCA containment pressure transient, to keep the containment pressure from exceeding the design pressure.

Any of the following combinations of equipment will provide sufficient heat removal capability to maintain the post-accident containment pressure below the design value, assuming that the core residual heat is released to the containment as steam.

a) All four containment cooling units

b) Both containment spray pumps, or

c) Two of the four containment cooling units and one containment spray pump.

After the injection operation it is expected that spray flow could be discontinued while maintaining containment pressure reduction with the containment fan cooler units. Details of the normal and emergency power sources for these ESF systems are presented in the discussion of the Electrical System, Section 8.

6.2.2.1.1 Containment Spray System

The primary purpose of the CSS is to spray cool water into the containment atmosphere when appropriate in the event of a LOCA and thereby ensure that containment pressure does not exceed its design value which is 42 psig at 263ºF (100 percent RH). This protection is afforded for all pipe break sizes up to and including the hypothetical instantaneous circumferential rupture of a reactor coolant pipe. Pressure and temperature transients for LOCA are presented in Section 6.2.1.1.1.1. Although the water in the core after a LOCA is quickly subcooled by the SIS, the CSS design is based on the conservative assumption that the core residual heat is released to the containment as steam.

The CSS was designed to spray at least 2322 gpm of borated water into the Containment Building whenever the coincidence of two sets of two out of three (Hi-Hi) containment pressure signals occurs or a manual signal is given. Either of two subsystems containing a pump and associated valving and spray headers is independently capable of delivering one-half of this flow, or, 1161 gpm.

The design basis was to provide sufficient heat removal capability to maintain the post-accident containment pressure below the design pressure, assuming that the core residual heat is released to the containment as steam. This requires a heat removal capacity of the subsystem, with either pump operating, at least equivalent to two fan-coolers heat removal capability at the containment design conditions.

A second purpose served by the CSS is to remove radioactive iodines and particulates from the containment atmosphere released during a LOCA (refer to Sections 6.5.2 and 15.6.5).

The spray system was designed to operate over an extended time period, following a primary coolant system failure as required to restore and maintain containment conditions at near atmospheric pressure. It has the capability of reducing the containment post-accident pressure and consequent containment leakage taking into account any reduction due to single failures of active components.

Portions of other systems which 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.

Those portions of the spray system located outside of the containment that are designed to circulate post-accident containment sump water must meet leakage rate limits to ensure LOCA dose acceptance criteria are met. Additionally, pressure relieving devices discharge into closed systems. Further discussion on leakage is provided in Section 6.3.2.5.5.

System active components are redundant. System piping located within the containment is redundant and separable in arrangement unless fully protected from damage which 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.

All portions of the system located within containment were 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.

6.2.2.1.2 Containment Air Recirculation Cooling System

The Containment Air Recirculation Cooling System was designed to recirculate and cool the containment atmosphere in the event of a LOCA and thereby ensure that the containment pressure cannot exceed its design value of 42 psig at 263ºF (100 percent relative humidity). Although the water in the core after a LOCA is quickly subcooled by the SIS, the Containment Air Recirculation Cooling System was designed on the conservative assumption that the core residual heat is released to the containment as steam. The fans and cooling coils continue to remove heat after the LOCA and reduce the containment pressure close to atmospheric within the first 24 hr.

The following objectives are met to provide the ESF functions:

a) Each of the four fan-cooler units (centrifugal fans and water cooled heat exchangers) is capable of transferring heat at the rate of 11,100 Btu/sec from the containment atmosphere following a Loss of Coolant Accident at the post-accident design conditions, i.e., a saturated air-steam mixture at 42 psig and 263ºF. This heat transfer rate was that assigned to the fan-cooler units in the analysis of containment and related heat removal system capability in Section 6.2.2.3.2.

   The establishment of basic heat transfer design parameters for the cooling coils of the fan-cooler units, and the calculation by computer of the overall heat transfer capacity are discussed in Section 6.2.2.3.2. Among the topics covered are selection of the tube side fouling factor, effect of air side pressure drop, effect of moisture entrainment in the air-steam mixture entering the fan-coolers, and calculation of the various air side to water side heat transfer resistances.

   During a postulated design basis LOCA, concurrent with a loss of off-site power (and failure of one train of the safety related AC power system), the service water system may temporarily be incapable of delivering 750 gpm to each containment air cooler due to flashing downstream of return line throttle valves. An evaluation of the consequences of flashing flow concluded that any two containment air coolers are capable of removing more heat from the containment atmosphere than is credited in the containment pressurization analysis (see Sections 6.2.2.3.2 and Figure 6.2.1-11).

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. 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.

In addition to the above design bases, the Containment Air Recirculation Cooling System was designed to possess sufficient margin to withstand an over-rated condition of 60 psig and 286ºF for one hour without loss of

operability. No specific criteria for heat removal capability are applied at the over-rated condition. The equipment was designed to operate at the post-accident conditions at 42 psig and 263ºF for three hours, followed by operation in an air-steam atmosphere at 20 psig, 219ºF for an additional 21 hr. The equipment design will permit subsequent operation in an air-steam atmosphere at 5 psig, 152ºF for an indefinite period.

All components are capable of withstanding or are protected from differential pressures which may occur during the rapid pressure rise to 42 psig in ten seconds.

During a postulated design basis LOCA concurrent with a loss of off-site power, service water flow to the containment air coolers would be interrupted and service water within the coolers could boil. An evaluation of this condition concluded that any resulting waterhammers would be no more severe than those generated when service water pumps restart following a loss of off-site power. The service water system and containment air cooling system are capable of withstanding these restart waterhammer conditions (see Section 9.2.1.4).

Portions of other systems which share functions and become part of this containment cooling system when required were designed to meet the criteria of this section. Neither a single active component failure in such systems during the injection phase nor an active or passive failure during the recirculation phase will degrade the heat removal capability of containment cooling.

Design provisions were made to the extent practical to facilitate access for periodic visual inspection of all important components of the Containment Air Recirculation Cooling System.

The Containment Air Recirculation Cooling System was designed to the extent practical so that the components can be tested periodically, and after any component maintenance, for operability and functional performance.

6.2.2.2 System Design

6.2.2.2.1 Containment Spray System

Adequate containment cooling and iodine and particulate removal are provided by the CSS shown in Figure 6.2.2-1. This system operates in sequential modes as follows:

a) Spray from the refueling water storage tank into the entire containment atmosphere using the containment spray pumps. During this mode, the contents of the spray additive tank (sodium hydroxide) are mixed into the spray stream to provide adequate iodine removal from the containment atmosphere.

b) Recirculation of water from the containment sump is provided by the diversion of a portion of the recirculation flow from the discharge of the residual heat removal (RHR) heat exchangers to the suction of the spray pumps after injection from the refueling water storage tank has been terminated.

The principal components of the CSS consist of two pumps, one spray additive tank, spray ring headers and nozzles, and the necessary piping and valves. The containment spray pumps and the spray additive tank are located in the Auxiliary Building and the spray pumps take suction directly from the refueling water storage tank.

The CSS also utilizes the two RHR pumps, two residual heat exchangers, and associated valves and piping of the SIS for the long term recirculation phase of containment cooling and iodine and particulate removal.

The spray system will be actuated by the coincidence of two sets of two out of three (Hi-Hi) containment pressure signals. This starting signal will start the pumps and open the discharge valves to the spray header and the valves associated with the spray additive tank. If required, the operator can manually actuate the entire system from the Control Room and, periodically, the operator will actuate system components to demonstrate operability.

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.

Recirculation Phase

After the injection operation it is expected that spray flow could be discontinued while maintaining containment pressure reduction with the containment fan cooler units, and returning all of the recirculated water to the core. In this mode the bulk of the core residual heat is transferred directly to the sump by the spilled coolant to be eventually dissipated through the residual heat exchanger once the sump water becomes heated. The heat removal capacity of two of the four fan coolers is sufficient to remove the corresponding energy addition to the vapor space resulting from steam boil off from the core assuming flow into the core from one RHR pump at the beginning of recirculation without exceeding containment design pressure; hence, it is not expected that continued spray operation would be required for containment cooling. If, for any reason, the containment pressure should be observed to increase recirculation spray flow may be initiated. The operator can direct part of the discharge flow from the residual heat exchangers to the suction of the spray pumps. With this mode of operation, core cooling can be maintained and containment pressure maintained below design even with no fan coolers operating.

There are two sump return lines which lead from the containment to the RHR pumps. Each line is located inside of a larger diameter guard pipe. The lines are separated by approximately 18 ft. The lines are designed to allow for 2 in. differential movement between the containment and pump chamber and are designed as Class I equipment.

The design of the ECCS Sumps are discussed in Section 6.3.2.2.2.

Recirculation may start with a water depth of 1.5 ft on the containment floor. This is equivalent to the amount of water in the primary systems plus 60 percent of the refueling water storage tank, or approximately 215,000 gallons of water at 263ºF.

Cooling Water

Component Cooling System

During the recirculation mode, the Component Cooling System is used to cool the recirculation fluid as it passes through the residual heat exchanger. This system is described in detail in Section 9.2.2.

One of the three component cooling pumps and one of the two component cooling heat exchangers provide the core and containment cooling function during recirculation.

Service Water System

The service water system is provided with redundant and independent loop headers and valves such that the two component cooling heat exchangers which are supplied with service water for cooling can have flow directed to them from the two independent headers. Two of the four service water pumps are required to operate during the recirculation phase. This system is described in detail in Section 9.2.1.

Change-Over from Injection Phase to Recirculation Phase

The sequence, from the time of the SI signal, for the change-over from the injection to the recirculation is described in Section 6.3.2.2.5.

Components

Materials, code requirements, and construction techniques for associated components, piping, and structures of the CSS are described in Section 6.1.1.1.2.

6.2.2.2.2 Containment Air Recirculation Cooling System

A schematic arrangement of a Containment Air Recirculation Cooling System is shown in Figure 6.2.2-2.

The air recirculation system consists of four air handling units, each including rack for roughing filters (pre-filters) air operated inlet dampers, failed open butterfly valves, cooling coils, fan and drive motor, duct distribution system, instrumentation, and controls. The units are located on the operating floor adjacent to the containment wall. The roughing filters are removed during all MODES of operation except during plant-shutdown. The filter pads should be replaced during plant shutdown conditions when activities within the containment may stir up dust which might deposit on the coils.

Each fan is designed to supply at least 65,000 cfm at design basis accident (DBA) conditions at approximately 20 in. s.p., 263ºF, 0.162 lb/ft^3 density. The fans are direct driven centrifugal type. Cooling coils are plate fin-tube type. Each air handling unit is capable of removing 40 x 10^6 Btu/hr from the containment atmosphere under DBA conditions. 750 minimum gpm of service (cooling) water is normally supplied to each unit not including the motor cooler. The design maximum service water inlet temperature is 100ºF.

A gravity operated damper in the fan discharge isolates any inactive air handling unit from the duct distribution system. The damper opens automatically when the fan is started. Duct work distributes the cooled air to the various containment compartments and areas. For plant shut-down condition the flow sequence through each air handling unit is as follows:

roughing filter (when installed), inlet damper valve, cooling coils, fan, outlet dampers, and discharge header for normal flow. For all other MODES of operation including post accident flow path, the inlet damper is closed and the flow enters the unit through a butterfly valve to the cooling coils.

Individual system components and their supports meet the requirement for Class I (Seismic) structures (Section 3.7) and each component is mounted to isolate it from fan vibration.

Actuation Provisions

The inlet dampers used to route air flow through the operating units have only two positions, full open or full closed. These dampers are air operated and spring loaded. Upon loss of control signal or control air, the spring actuates the normal damper to the closed position (fail-safe operation). The inlet butterfly valves used to route air flow through the operating units for accident conditions have only two positions, full close or full open. These valves are spring loaded; the spring maintains the emergency butterfly valve in the open position (fail-safe open operation) for all modes of operation except when manually closed as required for maintenance.

A high containment pressure signal automatically actuates the SI safety feature sequence which trips any open inlet dampers to the closed position, and starts any stopped fan cooler unit. The normal dampers have a 3-way selector switch, open-close-reset, to allow positioning the dampers as desired during normal plant operation. The inlet dampers close on an SI signal via one solenoid valve.

The fans are part of the ESF and either all four, or at least two of four fans will start after an accident, depending on the availability of emergency power (refer to Section 8.3).

Overload protection for the fan motors is provided at the switchgear by overcurrent trip devices in the motor feeder breakers. The breakers can be operated from the Control Room and can be reclosed from the Control Room following a motor overload trip.

Flow switches in the system, operating both normally and post-accident, indicate whether air is circulating in accordance with the design arrangement. Low flow alarms are provided in the Control Room.

Temperature elements (RTD's) are installed on the inlet and outlet (air side) of each fan cooler unit to provide data for monitoring cooling performance.

Flow Distribution and Flow Characteristics

The location of the distribution ductwork outlets, with reference to the location of the air handling unit return inlets, ensures that the air will be directed to all areas requiring ventilation before returning to the units. The distribution system is represented schematically by the Ventilation Systems Flow Diagram, Figure 9.4.1-2.

The air discharged inside the reactor coolant loop shield walls will circulate and rise above the operating floor through openings around the SG and return to the air handling unit inlets. The temperature of this air will be essentially the ambient existing in the containment vessel.

The steam-air mixture from the containment entering the fan-cooler units during the accident will be at approximately 263ºF and have a density of 0.162 lb per cubic foot. Part of the water vapor will condense on the cooling coils, and the air leaving the coils will be saturated at a temperature slightly below 263ºF.

The fluid will remain in this condition as it flows into the fan, but will pick up some sensible heat from the fan and fan motor before flowing into the distribution header. This sensible heat will increase the dry-bulb temperature slightly above 263ºF and will decrease the relative humidity slightly below 100 percent.

Cooling Water for the Fan Cooler Units

The cooling water requirements for all four fan cooling units during a major loss of primary coolant accident and recovery are supplied by two of the four service water pumps and one of the two service water booster pumps. The service water system is described in Section 9.2.1.

The cooling water discharges from the cooling coils to the discharge canal and is monitored for radioactivity by routing a small bypass flow from each unit through a common radiation monitor. Upon indication of radioactivity in the effluent, each cooler discharge line is monitored individually to locate the defective cooling coil. The service water system is pressurized inside the containment, but the pressure in certain portions will be below the containment design pressure of 42 psig. However, since the cooling coils and service water lines form a closed system inside the containment, no contaminated leakage is expected into these units. Isolation valves on the inlet and discharge of each fan cooler are located outside the containment and may be used to isolate individual fan cooler units in the event that radioactivity is detected by the radiation monitor.

Local flow and temperature indication is provided outside containment for service water flow from each cooling unit.

Local temperature elements (RTD's) on the water inlets and outlets and a local pressure differential indicator and pressure gauges are installed on each fan cooler unit to provide data for monitoring cooling coil performance.

Environmental Protection

All system control and instrumentation devices required for containment accident conditions are located to minimize the danger of control loss due to missile damage. Differential pressure switches across the fans indicate whether air is circulating in accordance with the design arrangement. Abnormal flow alarms are provided in the Control Room.

All fan parts, damper shaft and blade 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.

All of the air handling units are located outside the shield wall (which serves as a missile barrier) on the operating floor adjacent to the containment wall. The distribution header and service water cooling piping are also located outside the shield. This arrangement provides missile protection for all components.

Components

Roughing Filters

During reactor operation, the roughing filters are removed to reduce the amount of fibrous material in the containment. During outage conditions where activities in the containment may stir-up dust, the filter pads are installed, on both the normal dampers and over the butterfly valves, as required.

The roughing filter bank for the normal inlet dampers was designed for horizontal air flow, and can contain 54 individual filters, each of which is 2 ft sq by 2 in. thick. The filters are of fire resistance construction, with the media composed of glass fiber.

These filters when installed are in series with the inlet damper in the air inlet path and over the butterfly valve.

Fan-Motor Units

The four containment cooling fans are of centrifugal, non-overloading, direct drive type.

Each fan was designed for a minimum flow rate of 65,000 cfm when operating against the system resistance of approximately 20 in. s.p. existing during the DBA conditions (0.162 lb/ft^3 density, a containment pressure of 42 psig, and temperature of 263ºF). Each fan is also capable of circulating a minimum of 65,000 cfm at the containment over-rated condition.

The reactor containment fan cooler (RCFC) motors are Westinghouse, totally enclosed water cooled, 350 horsepower, induction type, 3 phase, 60 cycle, 720 rpm, Westinghouse, Thermalastic 460 volt with ample insulation margin. Significant motor details are as follows:

a) Insulation - Class B (NEMA rated total temperature 130ºC) Thermalastic. Basic structure high turn to turn and coil to ground insulation. It was impregnated and coated to give a homogeneous insulation system which is highly impervious to moisture. Internal leads and the terminal box-motor interconnection are given special design consideration to assure that the level of insulation matches or exceeds that of the basic motor system. At incident ambient and maximum rated load conditions (263ºF and 350 HP) the motor insulation hot spot temperature is not expected to exceed 107ºC.

b) Heat Exchanger - An air to water heat exchanger is connected to the motor to form an entirely enclosed cooling system. Air movement is through the heat exchanger and back to the motor. Two vent valves permit incident ambient (increasing containment pressure) to enter the motor air system so the bearings will not be subjected to differential pressure. It also assures pressure equalization as the containment pressure is reduced by the containment cooling systems. Water connections are welded or flanged throughout, and supply and discharge are common with the containment cooling water system, i.e., supplied from the service water header. The drain will be piped to the containment fan cooler drain system.

c) Bearings - The motors are equipped with high temperature grease lubricated ball bearings as would be required if the bearings were subjected to incident ambient temperatures.

d) Conduit (Connection) Box - The motor leads are brought out of the frame through a seal and into a cast iron, sealed explosion proof type of conduit box.

Cooling Coils

The coils are fabricated of copper plate fins vertically oriented on stainless steel tubes. The heat removal capability of the cooling coils is 40 x 10^6 Btu/hr per air handling unit at saturation conditions (263ºF, 42 psig).

The design internal pressure of the coil is 150 psig at 300ºF and the coils can withstand an external pressure of 60 psig at a temperature of 298ºF without damage.

Local flow and temperature indication of service water are provided at each air handling unit. Alarms indicating abnormal service water flow and radioactivity are provided in the Control Room.

The coils are provided with drain pans and drain piping to prevent flooding during accident conditions. This condensate is drained to the containment sump.

Ducting

The ducts are designed to withstand the sudden release of RCS energy and energy from associated chemical reactions without failure due to shock or pressure waves by incorporation of pressure-relieving devices along the ducts which open at slight overpressure, approximately 1.0 psi. The ducts are designed and supported to withstand thermal expansion during an accident.

The structural capability of the ductwork was analyzed to determine the maximum pressure differential that can be maintained across the ductwork without exceeding the maximum allowable stress of the containment air recirculation ductwork. In performing this analysis, the sheet metal duct walls were considered as membranes and the reinforcing members were considered as frame structures receiving its load from the sheet metal duct walls.

The results of the analysis of the reinforcing members and the duct walls indicate that the maximum allowable stress of 15,000 psi would be reached when the pressure differential across the containment air recirculation ductwork is 0.40 psi. The maximum allowable stress is well below the yield stress of 36,000 psi.

In order to ensure that the rapid pressurization of the containment following a LOCA would not interfere with the proper operation of the containment air recirculation system, pressure equalizing devices have been installed.

A computer program has been developed to calculate the pressure differential as a function of time across the walls of the duct and to determine the required relief panel areas and separation distances. The program assumes:

a) The ideal gas law for an isothermal process is used to calculate the pressure within the duct at any time

b) Air flows from the containment into the duct at a rate dependent upon the pressure differential, area of the panel and discharge coefficient of the panel

c) The panel area opens linearly with time after its set differential opening pressure is reached, and

d) The containment pressure transient can be approximated by several straight lines of different slope.

The duct was conservatively considered to be made up of several independent compartments whereby interflow between adjacent compartments of the duct is prohibited. The length of each compartment is the distance between adjacent panels. The differential pressures across the walls of the duct for different compartmental volumes and relief panel areas are calculated for the following conditions:

a) The relief panels will not open until a set pressure differential of 0.01 psig is reached.

b) A containment pressurization rate of 17.3 psi/sec exists up until 0.05 sec after the LOCA, and 15 psi/sec thereafter. The initial pressurization rate of 17.3 psi/sec is approximately 20 percent higher than the greatest pressurization rate occurring as a result of the double-ended pipe rupture. The 0.05 sec time duration of the initial pressurization rate is 100 percent higher than the actual duration in the double-ended pipe rupture. The conservative representation of the containment pressure transient assures conservatism of the calculated pressure differential across the duct walls.

The results of the analysis indicate that the greatest differential pressure exists across the duct walls of the compartment with the largest ratio of compartment volume to relief panel area servicing that compartment. The pressure differential across the duct walls have been calculated as a function of time for several panel areas, separation distances, and panel opening times and the design case is shown on Figures 6.2.2-3 through 6.2.2-5.

The results of the analysis were used to determine the number, separation distance, and size of the pressure relief panels to be installed. The separation distance between panels has been chosen to be 10 ft for the 72 in. x 72 in. duct.

The smaller ducts have pressure relief panels of either 24 in. x 24 in. or 24 in. x 12 in. size and are separated by distances of either 10 or 16 ft. The analysis has shown that these ducts would be subjected to lower pressure differentials than the 72 in. x 72 in. duct serviced by the 24 in. x 24 in. relief panel.

Each pressure relief panel has four louver blades interconnected by a linkage which is connected to an adjustable counterweight mechanism. The counterweight has been set in such a position that the panels open at a pressure differential of 0.01 psi.

The differential pressure produced across the ductwork within the crane wall has been analyzed by the same method used for the ductwork within the containment. The duct, which is 25 ft long and open at its end, has a cross sectional area of 5.5 sq ft. For purposes of this analysis this area served as the relief panel area. The pressurization rate within the compartments of the crane wall is considerably higher than the containment pressurization rate because of the much smaller free volume of the crane wall compartments. The maximum differential pressure calculated for this ductwork is 2.0 psi. This is based on a conservatively high value of crane wall pressurization rate. This segment of the duct will be reinforced so that the air recirculation capability within the crane wall will not be impaired.

Where flanged joints use gaskets, the material is suitable for temperatures to 300ºF.

Ducts are constructed of corrosion resistant material.

Air Operated Dampers

Air operator multi-bladed dampers are installed in the air inlet to each air handling unit. These dampers and normally open butterfly valves are used to route air flow through units that are operating. They have only two positions, fully open or fully closed; the damper operation is spring loaded to the closed position required for post-accident operation, the butterfly valves will remain open. Their design permits only nominal air leakage when closed.

Further information on the components of the Containment Air Recirculation Cooling System is given in Section 6.1.1.1.3.

6.2.2.3 Design Evaluation

6.2.2.3.1 Containment Spray System

During the injection phase following the maximum LOCA (i.e., during the time that the containment spray pumps take their suction from the refueling water storage tank) this system provides the design heat removal capacity for the containment. After the injection phase, each train of the recirculation system provides sufficient cooled recirculated water to keep the core flooded as well as providing, if required, sufficient flow to the suction of the containment spray pumps to maintain the containment pressure below the design value. This applies for all reactor coolant pipe sizes up to and including the hypothetical instantaneous circumferential rupture of a reactor coolant pipe. Only one pumping train and one heat exchanger are required to operate

for this capability at the earliest time recirculation is initiated. With a recirculation train and one spray pump in operation, no containment cooling fans would be required.

During the injection and recirculation phases the spray water is raised to the temperature of the containment in falling through the steam-air mixture. The minimum fall path of the droplets is approximately 80 ft from the lowest spray ring headers to the operating deck. 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 is reached in a distance of approximately five feet. Thus, the spray water reaches essentially the saturation temperature. The model for spray droplet heat removal is discussed below.

Containment Spray Droplet Heat Removal Model

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 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 6.2.2-2) where the drag coefficient C_D is a function of the Reynolds number (nomenclature used is given at the end of this discussion):

$$V^2 = \frac{4\,Dg\,(\rho - \rho_m)}{3\,C_D\,\rho_m} \tag{1}$$

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 6.2.2-3).

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

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

The Prandtl number and the Schmidt number for the conditions assumed are approximately 0.7 and 0.6, respectively. Both of these are sufficiently independent of pressure, temperature, and composition to be assumed constant under containment conditions (References 6.2.2-4 and 6.2.2-5). 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:

$$\frac{d}{dt}(Mu) = mh_g + q \tag{4}$$

$$\frac{d}{dt}(M) = m \tag{5}$$

where: $q = h_c A (T_s - T)$ (6)

$m = k_G A (P_s - P_v)$ (7)

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 percent effective in removing heat from the atmosphere.

Nomenclature

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

System Response

The starting sequence of the containment spray pumps and their related emergency power equipment was designed so that delivery of the minimum required flow is reached within 60 sec (see Section 8.3) which is the delay assumed for the starting of containment cooling (Section 6.2.1.1).

Single Failure Analysis

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 6.2.2-1.

The analysis of the LOCA presented in Section 6.2.1.1.3 reflects the single failure analysis.

Reliance on Interconnected Systems

The CSS initially operates independently of other ESF following a LOCA. It provides backup cooling to the Containment Air Recirculation Cooling. For extended operation in the recirculation mode, water is supplied through the RHR pumps. Spray pump cooling is supplied from the component cooling loop.

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 SI pumps. Sufficient flow instrumentation is provided so that the operator can perform appropriate flow adjustments with the remote valves in the flow path, as shown in Figure 6.2.2-1.

Shared Function Evaluation

Table 6.2.2-2 is an evaluation of the main components which have been discussed previously and a brief description of how each component functions during normal operation and during the accident.

6.2.2.3.2 Containment Air Recirculation Cooling System

The Containment Air Recirculation Cooling System provides the design heat removal capacity for the containment following a LOCA assuming that the core residual heat is released to the containment as steam. The system accomplishes this by continuously recirculating the air-steam mixture through coiling coils to transfer heat from containment to service water.

The performance of the Containment Recirculation Cooling System in pressure reduction is discussed below.

Air-Recirculation Fan-Coolers Heat Removal Capability Model

The ability of the containment air recirculation coolers to function properly in the accident environment was demonstrated by the Westinghouse computer code "HECO." The code determined the plate-fin cooling coil heat removal rate when operating in a saturated steam-air mixture.

In the code, a mass flow rate of cooling water was first established.

This determines the tube inside film coefficient. Next, the resistance to heat transfer between the cooling water and the outside of the fin collars was computed; including inside film coefficient, fouling factor, tube radial conduction, fin-collar interface resistance, and conduction across the fin collars.

A fouling factor of .001 hr-ft^2-ºF/Btu, under both normal and DBA conditions, was assumed for cooling coil design purposes. This value was conventionally used in sizing heat exchangers cooled by lake water at 125ºF or less (Reference 6.2.2-8), and is considered conservative for this application.

The analysis becomes iterative. Assuming an overall heat transfer rate Q_{tot}, the temperature at the outside of the fin collars was determined from Q_{tot} and the sum of the resistances cited above.

A second iterative procedure was then established. The variable whose value was assumed is the effective film coefficient between the fins and the gas stream, which involves the effect of convective heat transfer and mass transfer. With this value of $h_{effective}$, fin efficiency and the fin temperature distribution were determined. It was assumed that a condensate film exists on the vertical fins. An analysis was performed which relates this film thickness to the rate of removal due to gravity and shear, and the rate of addition of condensate by mass transfer from the bulk gas. In the process, from an energy balance, the temperature of the interface between the bulk gas and the condensate was determined; this was necessary for determining the mass transfer rate from the gas. When the thickness of the condensate film was known, the value of the assumed $h_{effective}$ was checked from the relation h_{eff} = K water/$_{film}$. If the assumed and computed values were not the same, a new guess was made and calculations repeated until the assumed and computed values were equal.

When this occurred, the heat transfer rate from the fins and fin collar was computed, using the standard equations for fin and fin collar heat transfer and the values of $h_{effective}$ and film-bulk gas interface temperature. If this value was not the same as Q_{tot}, initially assumed in order to determine fin collar temperature, the whole analysis was repeated with a new estimate of Q_{tot}. When, finally, the heat transfer rate to the cooling water from the fin collar equaled the resulting computed rate to the fin collar and fins from the gas, the effect of this heat transfer rate on the cooling water was computed. The water exit temperature was established and this value was used as the inlet temperature for the next heat exchanger pass. Also, the effect of convective heat transfer and condensate mass transfer were determined relative to the gas composition and thermodynamic state. The updated gas state was used as inlet conditions for the next pass. The process was now repeated for the second, third etc., passes until the gas exits the heat exchanger.

The mass transfer coefficients used in the "HECO" code were derived from analyses and reports of experimental data contained in References 6.2.2-6, 6.2.2-7, and 6.2.2-8. From Reference 6.2.2-6, the mass flow rate of condensate is defined by:

$$\dot{m} = \bar{h}_D \left(\rho_{sg} - \rho_{sw}\right) \qquad (8)$$

Nomenclature is defined at the end of the section.

From Reference 6.2.2-6, pp. 471-473, experimental data for mass and heat transfer are correlated by the expression.

$$\frac{\bar{h}_D}{u_S} (Sc)^{-2/3} = \bar{S}t\, (Pr)^{-2/3} \qquad (9)$$

as shown in Figure 16-10 of Reference 6.2.2-1. Thus

$$\bar{h}_D = u_s \bullet St \left(\frac{Sc}{Pr}\right)^{2/3} \qquad (10)$$

$$\bar{h}_D = \frac{u_s \bullet h}{\rho C \;\; u_s} \left(\frac{Sc}{Pr}\right)^{2/3}$$

As Reference 6.2.2-6 points out, for large partial pressures of the condensing components, Equation (10) must be corrected by a factor P_t/P_{am}. Thus h_D is defined by

$$\bar{h}_D = \frac{h}{\rho C} \frac{P_t}{P_{am}} \left(\frac{Sc}{Pr}\right)^{2/3} \qquad (11)$$

This is essentially the same result as reported by Reference 6.2.2-7, pg. 343 and Reference 6.2.2-9.

Reference 6.2.2-6 states that experiments show Equation (9) to be valid when the Schmidt number does not differ greatly from 1.0. Equations (9) and (11) are combined to give the mass transfer rate, which is

$$\dot{m} = \frac{h}{\rho C} \frac{P_t}{P_{am}} \left(\frac{Sc}{Pr}\right)^{2/3} \left(\rho_{sg} - \rho_{sw}\right) \qquad (12)$$

An approximation was made in assuming that $\left(\frac{Sc}{Pr}\right)^{2/3} \cong 1.0$ thus the local mass transfer rate was computed from

$$\dot{m} = \frac{h}{\rho C} \frac{P_t}{P_{am}} \left(\rho_{sg} - \rho_{sw}\right) \qquad (13)$$

The heat transfer rate due to condensation was computed from

$$q_1 = \frac{\dot{m} \ \lambda \, hP}{\rho \, CP_{am} \, t} \left(\rho_{sg} - \rho_{sw}\right) \qquad (14)$$

where: ρ_{sg} is evaluated at the local bulk gas temperature
ρ_{sw} is evaluated at the local gas-condensate interface temperature
λ is evaluated at the local gas-condensate interface temperature
P and C are evaluated at the local bulk gas temperature

The heat transfer coefficient, h, was determined from experiments on W plate-fin coils which are the same geometry as are used in this application.

The heat transfer rate, locally, was computed from

$$q_2 = h\,(T_g - T_i)(15)$$

The basis for selecting these values was that the authorities cited as references have shown, through analyses and through cited experiments, that the methods used are accurate.

The air ride pressure drop across the cooling coils under DBA condition was estimated to be approximately 1.9 in. of water or .07 psi. This will have negligible effect on the heat removal capability of the cooling coils.

The pressure of noncondensible gases were taken into consideration by virtue of the fact that the theory behind the analyses assumed that the condensible vapor must diffuse through a noncondensible gas.

Application of this method resulted in the fan-cooler heat removal rate per fan presented in Figure 6.2.2-6.

Nomenclature

$\dot{m}$ = mass flow rate of condensate, lbm/hr-ft^2

h_D = mass transfer coefficient, ft/hr

ρ_{sg} = density of saturated steam at local bulk gas temperature, lbm/ft^3

ρ_{sw} = density of saturated steam at local condensate-gas interface temperature, lbm/ft^3

u_s = free steam gas velocity, ft/min

Sc = Schmidt number, M/pD, dimensionless

μ = viscosity of bulk gas, lbm/ft-hr

ρ = bulk gas density, lbm/ft^3

D = gas-air diffusion coefficient, $\frac{ft^2}{hr}$ 12

St= Stanton number, $h/\rho cu_s$, dimensionless

h= convective heat transfer coefficient, Btu/hr-ft^2-ºF

C= specific heat of bulk gas, Btu/lbm-ºF

P_r= Prandtl number, $\mu c/k$, dimensionless

k= thermal conductivity of bulk gas, Btu/hr-ft-ºF

P_t= total gas pressure, lbf/ft^2

P_{am}= air log-mean $\frac{P_{aw} - P_{ag}}{\ln \frac{P_{aw}}{P_{ag}}}$ lbf/ft^2

P_{aw}= partial pressure of air at the local gas-condensate interface, lbf/ft^2

P_{ag}= partial pressure of air at the local bulk gas temperature, lbf/ft^2

λ = latent heat of vaporization (or condensation) at the local gas-condensate interface temperature, Btu/lbm

q_1= local heat transfer rate due to condensation, Btu/hr-ft^2

q_2= local heat transfer rate due to convection, Btu/hr-ft^2

T_g= local bulk gas temperature, ºF

T_i= local gas-condensate interface temperature, ºF

System Response

The starting sequence of the containment cooling fans and the related emergency power equipment is designed so that delivery of the minimum required air flow and cooling water flow is reached in 46 sec as shown in Section 8.3. In the analysis of the containment pressure transient, Section 6.2.1.1.3, a delay time of 60 sec was assumed for the initiation of containment cooling.

Single Failure Analysis

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 6.2.2-3.

Reliance on Interconnected Systems

The Containment Air Recirculation Cooling System is dependent on the operation of the electrical and service water systems. Cooling water to the coils is supplied from the service water system. Four service water pumps and two service water booster pumps are provided, only two and one of which respectively are required to operate during the post-accident period.

Shared Function Evaluation

Table 6.2.2-4 is an evaluation of the main components which have been discussed previously and a brief description of how each component functions during normal operation and during the accident.

Reliability Evaluation of the Fan Cooler Motor

The basic design of the motor and heat exchanger as described herein is such that the incident environment is prevented, in any major sense, from entering the motor winding or when entering in a very limited amount (equalizing motor interior pressure) the incoming atmosphere is directed to the heat exchanger coils where moisture is condensed out. If some quantity of moisture should pass through the coil, the changed motor interior environment would "clean up" in that interior air continually recirculates through the heat exchanger.

It should be noted that the motor insulation hot spot temperature is not expected to exceed 107ºC even under incident conditions. Considering that rated life could be expected with a continuous hot spot of 130ºC, using the industry accepted 10 degree rule (life is doubled for every 10ºC drop in temperature), the life expectancy would exceed by many times the expected life of motors applied elsewhere in the plant, even if the incident temperatures were experienced on a continuous basis.

During the lifetime of the plant, these motors perform the normal heat removal service and, as such, are only loaded to approximately 120-150 HP.

Motor insulation hot spot is expected to be from 15 to 20ºC below design level or approximately 90ºC with cooling water at maximum summer temperature. In summary, practically none of the insulation life due to thermal aging is used up in normal service and, at incident loading, the motor insulation should have greater than normal life. Incident high temperature, moisture, and load conditions last only a few hours.

The bearings are designed to perform in the incident ambient temperature conditions. However, it should be noted that the interior bearing housing details are cooled by the heat exchanger. It is expected that bearing temperatures would not exceed 125ºC by any significant amount, even under incident conditions.

The insulation has high resistance to moisture, and tests performed indicate the insulation system would survive the incident ambient moisture condition without failure. The heat exchanger system for preventing moisture from reaching the winding therefore provides a design margin. In addition, it should be noted that at the time of the postulated incident, the load on the fan motor would increase, internal motor temperature would increase, and would, therefore, tend to drive any moisture present out of the windings. Additionally, the motors are furnished with insulation margin beyond the operating voltage of 460 V.

Following the incident rise in pressure, a rather slow rise as far as equalizing pressure in the small volumes of the motor-heat exchanger is concerned, it is not expected that there will be significant mixing of the motor (closed system) environment and the containment ambient.

Also all hardware used in connection with the motor and heat exchanger is corrosion resistant.

The heat exchanger has been designed using a very conservative fouling factor. However, if surface fouling reduces the capability of the heat exchanger by one-half, the motor would still have a normal life expectancy, even under incident conditions.

6.2.2.4 Tests and Inspection

6.2.2.4.1 Containment Spray System

All components of the CSS can be inspected periodically to demonstrate system readiness. The pressure containing systems are inspected for leaks. The requirements for Inservice Testing of Class 1, 2, and 3 components are described in Section 3.9.6. The requirements for Inservice Inspection of Class 1 components are described in Section 5.2.4. The requirements for Inservice Inspection of Class 2 and 3 components are described in Section 6.6.

Component Testing

All active components in the CSS were tested both in pre-operational performance test in the manufacturer's shop and in-place testing after installation.

The containment spray pumps can be tested singly by opening the valves in the miniflow line. Each pump in turn can be started by operator action and checked for flow establishment. The spray injection valves can be tested with the pumps shut down.

The spray additive tank valves can be opened periodically for testing. The contents of the tank will be periodically sampled to determine that the proper solution is present.

Initially, the containment spray nozzle availability was tested by blowing smoke through the nozzles and observing the flow through the various nozzles in the containment.

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

System Testing

Permanent test lines for all containment spray loops were located so that the system, up to the isolation valves at the spray header, can be tested. These isolation valves can be checked separately.

The air test lines, for checking initially the spray nozzles, connect downstream of the isolation valves. Air flow through the nozzles is monitored by the use of hot air and infrared thermography.

During the initial pre-operational tests of the spray system, the flow bypass through the spray eductors was checked. This initial and all subsequent system tests were made with the spray additive tank isolation valves closed.

Operational Sequence Testing

The functional test of the SIS described in Section 6.3.4 demonstrated proper transfer to the emergency DG power source in the event of loss of power. A test signal simulating the containment spray signal has been used to demonstrate operation of the spray system up to the isolation valves on the pump discharge.

6.2.2.4.2 Containment air recirculation cooling system

Access is available for visual inspection of the containment air recirculation system components including fans, cooling coils, louvers, and ductwork.

The service water pumps and booster pumps which supply the cooling units, are in operation on an essentially continuous schedule during plant operation, and no additional periodic tests are required. The roughing filters are removed from the air recirculation cooling units during reactor operation.

Component Testing

The roughing filters used in the containment fan cooler system will be installed only during outage conditions. The filters are subjected to standard manufacturer's efficiency and production tests prior to shipment.

Reactor Containment Fan Cooler Motor Unit Tests

The testing program has been completed on the effects of radiation on the WF-8AC "Thermalastic" (Westinghouse Electric Corporation Trademark) epoxy insulation system used in the reactor containment fan cooler motor. Tests description and results are presented in Reference 6.2.2-10.

Fan Cooler Motor Insulation Irradiation Testing

This testing program is an extension of the work reported in Reference 6.2.2-11.

Irradiation of form wound motor coil sections was accomplished up to exposure levels exceeding that calculated for the design basis LOCA. Three coil samples received the following treatment sequence: Irradiation, high-potential test, vibration test, high-potential test, and breakdown voltage test. Nine coil samples received an alternate treatment sequence: Thermal aging, high-potential test, irradiation, high-potential test, vibration test. (Six of nine coil samples), high-potential test and breakdown voltage test.

All coil samples passed the high potential tests. The breakdown voltage levels of all coils were well in excess of those required by the design, and clearly indicate that the reactor containment fan cooler motor insulation system will perform satisfactorily following exposure to the radiation levels calculated for the DBA.

Reactor Containment Fan Cooler Motor Lubricant Irradiation Testing

The lubricant used in the containment fan cooler motors is qualified for its applicable service. Testing documentation is located in the EQ Central File.

RCFC Cooling Coil Test Summary

In the event of a LOCA of a pressurized water reactor system, compressed water at thermodynamic conditions of approximately 600ºF and 2250 psig would flash into the Containment Building. This condition causes the containment atmosphere to become a high pressure steam saturated environment, limited to a maximum pressure of 40 to 60 psig in most dry Containment Buildings. One of the active containment cooling systems employed to remove energy from the atmosphere and reduce the containment pressure is the RCFC System. An integral part of this system are plate-finned cooling coils. These heat exchangers remove sensible heat during normal operation, but become condensers in the post-accident environment. Because there was limited experimental information available concerning the performance of plate-finned cooling coils operating in a condensing environment in the presence of a noncondensible (air), Westinghouse undertook a demonstration test to establish the validity of its selection procedure (Reference 6.2.2-12).

The test method was to subject a scaled coil to a parametric test. These parameters were: containment pressure (with corresponding steam density and temperature), air flow rate, cooling water flow rate, cooling water temperature, and entrained water content. Each parametric test condition was then used as input to the computer program used in coil selections. The results of the test and the computer program predictions were compared to establish the applicability.

In all cases considered, the measured heat transfer rate is greater than that predicted by the computer code predictions. The range of parameters variations was selected to be consistent with the design points of the RCFC coils contained in actual plants. It is apparent that for this specific type of heat exchanger, functioning in the range of environments tested, no moisture separator is needed to protect the coils from excessive waterlogging due to entrained spray droplets.

The extension of the test to full size units is merely an increase in component size and total flow quantities, but not a change in controlling parameters. It is concluded that the test demonstrates that the computer code used to select cooling coil design is valid in defining the heat removal rates of plate-finned tube cooling coil assemblies of RCFC Systems. Therefore, these test demonstrate that Westinghouse fan cooler designs which are selected by this computer program will perform as required in the post-accident containment environment.

The air operated emergency butterfly valves on each air handling unit will be in the safe "open" position during normal power operation.

System Testing

Each fan cooling unit was tested after installation for proper flow and distribution through the duct distribution system. Three or four of the fan cooling units are used during normal operation. A fan not in use can be started from the Control Room to verify readiness.

Operational Sequence Testing

Periodic tests can be conducted to demonstrate proper transfer and sequencing of the fan motor supplies from the emergency DG in the event of loss of outside power as described in Section 6.3.4. These tests can be conducted at the same time as the DG are tested, as described in Section 8.3.

6.2.2.5 Instrumentation

The ESF Instrumentation System actuates (depending on the severity of the condition) the SIS, Containment Isolation, the Containment Air Recirculation Cooling System, and the CSS.

The ESF systems are actuated by the ESF actuation channels. Each coincidence network energizes an ESF actuation device that operates the associated ESF equipment, motor starters, and valve operators. The channels are designed to combine redundant sensors, and independent channel circuitry, coincident trip logic, and different parameter measurements so that a safe and reliable system is provided in which a single failure will not defeat the channel function. The action initiating sensors, bistables, and logic are shown in the figures included in the detailed ESF Instrumentation Description given in Section 7.3.

The ESF actuation circuits are designed on the same "de-energize to operate" principle as the reactor trip circuits with the exception of the containment spray actuation circuit which is energized to operate in order to avoid spray operation on inadvertent power failure.

The spray system will be actuated by the coincidence of two sets of two out of three (Hi-Hi) containment pressure signals. This starting signal will start the pumps and open the discharge valves to the spray header. The valves associated with the spray additive tank will be opened automatically.

The operator can manually actuate the entire system from the Control Room.

The containment air recirculation coolers are normally in use during plant operation. These units are in the automatic sequence which actuates the ESF upon receiving the necessary signals indicating an accident condition, e.g., a high containment pressure signal automatically actuates the SI safety feature sequence which trips any open inlet dampers to the closed position and starts any stopped fan cooler unit.

ESF Instrumentation Equipment

The following instrumentation ensures monitoring of the effective operation of the ESF.

Containment Pressure

Eight channels, monitoring containment pressure, and derived from three pressure taps, reflect the effectiveness of the containment and cooling systems and other ESF. High pressure indicates high temperatures and reduced pressure indicates reduced temperatures. Indicators and alarms are provided in the Control Room to inform the operator of system status and to guide actions taken during recovery operations. Containment pressure indication will be used to distinguish between various incidents.

Redundant containment pressure signals are provided to isolate the containment. The containment pressure is sensed by eight separate pressure transmitters located outside the containment. Containment pressure is communicated to the transmitters through three 1 in. stainless steel lines penetrating the containment vessel.

Each of the three pairs of differential pressure transmitters external to the containment in the Auxiliary Building have their own connection to the containment. Remote indicating facilities, and alarm and control signals are provided from each transmitter.

Remote indicating facilities have been provided which afford the operator the opportunity to read containment pressure.

Refueling Water Storage Tank Level

Level instrumentation on the refueling water storage tank consists of three channels. One channel provides a local indication. The second channel provides remote indication (on the control board) low level alarm, low-low level alarm and a high level alarm. The third channel provides remote indication on the control board.

Containment Spray Flow

Instrumentation monitoring containment spray and additive flow is described in Section 6.5.2.5.

Pump Energization

All pump motor power feed breakers indicate that they have closed by energizing indicating lights on the control board.

Valve Position

All ESF remote-operated valves have position indication on the control board to show proper positioning of the valves. Air-operated and solenoid-operated valves are selected to move in a preferred direction with the loss of air or power. After a loss of power to the motors, motor-operated valves remain in the same position as they were prior to the loss of power.

Air Coolers

The cooling water discharge flow of each of the coolers is alarmed in the Control Room if the flow is low. The transmitters are outside the reactor containment. In addition, the exit flow is monitored for radiation 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. Local water differential pressure and outlet water pressure indications are provided, in addition to, local temperature (RTD) indications on the water and air side inlets and outlets of each unit to provide data monitoring.

Containment Atmospheric Hydrogen

Two hydrogen concentration monitors are provided, with readout in the Control Room. RNP committed to maintain a containment hydrogen monitoring system designed to the Category 3 criteria of Regulatory Guide 1.97 as part of the justification for the removal of the requirements for these monitors from the Technical Specifications, which was approved in License Amendment No. 216.

Sump Instrumentation

The containment sump instrumentation consists of two analog instrument channels and two channels of eight-point level switches with gasketed junction boxes designed to operate in a post-accident environment. The indicators and alarm system are located in the Control Room.

Alarms

Visual and audible alarms are provided to call attention to abnormal conditions. The alarms are of the individual acknowledgement type; that is, the operator must recognize and silence the audible alarm for each alarm point.

TABLE 6.2.2-1

SINGLE FAILURE ANALYSIS - CONTAINMENT SPRAY SYSTEM

COMPONENT	MALFUNCTION	COMMENTS AND CONSEQUENCES
Spray Nozzles	Clogged	Large number of nozzles (116) renders clogging of a significant number of nozzles as incredible.
Pumps		
1) Containment Spray Pump	Fails to start	Two provided. Evaluation based on operation of one pump in addition to two out of four containment cooling fans operating during injection phase.
2) Residual Heat Removal Pump	Fails to start	Two provided. Evaluation based on operation of one pump and no containment cooling fans operating during recirculation phase.
3) Service Water Pump	Fails to start	Four provided. Operation of two pumps during recirculation required.
4) Component Cooling	Fails to start	Three provided. Operation of one pump during recirculation required.
Automatically operated Valves: (Open on coincidence of two - 2/3 high [Hi-Hi] containment pressure signals)		
1) Containment spray pump discharge isolation valve	Fails to open	Two provided. Operation of one required (per header).
Valves Operated From Control Room		

TABLE 6.2.2-1 (Cont'd)

COMPONENT		MALFUNCTION	COMMENTS AND CONSEQUENCES
(a)	Injection		
1)	Spray Additive Tank outlet isolation valve	Fails to open	Two provided. Operation of one required
(b)	Recirculation		
1)	Containment sump recirculation isolation	Fails to open	Two lines in parallel, each with two valves in series. One line required.
2)	Containment spray pump isolation valve from residual heat exchangers	Fails to open	Two valves provided. Operation of one required
3)	Residual heat removal pump recirculation line	Fails to close	Two valves in series, one required to close.

TABLE 6.2.2-2

SHARED FUNCTIONS EVALUATION CONTAINMENT SPRAY SYSTEM

COMPONENT	NORMAL OPERATING FUNCTION	NORMAL OPERATING ARRANGEMENT	ACCIDENT FUNCTION	ACCIDENT ARRANGEMENT
Spray Additive Tank	None	Discharge valves closed	Source of sodium hydroxide for spray water	Discharge valves open
Containment Spray Pumps (2)	None	Discharge valves closed	Supply spray water to containment atmosphere	Discharge valves open

NOTE: Refer to Section 6.2 for a brief description of the refueling water storage tank, residual heat removal pumps, conventional service water pumps, component cooling pump, residual heat exchangers, and component cooling heat exchangers which are also associated either directly or indirectly with the Containment Spray System.

TABLE 6.2.2-3

SINGLE FAILURE ANALYSIS - CONTAINMENT AIR RECIRCULATION COOLING SYSTEM

COMPONENT		MALFUNCTION	COMMENTS AND CONSEQUENCES
A.	Containment Cooling Fan	Fails to start	Four provided. Evaluation based on two fans and one containment spray pump operating during the injection phase.
B.	Service Water Pumps	Fails to start	Four provided. Two required for operation.
	Service Water Booster Pumps	Fails to start	Two provided. One required for operation.
C.	Automatically Operated Valves: (Open on automatic safeguards sequence signal)		
	Nuclear service water discharge from fan cooler units.	Fails to open	One valve per fan cooler unit. Operation of two units required.
D.	Automatically Operated Louvers: (Inlet damper closes and butterfly valve opens or remains open on automatic safeguards sequence signal)	Fails to open	Four fan-cooler units provided. Evaluation based on two units and one containment spray pump in operation during the injection phase.

TABLE 6.2.2-4

SHARED FUNCTION EVALUATION CONTAINMENT AIR RECIRCULATION COOLING SYSTEM

COMPONENT	NORMAL OPERATING FUNCTION	NORMAL OPERATING ARRANGEMENT	ACCIDENT FUNCTION	ACCIDENT ARRANGEMENT
Containment Fan Cooling Units (4)	Circulate and cool contain-ment atmosphere	Three or four fan units in service	Circulate and cool contain-ment atmosphere	Two fan units in service is required
Service Water Pumps (4)	Supply lake cooling water to fan units	Three pumps in service	Supply lake cooling water to fan units	Two pumps in service
Service Water Booster Pumps (2)	Supply lake cooling water to fan unit	One pump in service	Supply lake cooling water to fan units	One pump in service

TABLE 6.2.2-5

RCFC - MOTOR AND FAN BEARING LUBRICANT IRRADIATION TESTING

SAMPLE	MICRO-CONE PENETRATION				
	UNWORKED	60 STROKES	500 STROKES	1000 STROKES	50,000 STROKES
Unirradiated Chevron BRB-2	308	320	368	370	>400
Irradiated BRB-2 1.2 X 10^8R	300	300	308	324	400
Irradiated BRB-2 1.5 X 10^8R	308	288	292	298	364
Irradiated BRB-2 1.8 X 10^8R	340	320	304	296	280

6.2.3 SECONDARY CONTAINMENT FUNCTIONAL DESIGN

This Section does not apply to HBR 2.

6.2.4 CONTAINMENT ISOLATION SYSTEM (CIS)

6.2.4.1 Design Basis

Each system whose piping penetrates the containment leakage limiting boundary was designed to maintain or establish isolation of the containment from the outside environment under the following postulated conditions:

a) Any accident for which isolation was required (severely faulted conditions), and

b) A coincident independent single failure or malfunction (expected fault condition) occurring in any active system component within the isolated bounds.

Piping penetrating the containment was designed for pressures at least equal to the containment design pressure. Isolation valves were provided as necessary for all fluid system lines penetrating the containment to assure at least two barriers for redundance against leakage of radioactive fluids to the environment. 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.

These barriers, in the form of isolation valves or closed systems, were defined on an individual line basis. In addition to satisfying containment isolation criteria, the valving was designed to facilitate normal operation and maintenance of the systems and to ensure reliable operation of other engineered safeguards systems. With respect to numbers and locations of isolation valves, the criteria applied were generally those outlined by the six classes described in Section 6.2.4.2 below.

In general, isolation of a line outside the containment protects against rupture of the line inside concurrent with a LOCA, or closes off a line which communicates with the containment atmosphere in the event of a LOCA.

Isolation of a line inside the containment prevents flow from the RCS or any other large source of radioactive fluid in the event that a piping rupture outside the containment occurs. A piping rupture outside the containment at the same time as a LOCA was not considered credible, as the penetrating lines are seismic Class I design at least up to and including the second isolation barrier and were assumed to be an extension of containment.

The system was designed such that a single component failure will not prevent isolation, and no manual operation is required for immediate isolation. Automatic isolation is initiated by a containment isolation signal, Section 7.3, derived either from any automatic safety injection (SI) signal ("T" signal) or from a high containment pressure signal ("P" signal).

The containment isolation valves have been examined to assure that they are capable of withstanding the maximum potential seismic loads. To assure their adequacy in this respect:

1. Valves were located in a manner to reduce the accelerations on the valves. Valves suspended on piping spans were reviewed for adequacy for the loads to which the span would be subjected. Valves were mounted in the position recommended by the manufacturer.

2. Valve yokes were reviewed for adequacy and strengthened as required for the response of the valve operator to seismic loads.

3. Where valves are required to operate during seismic loading, the operate forces were reviewed to assure that system function is preserved. Seismic forces on the operating parts of the valve are small compared to the other forces present.

4. Control wires and piping to the valve operators were designed and installed to assure that the flexure of the line does not endanger the control system. Appendages to the valve, such as position indicators and operators, were checked for structural adequacy.

6.2.4.2 System Design

The six classes listed below are general categories into which lines penetrating containment may be classified. The seal water referred to in the listing of categories is provided by the Isolation Valve Seal Water (IVSW) System described in Section 6.8. The following notes apply to these classifications.

1. The "not missile protected" designation refers to lines that are not protected throughout their length inside containment against missiles generated as the result of a LOCA. These lines, therefore, were not assumed invulnerable to rupture as a result of a loss of coolant.

2. In order to qualify for containment isolation, valves inside the containment must be located outside the missile barrier for protection against loss of function following an accident.

3. Manual isolation valves that are locked closed or otherwise closed and under administrative control during power operation qualify as automatic trip valves.

4. A check valve qualifies as an automatic trip valve in certain incoming lines not requiring seal water injection.

5. The double disk type of gate valve was used to isolate certain lines. When sealed by water injection, this valve provides both barriers against leakage of radioactive liquids or containment atmosphere.

6. In lines isolated by globe valves and require seal water injection, seal water is injected between the valves, which are installed so that the zone between the seat and the stem packing contains seal water. Thus, any leakage past the seat or stem packing will be seal water and not containment atmosphere. When the valves are closed for containment isolation, the first isolation point is the valve plug in the valve closest to containment, and the water seal is applied between the valve plug and stem packing.

7. Excessive loss of seal water through an isolation valve that fails to close on signal, is prevented by the high resistance of the seal water injection line. A water seal at the failed valve is assured by proper slope of the protected line, or a loop seal, or by additional valves on the side of the isolation valves away from the containment.

8. Isolated lines between the containment and the second outside isolation barrier (valve or closed system) were designed to the same seismic criteria as the containment vessel, and were assumed to be an extension of containment.

9. The first outside isolation valve is located as close to the containment as possible unless a more remote location was dictated by equipment isolation requirements.

The six line classes listed below are general categories into which lines penetrating containment may be classified. For the purpose of line classification, the term “normally operating line” is defined as a line used during normal plant operations, including startup and shutdown. Reference 6.2.4-1 should be consulted for specific penetration classifications.

Class 1 Lines have the following characteristics:

1. The line is connected to the RCS, i.e., it normally experiences full RCS pressure and temperature.
2. The line is normally operating.
3. The line is outgoing, i.e., utilized to carry fluids out of containment.

At least two automatic trip valves in series with automatic seal water were provided. The valves typically were located outside containment.

Exceptions to the general classification are the residual heat removal loop outlet line (P-16) and the reactor coolant pump seal water return line (P-28). The two barriers for P-16 are a normally closed missile protected valve inside containment, and the closed residual heat removal loop outside containment. The two barriers for P-28 are an automatic trip valve (double disc gate valve with automatic IVSW) and the closed chemical and volume control system, both outside containment.

Class 2 Lines have the following characteristics:

1. The line is not a closed system inside containment.
2. The line is not connected to the RCS.
3. The line is normally operating.
4. The line is not missile protected.
5. The line is outgoing, i.e., utilized to carry fluids out of containment.

At least two automatic trip valves in series with automatic seal water injection were provided. The valves are typically located outside containment.

Class 3 Lines have the following characteristics:

1. The line is not a closed system inside containment.
2. The line is incoming, i.e., utilized to carry fluids into containment.
3. The line is not missile protected.

This line classification has the following two subcategories:

1. Open system outside containment and open system inside containment.
    a. For lines non-essential to plant operation following an accident two automatic trip valves in series, with automatic seal water injection were provided. The valves are typically located outside containment. Seal water was considered unnecessary for such lines connected to non-radioactive systems outside containment where a pressure gradient exists that opposes leakage from containment, e.g., the nitrogen supply lines to the pressurizer relief tank (P-2), the accumulators (P-65), the reactor coolant drain tank (P-4), the instrument air header (P-33), and plant air header (P-39). Containment fire water penetrations (P-73 and P-74) do not have seal water injection.
    b. For lines essential to plant operation following an accident, two manual valves in series with manual seal water injection were provided. The valves typically were located outside containment.

2. Closed system outside containment and open system inside containment.
    a. These configurations were provided, as a minimum, with one check valve or a normally closed isolation valve. The valves were located either inside or outside containment. Seal Water Injection is not required for lines in this category.

Class 4 Lines have the following characteristics:

1. The line is a closed system inside containment.
2. The line is normally operating.
3. The line is missile protected throughout its length.

At least one manual isolation valve located outside containment should be provided.

Class 5 Lines have the following characteristics:

1. The line is not a closed system inside containment.
2. The line is not normally operating.

Two isolation valves in series, or one isolation valve and one blind flange/mechanical connection should be provided. One valve or flange should be located inside containment, and the second valve or flange should be located outside containment. The containment sump recirculation lines (P-46 and P-47) are exceptions as both isolation valves are located outside containment.

Class 6 (Special Service)

There are a number of special groups of penetrating lines and containment access openings. These are discussed below.

Each ventilation purge duct penetration (P-37 and P-38) is provided with two tight-closing butterfly valves, which are closed during reactor power operation and are actuated to the closed position automatically upon a containment isolation or a containment high radiation signal. One valve is located inside and one valve is located outside the containment at each penetration.

The containment pressure and vacuum relief lines (P-41 and P-42) are similarly protected with two tight closing butterfly valves in series, one inside and one outside the containment. These valves also are actuated to the closed position upon a containment isolation or containment high radiation signal.

The equipment access hatch is a bolted, gasketed closure which is sealed during reactor operation. The personnel air locks consist of two doors in series with mechanical interlocks to assure that one door is closed at all times. Each air lock door and the equipment closure are provided with double gaskets to permit pressurization between the gaskets.

The fuel transfer tube penetration (P-32) inside the containment, Figure 3.8.1-16, is designed to present a missile protected and double barrier between the containment atmosphere and the atmosphere outside the containment. The penetration closure is treated in a manner similar to the equipment access hatch. The inside closure is a blind flange which contains two gaskets to complete the double barrier between the containment atmosphere and the inside of the fuel transfer tube. The interior of the fuel transfer tube is not pressurized. Seal water injection is not required for this penetration.

The containment radiation monitor inlet and outlet lines (P-35 and P-36) communicate with the containment atmosphere at all times (normally filled with air or vapor). In an accident condition the two containment isolation valves close.

The Reactor Vessel Level Instrumentation System (RVLIS) sensing lines (P-75 through P-80) are utilized post accident. Each line is isolated by a hydraulic isolator outside containment.

The containment pressure sensing lines (P-68, P-69 and P-70) are open to containment atmosphere and remain open to pressure transmitters post accident. Redundant, closed globe valves isolate the attached Post Accident Sampling System.

Figures 6.2.4-1 through 6.2.4-19 show the containment isolation provisions credited for each containment penetration. Figure 6.2.4-21 defines the nomenclature and symbols used on the aforementioned figures.

A summary of the fluid systems lines penetrating containment and the valves and closed systems employed for containment isolation is presented in Table 6.2.4-1. Each valve is described as to type, operator, position indication and open or closed status during normal operation, shutdown, and accident conditions. Information is also presented on valve preferential failure mode, automatic trip by containment isolation signal, and the fluid carried by the line.

Containment isolation valves were provided with actuation and control equipment appropriate to the valve type. For example, air-operated globe and diaphragm (Saunders Patent) valves are generally equipped with air diaphragm operators, with fail-safe operation assured by the control devices in the instrument air supply to the valve. Motor-operated gate valves are capable of being supplied from reliable onsite emergency power as well as from their normal power source.

Automatically operated containment isolation valves are actuated to the closed position by one of two separate containment isolation signals. The first of these signals is derived in conjunction with automatic safety injection actuation, and trips the majority of the automatic isolation valves. These are valves in the so-called "non-essential" process lines penetrating the containment. This was defined as "Phase A" isolation, and the trip valves were designated by the letter "T" in the isolation diagrams (Figures 6.2.4-1 through 6.2.4-20). This signal also initiates automatic seal water injection. The second, or "Phase B", containment isolation signal was derived upon actuation of the containment spray system, and trips the automatic isolation valves in the so-called "essential" process lines penetrating the containment. "Essential" process lines are those providing cooling and seal water flow through the reactor coolant pumps. These services should not be interrupted unless absolutely necessary while the reactor coolant pumps are operating. These trip valves were designated by the letter "P" in the isolation diagrams.

Some automatically tripped isolation valves are actuated to the closed position by the containment ventilation isolation signal. These valves were designated by the letter "V" in the isolation diagrams. The "V" signal is derived from Safety Injection, Containment High Radiation, or manually.

Manual containment isolation signals can be generated from the Control Room. These signals perform the same functions as the automatically derived "T" signal (i.e., "Phase A" isolation and automatic seal water injection) and "P" signal (i.e., "Phase B" isolation).

Non-automatic isolation valves, i.e., remote stop valves and manual valves, were used in lines which 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.

"Non-essential" lines are defined as lines which are not required to mitigate or limit an accident and which, if required at all, would be required for long term recovery only. "Essential" lines are defined as lines required to mitigate an accident or which, if unavailable, could increase the magnitude of the event.

Standard closing times available with commercial valve modes were adequate for the sizes of containment isolation valves used. Valves equipped with air-diaphragm operators generally close in approximately two seconds. The typical closing time available for large motor-operated gate valves was ten seconds.

The large butterfly valves used to isolate the containment ventilation purge ducts were equipped with air-cylinder operators, with spring returns capable of closing the valves in two seconds. These valves fail to the closed position on loss of control signal or instrument air.

The following types of isolation valves were generally employed outside the containment:

1. Diaphragm valves (Saunders Patent)

2. Globe valves

3. Double disk gate valves

4. Regular gate valves, and

5. Butterfly valves.

Isolation valves with packed stems were provided with stem leak offs if the following operating conditions were satisfied with the exception of those valves which have had their leak off lines capped:

1. Line size is 2 in. or larger

2. Operating temperature can exceed 212ºF, and

3. The fluid is radioactive.

All air and motor operated containment isolation valves can be remotely operated from the Control Room. The open or closed conditions of these valves are displayed visually in the Control Room, with the exception of MS-353A/B/C. Post Accident Venting valves associated with penetrations P-40 and P-41 are exceptions to this criteria.

Only the valves located inside the containment which were missile protected can be considered as available for containment isolation. These valves were located outside the missile barrier.

Typically, lines penetrating the containment which normally carry radioactive fluids or that can communicate with the containment atmosphere following an accident were provided with radiation shielding in areas where personnel access is possible. Manually operated valves in the non-radioactive seal water injection lines were located outside the shielding.

Valves that are normally open during power operation and which must be closed for containment isolation are actuated to the closed position on receipt of a containment isolation signal.

Redundant electrical control circuits were provided for all remotely operated containment isolation valves. If the normal power supply for the control circuits fails, they may be energized by an emergency power supply. Duplicate cabling to the valve operators was not provided.

All air operated isolation valves fail closed on loss of control signal or control air. This is not detrimental to power operation. If one of the isolation valves should fail closed, operation of the connected systems either is not affected or can be modified until repairs are made.

It was necessary to demonstrate that containment isolation barriers were leak-tight. The closed systems that back up the containment isolation valves have adequate capability for flow toward the containment or adequate design to contain any radioactivity introduced into the system as the result of an accident. The water seal maintained between certain closed isolation valves by seal water injection was designed to prevent leakage of containment atmosphere to the environment by ensuring that any leakage through the valve seats or past stem packing is seal water, not containment atmosphere.

In general, vertical water legs were not used to seal the closed isolation valves. However, on lines isolated by two remotely operated valves in series, a loop seal or vertical water leg was installed between the isolation valves and the containment. This prevents loss of the water seal provided by seal water injection if the first outside isolation valve fails to close and the line is exposed to the containment atmosphere. Presence of water in the loop seal or vertical leg is assured by the inflow of seal water.

Penetrating lines other than those associated with the engineered safety features (ESF) which continue to be used, at least for a time, after containment isolation include:

1. Main steam headers

2. Auxiliary feedwater headers

3. Reactor coolant pump cooling water supply lines

4. Reactor coolant pump cooling water return lines

5. Reactor coolant pump seal water supply lines

6. Containment air sample in if containment pressure <5 psig,

7. Containment air sample out if containment pressure <5 psig, and

8. Reactor vessel level instrumentation system lines.

Automatic isolation valve sizes are listed in Table 6.2.4-2.

6.2.4.3 Tests and Inspections

The HBR 2 containment structure was designed such that the maximum allowable containment vessel leakage rate shall not exceed 0.1 percent per day of the containment atmosphere at 42 psig and 263ºF which are the maximum conditions of the DBA.

Leakage from the containment to the outside could occur in the following locations:

1. Containment Penetrations (L_{pen})

2. Containment Liner Welds (L_c)

3. Containment Liner Plates (L_L), and

4. Containment Isolation Valves (L_{iso}).

The leakage from the penetrations (L_{pen}) may be continuously or intermittently monitored by the PPS as described in Section 6.9. The PPS can provide pressurization to several volumes formed by double containment isolation valves or by double gasketed seals. These include the spaces between butterfly type isolation valves in the purge supply and exhaust lines, containment pressure and vacuum relief lines, the double isolation valves in the containment radiation monitor inlet and outlet lines, the plant air supply header and the post accident venting line, and into the spaces formed by double gaskets in the fuel transfer tube and on the equipment hatch and personnel lock doors. Leakage designated by L_{pen} was defined to

include leakage from these volumes as well as from the penetration sleeves. In this context the word "penetration" also includes these volumes. The PPS is used to perform a sensitive leak rate test of these volumes to verify that leakage to the outside does not exceed the design limits at accident pressure (i.e., 42 psig).

Containment liner weld channels were installed on all liner welds to provide the means for a sensitive leak rate test to determine liner weld leakage (L_c). However, the liner weld leakage is no longer determined and the integrity of the containment welds was verified by periodic integrated leakage rate testing.

Containment isolation valves were individually tested prior to the preoperational leak rate tests to assure proper seating. The design of the lines which penetrate the containment boundary provide isolation valves and additional positive means for limiting the leakage (L_{iso}) which can occur from the containment atmosphere through these lines in the post-accident condition. Table 6.2.4-1 lists each fluid line which penetrates the containment wall and indicates the additional positive barriers which will minimize leakage through these lines from the containment following an accident. These positive barriers include injection of IVSW System water at a pressure greater than accident pressure between the seats and stem packing of the globe and double disc types of isolation valves and into piping between closed diaphragm type isolation valves.

Other lines are all located outside the missile barrier and are connected to closed systems within the containment, or are part of a system with design pressure greater than the design pressure of the containment. Therefore, the isolation valve arrangement and these positive barriers will assure minimal leakage after a DBA through these potential leak paths.

No leakage was expected through the liner plates (L_L). However, any liner plate leakage will be measured as part of the preoperational integrated leakage rate test. The containment liner has insulation from the area of the "spring line" to the base mat. This polyvinyl chloride (PVC) foam insulation has a sheet stainless steel outer covering. Any physical damage to this insulation and thus to the underneath liner would be readily observable.

Following the preoperational tests, periodic inspection of the containment wall was conducted to ensure that no physical damage to the liner has occurred. Evidence of damage would be examined to determine the necessary methods for assuring that the liner plate(s) in the affected area will not leak at containment design conditions. Therefore, no periodic leak rate testing of the liner plates is required unless physical damage was evident.

The preoperational integrated leak rate test was conducted with containment atmosphere at approximately 42 psig and 90ºF. The corresponding test leakage

rate limit was 0.0761 percent of the containment free volume per day. The total integrated leakage rate of the containment vessel was described by the following equation:

$$L_{pm} = L_{pen} + L_c + L_{iso} + L_L \qquad (1)$$

The actual integrated leak rate (L_{pm}) measurement was preceded by individual leak testing of containment isolation valves and penetrations to determine and correct, if necessary, leakage paths which might be present. The integrated leak rate (L_{pm}) was determined during a twenty-four hour test period by taking hourly readings of containment internal pressure, temperature, and dew point temperature. These parameters were then introduced into an equation derived from the ideal gas law to establish the weight of air lost per unit time. The limit on weight lost per unit time at the test conditions of 42 psig and 90ºF is 446 lb/day.

The error (p) associated with the integrated leak rate measurement must be considered to evaluate the results of the test. Therefore, the following relationship must be satisfied to assure that the containment vessel meets its design criterion:

$$L_{pm} + \sigma_p \leq 0.0761 \text{ percent per day at 42 psig and 90ºF} \qquad (2)$$

The integrated leak rate test (Test Case I) was conducted with the containment isolation valves in their post-accident condition without utilization of the IVSW System, and without PPS pressure to the penetrations. Pressure buildup was observed to evaluate leakage from the containment. This measurement of the integrated leakage rate along with the preoperational sensitive leakage rate tests provides a basis for evaluation of operational sensitive leak rate tests, to confirm periodically that the containment leakage rate is within the design limit.

After the twenty-four hour integrated leakage rate test, a controlled leakage rate equal to 0.0761 percent per 24 hr was superimposed on the containment for 12 hr. This procedure is intended to validate the method and instrumentation used for the integrated leak rate test.

Leakage through containment penetrations (L_{pen-2}) to the outside was measured by the first phase of the preoperational sensitive leak rate test following the integrated leak rate test.

The containment pressure was then reduced to 41 psig and the penetrations were pressurized to 42 psig (Test Case II). This arrangement prevents inleakage from the containment atmosphere to the test channel volume during the sensitive leak test. Leakage to the outside was measured using the flow instrumentation of the PPS, and was subject to the measurement error, σ_{pen-2}. The containment pressure was then reduced to 0 psig and the penetrations were pressurized to 42 psig (Test Case III). The leakages from these volumes, L_{pen-3}, respectively, were then measured. These leakages were subject to measurement error, σ_{pen-3}, and represents the total leakage from these volumes to both the containment interior and to the outside environment.

Therefore, these sensitive leak rate tests permitted determination of leakage through the outer and inner barriers of the penetrations.

The integrated leak rate tests are summarized as follows:

1. <u>Test Case I</u>

   Condition: Containment at 42 psig, penetrations at 0 psig.

   $L_{pen-1} + \sigma_{pen-1}$ describes penetration inleakage.

2. <u>Test Case II</u>

   Conditions: Containment at 41 psig, penetrations at 42 psig.

   $L_{pen-2} + \sigma_{pen-2}$ describes penetration leakage to the outside assuming no leakage to the containment.

3. <u>Test Case III</u>

   Conditions: Containment at 0 psig, penetrations at 42 psig.

   $L_{pen-3} + \sigma_{pen-3}$ describes penetration leakage.

Leakage to outside from the penetrations was assumed to be equal for Test Cases II and III since the penetrations were at 42 psig.

A value for leakage through liner plates and isolation valves ($L_L + L_{iso}$) can be obtained from the preoperational integrated leak rate and sensitive leak rate tests by substituting the measured integrated leakage into Equation. (1)

$$L_{pm} + \sigma_p = L_{pen} + L_c + L_{iso} + L_L \tag{3}$$

where $L_{pen} + L_c$ was obtained from the sensitive leak rate tests (Test Case II). Therefore:

$$L_{iso} + L_L = L_{pm} + \sigma_p - [\,(L_{pen-2}\ \ \sigma_{pen-2}) + (L_{c-2}\ \ \sigma_{c-2})\,] \tag{4}$$

Therefore, the sum of leakages through isolation valves and liner plates ($L_{iso} + L_L$) can be estimated during the preoperational leak rate testing.

Periodic sensitive leak rate tests are performed on the penetrations and isolation valve and seals receiving PPS pressure to assure that the leakage from these most probable leak paths has not increased sufficiently since the preoperational testing to result in a containment leak rate exceeding 0.1 percent of the containment volume at design conditions. This periodic sensitive leak rate test (Test Case IV) was performed with the containment at zero psig and the penetrations, isolation valves and seals at 42 psig, which were the same test conditions at Test Case III discussed above. Leakage from the weld channels and penetrations, L_{c-4} and L_{pen-4}, and their respective measurement errors, σ_{c-4} and σ_{pen-4}, can be represented by:

$L_{c-4} + \sigma_{c-4}$ describes weld channel leakage.

$L_{pen-4} + \sigma_{pen-4}$ describes penetration leakage.

The current status of the containment leakage can then be established by comparing the results of the periodic operational sensitive leak test with preoperational leak rate data. From preoperational sensitive leak rate testing, $L_{pen-3} + L_{c-3}$ was measured. These measurements include leakage from the penetrations to the inside and outside of the containment, and leakage from the isolation valves and seals. Any increase in the periodic sensitive leak rates $L_{pen-4} + L_{c-4}$ over preoperational sensitive leak rates was assumed to be leakage to the outside. This assumption is conservative in that any leakage increase detected in periodic sensitive leak rate testing might consist of increased leakage to the containment interior as well as increased leakage to the environment. The periodic sensitive leak rates, $L_{pen-4} + L_{c-4}$, were measured to establish the current condition of the containment.

Therefore, the total containment leakage (L_p) as a result of increased leakage detected by periodic sensitive leak rate testing would be:

$$L_{pm} + \sigma_p + [\,(L_{pen-4} + \sigma_{pen-4}) - (L_{pen-3} \pm \sigma_{pen-3})\,] \qquad (5)$$

$$[\,(L_{c-4} + \sigma_{c-4}) - (L_{c-3} \pm \sigma_{c-3})\,] \leq 0.0761 \text{ percent}$$

of the containment volume at 42 psig and 90ºF.

Equation (5) represents the criterion which must be satisfied as a result of the periodic sensitive leak rate tests to verify that the containment leak rate does not exceed its design limit. The equation takes into account the preoperational integrated leak rate and the relative changes in sensitive leak rates as determined by periodic sensitive leakage measurements.

Initial failure of the sensitive leak rate test to verify this relationship will result in efforts to reduce leakage from the penetrations, and isolation valves and seals so that a sensitive leak rate retest will demonstrate compliance with the design limit.

Periodic sensitive leak rate testing and preoperational leak rate testing discussed above describe a reasonable approach to assuring that the containment leakage is maintained below design limits during the life of the plant. Periodic sensitive leak testing provides a very accurate method of monitoring changes in the leakage characteristics of the containment. An integrated leak rate test was performed if major maintenance or modification to the containment was made.

The requirements for Inservice Testing of Class 1, 2, and 3 components are described in Section 3.9.6. The requirements for Inservice Inspection of Class 1 components are described in Section 5.2.4. The requirements for Inservice Inspection of Class 2 and 3 components are described in Section 6.6.

6.2.4.4 Gas Analyzer Isolation Valves

The control circuitry associated with the containment isolation valves in the Gas Analyzer Sample Line from the Pressurizer Relief Tank (RC-516 and RC-553) and the containment isolation valves in the Gas Analyzer Sample Line from the Reactor Coolant Drain Tank (WD-1789 and WD-1794) is such that the valves will close in any and all of the following cases:

1. "CLOSE" command from the Gas Analyzer Panel

2. Containment Phase A Isolation Signal, and

3. Loss of Power.

If the valves have closed as a result of a "CLOSE" command from the Gas Analyzer Panel, they can be reopened by an "OPEN" command from the Gas Analyzer Panel as long as there has been no Containment Phase A Isolation Signal or Loss of Power. In the latter two cases, the valves can only be reopened by first resetting the Phase A Isolation Signal and/or reestablishing power to the circuitry.

The valves will continue to remain closed until the corresponding Isolation Reset Pushbuttons (one per valve) are depressed. Once this is accomplished, the valves will reopen. These valves were initially resettable in a ganged fashion, but are presently resettable on a valve-by-valve basis. These valve systems have been modified such that the resetting of containment isolation will not result in automatic reopening.

TABLE 6.2.4-1

CONTAINMENT PIPING PENETRATIONS AND VALVING

PENE .NO.	PENETRATION AND SYSTEM	FIG. 6.2.4-	VALVE OR BARRIER (SEE FIGURE)	VALVE TYPE	OPER. TYPE	POSIT. INDIC. IN CONT. ROOM	NORMAL POSIT.	POSIT. DURING SHUTDOWN	POSIT. AFTER ACCIDENT	POSIT. ON POWER FAIL.	CONT. ISOL. TRIP	SEAL WATER INJ.	USED AFTER ACCID.	FLUID G-GAS W-WATER	NOTES
1	Pressurizer Relief Tank Gas Analyzer Line	-1	RC-553 RC-516	Globe Globe	Air Air	Yes Yes	Closed Closed	Closed Closed	Closed Closed	FC FC	T T	A A	No No	G G	NE
2	Pressurizer Relief Tank N_2 Supply Line	-1	RC-518 RC-550	Check Dia.	- Air	No Yes	Closed Open	Closed Open	Closed Closed	- FC	No T	- -	No No	G G	NE
3	Pressurizer Relief Tank Makeup	-1	RC-519B RC-519A	Dia. Dia.	Air Air	Yes Yes	Closed Closed	Closed Closed	Closed Closed	FC FC	T T	A	No No	W W	NE
4	Primary System Vent Header and N_2 Supply Line	-2	WD-1786 WD-1787 WD-1793 WD-1713	Dia. Dia. Dia. Check	Air Air Man. -	Yes Yes No No	Open Open Closed Closed	Open Open Closed Closed	Closed Closed Closed Closed	FC FC - -	T T No No	A A - -	No No No No	G G G G	NE
5	Reactor Coolant Drain Tank Gas Analyzer Line	-2	WD-1794 WD-1789	Dia. Dia.	Air Air	Yes Yes	Closed Closed	Closed Closed	Closed Closed	FC FC	T T	A A	No No	G G	NE
6	Reactor Coolant Drain Tank Pump Discharge Line	-3	WD-1721 WD-1722	Dia. Dia.	Air Air	Yes* Yes*	Open Open	Open Open	Closed Closed	FC FC	T T	A A	No No	W W	*Ganged indication in CR, individual indication at WDSP,NE
7	Main Stream Header	-3	MS-V1-3A MS-353A MS-10A MS-19 MS-21 RV1-1 MS-262A C.S.	SDSV DDV Globe Globe Globe PORV Gate -	Air Mot. Man. Man. Man. Air Man. -	Yes No No No No Yes No -	Open L.C. L.C. LC LC Closed L.O. -	Closed Closed Closed Closed** Closed** Closed Open -	Open* Closed Closed Closed Closed Closed Open -	FC As is - - - FC - -	No* No No No No No No -	- - - - - - - -	Yes* No No No No Maybe Yes -	G G G G G G G -	*Automatic isolation for MSLB,E **May be opened for RCS temperature control

TABLE 6.2.4-1 (Continued)

PENE. NO.	PENETRATION AND SYSTEM	FIG. 6.2.4-	VALVE OR BARRIER (SEE FIGURE)	VALVE TYPE	OPER. TYPE	POSIT. INDIC. IN CONT. ROOM	NORMAL POSIT.	POSIT. DURING SHUTDOWN	POSIT. AFTER ACCIDENT	POSIT. ON POWER FAIL.	CONT. ISOL. TRIP	SEAL WATER INJ.	USED AFTER ACCID.	FLUID G-GAS W-WATER	NOTES
8	Main Stream Header	-3	MS-V1-3B	SDSV	Air	Yes	Open	Closed	Open*	FC	No*	-	Yes*	G	*Automatic isolation for MSLB,E
			MS-353B	DDV	Mot.	No	L.C.	Closed	Closed	As is	No	-	No	G	
			MS-11A	Globe	Man.	No	L.C.	Closed	Closed	-	No	-	No	G	
			MS-28	Globe	Man.	No	L.C.	Closed**	Closed	-	No	-	No	G	**May be opened for RCS temperature control
			MS-30	Globe	Man.	No	L.C.	Closed**	Closed	-	No	-	No	G	
			RV1-2	PORV	Air	Yes	Closed	Closed	Closed	FC	No	-	Maybe	G	
			MS-262B	Gate	Man.	No	L.O.	Open	Open	-	No	-	Yes	G	
			C.S.	-	-	-	-	-	-	-	-	-	-	-	
9	Main Stream Header	-3	MS-V1-3C	SDSV	Air	Yes	Open	Closed	Open*	FC	No*	-	Yes*	G	*Automatic isolation for MSLB,E
			MS-353C	DDV	Mot.	No	L.C.	Closed	Closed	As is	No	-	No	G	
			MS-12A	Globe	Man.	No	L.C.	Closed	Closed	-	No	-	No	G	
			MS-37	Globe	Man.	No	L.C.	Closed**	Closed	-	No	-	No	G	**May be opened for RCS temperature control
			MS-39	Globe	Man.	No	L.C.	Closed**	Closed	-	No	-	No	G	
			RV1-3	PORV	Air	Yes	Closed	Closed	Closed	FC	No	-	Maybe	G	
			MS-262C	Gate	Man.	No	L.O.	Open	Open	-	No	-	No	G	
			C.S.	-	-	-	-	-	-	-	-	-	-	-	
10	Feedwater	-4	FW-8A	Check	-	No	L.O.	Open	Open*	-	No	-	Yes*	W	*Isolated for MSLB
			FW-201	Gate	Man.	No	L.C.	Closed**	Closed	-	No	-	Yes*	W	**Open during wet layup
			C.S.	-	-	-	-	-	-	-	-	-	-	-	activities, E
11	Feedwater	-4	FW-8B	Check	-	No	L.O.	Open	Open*	-	No	-	Yes*	W	*Isolated for MSLB
			FW-203	Gate	Man.	No	L.C.	Closed**	Closed	-	No	-	Yes*	W	**Open during wet layup
			C.S.	-	-	-	-	-	-	-	-	-	-	-	activities, E
12	Feedwater	-4	FW-8C	Check	-	No	L.O.	Open	Open*	-	No	-	Yes*	W	*Isolated for MSLB
			FW-205	Gate	Man.	No	L.C.	Closed**	Closed	-	No	-	Yes*	W	**Open during wet layup
			C.S.	-	-	-	-	-	-	-	-	-	-	-	activities, E

TABLE 6.2.4-1 (Continued)

PENE. NO.	PENETRATION AND SYSTEM	FIG. 6.2.4-	VALVE OR BARRIER (SEE FIGURE)	VALVE TYPE	OPER. TYPE	POSIT. INDIC. IN CONT. ROOM	NORMAL POSIT.	POSIT. DURING SHUTDOWN	POSIT. AFTER ACCIDENT	POSIT. ON POWER FAIL.	CONT. ISOL. TRIP	SEAL WATER INJ.	USED AFTER ACCID.	FLUID G-GAS W-WATER	NOTES
13	Steam Generator Blowdown	-4	FCV-1931A	DDV	Air	Yes	Open	Closed	Closed	FC	T	A	No	W	NE
			FCV-1931B	DDV	Air	Yes	Open	Closed	Closed	FC	T	A	No	W	
			FCV-1934A	Gate	Air	Yes	Open	Closed	Closed	FC	T	A	No	W	
			FCV-1934B	Gate	Air	Yes	Open	Closed	Closed	FC	T	A	No	W	
14	Steam Generator Blowdown	-4	FCV-1932A	DDV	Air	Yes	Open	Closed	Closed	FC	T	A	No	W	NE
			FCV-1932B	DDV	Air	Yes	Open	Closed	Closed	FC	T	A	No	W	
			FCV-1935A	Gate	Air	Yes	Open	Closed	Closed	FC	T	A	No	W	
			FCV-1935B	Gate	Air	Yes	Open	Closed	Closed	FC	T	A	No	W	
15	Steam Generator Blowdown	-4	FCV-1930A	DDV	Air	Yes	Open	Closed	Closed	FC	T	A	No	W	NE
			FCV-1930B	DDV	Air	Yes	Open	Closed	Closed	FC	T	A	No	W	
			FCV-1933A	Gate	Air	Yes	Open	Closed	Closed	FC	T	A	No	W	
			FCV-1933B	Gate	Air	Yes	Open	Closed	Closed	FC	T	A	No	W	
16	Residual Heat Removal Loop Out	-5	RHR-751	DDV	Motor	Yes	Closed	Open*	Closed	As is	No	-	No	W	*If in cold S/D,E
			C.S.	-	-	-	-	-	-	-	-	-	-	-	
17	Residual Heat Removal Loop In	-5	RHR-744A	Gate	Mot.	Yes	Closed	Open**	Open*	As is	No*	-	Yes	W	*Open on SI Signal
			RHR-744B	Gate	Mot.	Yes	Closed	Open**	Open*	As is	No*	-	Yes	W	**If in cold S/D,E
			C.S.	-	-	-	-	-	-	-	-	-	-	-	
18	Reactor Coolant Pump Cooling Water In	-5	CC-716B	DDV	Mot.	Yes	Open	Open	Closed*	As is	P	A	Yes	W	*Closes on P Signal, E
			C.S.	-	-	-	-	-	-	-	-	-	-	-	
19	Reactor Coolant Pump Cooling Water Out	-6	CC-730	DDV	Mot.	Yes	Open	Open	Closed*	As is	P	A	Yes	W	*Closes on P Signal, E
			C.S.	-	-	-	-	-	-	-	-	-	-	-	
20	Reactor Coolant Pump Cooling Water Out	-6	FCV-626	Gate	Mot.	Yes	Open	Open	Closed*	As is	P	A	Yes	W	*Closes on P Signal, E
			CC-932	DDV	Man	No	LC	LC	LC	-	No	A	No	W	*Used for Safe Shutdown
			C.S.	-	-	-	-	-	-	-	-	-	-	-	
21	Excess Letdown Heat Exchanger Cooling Water In	-6	CC-737A	Gate	Man.	No	Open	Open	Open	-	No	-	No	W	NE
			C.S.	-	-	-	-	-	-	-	-	-	-	-	

TABLE 6.2.4-1 (Continued)

PEN E. NO.	PENETRATION AND SYSTEM	FIG. 6.2.4-	VALVE OR BARRIER (SEE FIGURE)	VALVE TYPE	OPER. TYPE	POSIT. INDIC. IN CONT. ROOM	NORMAL POSIT.	POSIT. DURING SHUTDOWN	POSIT. AFTER ACCIDENT	POSIT. ON POWER FAIL.	CONT. ISOL. TRIP	SEAL WATER INJ.	USED AFTER ACCID.	FLUID G-GAS W-WATER	NOTES
22	Excess Letdown Heat Exchanger Cooling Water Out	-7	CC-739	Gate	Air	Yes	Open	Open	Closed	FC	T	-	No	W	NE
			C.S.	-	-	-	-	-	-	-	-	-	-	-	
23	Letdown Line	-7	CVC-204A	Globe	Air	Yes	Open	Closed	Closed	FC	T	A	No	W	NE
			CVC-204B	Globe	Air	Yes	Open	Closed	Closed	FC	T	A	No	W	
24	Charging Line	-7	CVC-282	Globe	Man.	No	Open	Open	Closed*	-	No	M	No*	W	*May be used for High Pressure Safety Injection, Line is isolated after charging is shutdown to provide long term recovery, E
			CVC-202A	Gate	Man.	No	Open	Open	Closed*	-	No	M	No*	W	
			CVC-309A	Globe	Man.	No	Closed	Closed	Closed*	-	No	M	No*	W	
25	Reactor Coolant Pump Seal Water Supply Line	-8	CVC-297C	Needle	Man.	No	Throt.	Open	Closed*	-	No	M	Yes*	W	*CVC-293A or CVC-293C may be open depending on whether seal injection filter A or B is insecvice. Line is isolated after RCP is shutdown, to provide long term recovery, E
			CVC-293C	Globe	Man.	No	Closed	Closed	Closed*	-	No	M	Yes*	W	
			CVC-293A	Globe	Man.	No	Open	Open	Closed*	-	No	M	Yes*	W	
			CVC-295	Gate	Man.	No	Closed	Closed	Closed	-	No	M	Yes*	W	
26	Reactor Coolant Pump Seal Water Supply Line	-8	CVC-297B	Needle	Man.	No	Throt.	Open	Closed*	-	No	M	Yes*	W	*CVC-293A or CVC-293C may be open depending on whether seal injection filter A or B is insecvice. Line is isolated after RCP is shutdown, to provide long term recovery, E
			CVC-293C	Globe	Man.	No	Closed	Closed	Closed*	-	No	M	Yes*	W	
			CVC-293A	Globe	Man.	No	Open	Open	Closed*	-	No	M	Yes*	W	
			CVC-295	Gate	Man.	No	Closed	Closed	Closed	-	No	M	Yes*	W	
27	Reactor Coolant Pump Seal Water Supply Line	-8	CVC-297A	Needle	Man.	No	Throt.	Open	Closed*	-	No	M	Yes*	W	*CVC-293A or CVC-293C may be open depending on whether seal injection filter A or B is insecvice. Line is isolated after RCP is shutdown, to provide long term recovery, E
			CVC-293C	Globe	Man.	No	Closed	Closed	Closed*	-	No	M	Yes*	W	
			CVC-293A	Globe	Man.	No	Open	Open	Closed*	-	No	M	Yes*	W	
			CVC-295	Gate	Man.	No	Closed	Closed	Closed	-	No	M	Yes*	W	
28	Reactor Coolant Pump Seal Water Return Line	-8	CVC-381	DDV	Mot.	Yes	Open	Open	Closed*	As is	P	A	No	W	*Closes on P Signal, NE
			C.S.	-	-	-	-	-	-	-	-	-	-	-	

TABLE 6.2.4-1 (Continued)

PENE. NO.	PENETRATION AND SYSTEM	FIG. 6.2.4	VALVE OR BARRIER (SEE FIGURE)	VALVE TYPE	OPER TYPE	POS. IND. IN CONT ROOM	NORMAL POSIT.	POSIT. DURING SHUTDOWN	POSIT. AFTER ACCIDENT	POSIT. ON POWER FAIL.	CONT. ISOL. TRIP	SEAL H20 INJ.	USED AFTER ACCID.	FLUID G-GAS W-WATER	NOTES
29	RCS Sample System Line	-8	PS-956A PS-956B	Globe Globe	Air Air	Yes Yes	Closed Closed	Closed Closed	Closed Closed	FC FC	T T	A A	No No	W W	NE
30	RCS Sample System Line	-8	PS-956C PS-956D	Globe Globe	Air Air	Yes Yes	Closed Closed	Closed Closed	Closed Closed	FC FC	T T	A A	No No	W W	NE
31	Reactor Coolant System Sample Line	-8	PS-956E PS-956F	Globe Globe	Air Air	Yes Yes	Closed Closed	Closed Closed	Closed Closed	FC FC	T T	A A	No No	W W	NE
32	Fuel Transfer Tube	-9	DGBF	-	-	No	Closed	Closed*	Closed	-	No	-	No	W	*Open for fuel transfer, NE
33	Instrument Air Header	-9	IA-525 PCV-1716	Check Globe	- Air	No Yes	Open Open	Open Open	Closed Closed	- FC	No T	- -	No No	G G	NE
34A	Post Accident Venting Valve Operator N_2 Supply	-10	PAV-37 C.S.	Globe -	Man. -	No -	L.C. -	Closed -	Closed* -	- -	No -	- -	Yes* -	G -	*Open for Post Accident Venting, NE
34B	Post Accident Venting Valve Operator N_2 Supply	-10	PAV-35 C.S.	Globe -	Man. -	No -	L.C. -	Closed -	Closed* -	- -	No -	- -	Yes* -	G -	*Open for Post Accident Venting, NE
34C	Post Accident Venting Valve Operator N_2 Supply	-10	PAV-33 C.S.	Globe -	Man. -	No -	L.C. -	Closed -	Closed* -	- -	No -	- -	Yes* -	G -	*Open for Post Accident Venting, NE
34D	Post Accident Venting Valve Operator N_2 Supply	-10	PAV-31 C.S.	Globe -	Man. -	No -	L.C. -	Closed -	Closed* -	- -	No -	- -	Yes* -	G -	*Open for Post Accident Venting, NE
35	Containment Air Sample In	-10	RMS-3 RMS-4	Globe Globe	Air Air	Yes Yes	Open Open	Open Open	Closed* Closed*	FC FC	T T	- -	No* No*	G G	*Open for Post Accident Sample, NE
36	Containment Air Sample Out	-10	RMS-1 RMS-2	Globe Globe	Air Air	Yes Yes	Open Open	Open Open	Closed* Closed*	FC FC	T T	- -	No* No*	G G	*Open for Post Accident Sample, NE

TABLE 6.2.4-1 (Continued)

PENE. NO.	PENETRATION AND SYSTEM	FIG. 6.2.4-	VALVE OR BARRIER (SEE FIGURE)	VALVE TYPE	OPER. TYPE	POSIT. INDIC. IN CONT. ROOM	NORMAL POSIT.	POSIT. DURING SHUTDOWN	POSIT. AFTER ACCIDENT	POSIT. ON POWER FAIL.	CONT. ISOL. TRIP	SEAL WATER INJ.	USED AFTER ACCID.	FLUID G-GAS W-WATER	NOTES
37	Containment Purge Supply Duct	-11	V12-7	Btrfly	Air	Yes	Closed	Open*	Closed	FC	V	-	No	G	*If cold and not refueling with high humidity, NE
			V12-6	Btrfly	Air	Yes	Closed	Open*	Closed	FC	V	-	No	G	
38	Containment Purge Exhaust Duct	-11	V12-9	Btrfly	Air	Yes	Closed	Open*	Closed	FC	V	-	No	G	*If cold and not refueling with high humidity, NE
			V12-8	Btrfly	Air	Yes	Closed	Open*	Closed	FC	V	-	No	G	
39	Plant Air Supply Header	-11	SA-44	Dia.	Man.	No	L.C.	Closed	Closed	-	No	-	No	G	NE
			SA-43	Dia.	Man.	No	L.C.	Closed	Closed	-	No	-	No	G	
40	Post Accident Venting	-12	V12-18	Dia.	Air	No	Closed	Closed	Closed*	FC	T	-	Yes	G	*Open for Cont. Air Exhaust 'B',NE
			V12-19	Dia.	Air	No	Closed	Closed	Closed*	FC	T	-	Yes	G	
41	Containment Pressure Relief	-12	V12-11	Btrfly	Air	Yes	Closed	Closed	Closed	FC	V	-	No	G	*Open for Post Accident Cont. Vent, NE
			V12-10	Btrfly	Air	Yes	Closed	Closed	Closed	FC	V	-	No	G	
			V12-14	Dia.	Air	No	Closed	Closed	Closed*	FC	No	-	Yes	G	
			V12-15	Dia.	Air	No	Closed	Closed	Closed*	FC	No	-	Yes	G	
42	Containment Vacuum Relief	-12	V12-13	Btrfly	Air	Yes	Closed	Closed	Closed	FC	V	-	No	G	NE
			V12-12	Btrfly		Air	Closed	Closed	Closed	FC	V	-	No	G	
43	Safety In-jection Line	-13	SI-869	DDV	Mot.	Yes	Closed	Closed	Closed*	As is	No	M	Yes	W	*Open for Hot Leg Injection, E
			C.S.	-	-	-	-	-	-	-	-	-	-	-	
44	Containment Spray Header	-13	SI-891A	DDV	Man.	No	L.O.	Open	Open	-	No	M	Yes	W	E
45	Containment Spray Header	-13	SI-891B	DDV	Man.	No	L.O.	Open	Open	-	No	M	Yes	W	E
46	Containment Sump Recirculation Line	-13	SI-860A	DDV	Mot.	Yes	Closed	Closed	Closed*	As is	No	-	Yes	W	*Open for RHR Recirculation, E
			SI-861A	DDV	Mot.	Yes	Closed	Closed	Closed*	As is	No	-	Yes	W	
47	Containment Sump Recirculation Line	-13	SI-860B	DDV	Mot.	Yes	Closed	Closed	Closed*	As is	No	-	Yes	W	*Open for RHR Recirculation, E
			SI-861B	DDV	Mot.	Yes	Closed	Closed	Closed*	As is	No	-	Yes	W	

TABLE 6.2.4-1 (Continued)

PENE. NO.	PENETRATION AND SYSTEM	FIG. 6.2.4-	VALVE OR BARRIER (SEE FIGURE)	VALVE TYPE	OPER. TYPE	POSIT. INDIC. IN CONT. ROOM	NORMAL POSIT.	POSIT. DURING SHUTDOWN	POSIT. AFTER ACCIDENT	POSIT. ON POWER FAIL.	CONT. ISOL. TRIP	SEAL WATER INJ.	USED AFTER ACCID.	FLUID G-GAS W-WATER	NOTES
48	Safety Injection	-14	SI-895V	Globe	Man.	No	L.C.	Closed	Closed	-	No	A	No	W	NE
	Test Line		SI-898F	Globe	Man.	No	L.C.	Closed	Closed	-	No	A	No	W	
49	Ventilation System	-14	V6-33B	Btrfly	Mot.	Yes	Open	Open	Open	As is	No	-	Yes	W	E
	Cooling Water In		V6-33F	Btrfly	Mot.	Yes	Open	Open	Open	As is	No	-	Yes	W	
			C.S.	-	-	-	-	-	-	-	-	-	-	-	
50	Ventilation System	-14	V6-33A	Btrfly	Mot.	Yes	Open	Open	Open	As is	No	-	Yes	W	E
	Cooling Water In		C.S.	-	-	-	-	-	-	-	-	-	-	-	
51	Ventilation System	-14	V6-33D	Btrfly	Mot.	Yes	Open	Open	Open	As is	No	-	Yes	W	E
	Cooling Water In		V6-33E	Btrfly	Mot.	Yes	Open	Open	Open	As is	No	-	Yes	W	
			C.S.	-	-	-	-	-	-	-	-	-	-	-	
52	Ventilation System	-14	V6-33C	Btrfly	Mot.	Yes	Open	Open	Open	As is	No	-	Yes	W	E
	Cooling Water In		C.S.	-	-	-	-	-	-	-	-	-	-		
53	Ventilation System	-15	V6-34B	Btrfly	Mot.	Yes	Open	Open	Open	As is	No	-	Yes	W	E
	Cooling Water Out		V6-35B	Globe	Mot.	Yes	Open	Open	Open	As is	No	-	Yes	W	
			C.S.	-	-	-	-	-	-	-	-	-	-	-	
54	Ventilation System	-15	V6-34C	Btrfly	Mot.	Yes	Open	Open	Open	As is	No	-	Yes	W	E
	Cooling Water Out		V6-35C	Globe	Mot.	Yes	Open	Open	Open	As is	No	-	Yes	W	
			C.S.	-	-	-	-	-	-	-	-	-	-	-	
55	Ventilation System	-15	V6-34D	Btrfly	Mot.	Yes	Open	Open	Open	As is	No	-	Yes	W	E
	Cooling Water Out		V6-35D	Globe	Mot.	Yes	Open	Open	Open	As is	No	-	Yes	W	
			C.S.	-	-	-	-	-	-	-	-	-	-	-	
56	Ventilation System	-15	V6-34A	Btrfly	Mot.	Yes	Open	Open	Open	As is	No	-	Yes	W	E
	Cooling Water Out		V6-35A	Globe	Mot.	Yes	Open	Open	Open	As is	No	-	Yes	W	
			C.S.	-	-	-	-	-	-	-	-	-	-	-	

TABLE 6.2.4-1 (Continued)

PENE. NO.	PENETRATION AND SYSTEM	FIG. 6.2.4-	VALVE OR BARRIER (SEE FIGURE)	VALVE TYPE	OPER. TYPE	POSIT. INDIC. IN CONT. ROOM	NORMAL POSIT.	POSIT. DURING SHUTDOWN	POSIT. AFTER ACCIDENT	POSIT. ON POWER FAIL.	CONT. ISOL. TRIP	SEAL WATER INJ.	USED AFTER ACCID.	FLUID G-GAS W-WATER	NOTES
53A	Spare	-15	Cap	-	-	-	-	-	-	-	-	-	-	G	NE
54A	Spare		Cap	-	-	-	-	-	-	-	-	-	-	G	NE
55A	Spare		Cap	-	-	-	-	-	-	-	-	-	-	G	NE
56A	Spare		Cap	-	-	-	-	-	-	-	-	-	-	G	NE
57	Auxiliary Feedwater Header	-15	V2-16A	DDV	Mot.	Yes	Closed	Closed	Open*	As is	No	-	Yes	W	*Open on SI signal, E
			C.S.	-	-	-	-	-	-	-	-	-	-	-	
58	Auxiliary Feedwater Header	-15	V2-16B	DDV	Mot.	Yes	Closed	Closed	Open*	As is	No	-	Yes	W	*Open on SI signal, E
			C.S.	-	-	-	-	-	-	-	-	-	-	-	
59	Auxiliary Feedwater Header	-15	V2-16C	DDV	Mot.	Yes	Closed	Closed	Open*	As is	No	-	Yes	W	*Open on SI signal, E
			C.S.	-	-	-	-	-	-	-	-	-	-	-	
60	Accumulator Sample Line	-16	PS-956G	Globe	Air	Yes	Closed	Closed	Closed	FC	T	A	No	W	NE
			PS-956H	Globe	Air	Yes	Closed	Closed	Closed	FC	T	A	No	W	
61	Containment Sump Pumps Discharge Line	-16	WD-1728	Dia.	Air	Yes*	Open	Open	Closed	FC	T	A	No	W	*Ganged indication in CR, individual indication at WDSP,NE
			WD-1723	Dia.	Air	Yes*	Open	Open	Closed	FC	T	A	No	W	
62	Boron	-16	SI-870A	Gate	Mot.	Yes	Closed	Open	Open	As is	No	-	Yes	W	E
63	Injection		SI-870B	Gate	Mot.	Yes	Closed	Open	Open	As is	No	-	Yes	W	E
64	Lines														
			C.S.	-	-	-	-	-	-	-	-	-	-	-	
65	Accumulator Nitrogen Supply	-17	SI-855	Globe	Air	Yes	Open	Open	Closed	FC	T	-	No	G	
			SI-909	Check	-	No	Open	Open	Closed	-	No	-	No	G	NE
66	Containment Test Channel Line	-17	PP-285D	Globe	Man.	No	L.C.	Closed	Closed	-	No	-	No	G	Abandoned
			PP-284D	Globe	Man.	No	L.C.	Closed	Closed	-	No	-	No	G	
67	Containment Controlled Leak	-17	VCT-13	Gate	Man.	No	L.C.	Closed	Closed	-	No	-	No	G	
			Cap	-	-	-	-	-	-	-	-	-	-	G	NE

TABLE 6.2.4-1 (Continued)

PENE. NO.	PENETRATION AND SYSTEM	FIG. 6.2.4-	VALVE OR BARRIER (SEE FIGURE)	VALVE TYPE	OPER. TYPE	POSIT. INDIC. IN CONT. ROOM	NORMAL POSIT.	POSIT. DURING SHUTDOWN	POSIT. AFTER ACCIDENT	POSIT. ON POWER FAIL.	CONT. ISOL. TRIP	SEAL WATER INJ.	USED AFTER ACCID.	FLUID G-GAS W-WATER	NOTES
68	Containment Pressure Sensing Lines	-18	VCT-20 PAS-5 PAS-6	Globe Globe Globe	Man. Man. Man.	No No No	L.C. L.C. L.C.	Closed Closed Closed	Closed* Closed* Closed*	- - -	No No No	- - -	No No No	G G G	E *May be opened for Post Accident sample.
69	Containment Pressure Sensing Lines	-18	VCT-19 PAS-3 PAS-4	Globe Globe Globe	Man. Man. Man.	No No No	L.C. L.C. L.C.	Closed Closed Closed	Closed* Closed* Closed*	- - -	No No No	- - -	No No No	G G G	E *May be opened for Post Accident sample.
70	Containment Pressure Sensing Lines	-18	VCT-18 PAS-1 PAS-2	Globe Globe Globe	Man. Man. Man.	No No No	L.C. L.C. L.C.	Closed Closed Closed	Closed* Closed* Closed*	- - -	No No No	- - -	No No No	G G G	E *May be opened for Post Accident sample.
71	Penetration Pressure System Air Supply	-18	PP-275D C.S.	Globe -	Man. -	No -	Open -	Open -	Open -	- -	No -	- -	Yes -	G -	E
72	Deadweight Tester Line	-18	CAP RC-582	- Gate	- Man.	- No	- L.C.	- Closed	- Closed	- -	- No	- -	- No	W W	Abandoned
73	Fire Water	-19	FP-258 FP-256	Gate Gate	Mot. Mot.	Yes Yes	Open Open	Open Open	Closed Closed	As is As is	T T	- -	No No	W W	NE
74	Fire Water	-19	FP-249 FP-248	Gate Gate	Mot. Mot.	Yes Yes	Open Open	Open Open	Closed Closed	As is As is	T T	- -	No No	W W	NE
75	RVLIS Sensing	-19	LIS511AB	*	-	No	-	-	-	-	No	-	Yes	W	*Isolator, NE
76	RVLIS Sensing	-19	LIS511AA	*	-	No	-	-	-	-	No	-	Yes	W	*Isolator, NE
77	RVLIS Sensing	-19	LIS511AC	*	-	No	-	-	-	-	No	-	Yes	W	*Isolator,NE

Error! Bookmark not defined.

TABLE 6.2.4-1 (Continued)

PENE. NO.	PENETRATION AND SYSTEM	FIG. 6.2.4-	VALVE OR BARRIER (SEE FIGURE)	VALVE TYPE	OPER. TYPE	POSIT. INDIC. IN CONT. ROOM	NORMAL POSIT.	POSIT. DURING SHUTDOWN	POSIT. AFTER ACCIDENT	POSIT. ON POWER FAIL.	CONT. ISOL. TRIP	SEAL WATER INJ.	USED AFTER ACCID.	FLUID G-GAS W-WATER	NOTES
78	RVLIS Sensing	-19	LIS511BB	*	-	No	-	-	-	-	No	-	Yes	W	*Isolator, NE
79	RVLIS Sensing	-19	LIS511BA	*	-	No	-	-	-	-	No	-	Yes	W	*Isolator, NE
80	RVLIS Sensing	-19	LIS511BC	*	-	No	-	-	-	-	No	-	Yes	W	*Isolator, NE

Table Legend

A- Automatic Seal Water Injection
M- Manual Seal Water Injection
PPS- Penetration Pressurization System
FC- Fail Closed
Dia.- Diaphragm Valve
DDV- Double Disc Gate Valve
C.S.- Closed System
SDSV- Swing Disc Stop Valve
Btrfly- Butterfly Valve
PORV- Power Operated Relief Valve
CR- Control Room
DGBF- Double Gasketed Blind Flange
E- Essential
NE- Non-essential

Mot.- Motor Operator
Man.- Manual Operator
L.O.- Locked Open
L.C.- Locked Closed
T- Containment Isolation (Phase A) Signal
P- High-High Containment Pressure (Phase B) Signal
V- Containment Ventilation Isolation Signal
RHR- Residual Heat Removal
RCP- Reactor Coolant Pump
SI- Safety Injection
WDSP- Waste Disposal System Panel
S/D- Shutdown

TABLE 6.2.4-2

AUTOMATIC ISOLATION VALVE SIZES

Penetration	Valve Numbers	Valve Size
P-1	RC-553, RC-516	3/8
P-2	RC-518, RC-550	3/4
P-3	RC-519B, RC-519A	3
P-4	WD-1786, WD-1787	1
	WD-1793, WD-1713	1
P-5	WD-1794, WD-1789	3/4
P-6	WD-1721, WD-1722	3
P-13	FCV-1931A, FCV-1931B	3
	FCV-1934A, FCV-1934B	3/4
P-14	FCV-1932A, FCV-1932B	3
	FCV-1935A, FCV-1935B	3/4
P-15	FCV-1930A, FCV-1930B	3
	FCV-1933A, FCV-1933B	3/4
P-17	RHR-744A, RHR-744B	10
P-18	CC-716B	6
P-19	CC-730	6
P-20	FCV-626	3
P-22	CC-739	3
P-23	CVC-204A, CVC-204B	2
P-28	CVC-381	3
P-29	PS-956A, PS-956B	3/8
P-30	PS-956C, PS-956D	3/8
P-31	PS-956E, PS-956F	3/8
P-33	IA-525, PCV-1716	2
P-34A	PAV-37	3/8
P-34B	PAV-35	3/8
P-34C	PAV-33	3/8
P-34D	PAV-31	3/8
P-35	RMS-3, RMS-4	1
P-36	RMS-1, RMS-2	1

TABLE 6.2.4-2 (Continued)

Penetration	Valve Numbers	Valve Size
P-37	V12-7, V12-6	42
P-38	V12-9, V12-8	42
P-39	SA-44, SA-43	2
P-40	V12-18, V12-19	3
P-41	V12-11, V12-10	6
P-42	V12-12, V12-13	6
P-48	SI-895V, SI-898F	3/4
P-60	PS-956G, PS-956H	3/8
P-61	WD-1728, WD-1723	2
P-62,63,64	SI-870A, SI-870B SI-895T	3 3/4
P-65	SI-909, SI-855	1
P-73	FP-258, FP-256	4
P-74	FP-249, FP-248	4

Note 1:By definition, manual containment isolation valves, locked closed or under administrative control, qualify as automatic isolation valves. Valves meeting this definition are included in Table 6.2.4-2.

6.2.5 COMBUSTIBLE GAS CONTROL IN CONTAINMENT

6.2.5.1 Design Basis

Following a design basis accident (DBA), hydrogen gas may be generated inside the containment by reactions such as radiolysis of aqueous solutions in the sump and core, zirconium metal with water, and corrosion of materials of construction.

Prior to October 2003, 10 CFR 50.44 required controls to ensure the containment hydrogen concentration remained below combustible concentrations. Such controls included the use of a hydrogen recombiner or hydrogen purge. An October 2003 revision to 10 CFR 50.44 eliminated the need for such controls for recovery from design basis accidents for containments similar to the HBR 2 containment. Therefore, HBR 2 no longer maintains access to a hydrogen recombiner and does not require hydrogen purging to recover from a design basis accident. The Post-Accident Containment Venting System is available for reduction of the containment hydrogen concentration if desired. The Post-Accident Containment Venting System is described in Section 9.4.3.

6.2.6 CONTAINMENT LEAKAGE TESTING

The containment design leakage testability includes the necessary provisions to enable tests to comply with:

a) American Nuclear Society (ANS) 7.62 Leakage Testing for Containment Structures for Nuclear Reactors (July 14, 1967), and

b) Atomic Energy Commission (AEC) Technical Safety Guide 7.5.1, "Containment Leakage Testing and Surveillance Requirements," (December 15, 1966).

c) ANSI/ANS-56.8-2002, Containment System Leakage Testing Requirements

The preoperational leakage rate test demonstrated the adequacy of the Containment Building to meet the requirements assumed in the plant safety analysis, and that the Containment Building was ready to be placed in service (References 6.2.6-1 and 6.2.6-2).

During a refueling plant shutdown, 1973-1974 time frame, bulges in the containment steel liner were observed. A full pressure containment integrity test has been performed since the bulges were observed and no damage to either the liner or liner anchor studs was found. Additionally, an analysis was performed by Ebasco Services, Inc. to determine the response of the bulged liner during normal and accident conditions. This analysis showed that the bulged liner and its anchor studs are effective to meet their functional requirements during a loss-of-coolant accident (LOCA) or normal operating conditions. Documentation was provided, HBR 2, Docket No. 50-263, to assure that the bulged areas are stable and will maintain containment integrity during normal and accident conditions.

6.2.6.1 Results of Integrated Leakage Rate and Sensitive Leakage Rate Test

This section presents the results of the HBR 2 Reactor Building Integrated Leak Rate Test which was performed in February, 1978. Completion of this test and submittal of the results were in accordance with HBR 2 Technical Specifications 4.4.1.1 and 6.9.3.A.

The results presented indicated a leakage rate at the upper bound of the 95 percent confidence interval, well below the allowable leakage rate of 0.0424 percent by weight per day at the 21.0 psig test pressure (Reference 6.2.6-3).

All containment isolation valves isolated during the test were locally tested consistent with the requirements of 10CFR50, Appendix J and HBR 2 Technical Specification 4.4.1.1.d. The combined leakage from these valves was determined to have a negligible impact on the measured integrated leakage rate.

The results included in the report indicate that the containment leakage rate is well within acceptable limits and can perform its designed function in the unlikely event of a major accident. In accordance with the results of this test, 10CFR50, Appendix J, and Technical Specification 4.4.1.1.g, the next integrated leak rate test will be performed at the end of the current ten-year in-service inspection interval.

6.2.6.2 Containment Penetration Leakage Rate Test

A summary of the fluid systems lines penetrating containment and the valves and closed systems employed for containment isolation is presented in Table 6.2.4-1. The test methods used to determine containment penetration leakage rates are described in Section 6.2.4.3.

6.2.6.3 Containment Isolation Valve Leakage Rate Test

A summary of the fluid systems lines penetrating containment and the valves and closed systems employed for containment isolation is presented in Table 6.2.4-1.

During the refuelings since the Integrated Leak Rate Test of 1974, leakage measurements were made on all isolation valves subject to Type C testing. The leakage reported excludes leakage from containment isolation valves that are sealed with fluid from a seal system in accordance with 10CFR50, Appendix J, Section III C.3. The leakage from these valves did not exceed that specified in the Technical Specifications and the isolation valve seal water system is sufficient to ensure the sealing function for at least 30 days at a pressure of 1.10P_a. The only leakage measurement not subject to the above was the containment pressure manometer line isolation valve leakage measurement.

By letter of March 9, 1978, the NRC formally notified Carolina Power and Light Company (CP&L) of a change in the interpretation of 10CFR50, Appendix J, as it relates to "Type C" local leak testing at HBR. This change in interpretation required "Type C" local tests be performed on all containment isolation valves which receive seal water during containment isolation from the Isolation Valve Seal Water (IVSW) System.

Previous to this interpretation, pursuant to an consistent with Appendix J, Subparagraph III. C.3 and subsection 4.4.2 of the HBR Technical Specifications, the subject isolation valves have been leak tested during performance of the refueling interval periodic test of the IVSW system. The capability of the IVSW System to provide seal water and the total leakage from all the isolation valves receiving seal water, regardless of direction, at the seal water pressure of 46 psig is checked during this test. Pursuant to the exception noted in subparagraph III. C.3. of 10CFR50, Appendix J, relating to seal systems, no individual local leak tests were previously required and none were performed.

Consistent with the requirements of the March 9, 1978 letter, "Type C" local leak tests were performed on the isolation valves which receive seal water from the IVSW System. The tests were completed and results of the tests are presented in Reference 6.2.6-3.

Tests were performed using instrument air at 42 psig as the test medium. Test methods involved the use of the "in-leakage" and "out-leakage" measurements. Using the "in-leakage" method, the interspace between valves in series or between the seats of double disk valves was pressurized with air to the test pressure and the makeup air to the interspace volume was measured as leakage from the valves under test. Using the "out-leakage" method, a constant test pressure was maintained upstream of the valve tested and leakage was measured through a downstream connection. Valves were tested in a conservative direction

or in the direction of accident flow. Test procedures were developed using the guidelines of 10CFR50, Appendix J, and proposed standard ANS-56.8 Draft 1, Revision 3, "Containment System Leakage Testing Requirements."

Results of the leak tests are presented both by valve and by penetration. The total leakage presented is the total from all penetrations and is consistent with the reporting methods of Draft 1 of ANS-56.8. Valve acceptance criteria were based on a total leakage of 150 scc/min per inch of valve diameter. This limit was taken from Section XI of the 1977 Edition of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (B&PV) Code. Final acceptance criteria were based on the total leakage of all valves not exceeding the sum of the individual acceptance values and no single valve exceeding $0.05L_a$*. Certain valves had leakage which exceeded the individual valve acceptance criteria; however, the total leakage from all valves was well within the final acceptance criteria and total leakage from all penetrations was well below the 10CFR50, Appendix J, ($0.6L_a$*) limit.

Prior to receiving official notification that local leak rate tests of the IVSW supplied isolation valves were required, the refueling interval test on the IVSW System had been performed. The system had performed satisfactorily although leakage greater than the acceptance criteria leak rate was detected through some of the valves. Some adjustments had been made for some of the valves and maintenance was scheduled for the remaining leaking valves. Prior to this maintenance, the individual "Type C" tests were performed on all valves serviced by the IVSW System, including those requiring maintenance. During the "Type C" tests, all the valves which had passed the IVSW test showed leakage rates well within the individual valve acceptance criteria established for the local tests. Correspondingly, the only valves which indicated significant leakage had previously been identified by the routine refueling interval test. After concluding the "Type C" testing, including required maintenance, the leakage from the IVSW System was again checked and all header leakages were well within the limits established based on valve design.

6.2.6.4 Scheduling and Reporting of Periodic Tests

Containment integrated leakage rate testing is performed in conformance with the requirements of 10CFR50, Appendix J, Option B in accordance with Nuclear Energy Institute (NEI) 94-01, Revision 3-A, and the conditions and limitations specified in NEI 94-01, Revision 2-A, and the Technical Specifications.

6.2.6.5 Special Testing Requirements

The HBR plant does not have a subatmospheric containment or a dual containment. Therefore, this section is not required for HBR 2.

*L_a is the allowable leakage as defined by Technical Specifications.

REFERENCES: SECTION 6.2

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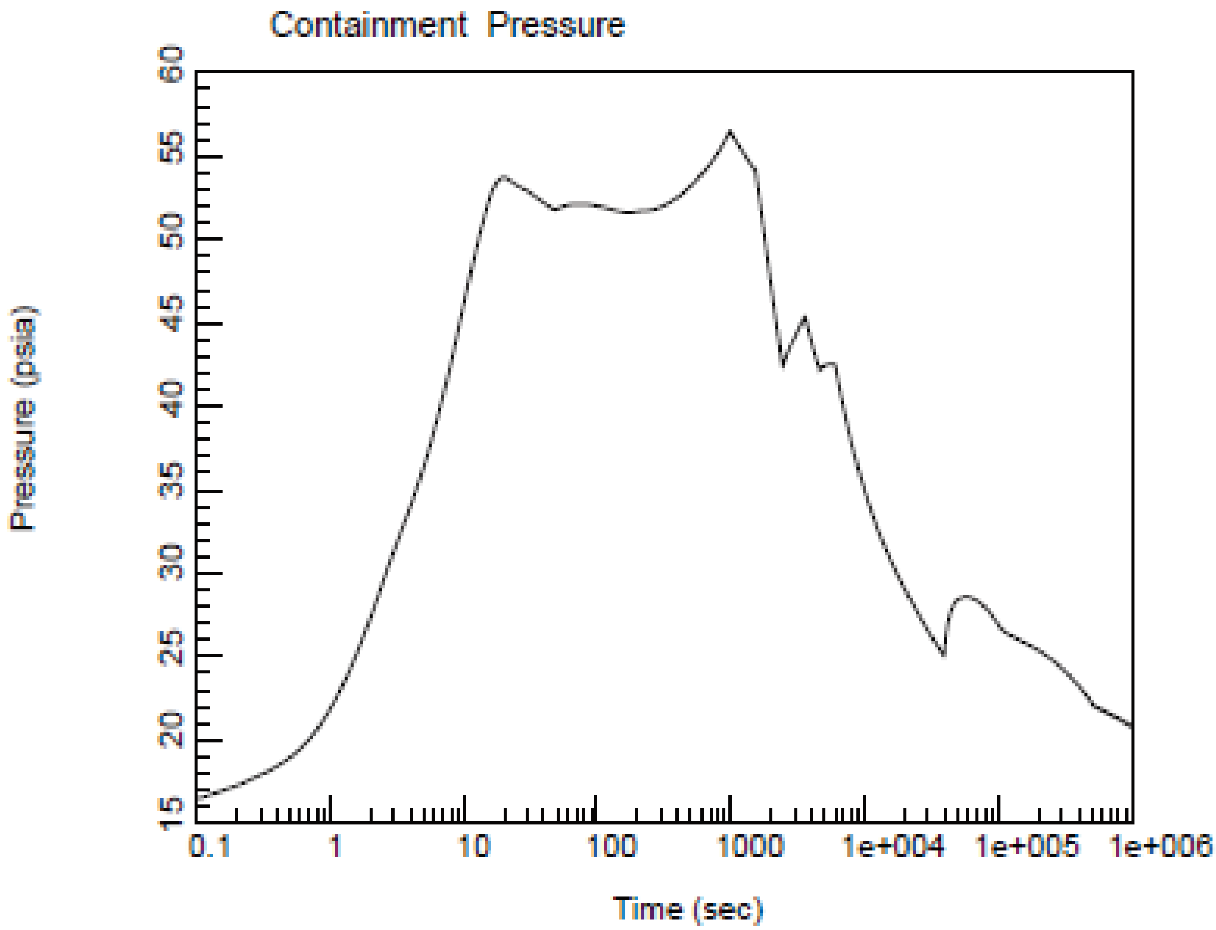
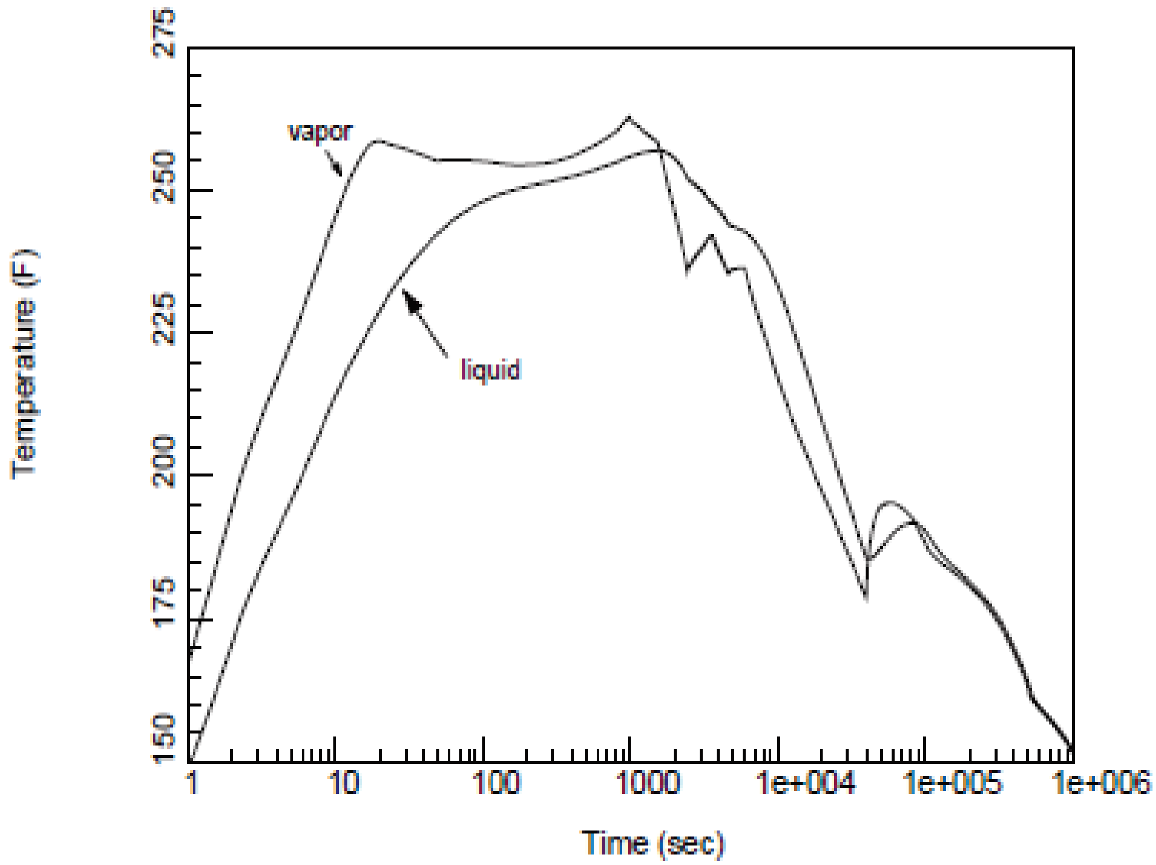


REVISION NO. 25

H. B. Robinson Unit 2 Carolina Power & Light Company UPDATED FINAL SAFETY ANALYSIS REPORT	DOUBLE-ENDED PUMP SUCTION BREAK WITH MINIMUM SAFEGUARDS PRESSURE RESPONSE	FIGURE No. 6.2.1-1



REVISION NO. 25

H. B. Robinson Unit 2 Carolina Power & Light Company UPDATED FINAL SAFETY ANALYSIS REPORT	DOUBLE-ENDED PUMP SUCTION BREAK WITH MINIMUM SAFEGUARDS CONTAINMENT TEMPERATURE RESPONSE	FIGURE No. 6.2.1-2

FIGURE 6.2.1-3

DOUBLE-ENDED PUMP SUCTION BREAK WITH
MAXIMUM SAFEGUARDS PRESSURE RESPONSE

DELETED IN REVISION NO. 21

FIGURE 6.2.1-4

DOUBLE-ENDED PUMP SUCTION BREAK WITH MAXIMUM SAFEGUARDS CONTAINMENT TEMPERATURE RESPONSE

DELETED IN REVISION NO. 21

FIGURE 6.2.1-5

DOUBLE-ENDED HOT LEG BREAK WITH
MINIMUM SAFEGUARDS PRESSURE RESPONSE

DELETED IN REVISION NO. 21

FIGURE 6.2.1-6

DOUBLE-ENDED HOT LEG BREAK WITH
MINIMUM SAFEGUARDS TEMPERATURE RESPONSE

DELETED IN REVISION NO. 21

FIGURES 6.2.1-7 THROUGH 6.2.1-12

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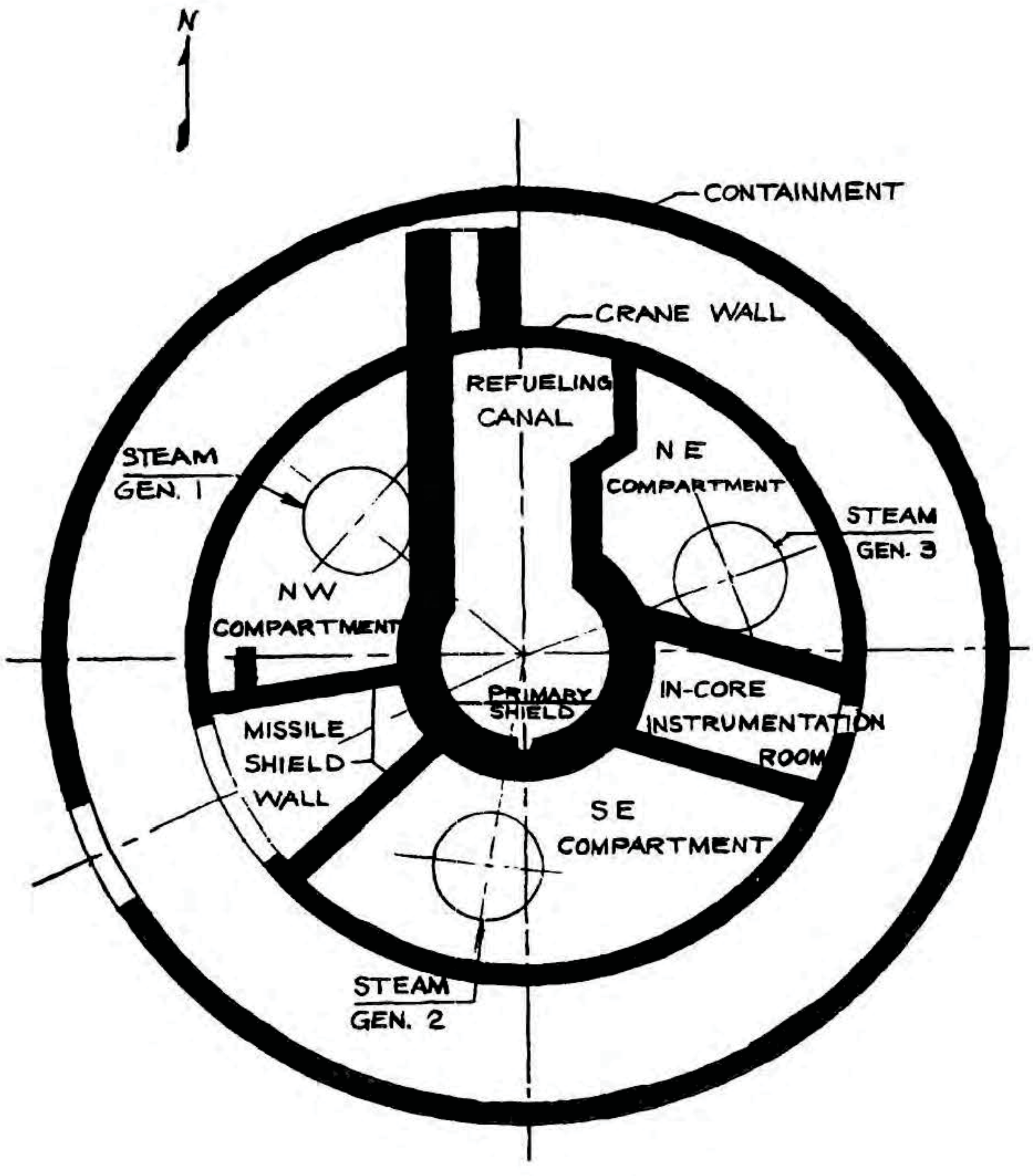
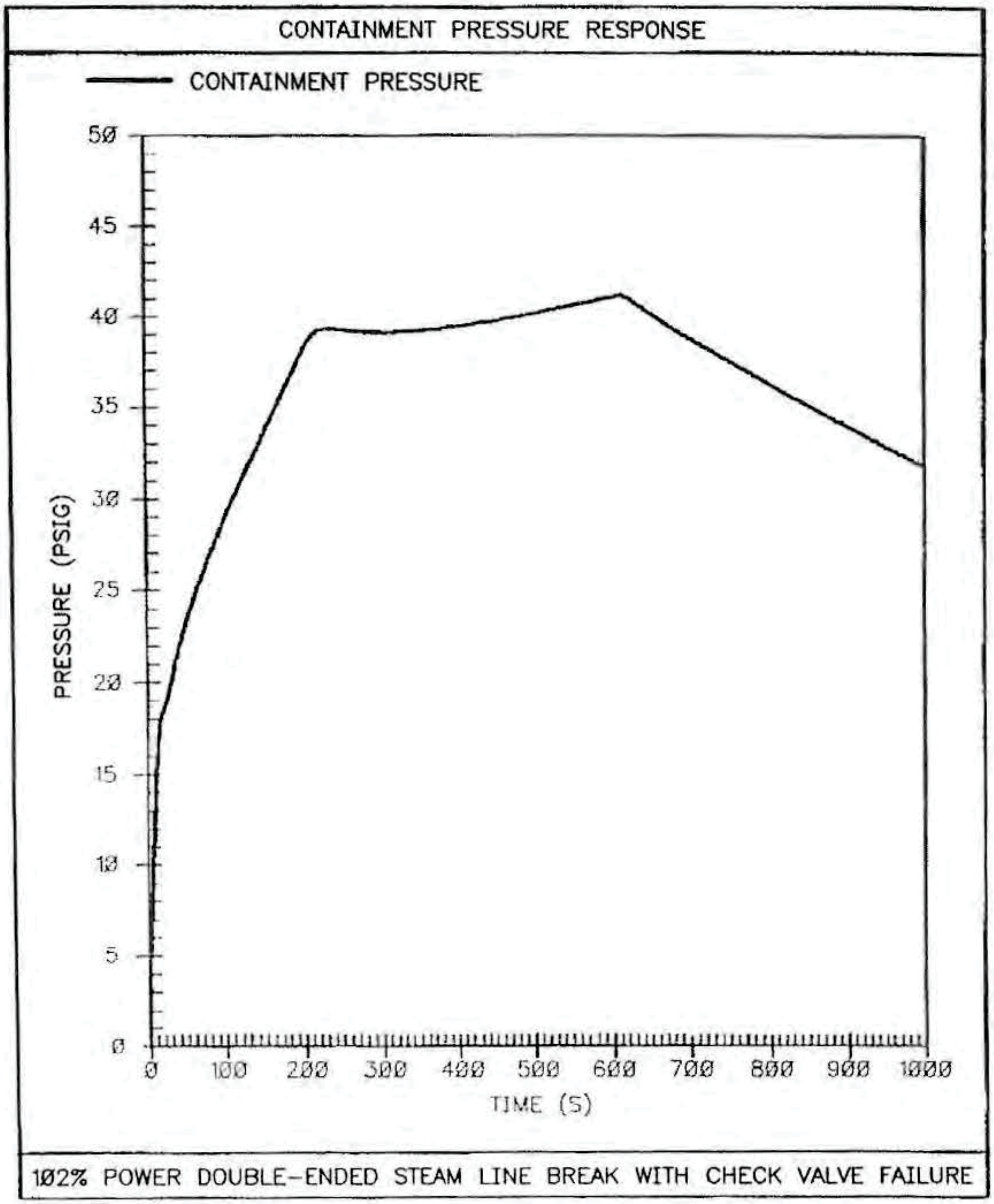


H. B. ROBINSON UNIT 2 Carolina Power & Light Company UPDATED FINAL SAFETY ANALYSIS REPORT	CONTAINMENT SUBCOMPARTMENTS	FIGURE 6.2.1 - 13

FIGURES 6.2.1-14
AND 6.2.1-15

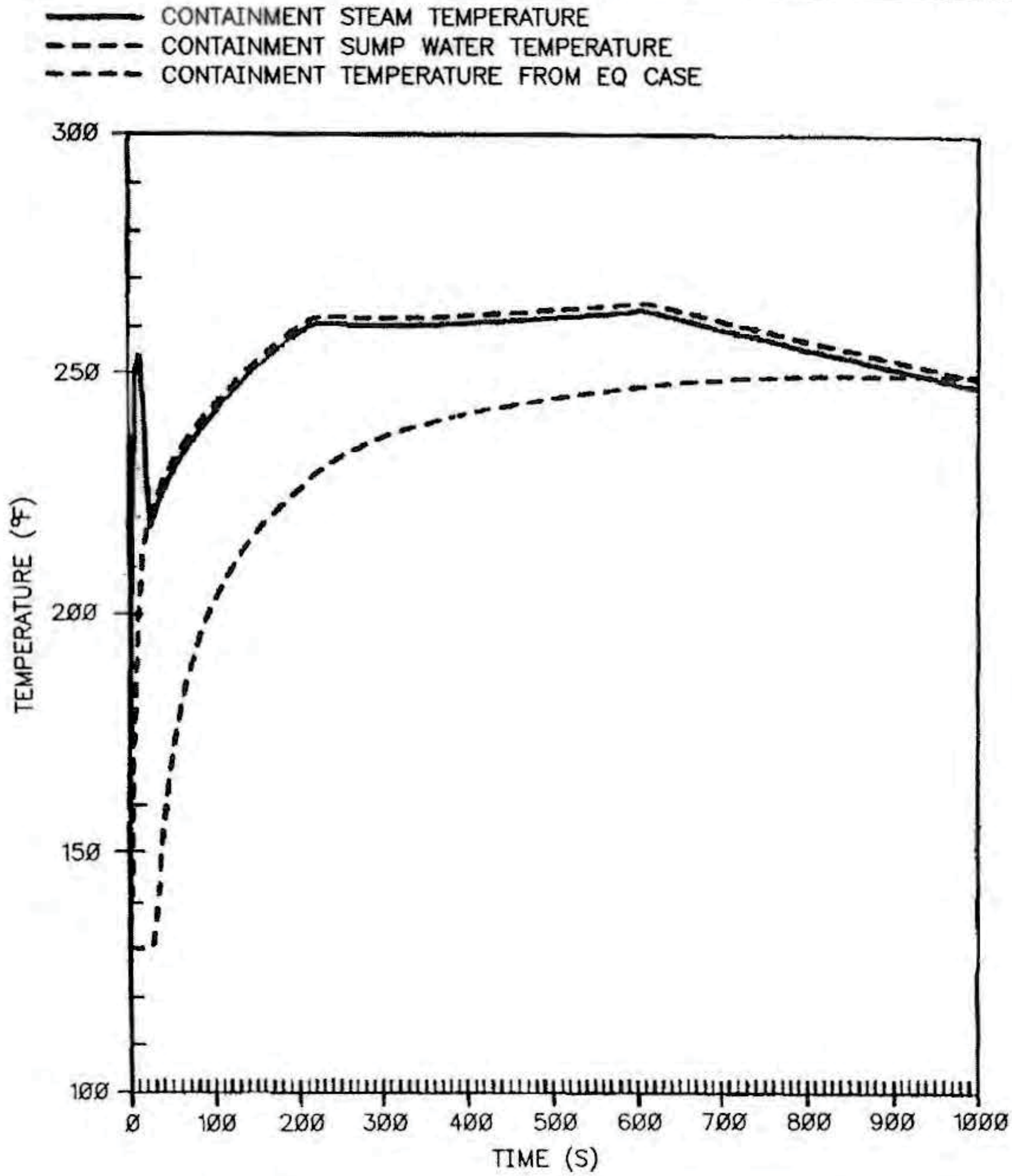
DELETED



Revision No. 19

H.B. ROBINSON UNIT 2 Carolina Power & Light Company UPDATED FINAL SAFETY ANALYSIS REPORT	CONTAINMENT PRESSURE 102% OF 2300 MWT CHECK VALVE FAILURE	FIGURE 6.2.1.4-1

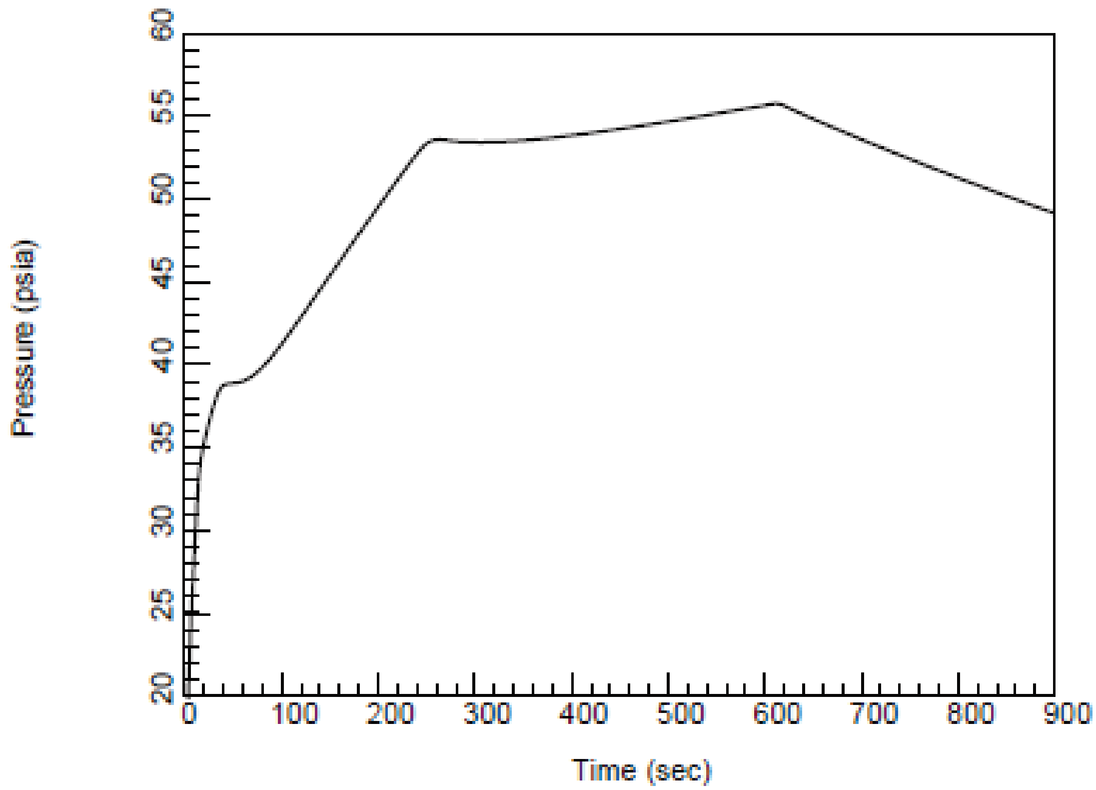
CONTAINMENT TEMPERATURE RESPONSE



102% POWER DOUBLE-ENDED STEAM LINE BREAK WITH CHECK VALVE FAILURE

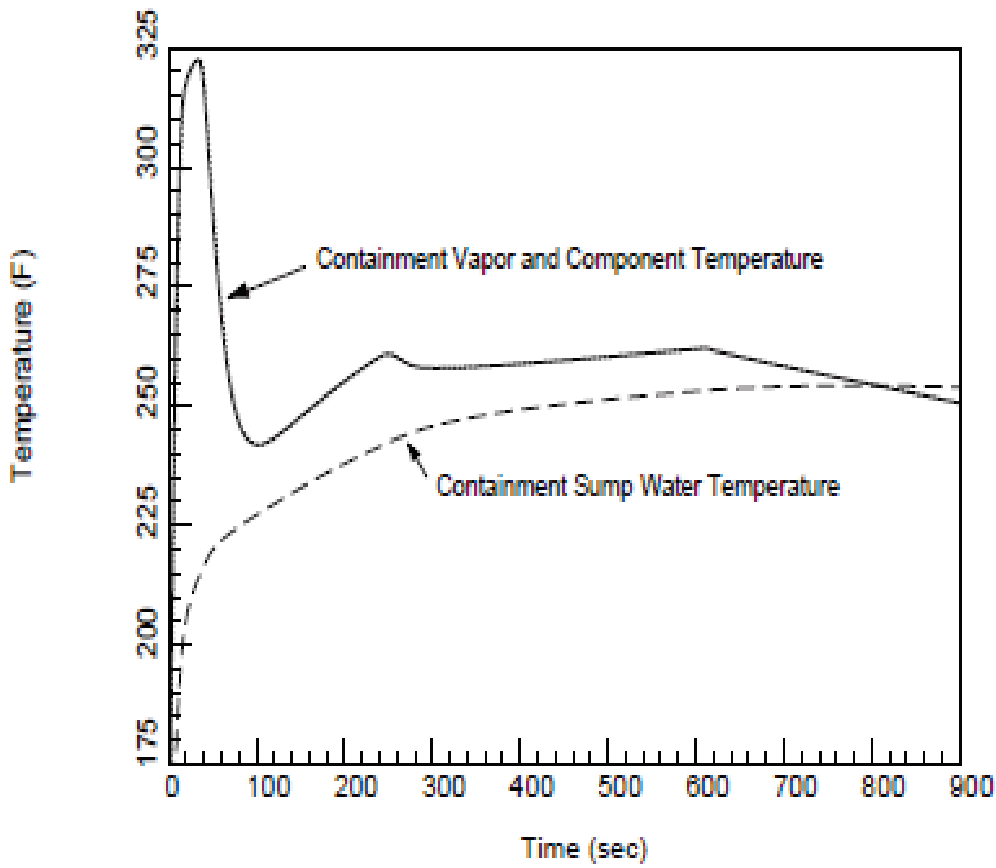
Revision No. 19

H.B. ROBINSON UNIT 2 Carolina Power & Light Company UPDATED FINAL SAFETY ANALYSIS REPORT	CONTAINMENT TEMPERATURES 102% OF 2300 MWT CHECK VALVE FAILURE	FIGURE 6.2.1.4–2



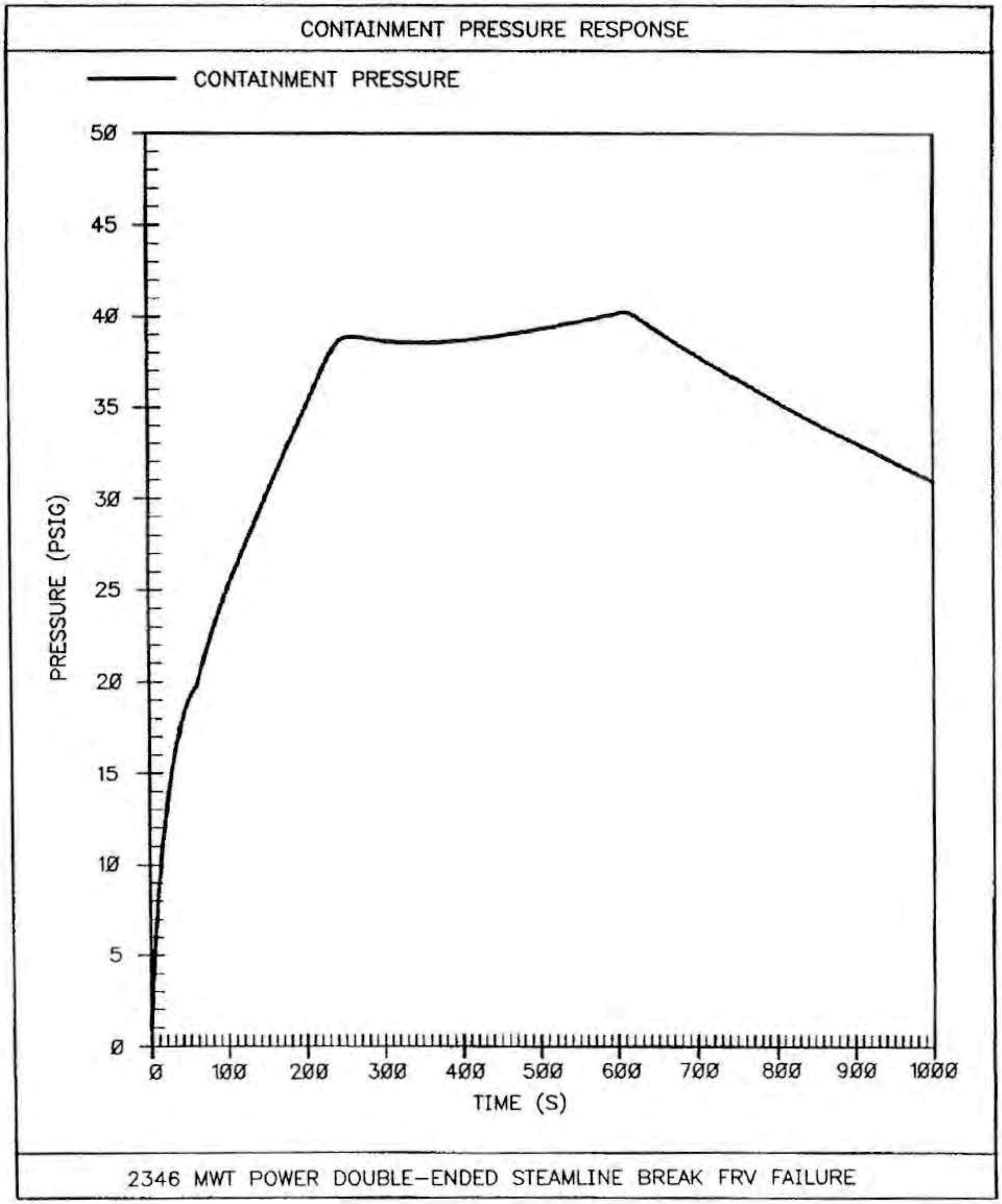
REVISION NO. 25

H. B. Robinson Unit 2 Carolina Power & Light Company UPDATED FINAL SAFETY ANALYSIS REPORT	HZP CASE WITH CHECK VALVE FAILURE – CONTAINMENT PRESSURE	FIGURE No. 6.2.1.4-3



REVISION NO. 25

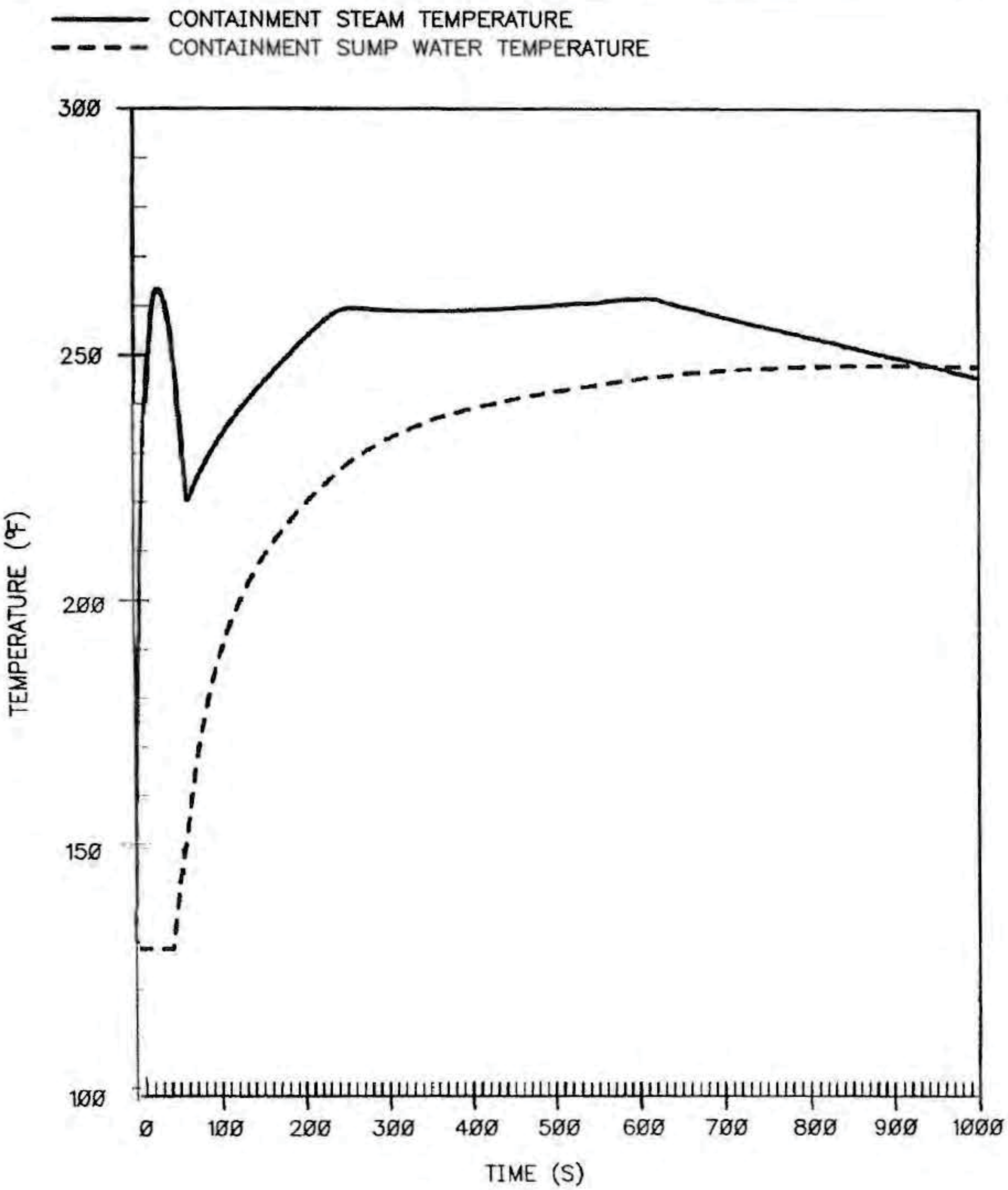
H. B. Robinson Unit 2 Carolina Power & Light Company UPDATED FINAL SAFETY ANALYSIS REPORT	HZP CHECK VALVE FAILURE – CONTAINMENT TEMPERATURES	FIGURE No. 6.2.1.4-4



Revision No. 19

H.B. ROBINSON UNIT 2 Carolina Power & Light Company UPDATED FINAL SAFETY ANALYSIS REPORT	CONTAINMENT PRESSURE 102% OF 2300 MWT FRV FAILURE CASE	FIGURE 6.2.1.4-5

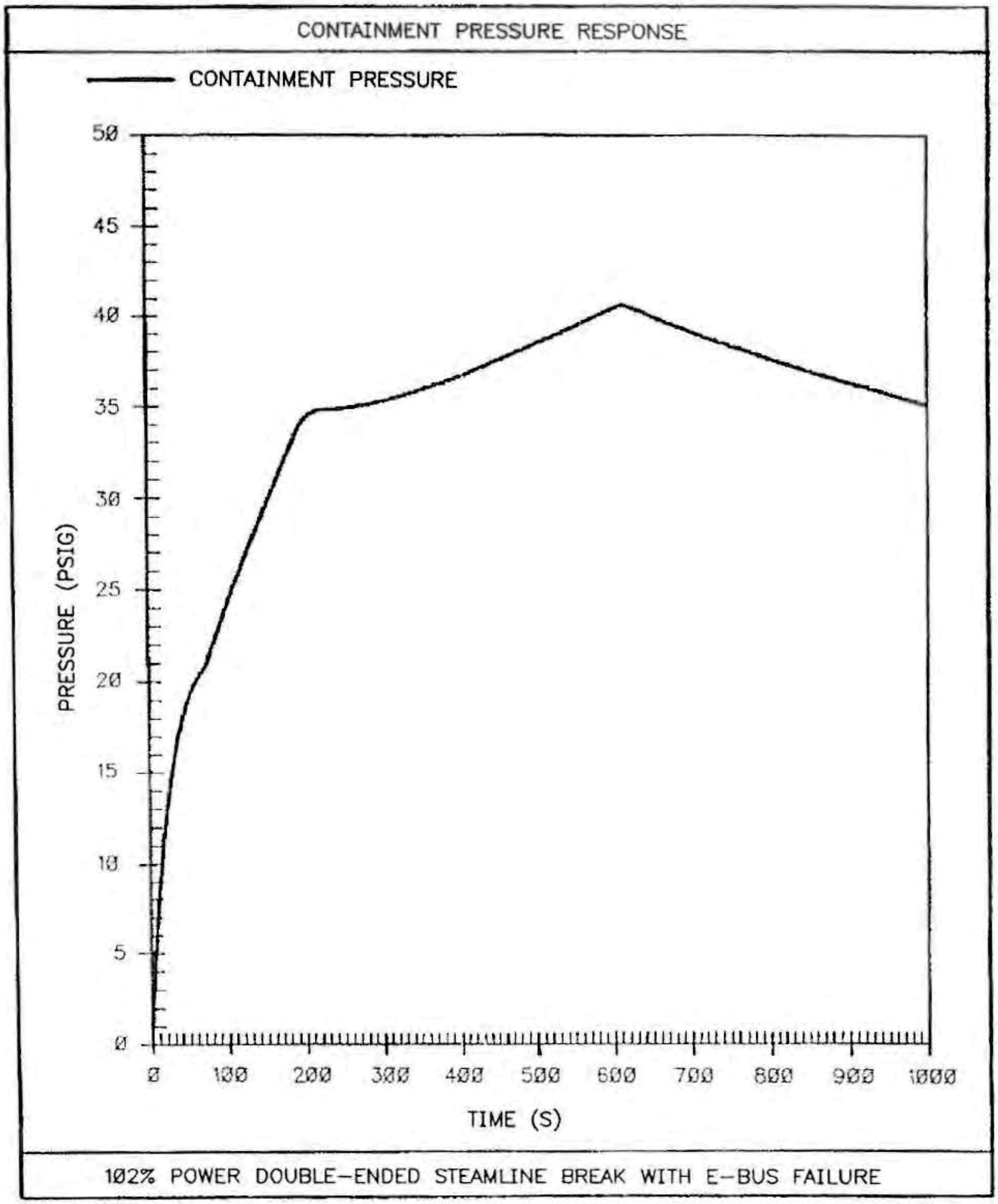
CONTAINMENT TEMPERATURE RESPONSE



2346 MWT POWER DOUBLE-ENDED STEAMLINE BREAK FRV FAILURE

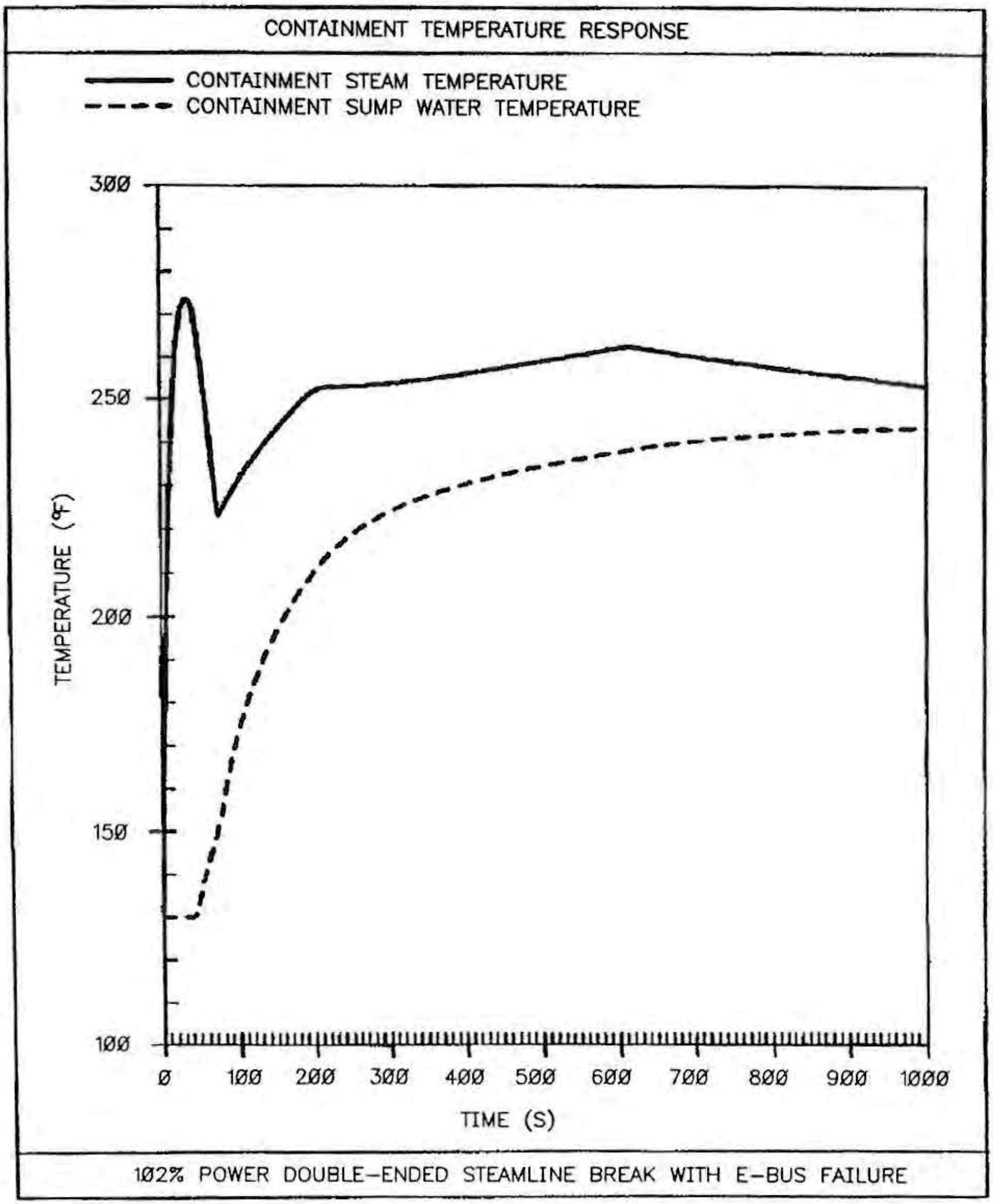
Revision No. 19

H.B. ROBINSON UNIT 2 arolina Power & Light Company UPDATED FINAL SAFETY ANALYSIS REPORT	CONTAINMENT TEMPERATURES 102% OF 2300 MWT FRV FAILURE CASE	FIGURE 6.2.1.4-6



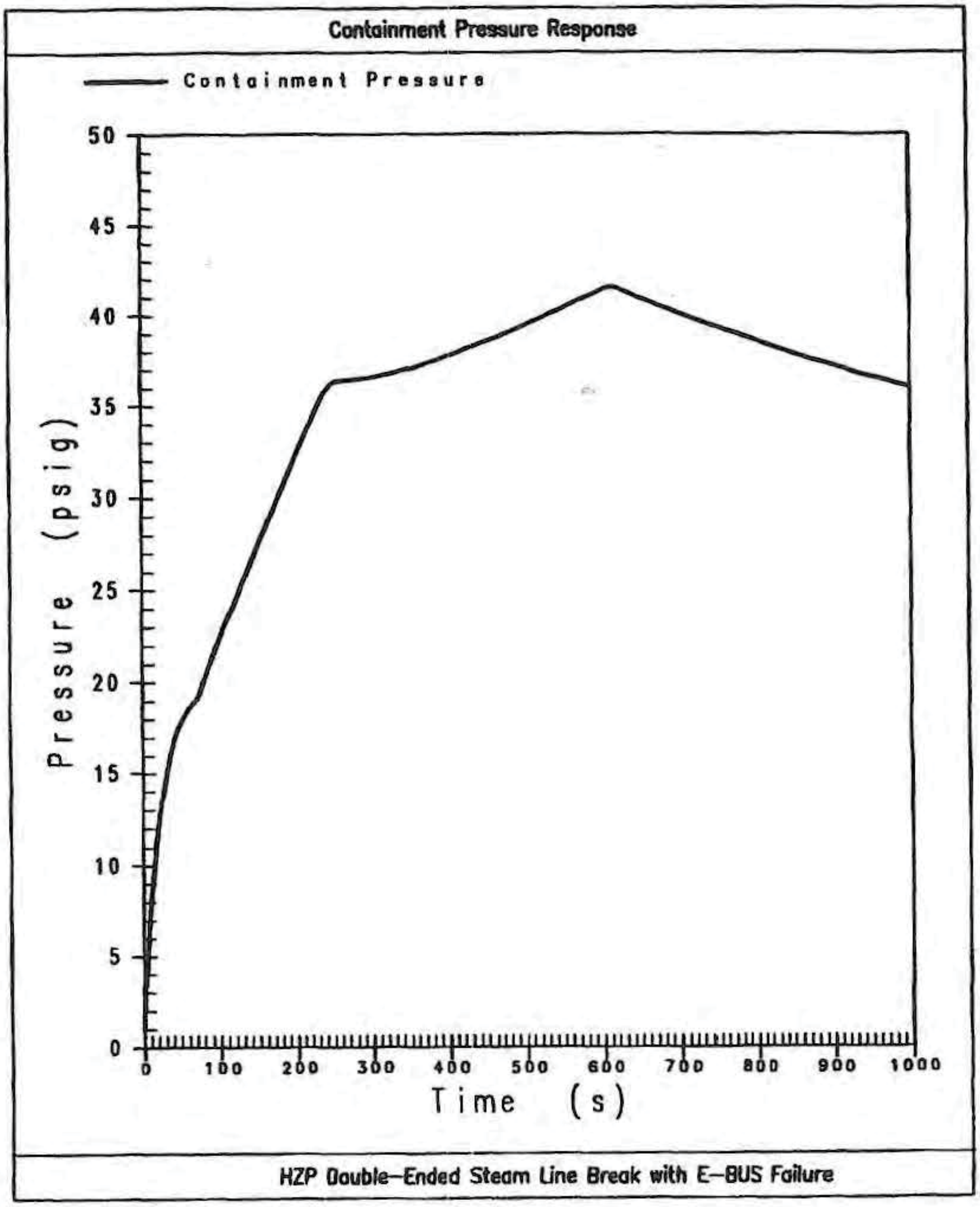
Revision No. 19

H.B. ROBINSON UNIT 2 Carolina Power & Light Company UPDATED FINAL SAFETY ANALYSIS REPORT	CONTAINMENT PRESSURE 102% OF 2300 MWT E-BUS FAILURE CASE	FIGURE 6.2.1.4-7

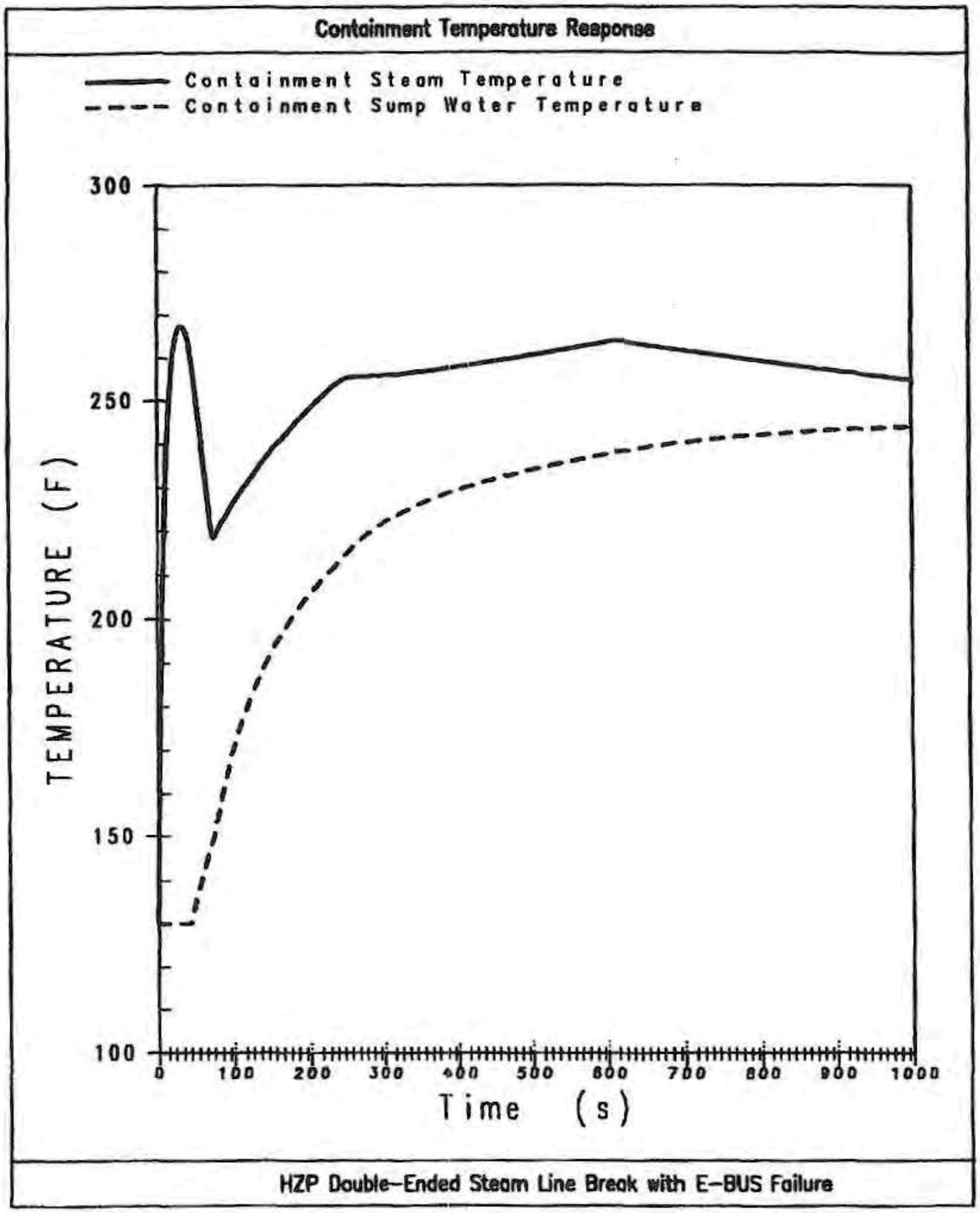


Revision No. 19

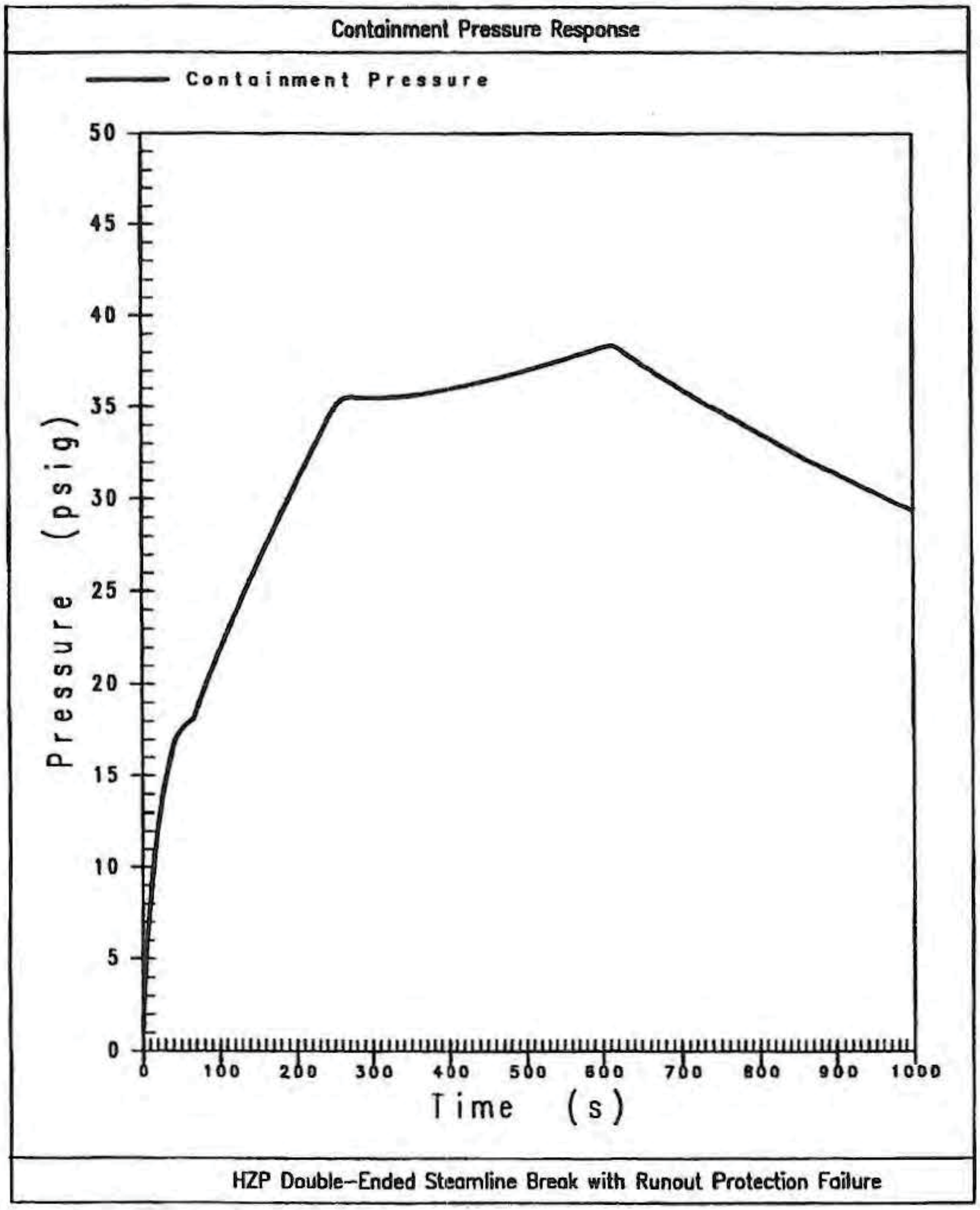
H.B. ROBINSON UNIT 2 Carolina Power & Light Company UPDATED FINAL SAFETY ANALYSIS REPORT	CONTAINMENT TEMPERATURE 102% OF 2300 MWT E-BUS FAILURE CASE	FIGURE 6.2.1.4-8



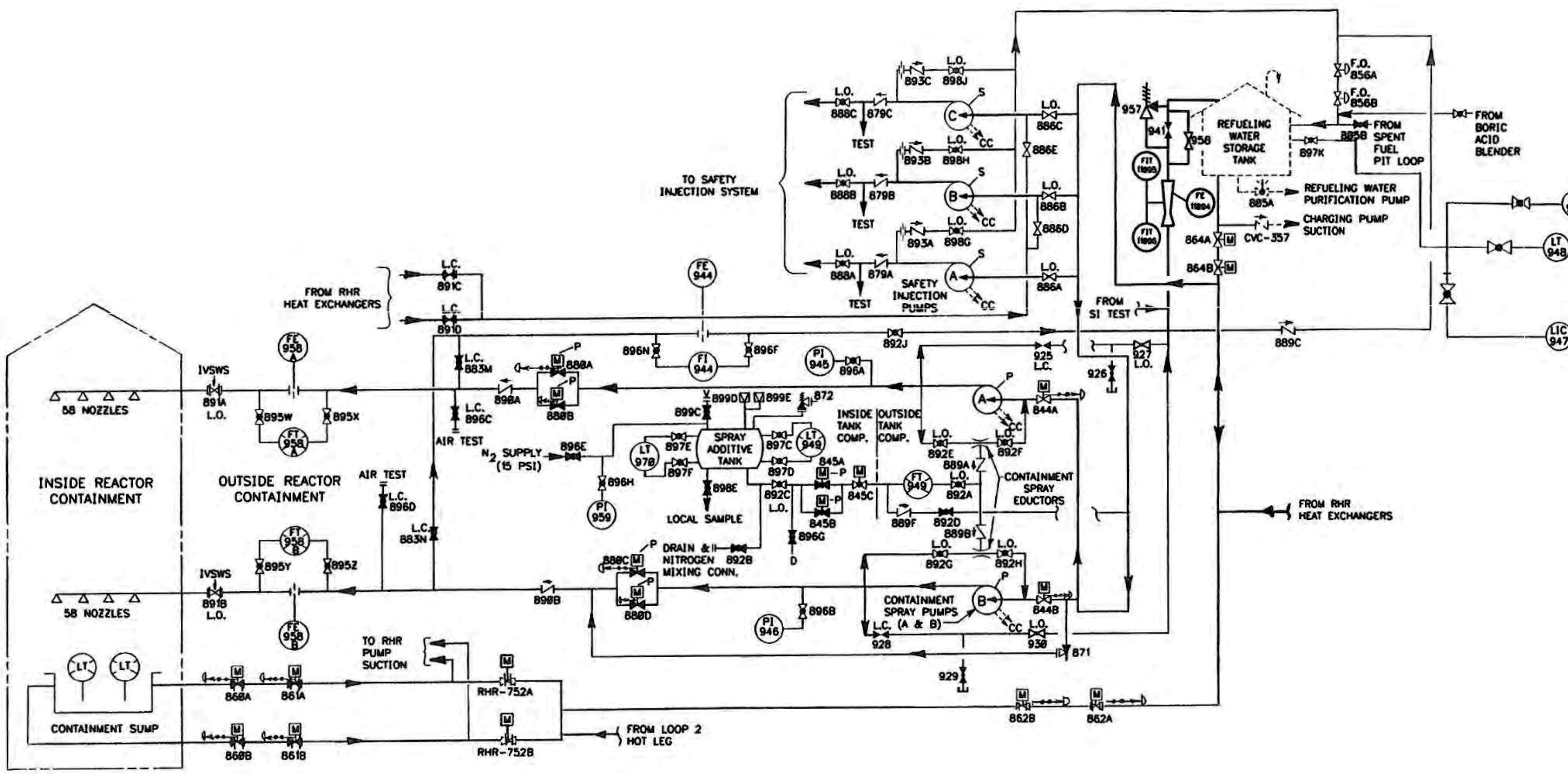
H. B. Robinson Unit 2 Carolina Power & Light Company UPDATED FINAL SAFETY ANALYSIS REPORT	HZP E-Bus Failure Case - Containment Pressure	FIGURE 6.2.1.4-9 Rev. 17



H. B. Robinson Unit 2 Carolina Power & Light Company UPDATED FINAL SAFETY ANALYSIS REPORT	HZP E-Bus Failure Case - Containment Temperature	FIGURE 6.2.1.4-10 Rev. 17



H. B. Robinson Unit 2 Carolina Power & Light Company UPDATED FINAL SAFETY ANALYSIS REPORT	HZP Runout Protection Failure Case - Containment Pressure	FIGURE 6.2.1.4-11 Rev. 17



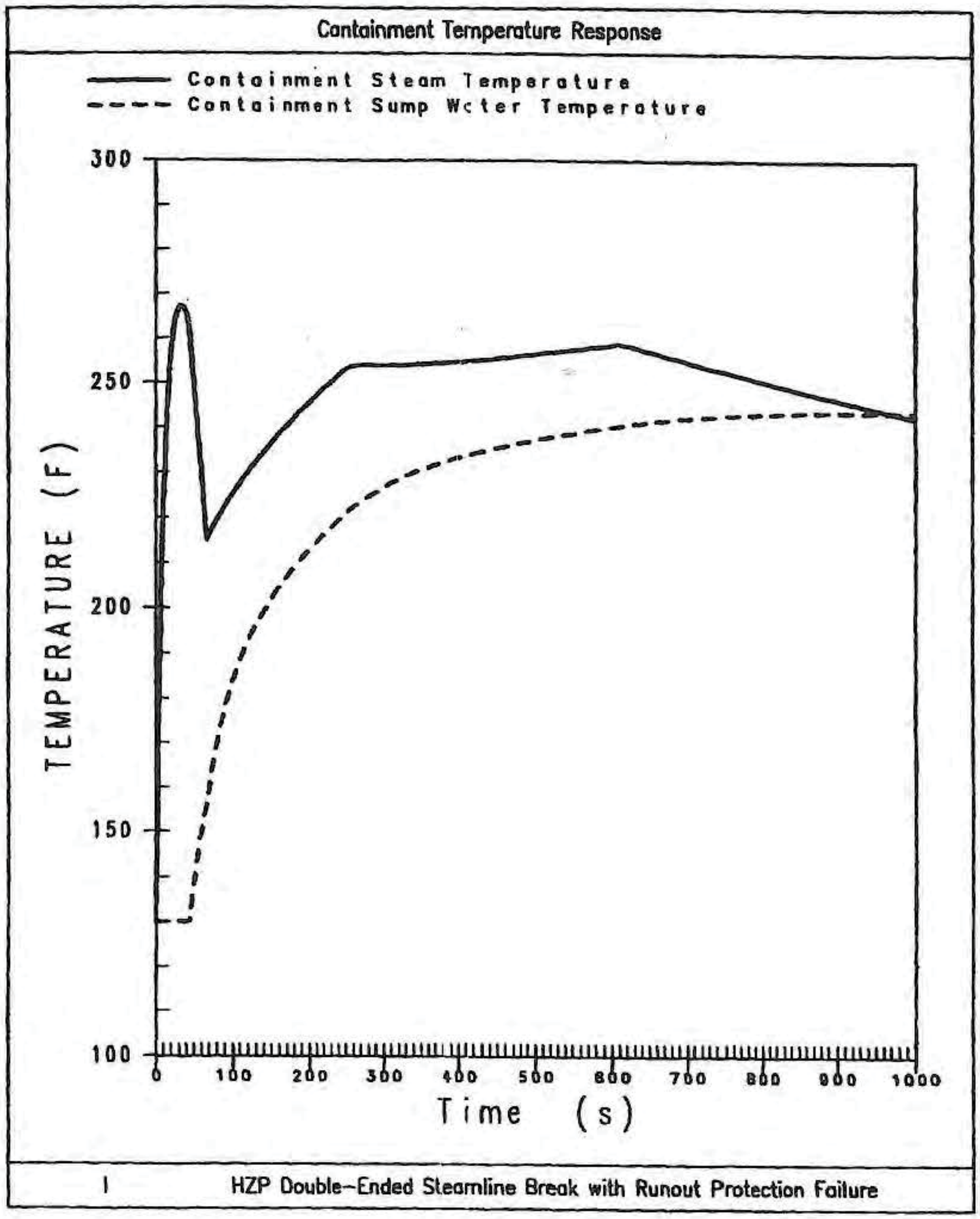
REF. DWG. CP-200-5379-1082

H. B. ROBINSON
UNIT 2
Carolina Power & Light Company
UPDATED FINAL SAFETY ANALYSIS REPORT

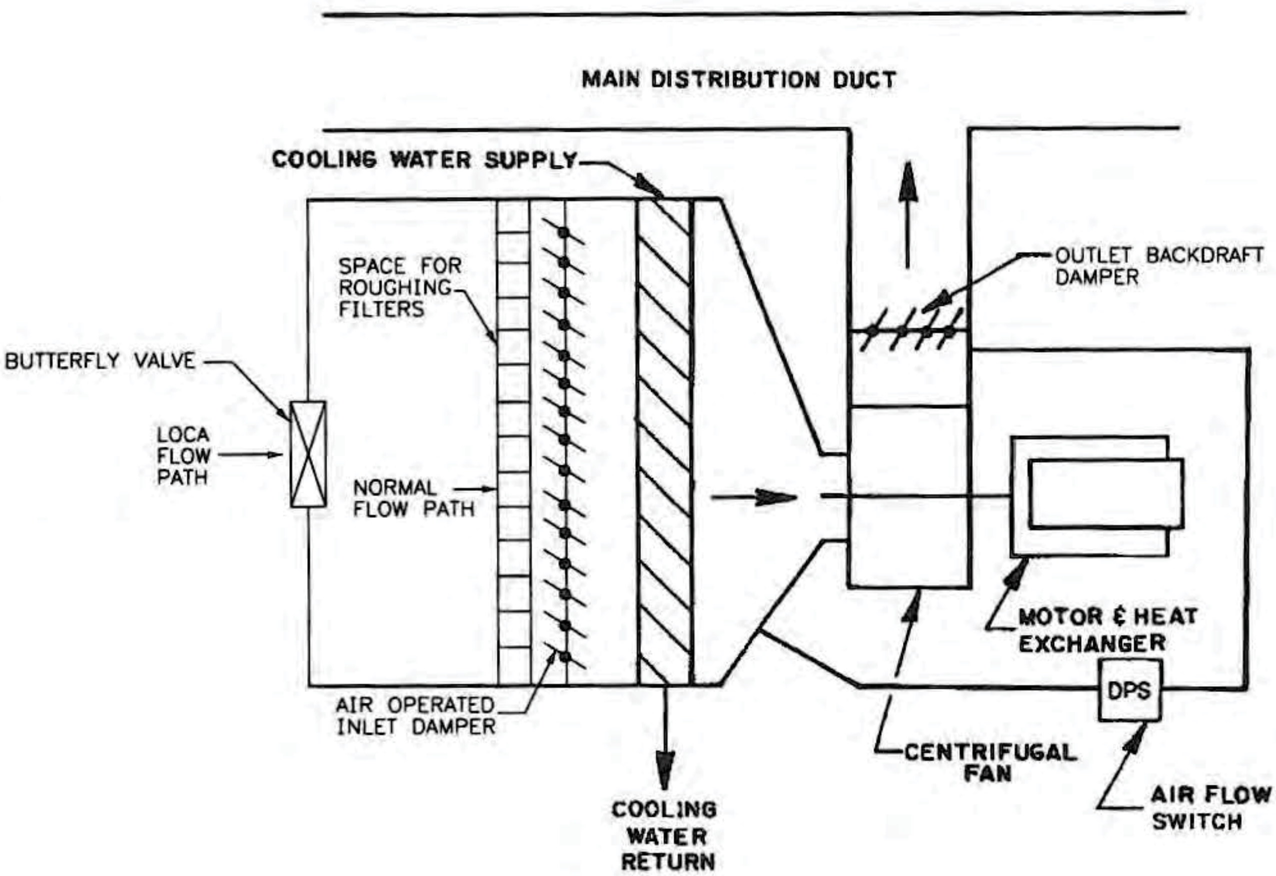
FLOW DIAGRAM
CONTAINMENT SPRAY

FIGURE 6.2.2-1

REVISION NO. 21

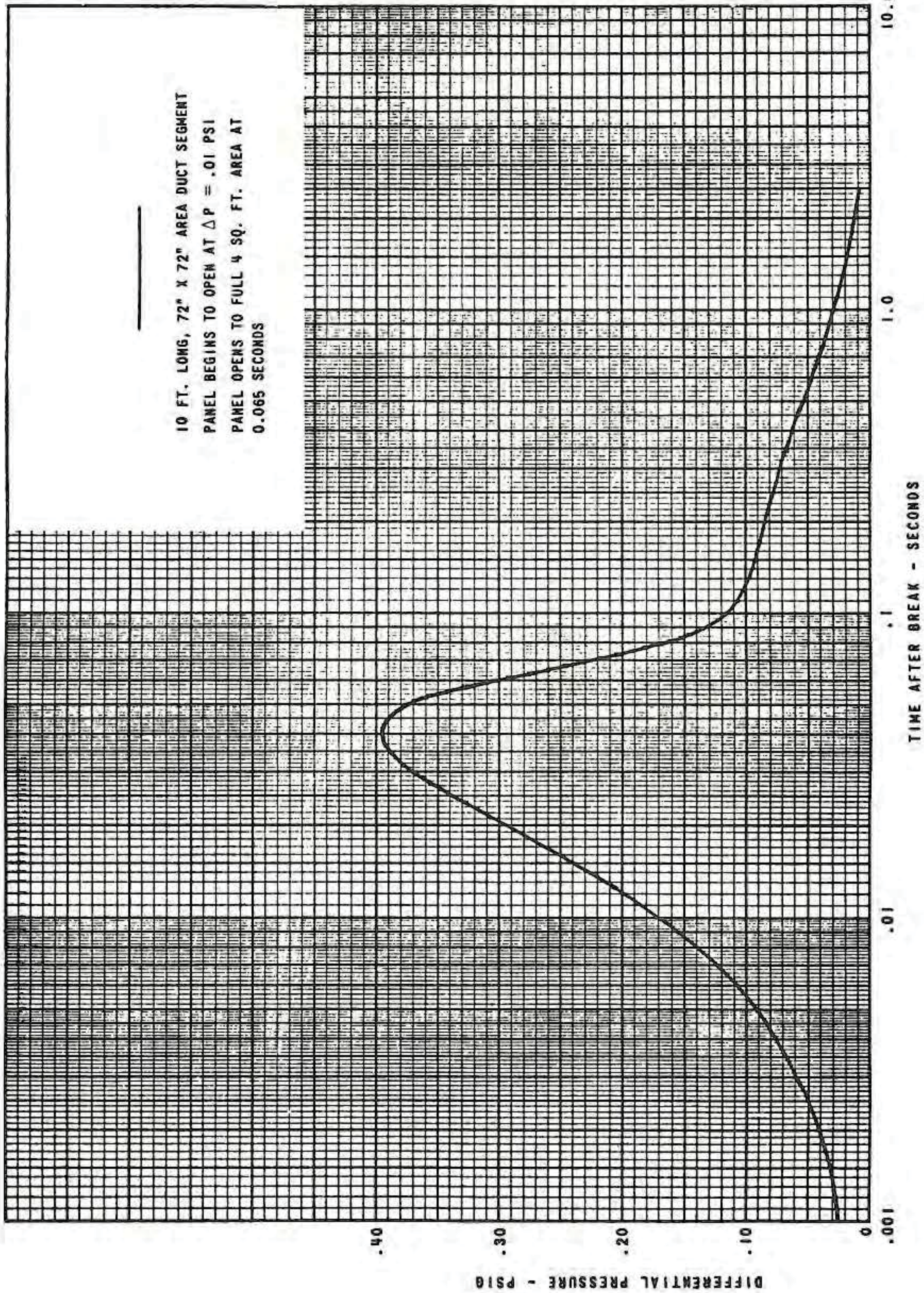


H. B. Robinson Unit 2 Carolina Power & Light Company UPDATED FINAL SAFETY ANALYSIS REPORT	**HZP Runout Protection Failure Case - Containment Temperature**	**FIGURE 6.2.1.4-12 Rev. 17**

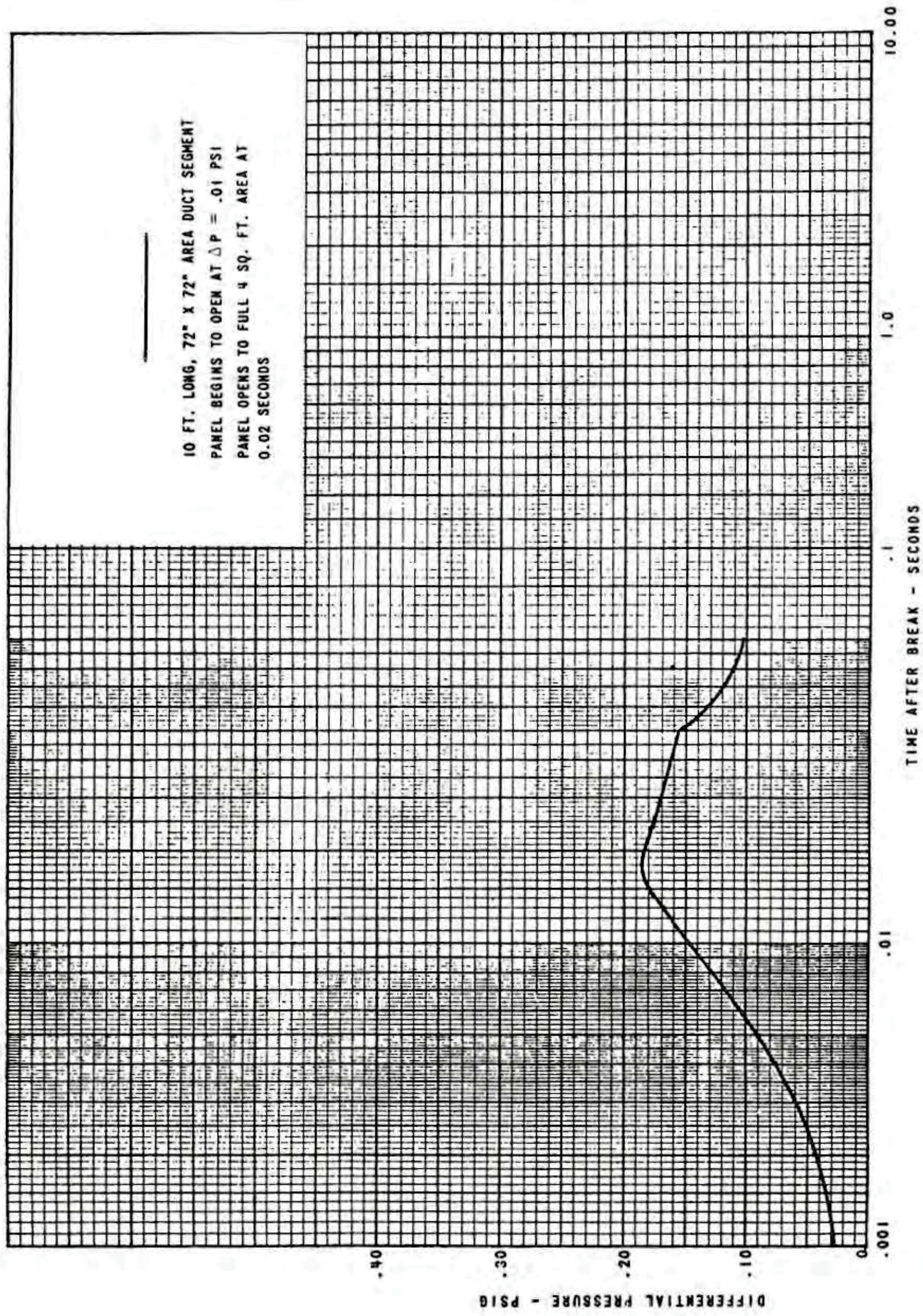


REVISION NO. 15

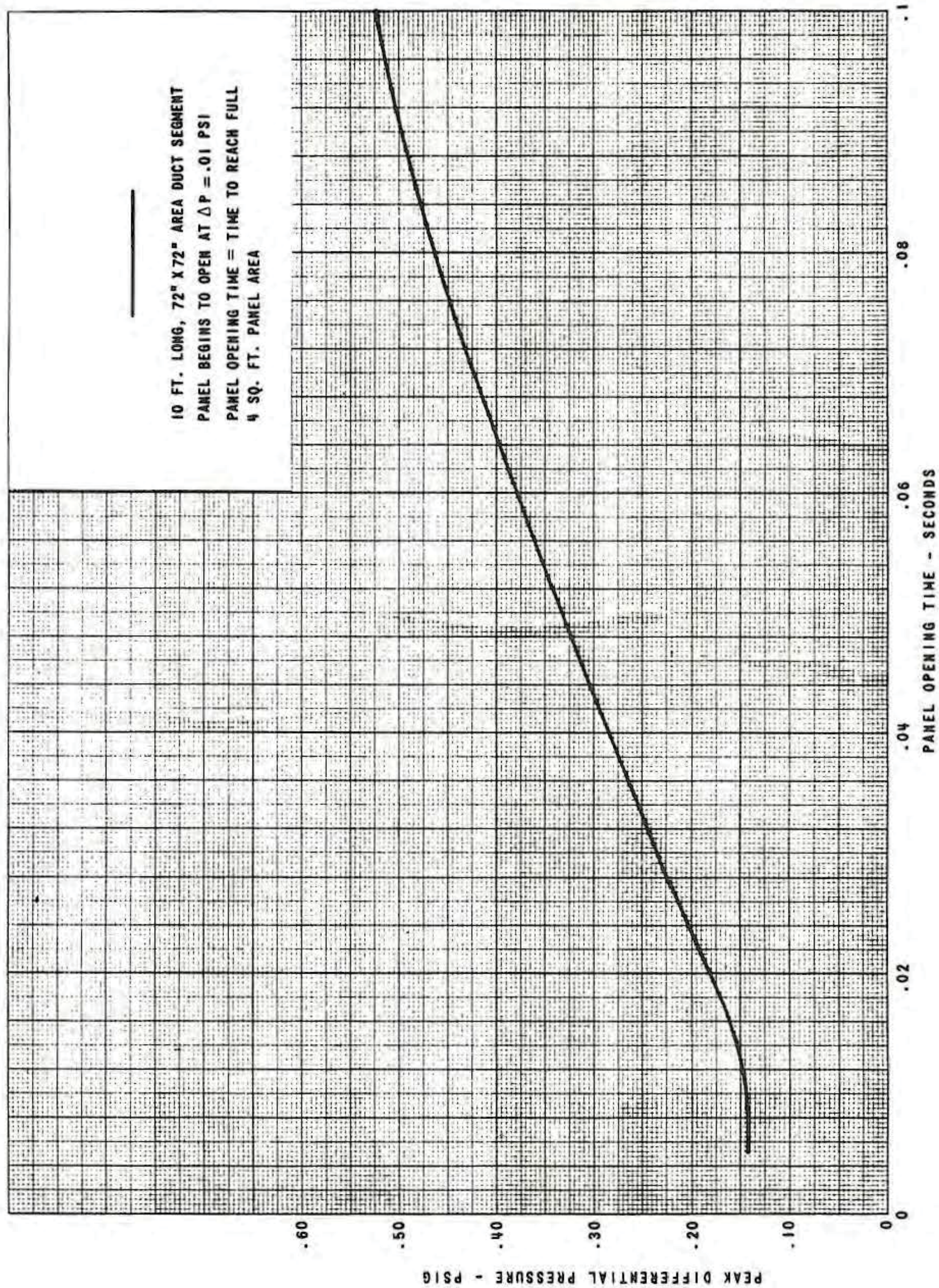
H. B. ROBINSON UNIT 2 Carolina Power & Light Company UPDATED FINAL SAFETY ANALYSIS REPORT	CONTAINMENT AIR RECIRCULATION COOLING SYSTEM	FIGURE 6.2.2 - 2



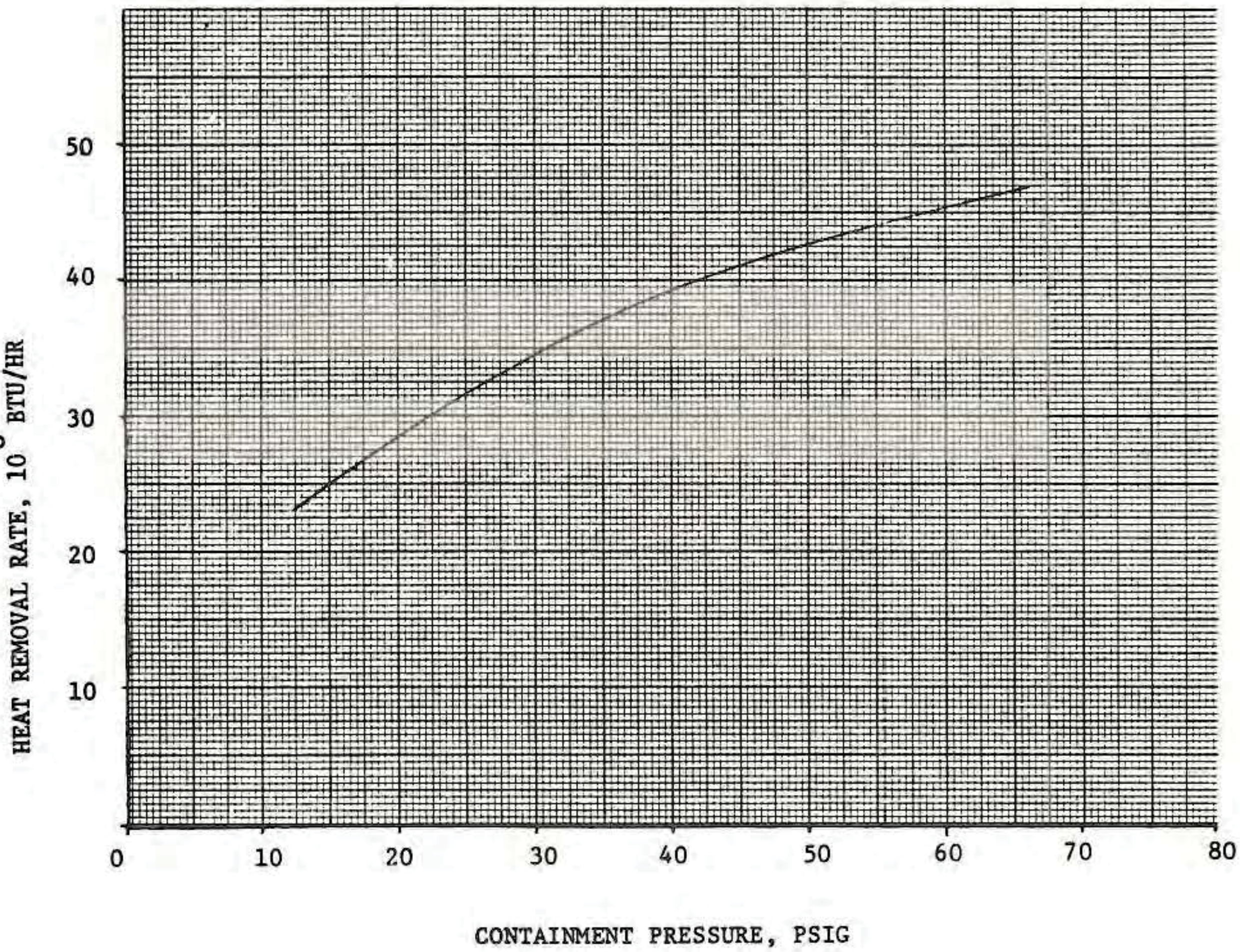
H. B. ROBINSON UNIT 2 Carolina Power & Light Company UPDATED FINAL SAFETY ANALYSIS REPORT	DIFFERENTIAL PRESSURE VS TIME FOR PANEL OPENING TIME = 0.065 SECONDS	FIGURE 6.2.2 - 3



H. B. ROBINSON UNIT 2 Carolina Power & Light Company UPDATED FINAL SAFETY ANALYSIS REPORT	DIFFERENTIAL PRESSURE VS TIME FOR PANEL OPENING TIME = 0.02 SECONDS	FIGURE 6.2.2 - 4



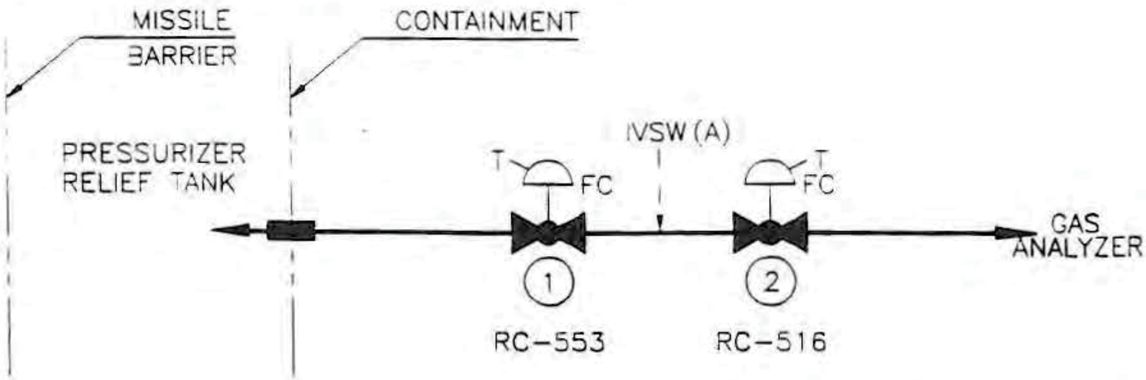
H. B. ROBINSON UNIT 2 Carolina Power & Light Company UPDATED FINAL SAFETY ANALYSIS REPORT	PEAK DIFFERENTIAL ACROSS DUCT WALLS VS PANEL OPENING TIME	FIGURE 6.2.2 - 5



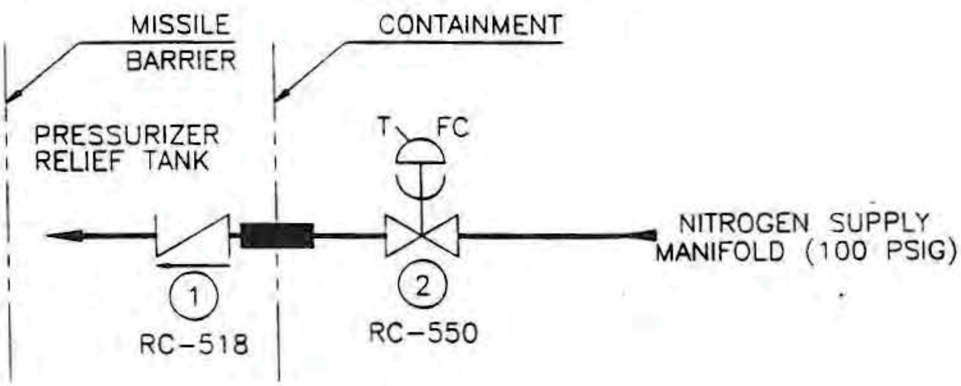
AMENDMENT NO. 6

H. B. ROBINSON UNIT 2 Carolina Power & Light Company UPDATED FINAL SAFETY ANALYSIS REPORT	FAN COOLER HEAT REMOVAL AS A FUNCTIO. OF CONTAINMENT PRESSURE	FIGURE 6.2.2 - 6

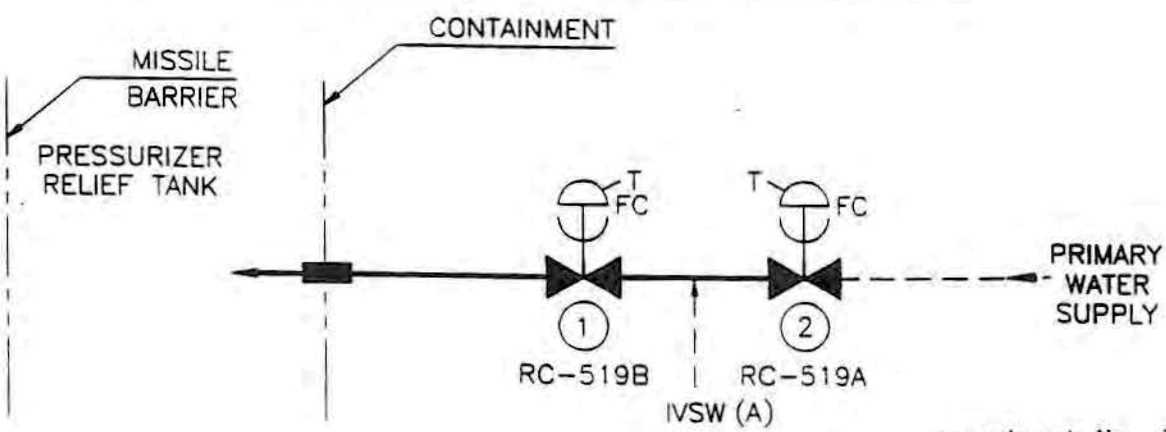
PENETRATION NO. 1 – PRESSURIZER RELIEF TANK GAS ANALYZER LINE



PENETRATION NO. 2 – PRESSURIZER RELIEF TANK NITROGEN SUPPLY LINE



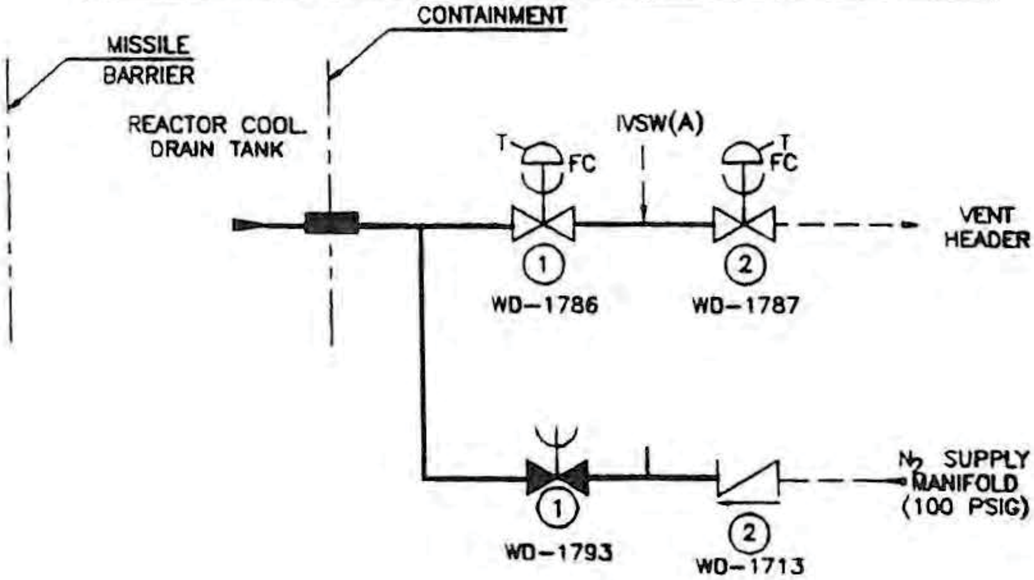
PENETRATION NO. 3 – PRESSURIZER RELIEF TANK MAKEUP



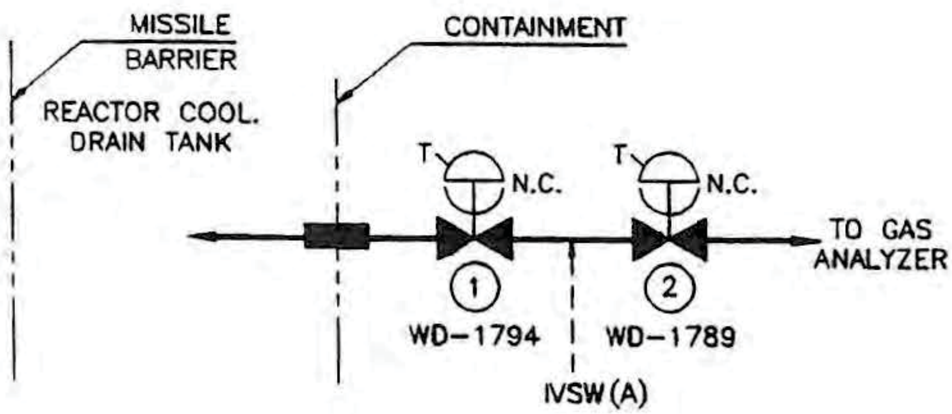
Amendment No. 12

H.B.ROBINSON UNIT 2 Carolina Power & Light Company UPDATED FINAL SAFETY ANALYSIS REPORT	CONTAINMENT ISOLATION VALVES PENETRATIONS P-1, P-2, P-3	FIGURE 6.2.4-1

PENETRATION NO. 4 – PRIMARY SYSTEM VENT HEADER (& N_2 SUPPLY LINE)



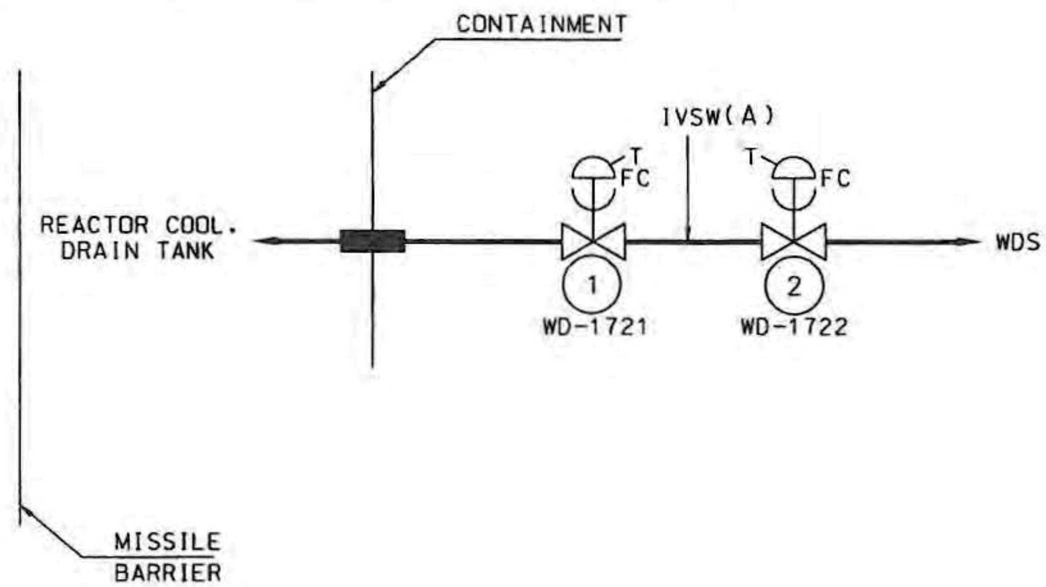
PENETRATION NO. 5 – REACTOR COOLANT DRAIN TANK GAS ANALYZER LINE



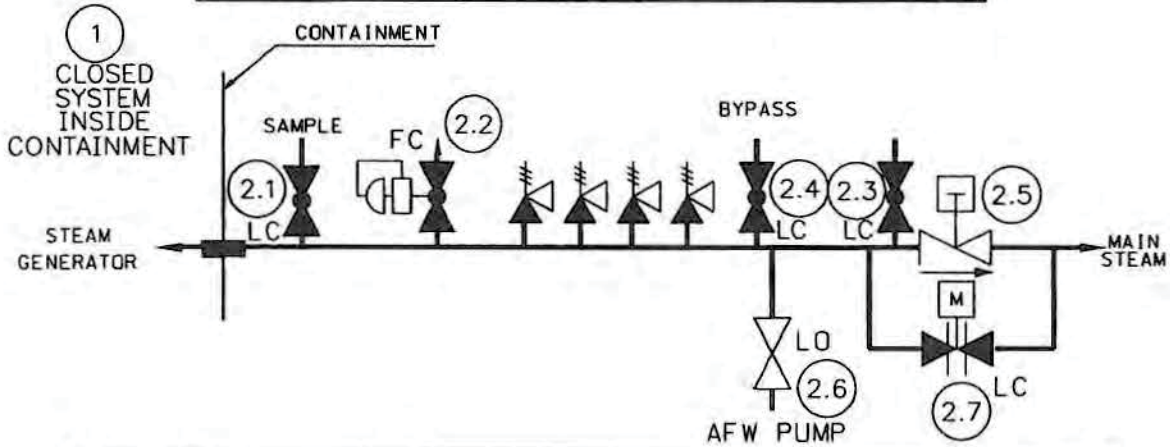
REVISION NO. 23

H.B.ROBINSON UNIT 2 Carolina Power & Light Company UPDATED FINAL SAFETY ANALYSIS REPORT	CONTAINMENT ISOLATION VALVES PENETRATIONS P-4, P-5	FIGURE 6.2.4-2

PENETRATION NO. 6 - REACTOR COOLANT DRAIN TANK PUMP DISCHARGE LINE



PENETRATIONS NO. 7, 8, 9 - MAIN STEAM HEADER



BARRIER	P-7	P-8	P-9
2.1	MS-10A	MS-11A	MS-12A
2.2	RV1-1	RV1-2	RV1-3
2.3	MS-21	MS-30	MS-39
2.4	MS-19	MS-28	MS-37
2.5	MS-V1-3A	MS-V1-3B	MS-V1-3C
2.6	MS-262A	MS-262B	MS-262C
2.7	MS-353A	MS-353B	MS-353C

Revision No. 17

H.B. ROBINSON
UNIT 2

Carolina Power & Light Company

UPDATED FINAL
SAFETY ANALYSIS REPORT

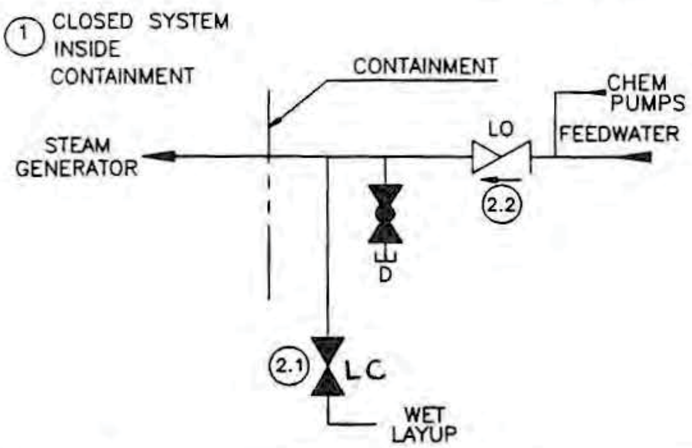
CONTAINMENT ISOLATION VALVES

PENETRATIONS P-6, P-7, P-8, P-9

FIGURE

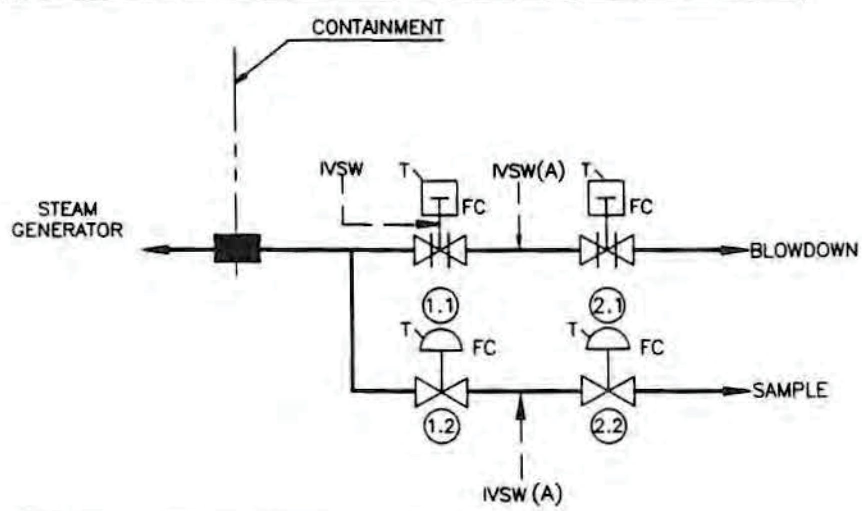
6.2.4-3

PENETRATIONS NO. 10, 11, 12 – FEEDWATER



BARRIER	P–10	P–11	P–12
2.1	FW–201	FW–203	FW–205
2.2	FW–8A	FW–8B	FW–8C

PENETRATIONS NO. 13, 14, 15 – STEAM GENERATOR BLOWDOWN

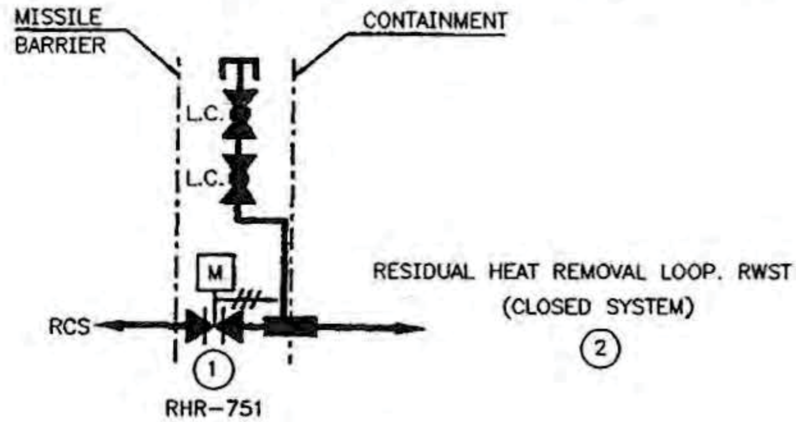


BARRIER	P–13	P–14	P–15
1.1	FCV–1931A	FCV–1932A	FCV–1930A
2.1	FCV–1931B	FCV–1932B	FCV–1930B
1.2	FCV–1934A	FCV–1935A	FCV–1933A
2.2	FCV–1934B	FCV–1935B	FCV–1933B

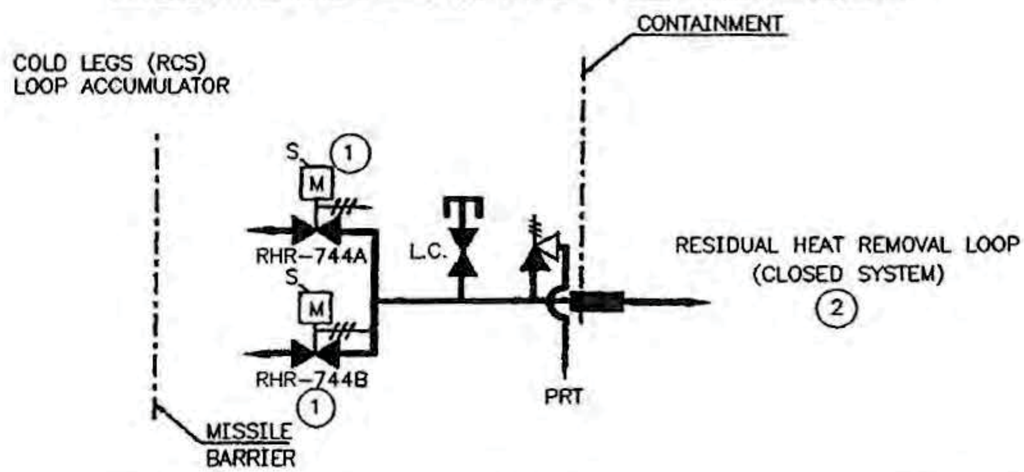
Revision No. 15

H.B.ROBINSON UNIT 2 Carolina Power & Light Company UPDATED FINAL SAFETY ANALYSIS REPORT	CONTAINMENT ISOLATION VALVES PENETRATIONS P-10, P-11, P-12, P-13, P-14, P-15	FIGURE 6.2.4-4

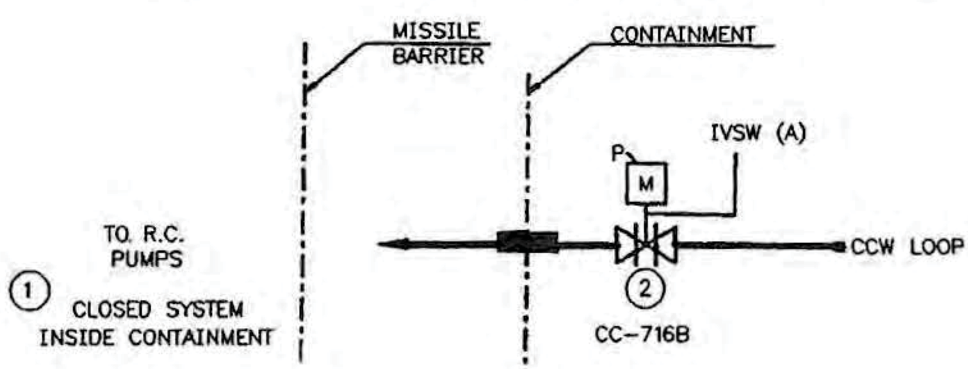
PENETRATION NO. 16 – RESIDUAL HEAT REMOVAL LOOP OUT



PENETRATION NO. 17 – RESIDUAL HEAT REMOVAL LOOP IN



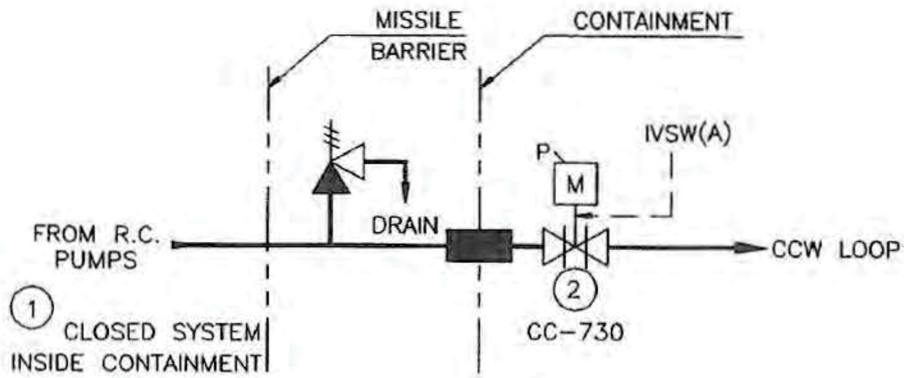
PENETRATION NO. 18 – REACTOR COOLANT PUMP COOLING WATER IN



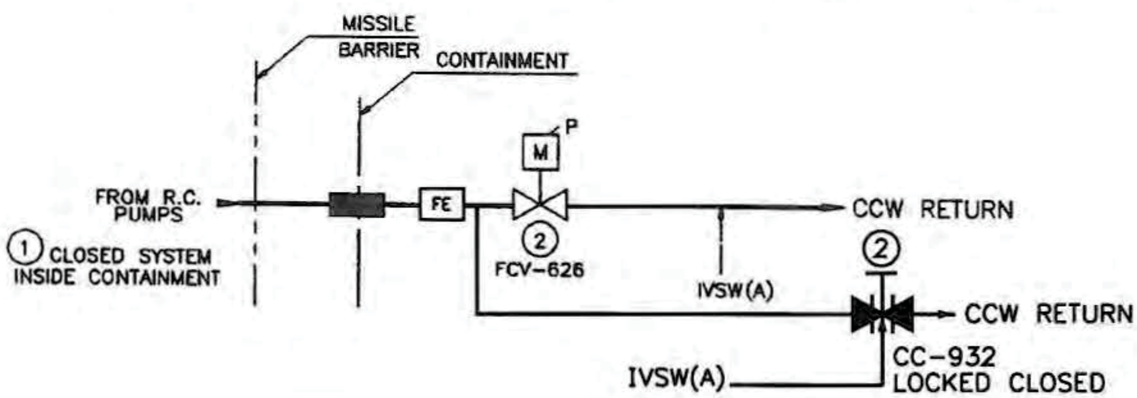
REVISION NO. 22

H.B. ROBINSON UNIT 2 Carolina Power & Light Company UPDATED FINAL SAFETY ANALYSIS REPORT	CONTAINMENT ISOLATION VALVES PENETRATIONS P–16, P–17, P–18	FIGURE 6.2.4–5

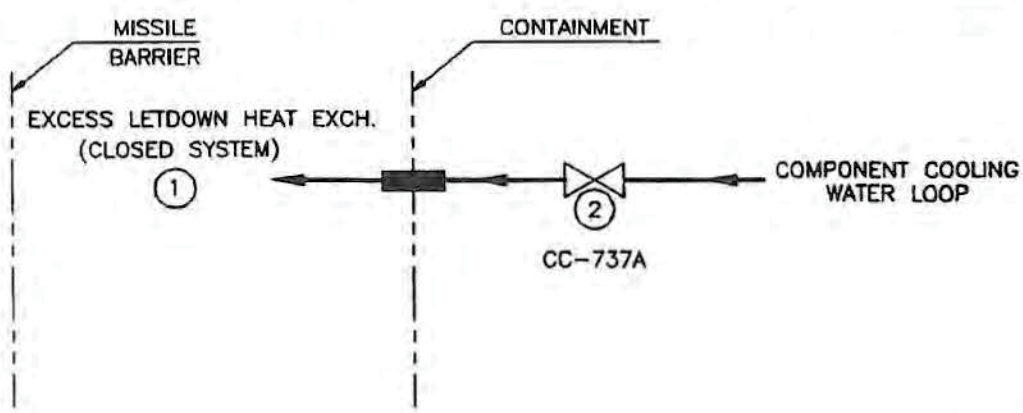
PENETRATION NO. 19 – REACTOR COOLANT PUMP COOLING WATER OUT



PENETRATION NO. 2Ø – REACTOR COOLANT PUMP COOLING WATER OUT

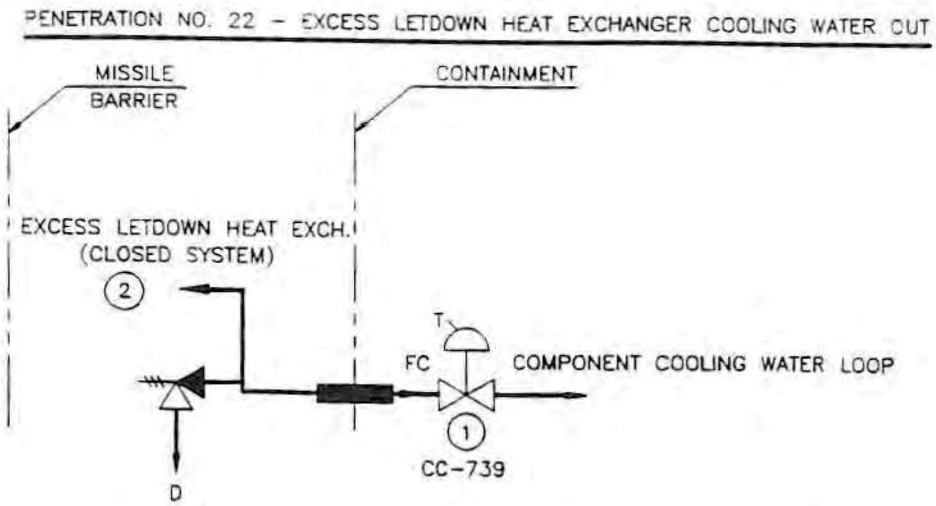


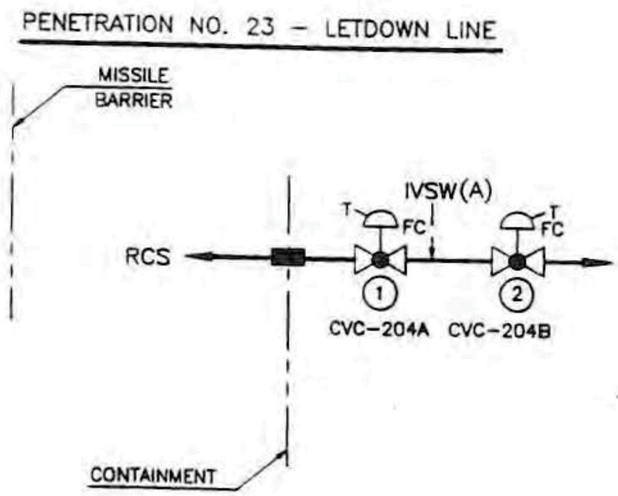
PENETRATION NO. 21 – EXCESS LETDOWN HEAT EXCHANGER COOLING WATER IN

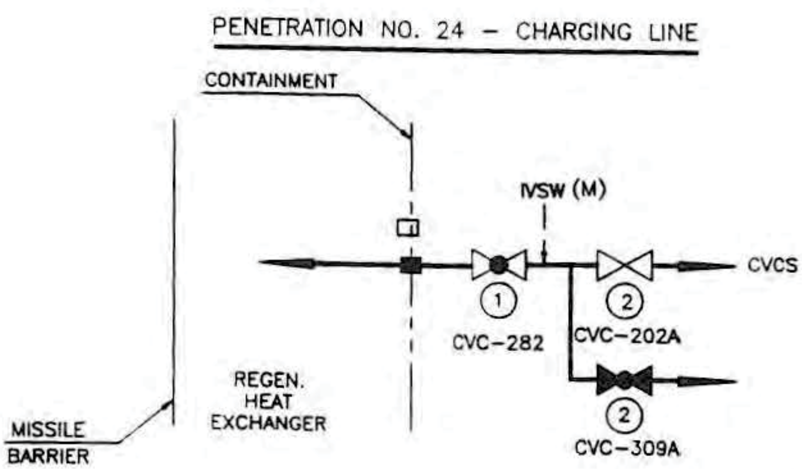


REVISION NO. 21

H.B. ROBINSON UNIT 2 Carolina Power & Light Company UPDATED FINAL SAFETY ANALYSIS REPORT	CONTAINMENT ISOLATION VALVES PENETRATIONS P-19, P-2Ø, P-21	FIGURE 6.2.4-6



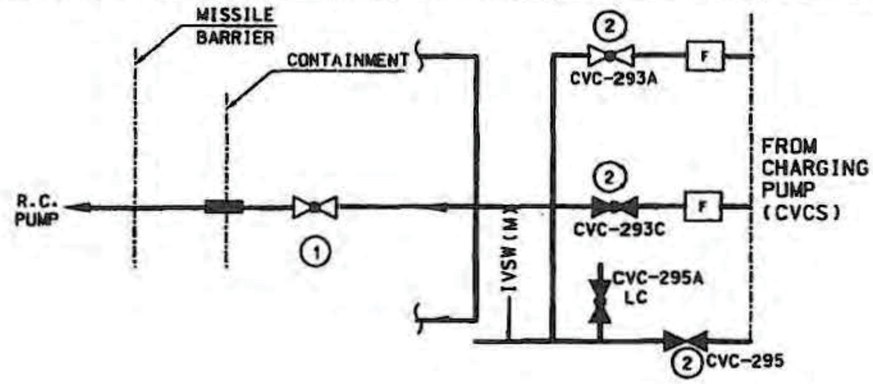




Amendment No. 12

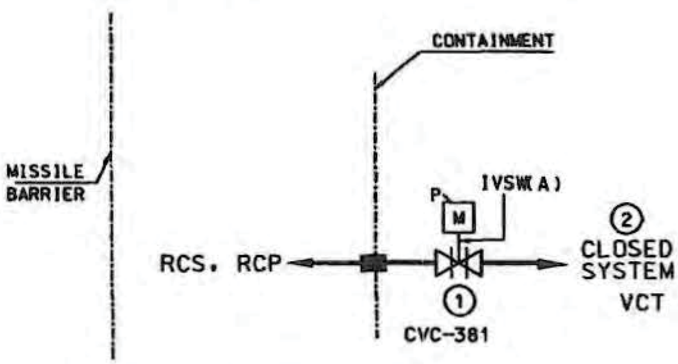
H.B.ROBINSON UNIT 2 Carolina Power & Light Company UPDATED FINAL SAFETY ANALYSIS REPORT	CONTAINMENT ISOLATION VALVES PENETRATIONS P-22, P-23, P-24	FIGURE 6.2.4-7

PENETRATIONS NO. 25, 26, 27 - REACTOR COOLANT PUMP SEAL WATER SUPPLY LINE

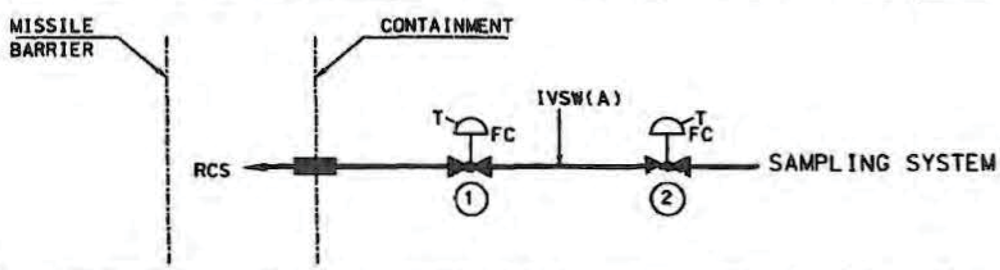


BARRIER	P-25	P-26	P-27
1	CVC-297C	CVC-297B	CVC-297A

PENETRATION NO. 28 - REACTOR COOLANT PUMP SEAL WATER RETURN LINE



PENETRATIONS NO. 29, 30, 31 - REACTOR COOLANT SYSTEM SAMPLE LINE

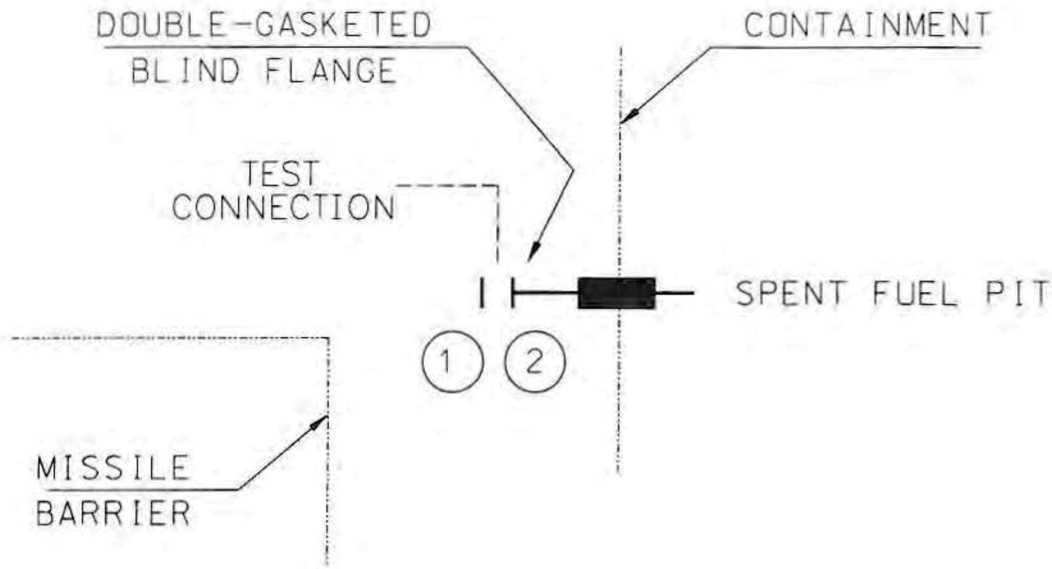


BARRIER	P-29	P-30	P-31
1	PS-956A	PS-956C	PS-956E
2	PS-956B	PS-956D	PS-956F

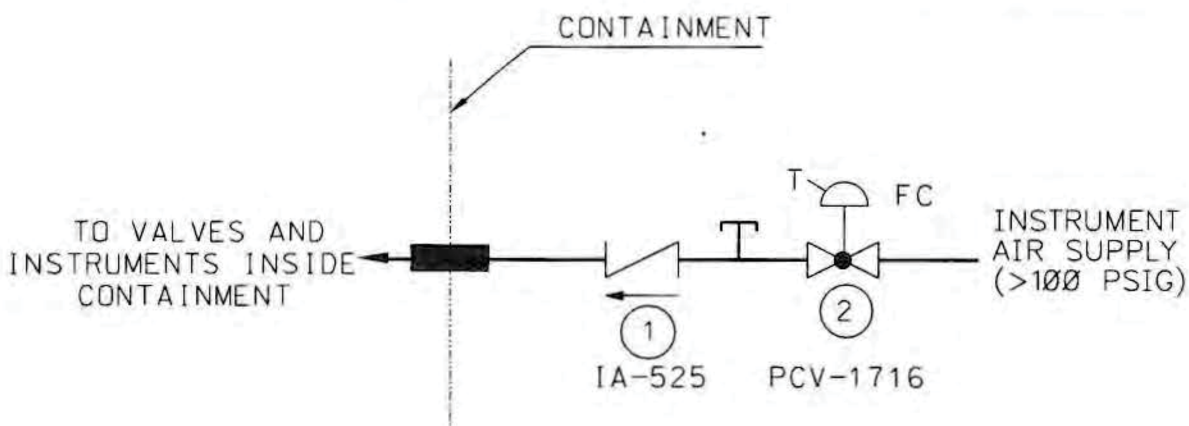
Revision No. 17

H.B. ROBINSON UNIT 2 Carolina Power & Light Company UPDATED FINAL SAFETY ANALYSIS REPORT	CONTAINMENT ISOLATION VALVES PENETRATIONS P-25, P-26, P-27, P-28, P-29, P-30, P-31	FIGURE 6.2.4-8

PENETRATION NO. 32 - FUEL TRANSFER TUBE



PENETRATION NO. 33 - INSTRUMENT AIR HEADER

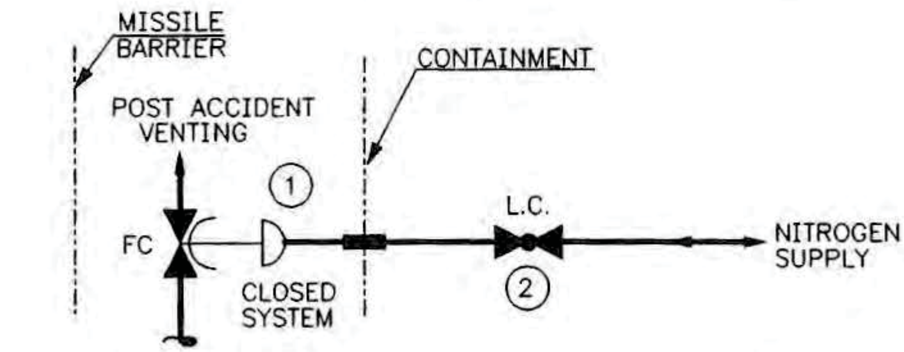


Revision No. 14

H.B. ROBINSON UNIT 2 Carolina Power & Light Company UPDATED FINAL SAFETY ANALYSIS REPORT	CONTAINMENT ISOLATION VALVES PENETRATIONS P-32, P-33	FIGURE 6.2.4-9

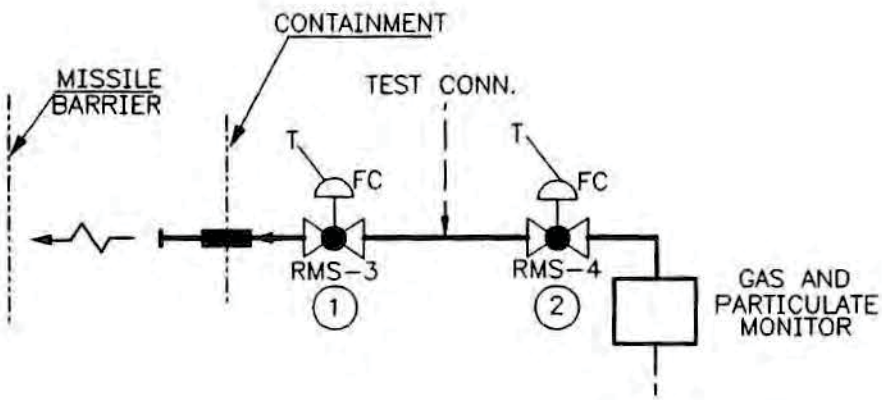
PENETRATION NO. 34A, 34B, 34C, 34D

POST ACCIDENT VENTING VALVE OPERATOR NITROGEN SUPPLY

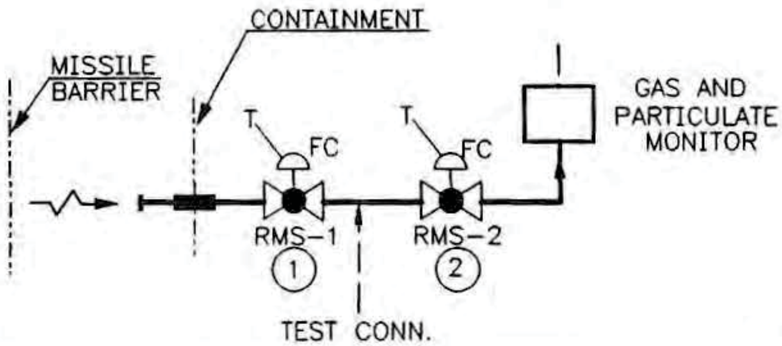


BARRIER	P-34A	P-34B	P-34C	P-34D
2	PAC-37	PAV-35	PAV-33	PAV-31

PENETRATION NO. 35 - CONTAINMENT AIR SAMPLE IN



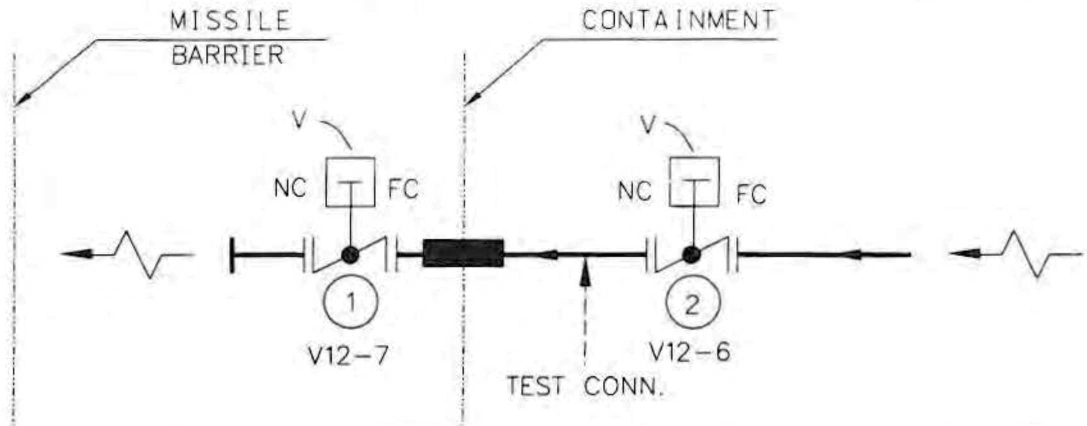
PENETRATION NO. 36 - CONTAINMENT AIR SAMPLE OUT



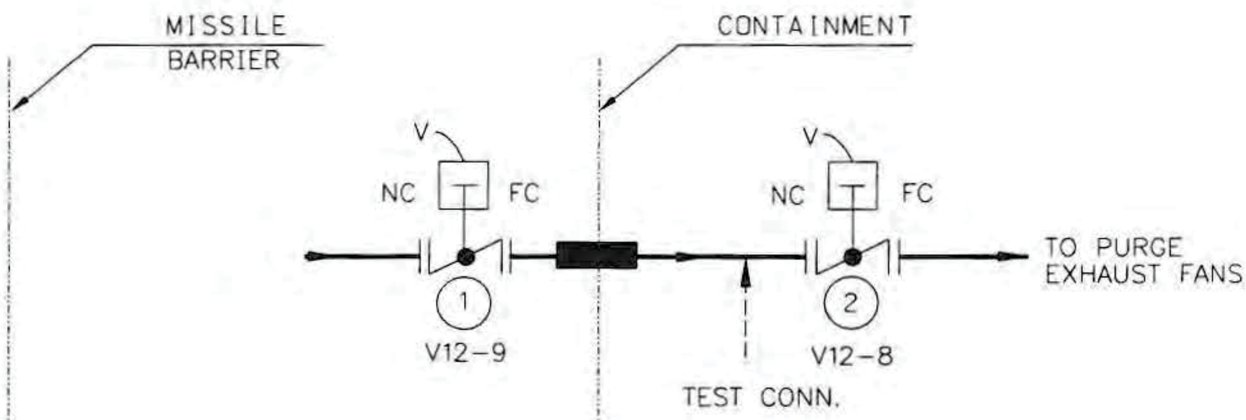
REVISION NO. 20

H.B. ROBINSON UNIT 2 Carolina Power & Light Company UPDATED FINAL SAFETY ANALYSIS REPORT	CONTAINMENT ISOLATION VALVES PENETRATIONS P-34A, P-34B, P-34C, P-34D, P-35, P-36	FIGURE 6.2.4-10

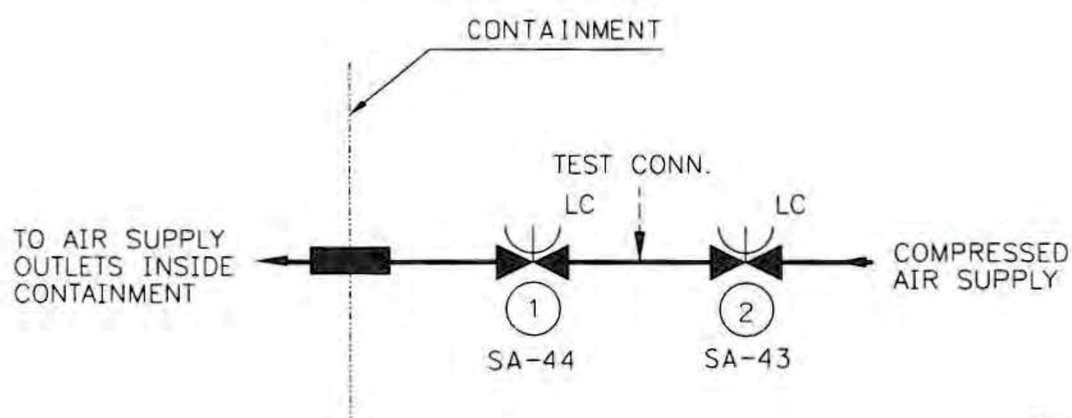
PENETRATION NO. 37 – CONTAINMENT PURGE SUPPLY DUCT



PENETRATION NO. 38 – CONTAINMENT PURGE EXHAUST DUCT



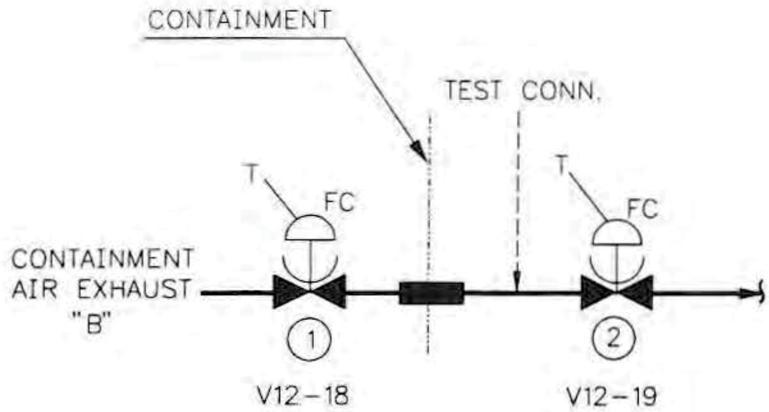
PENETRATION NO. 39 – PLANT AIR SUPPLY HEADER



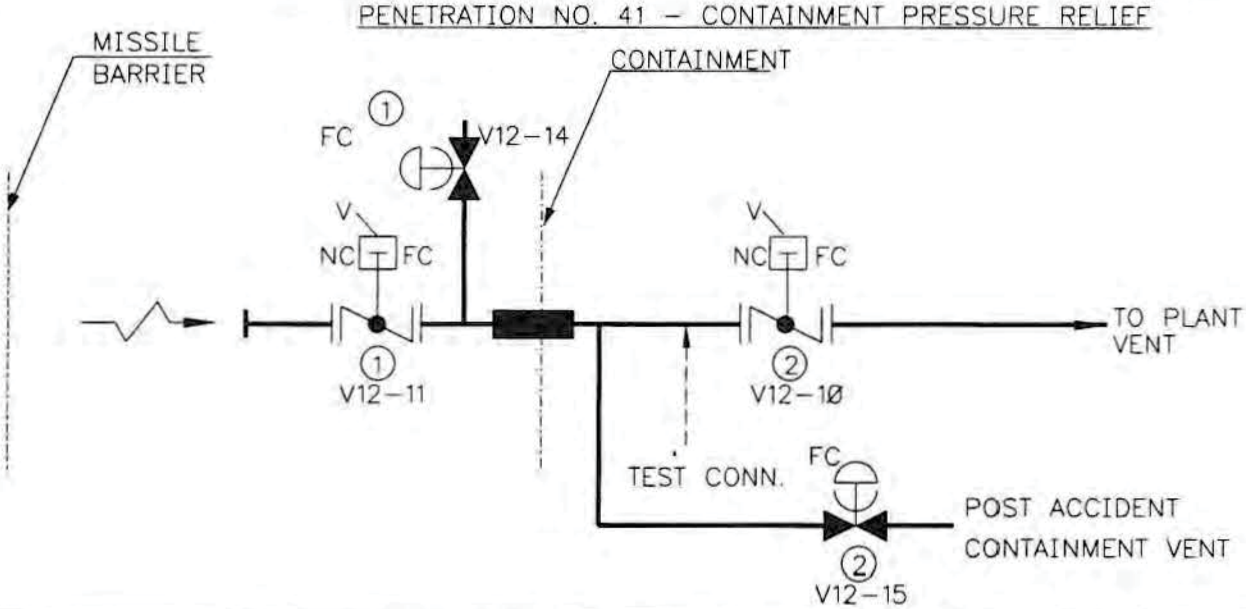
Revision No. 14

H.B. ROBINSON UNIT 2 Carolina Power & Light Company UPDATED FINAL SAFETY ANALYSIS REPORT	CONTAINMENT ISOLATION VALVES PENETRATIONS P-37, P-38, P-39	FIGURE 6.2.4-11

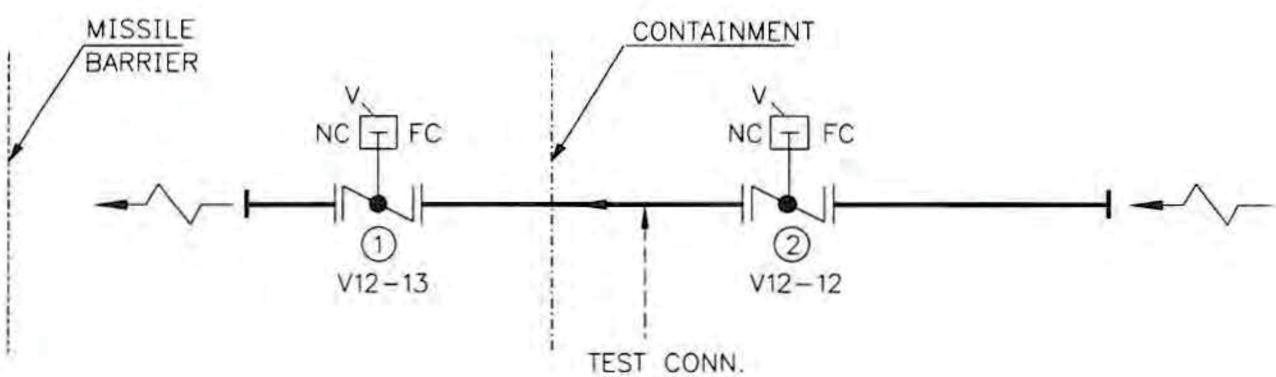
PENETRATION NO. 40 – POST ACCIDENT VENTING



PENETRATION NO. 41 – CONTAINMENT PRESSURE RELIEF



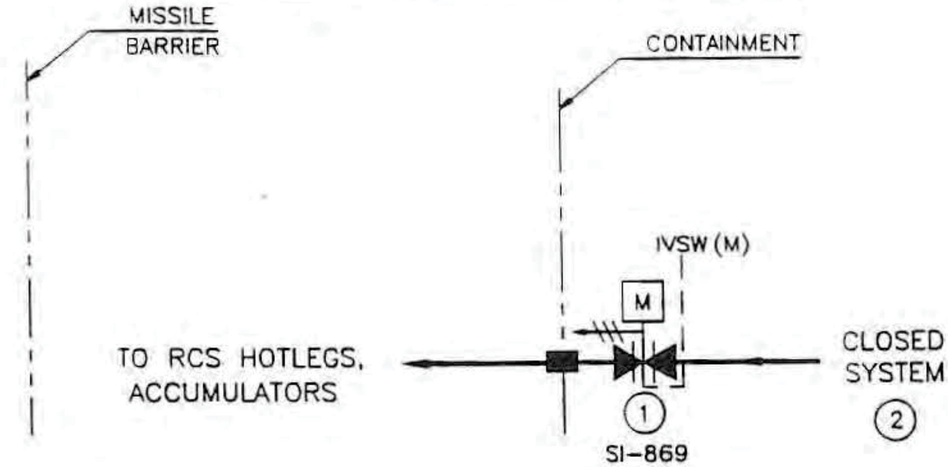
PENETRATION NO. 42 – CONTAINMENT VACUUM RELIEF



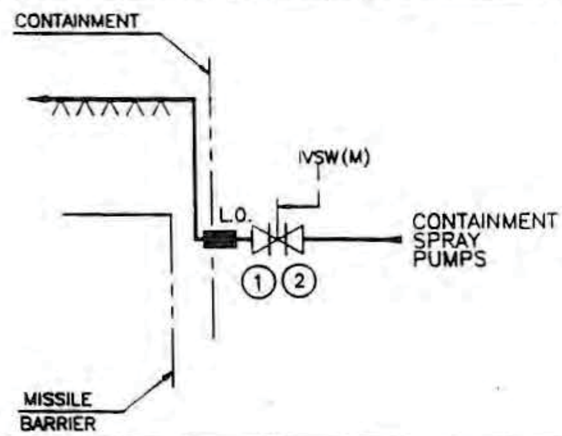
Revision No. 14

H.B. ROBINSON UNIT 2 Carolina Power & Light Company UPDATED FINAL SAFETY ANALYSIS REPORT	CONTAINMENT ISOLATION VALVES PENETRATIONS P–40, P–41, P–42	FIGURE 6.2.4–12

PENETRATION NO. 43 – SAFETY INJECTION LINE

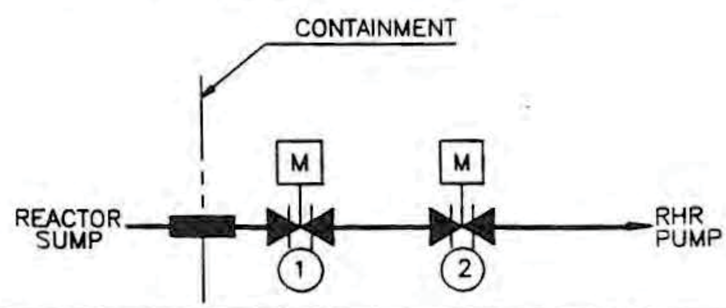


PENETRATIONS NO. 44, 45 – CONTAINMENT SPRAY HEADER



BARRIER	P–44	P–45
1, 2	SI–891A	SI–891B

PENETRATIONS NO. 46, 47 – CONTAINMENT SUMP RECIRCULATION LINE

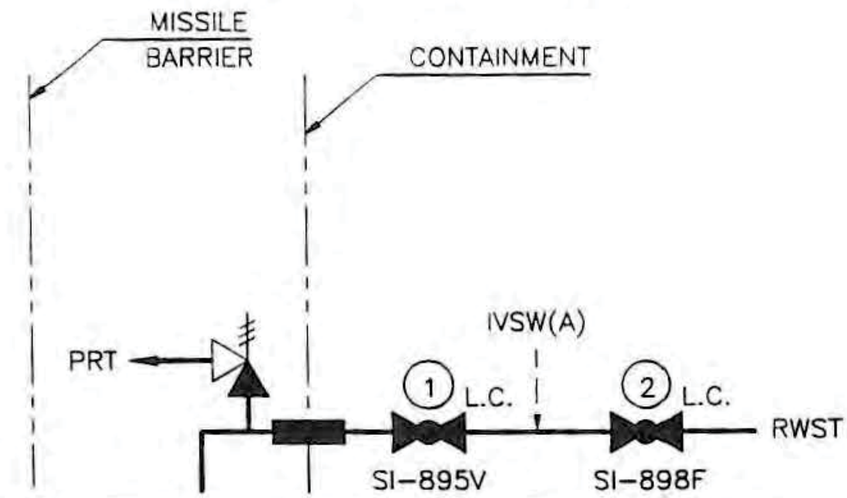


BARRIER	P–46	P–47
1	SI–860A	SI–860B
2	SI–861A	SI–861B

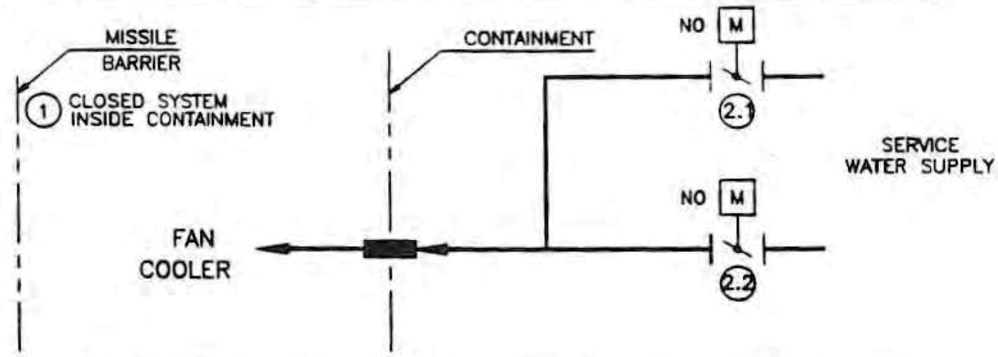
Amendment No. 12

H.B.ROBINSON UNIT 2 Carolina Power & Light Company UPDATED FINAL SAFETY ANALYSIS REPORT	CONTAINMENT ISOLATION VALVES PENETRATIONS P-43, P-44, P-45, P-46, P-47	FIGURE 6.2.4-13

PENETRATION NO. 48 – SAFETY INJECTION TEST LINE

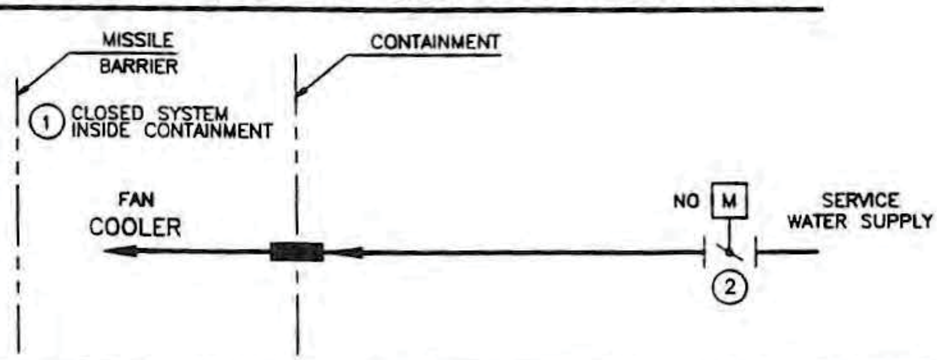


PENETRATIONS NO. 49, 51 – VENTILATION SYSTEM COOLING WATER IN



BARRIER	P–49	P–51
2.1	V6–33B	V6–33D
2.2	V6–33F	V6–33E

PENETRATIONS NO. 50, 52 – VENTILATION SYSTEM COOLING WATER IN

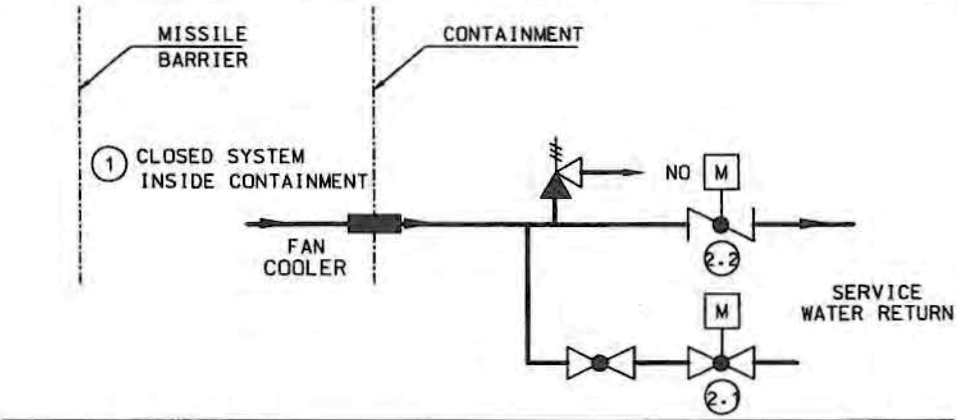


BARRIER	P–50	P–52
2	V6–33A	V6–33C

Amendment No. 12

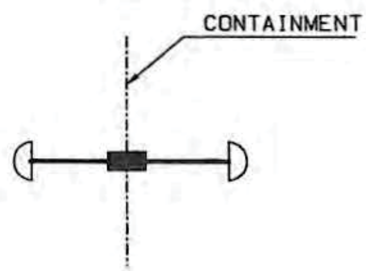
H.B.ROBINSON UNIT 2 Carolina Power & Light Company UPDATED FINAL SAFETY ANALYSIS REPORT	CONTAINMENT ISOLATION VALVES PENETRATIONS P-48, P-49, P-50, P-51, P-52	FIGURE 6.2.4-14

PENETRATIONS NO. 53, 54, 55, 56 - VENTILATION SYSTEM COOLING WATER OUT

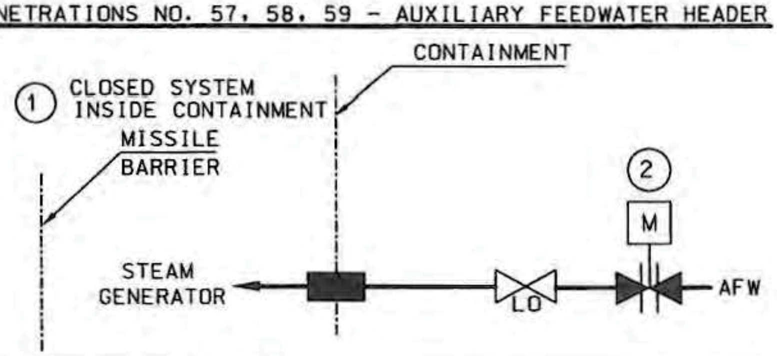


BARRIER	P-53	P-54	P-55	P-56
2.1	V6-35B	V6-35C	V6-35D	V6-35A
2.2	V6-34B	V6-34C	V6-34D	V6-34A

PENETRATIONS NO. 53A, 54A, 55A, 56A - SPARE



PENETRATIONS NO. 57, 58, 59 - AUXILIARY FEEDWATER HEADER

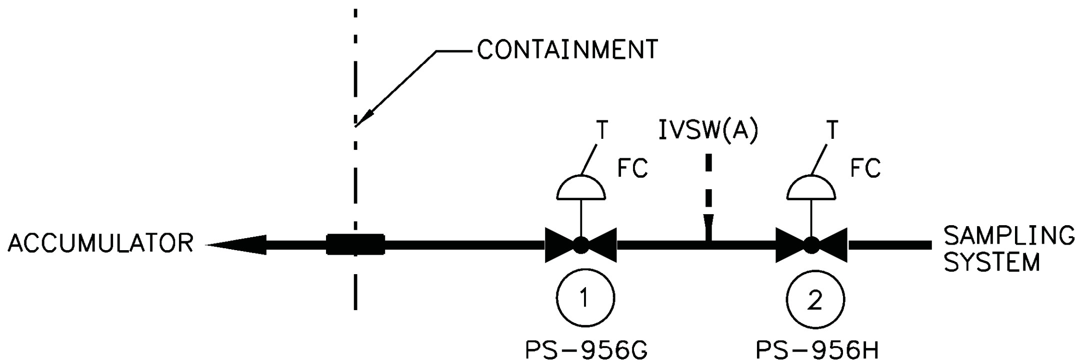


BARRIER	P-57	P-58	P-59
2	V2-16B	V2-16A	V2-16C

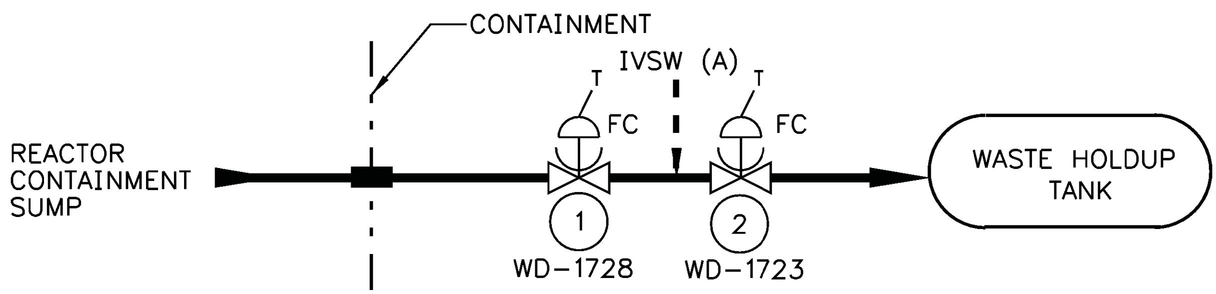
Revision No. 17

H.B. ROBINSON UNIT 2 Carolina Power & Light Company UPDATED FINAL SAFETY ANALYSIS REPORT	CONTAINMENT ISOLATION VALVES PENETRATIONS P-53, P-53A, P-54, P-54A, P-55, P-55A, P-56, P-56A, P-57, P-58, P-59	FIGURE 6.2.4-15

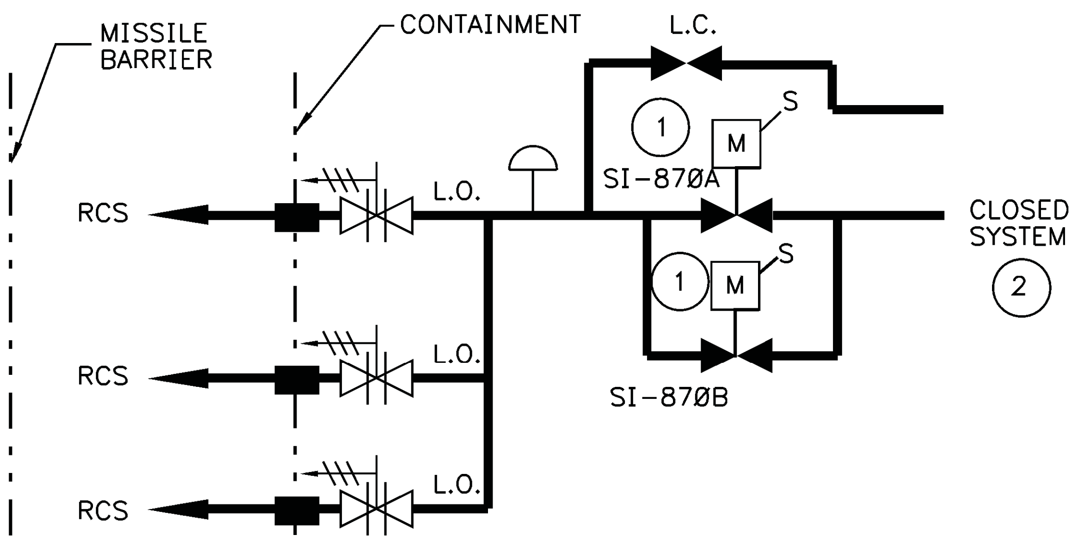
PENETRATION NO. 6Ø – ACCUMULATOR SAMPLE LINE



PENETRATION NO. 61 – CONTAINMENT SUMP PUMPS DISCHARGE LINE



PENETRATIONS NO. 62, 63, 64 – BORON INJECTION LINES

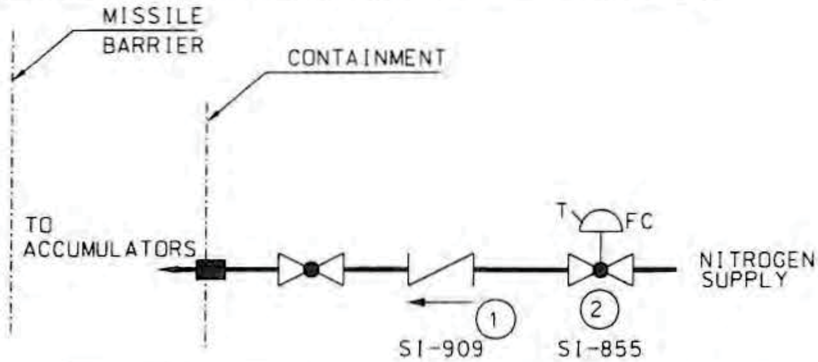


- SI–87ØA AND SI–87ØB CAN BE MANUALLY CLOSED BY OPERATOR AFTER RWST HAS BEEN EMPTIED BY SAFETY INJECTION IF THE MOV IS NOT OPERABLE.

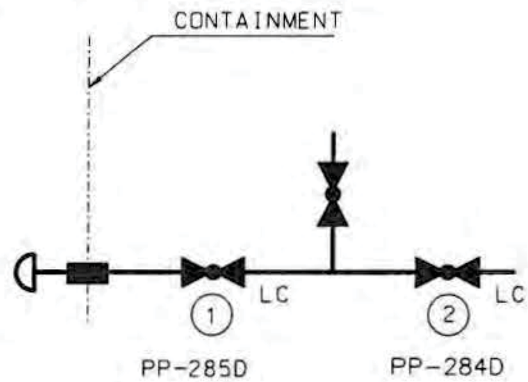
Revision No. 26

H.B. ROBINSON UNIT 2 DUKE ENERGY UPDATED FINAL SAFETY ANALYSIS REPORT	CONTAINMENT ISOLATION VALVES PENETRATIONS P–6Ø, P–61, P–62, P–63, P–64	FIGURE 6.2.4–16

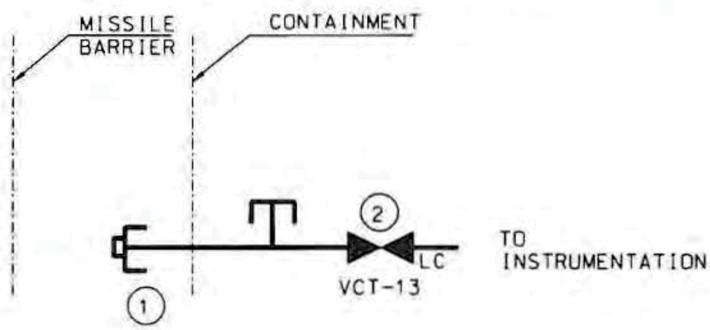
PENETRATION NO. 65 - ACCUMULATOR NITROGEN SUPPLY



PENETRATION NO. 66 - CONTAINMENT TEST CHANNEL LINE (ABANDONED)



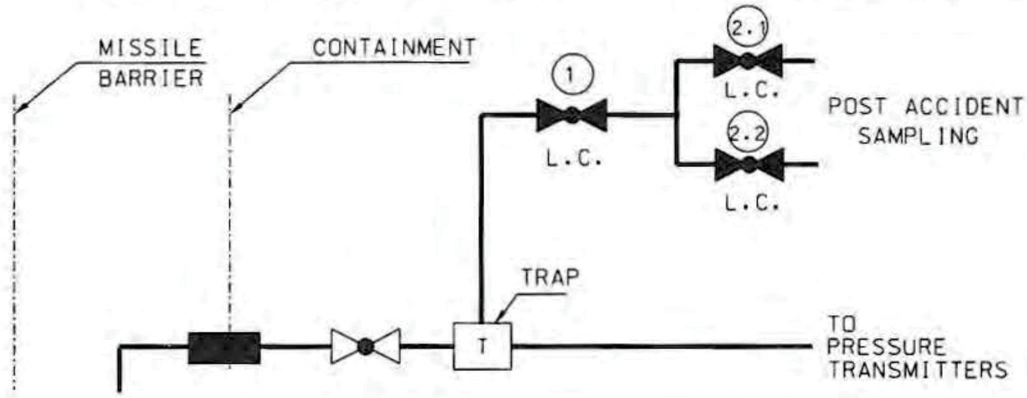
PENETRATION NO. 67 - CONTAINMENT CONTROLLED LEAK



Revision No. 14

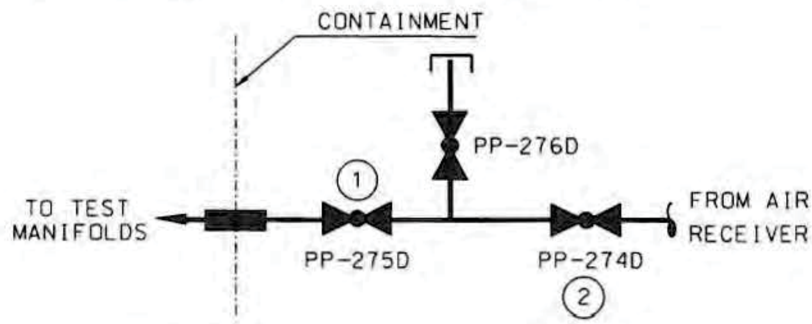
H.B. ROBINSON UNIT 2 Carolina Power & Light Company UPDATED FINAL SAFETY ANALYSIS REPORT	CONTAINMENT ISOLATION VALVES PENETRATIONS P-65, P-66, P-67	FIGURE 6.2.4-17

PENETRATIONS NO. 68, 69, 70 - CONTAINMENT PRESSURE SENSING LINES

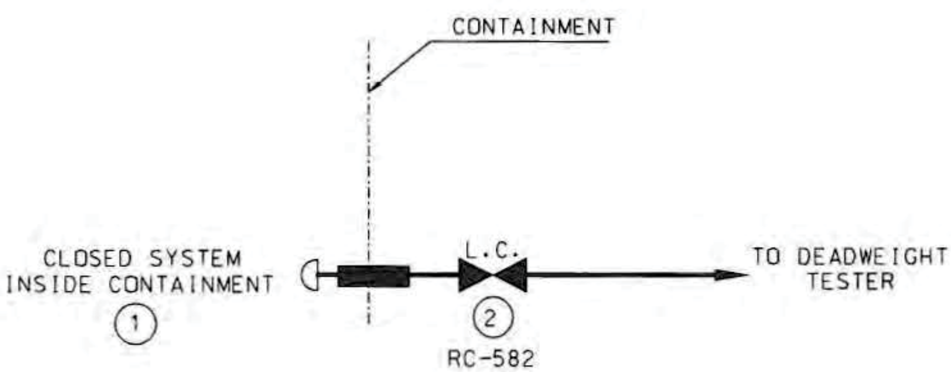


BARRIER	P-68	P-69	P-70
1	VCT-20	VCT-19	VCT-18
2.1	PAS-5	PAS-3	PAS-1
2.2	PAS-6	PAS-4	PAS-2

PENETRATION NO. 71 - PENETRATION PRESSURIZATION SYSTEM AIR SUPPLY



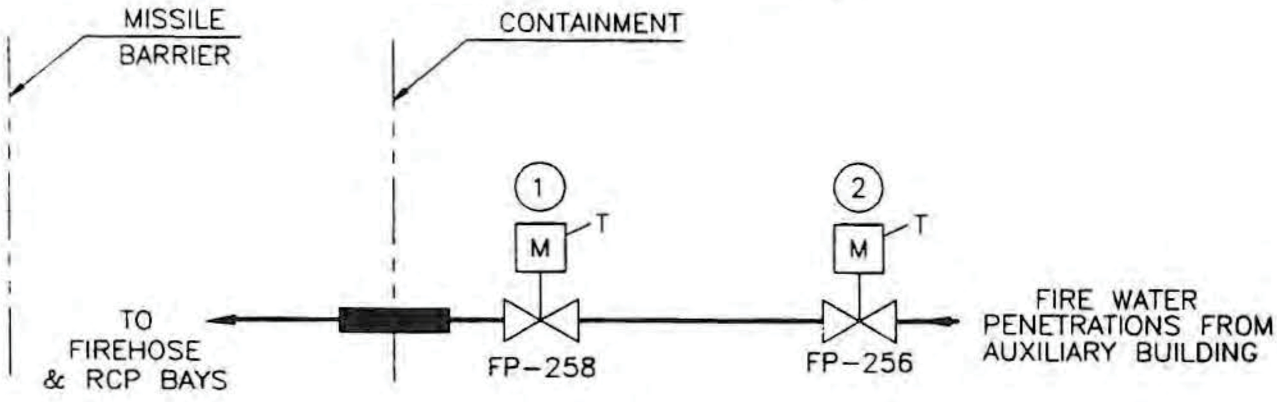
PENETRATION NO. 72 - DEADWEIGHT TESTER LINE



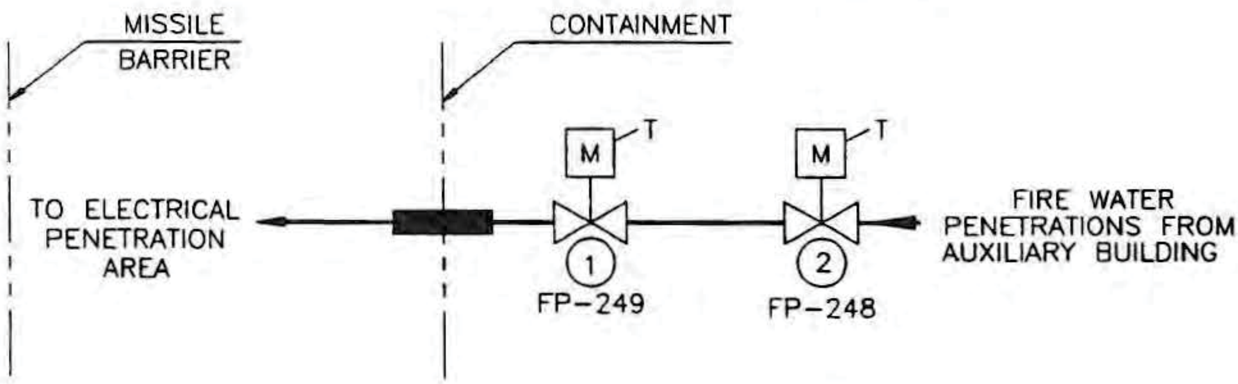
Revision No. 15

H.B. ROBINSON UNIT 2 Carolina Power & Light Company UPDATED FINAL SAFETY ANALYSIS REPORT	CONTAINMENT ISOLATION VALVES PENETRATIONS P-68, P-69, P-7Ø, P-71, P-72	FIGURE 6.2.4-18

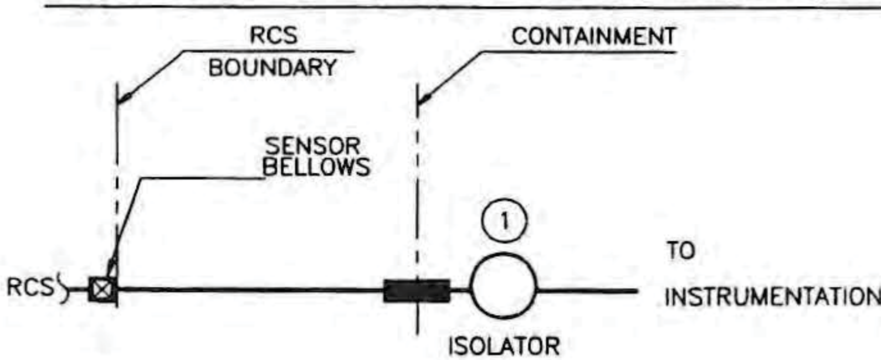
PENETRATION NO. 73 – FIRE WATER



PENETRATION NO. 74 – FIRE WATER



PENETRATIONS NO. 75, 76, 77, 78, 79, 80 – RVLIS SENSING LINE

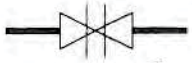
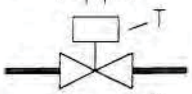
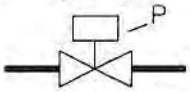
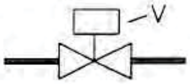


BARRIER	P–75	P–76	P–77	P–78	P–79	P–80
1	LIS511AB	LIS511AA	LIS511AC	LIS511BB	LIS511BA	LIS511BC

REVISION No. 15

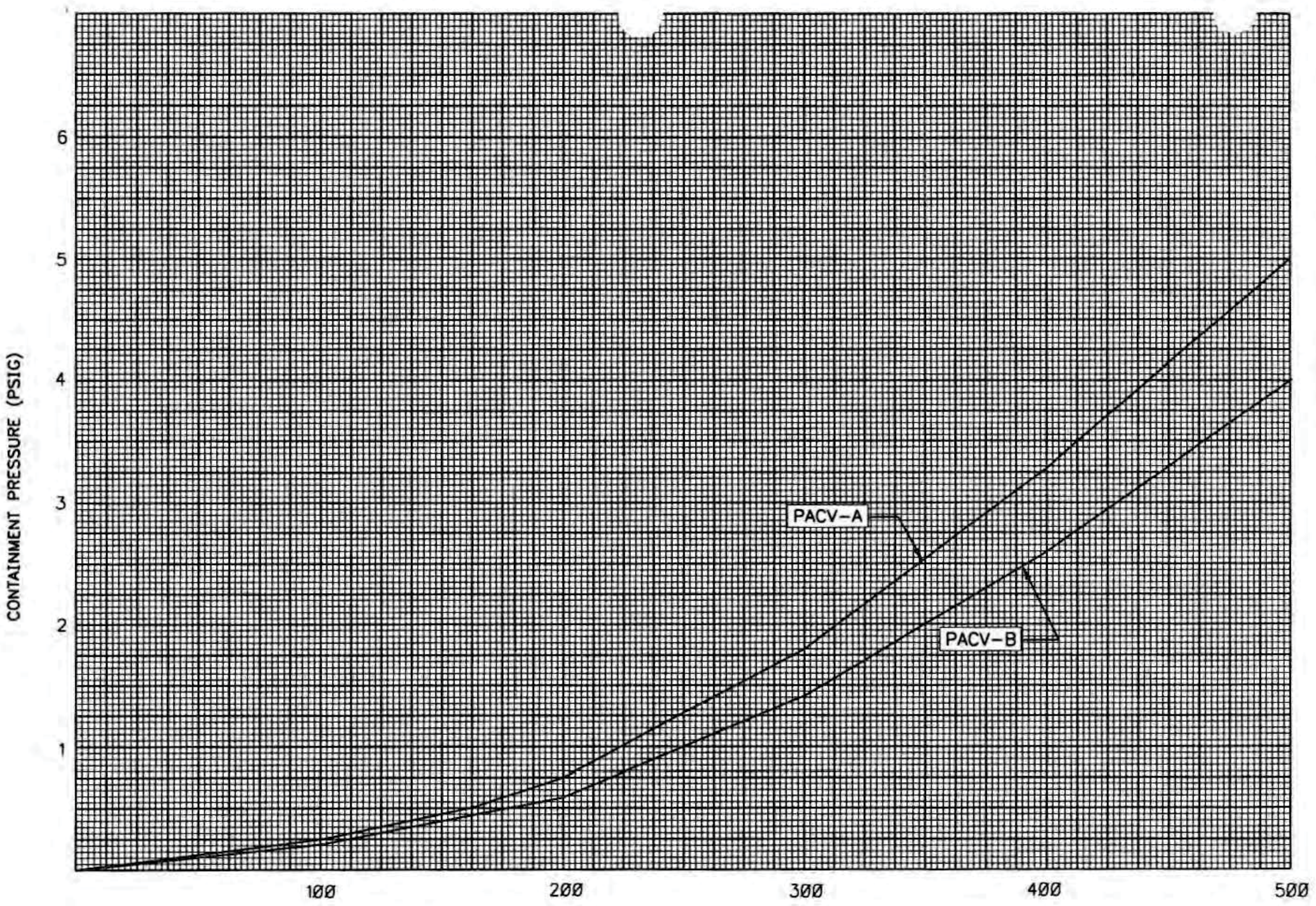
H.B.ROBINSON UNIT 2 Carolina Power & Light Company UPDATED FINAL SAFETY ANALYSIS REPORT	CONTAINMENT ISOLATION VALVES PENETRATIONS P-73, P-74, P-75, P-76, P-77, P-78, P-79, P-80	FIGURE 6.2.4-19

Figure 6.2.4-20 was deleted by Amendment No. 12

	INDICATES NORMALLY CLOSED (N.C.) VALVE
	INDICATES NORMALLY OPEN (N.O.) VALVE
	DOUBLE DISC GATE VALVE
	INDICATES VALVE CLOSES ON T SIGNAL (CONTAINMENT ISOLATION SIGNAL, PHASE A. DERIVED FROM SAFETY INJECTION OR MANUALLY.)
	INDICATES VALVE CLOSES ON P SIGNAL (CONTAINMENT ISOLATION SIGNAL, PHASE B. HIGH HIGH CONTAINMENT PRESSURE)
	INDICATES VALVE CLOSES ON V SIGNAL (CONTAINMENT VENTILATION ISOLATION SIGNAL. DERIVED FROM SAFETY INJECTION, CONTAINMENT HIGH RADIATION OR MANUALLY.)
	CONTAINMENT PENETRATION
	SCREWED CAP
	WELDED CAP
	SCREWED CAP W/THREADED PLUG
	BLIND FLANGE
F	FILTER
1 2	CONTAINMENT BARRIERS
F.C.	VALVE FAILS CLOSED
F.O	VALVE FAILS OPEN
L.C.	VALVE IS LOCKED CLOSED
L.O.	VALVE IS LOCKED OPEN
FAI	VALVE FAILS AS IS
C.S.	CLOSED SYSTEM
N/A	NOT APPLICABLE
IVSW	ISOLATION VALVE SEAL WATER
AUTO.	AUTOMATIC IVSW ACTUATION
MAN.	MANUAL IVSW ACTUATION
V	VENT VALVE
D	DRAIN VALVE

Revision No. 14

H.B. ROBINSON UNIT 2 Carolina Power & Light Company UPDATED FINAL SAFETY ANALYSIS REPORT	CONTAINMENT ISOLATION VALVES LEGEND FOR SYMBOLS USED ON FIGURES 6.4.2-1 THROUGH 6.4.2-19	FIGURE 6.2.4-21



REVISION NO. 15

H.B. ROBINSON UNIT 2 Carolina Power & Light Company UPDATED FINAL SAFETY ANALYSIS REPORT	POST-ACCIDENT VENTING SYSTEM SYSTEM RESISTANCE CURVE	FIGURE 6.2.5-2

6.3 EMERGENCY CORE COOLING SYSTEM

6.3.1 DESIGN BASIS

6.3.1.1 Summary Description

Adequate emergency core cooling is provided by the Safety Injection System (SIS) [which constitutes the Emergency Core Cooling System (ECCS)], whose components operate in three modes. These modes are delineated as passive accumulator injection, active safety injection (SI), and residual heat removal recirculation.

The primary purpose of the SIS is to automatically deliver cooling water to the reactor core in the event of a loss-of-coolant accident (LOCA). This limits the fuel cladding temperature and thereby ensures that the core will remain intact and in place, with its heat transfer geometry preserved. This protection is afforded for:

a) All pipe break sizes up to and including the hypothetical instantaneous circumferential rupture of a reactor coolant loop, assuming unobstructed discharge from both ends.

b) A loss of coolant associated with the rod ejection accident.

c) A steam generator (SG) tube rupture.

The principal components of the SIS which provide emergency core cooling immediately following a loss of coolant are the accumulators (one for each loop), the two SI (high head) pumps with Pump B to act as a maintenance replacement for Pumps A and C, and the two residual heat removal (low head) pumps.

The SIS operates in the following possible modes:

a) Injection of borated water by the passive accumulators.

b) Injection of borated water from the refueling water storage tank by the SI pumps.

c) Injection by the residual heat removal pumps, which also draw borated water from the refueling water storage tank.

d) Recirculation of spilled coolant, injected water, and Containment Spray System (CSS) drainage back to the reactor from the containment sump by the residual heat removal pumps.

The initiation signal for core cooling by the SI pumps and the residual heat removal pumps in the SIS is actuated by any of the following:

a) Low pressurizer pressure (2/3)

b) High containment pressure (2/3, Hi level-approximately 10 percent of containment design pressure)

c) High steam line differential pressure (2/3 per line in 1/3 lines)

d) High steam flow (1/2 per line in 2/3 lines) with low T_{avg} (2/3 loops) or low steam line pressure (2/3 lines), and

e) Manual Actuation (1/2 pushbuttons).

Automatic initiation of SI due to pressurizer low pressure and high steam line differential pressure may be manually blocked when the plant is below 2000 psi. Initiation due to high steam line flow coincident with low steam line pressure or low T_{avg} can be blocked when T_{avg} is below 543ºF.

6.3.1.2 Design Basis for Functional Requirements

The ECCS complies with the functional criteria for ECCS derived from 10CFR50, Appendix K, as delineated in 10CFR50.46. The conditions relating to peak cladding temperature, maximum cladding oxidation, maximum hydrogen generation, coolable geometry, and long-term cooling are all met with adequate margin relative to the specified limits.

6.3.1.3 Design Basis for Reliability

For any rupture of a steam pipe and the associated uncontrolled heat removal from the core, the SIS adds shutdown reactivity so that, with a stuck rod, no offsite power, and minimum engineered safety features, there is no consequential damage to the Reactor Coolant System (RCS) and the core remains in place and intact.

Redundancy and segregation of instrumentation and components are incorporated in the design to assure that postulated malfunctions will not impair the ability of the system to meet the design objectives. The system is effective in the event of loss of normal plant auxiliary power coincident with the loss of coolant, and can accommodate the failure of any single component or instrument channel to respond actively in the system. During the recirculation phase of a LOCA, the system can accommodate a loss of any part of the flow path, since backup alternative flow path, capability is provided.

6.3.1.4 ECCS Protection from Physical Damage

Pipe whip protection for ECCS components is provided in accordance with General Design Criteria 40 and 42 (Section 3.6).

Protection of ECCS components against seismic loads is discussed in Section 3.2 and 3.7.

Protection of ECCS components against missiles is discussed in Section 3.5.

Protection is provided for ECCS components against loads which may result from the effects of a LOCA.

The accumulators, which are passive components, discharge into the cold legs of the reactor coolant piping when RCS pressure decreases to 660 psig, thus assuring rapid core cooling for large pipe breaks. They are located inside the containment, but outside the crane wall. Therefore each accumulator is protected against possible missiles.

6.3.1.5 ECCS Environmental Design Basis

The ability of the ECCS to operate in the harsh environmental conditions that may exist during operation of the ECCS is discussed in Section 3.11.

6.3.2 SYSTEM DESIGN

6.3.2.1 Schematic Piping and Instrument Diagrams

The SIS flow diagrams are shown in Figures 6.3.2-1 and 6.3.2-2. The initiating systems for SIS are discussed in Section 7.3.

6.3.2.2 Equipment and Component Descriptions

6.3.2.2.1 Injection Phase

The SI signal opens the SIS isolation valves and starts two SI pumps and the residual heat removal (RHR) pumps. The items on Figures 6.3.2-1 and 6.3.2-2 marked with an "S" receive the SI signal. The high head SI pumps take suction from the refueling water storage tank (RWST).

The RHR pumps deliver to all three cold legs through the piping between the accumulators and the cold legs. The high head SI pumps deliver automatically into a header connected to the cold legs and one connected to the hot legs. The header to the cold legs contains the BIT. Downstream of the BIT, the header divides into three injection lines connecting to the pipes from the accumulators close to the RCS cold leg piping. The capability is provided to manually isolate the pumps on separate headers from the reactor turbine generator board (RTGB), thereby ensuring the delivery of full flow from at least one pump for the special case of a broken header.

For large pipe breaks, the RCS would be depressurized and voided of coolant rapidly (about 25 sec for the largest break). A high flow rate is required to quickly recover the exposed fuel rods and limit possible core damage. To achieve this objective, one RHR pump (high flow, low head) and one SI pump (high head, low flow) are required to deliver borated water to the cold legs of the reactor coolant loops. For the slower and less extensive depressurization of the RCS resulting from a small break LOCA, initial recovery depends on the high head SI delivery. In order to provide for a single active failure, two SI pumps are started automatically. Two trains are available in order to provide for an active component failure. Delivery from these pumps supplements the accumulator discharge.

Because the injection phase of the accident is terminated before the RWST is completely emptied, all pipes are kept filled with water before recirculation is initiated. Water level indication and alarms on the RWST give the operator ample warning to terminate the injection phase. Additional level indicators are provided for the containment sump, which also give backup indication when injection can be terminated and recirculation initiated.

For small pipe breaks, the depressurization of the RCS by the SIS can be augmented by dumping steam to the atmosphere or the condenser, and addition of auxiliary feedwater to the SG. Use of steam dump is not required to meet the core cooling objectives. It is intended that for small breaks (4 in. and smaller), steam dump will be employed to facilitate the recovery from the accident, and to reduce the reactor coolant system pressure to the cut-in pressure of the RHR pumps.

The main steam isolation valves do not get a containment isolation signal. However, they do close automatically on a steam line break.

Since leakage between the RCS and the SG during operation is possible, careful consideration is given to the effect of any possible radioactive leakage into the SG. Manual steam dump to the atmosphere will not be initiated unless it can be assured that there has been no measured contamination of the SG as a result of the LOCA.

Breaks large enough to release fission products from the core are characterized by a rapid depressurization of the RCS and uncovering of the core, followed by an increase in fuel clad temperature causing the cladding to burst. For these breaks, the reactor coolant pressure would fall below that of the SG before the SG is pressurized to the SG safety valves' setpoint. There would be no leakage of radioactivity to the atmosphere.

Before initiating any cooldown of the SG either by atmospheric steam dump or steam dump to the condenser, the operator would check the activity in the SG. The operator would open the blowdown sample lines one at a time from the control room and observe the readings on the radiation monitor or have the Environmental and Chemistry Technician obtain a sample using the Primary Sample System. If the readings showed an increase over the normal operating level, steam dump would not be permitted from the ruptured SG (unless all SGs are ruptured) and the SG would remain isolated for the duration of the accident.

When steam dump cooldown is used for small breaks (4 in. and smaller), the steam will be dumped to the condenser when outside power is available, or directly to the atmosphere when outside power is not available. The expected peak fuel clad temperatures for break sizes 4 in. and smaller are limited to a value below which cladding bursting is expected. When steam dump is initiated, the only activity that could be leaked into the steam would be dumped to the condenser if outside power is available. In that case, the air ejector radiation monitor would provide additional information that activity carryover to the secondary side had not occurred as a result of the accident.

6.3.2.2.2 Recirculation Phase

After the injection phase of SIS operation, coolant spilled from the break and water collected from the containment spray is cooled and returned to the RCS by the recirculation system.

When the break is large, RCS depressurization occurs due to the large rate of mass and energy loss through the break to the containment. The system is arranged so that the RHR pumps take suction from the sump in the containment floor and deliver spilled reactor coolant and borated refueling water back to the core through the residual heat exchangers. The system is arranged to allow either of the RHR pumps to take over the recirculation function.

There are two sump return lines which lead from the containment to the RHR pumps. Each line is located inside of a larger diameter guard pipe. The lines are separated by approximately 18 ft. The lines are designed to allow for 2 in. differential movement between the containment and pump chamber and are designed as Seismic Class I equipment.

Debris is removed from water entering the RHR pump suction piping by the containment sump strainer. The containment sump strainer consist of multiple high-performance top hat style assemblies (Fig. 6.3.2-3) which provide a net effective surface area in excess of 4,100 ft^2. The strainer assembly inside the crane wall is physically located under the refueling canal. The top hat modules are bolted, horizontally, to both sides of a 15-inch square plenum (manifold) box. The plenum is supported approximately one inch above the nominal containment floor elevation of 228.0 ft. In this design, water enters through the perforated plate surfaces of the strainers. The perforated plate surface consists of 3/32" diameter holes, spaced to give a free surface area of 32.7%. The flow then travels through the plenum to a sump box covering the RHR suction nozzles (Fig.6.3.2 3). The top-hats include a "Bypass-Eliminator" which minimizes the amount of fibrous debris that passes through the strainer. The "Bypass-Eliminator" minimizes the impact on down stream components.

The adequacy of the surface area is determined by the pressure drop across the strainer assembly following a worst case accident scenario (with worst case debris generation), which is a Large Break LOCA.

Values for the worst case debris generation scenario and the subsequent transport of this debris to the ECCS containment sump strainer are utilized to determine the head loss across the strainer surface area at worst case flows. The head loss across the debris bed on the strainer is combined with the clean strainer head loss. The clean strainer head loss (CSHL) is the head loss calculated to occur as water flows from the top hats, through the sump box to the RHR (LHSI) pump suction inlets.

In addition to the strainer surface area, another design consideration for the strainer is the interstitial volume. The head loss for the sump strainer design is calculated utilizing an even distribution of debris material on the strainer. This approach is that the interstitial volume is greater than the volume of debris that will reach the sump strainer. For the horizontal top hats, the interstitial volume is equal to the open volume between the sump strainer top hat modules and the volume between the perforated plate tubes that can become filled with debris. The ECCS containment sump strainer design has an interstitial volume of approximately 530 ft^3. The total volume of debris that could reach the strainer and fill the interstitial volume is less than 530 ft^3. The interstitial volume is greater than the volume of the debris that will reach the sump strainer, so not enough debris will exist to fill in the open volume within the sump strainer modules or between the sump strainer modules. Therefore, the strainer is adequately designed to

accommodate the predicted debris generation.

Recirculation may start with a water depth of 1.5 ft on the containment floor. This is equivalent to the amount of water in the primary systems plus 60 percent of the RWST contents, or approximately 215,000 gal of water at 263 F.

6.3.2.2.3 Net Positive Suction Head (NPSH) Requirements

During the safety injection phase the worst case conditions for determining NPSH requirements occur with the single failure of a high head pump resulting in the following:

1. 1 high head pump at approximately 650 gpm,
2. 2 low head pumps at approximately 4400 gpm total, and
3. 2 containment spray pumps at approximately 2500 gpm total.

A quantitative analysis of the available and required NPSH for the SI, RHR and containment spray pumps for both the safety injection phase (with suction from the RWST) and the recirculation phase (suction from the containment sump) shows:

1. During the safety injection phase with suction from the RWST, (at the RWST low level setpoint), operating as described above, the following applies:

	NPSH, ft	
Pump	Required	Available
High head, 1 pump (most limiting)	31	Approx 34
Low head (RHR), 2 pumps	10	Approx 54
Containment spray, 2 pumps	20	Approx 35

From this it can be seen that the high head pump is the controlling component for NPSH. The safety injection phase will be terminated just before the RWST level decreases to the point at which the available NPSH is reduced to the required NPSH at the runout flow. Transition to recirculation from the containment sump will commence at this point.

2. During the transition to the recirculation phase, conditions are such that one high head pump and one containment spray pump are operating. During this period the worst case NPSH conditions occur at the RWST low-low level setpoint as follows:

	NPSH, ft	
Pump	Required	Available
1 high head pump	31	Approx 32
1 containment spray pump	20	Approx 34

3. During the recirculation phase (from containment sump) the following applies:

a. High head SI pumps - During recirculation via the high head pump, this pump and the RHR pump would be aligned in series, with the RHR pump (which has a design discharge head of 240 ft) boosting the suction of the high head pump. Thus, no NPSH problems would be experienced.

b. Containment spray pump - Same as high head SI pump.

c. RHR (low head) pump - During recirculation from the containment sump the minimum available NPSH with 1.5 ft of water on the containment floor is more than the required NPSH.

The high head recirculation flow path via the high head SI pumps is only required for the range of small break sizes for which the RCS pressure remains in excess of the shut-off head of the RHR pumps at the end of the safety injection phase.

Those portions of the SIS located outside of the containment that are designed to circulate post-accident containment sump water must meet leakage rate limits to ensure LOCA dose acceptance criteria are met. Additionally, pressure relieving devices discharge into closed systems or areas monitored for radwaste leakage by radiation monitoring instrumentation with reliable power sources.

Recirculation loop leakage is discussed in Section 6.3.2.5.5.

For the recirculation phase of the accident, the reactor coolant water which eventually is located on the containment floor is recirculated through the sump line from the containment to the suction of the RHR pump. Two independent and redundant recirculation lines are provided. Each line has two motor-operated valves. Both valves are located adjacent to the containment penetration in the RHR pit such that the line outside the containment can be isolated in the event of a passive failure. During recirculation, one recirculation train, which includes either of the two RHR pumps and either of the two residual heat exchangers, will be in service. The flow will go from the discharge of the RHR pump through the residual heat exchanger and then into the reactor via either the low head injection path or the high head injection path via the SI pumps. The high head injection path is provided in the event of a small break in which the pressure in the RCS is higher than the shut-off head of the RHR pumps.

In the event of a failure in the operating train during recirculation, the capability exists to switch to the other independent recirculation flow path; i.e., through the high head SI pumps to provide core cooling.

In the long term (post-accident) phase, injection through a separate header into the hot legs is possible by manual remote Control Room switch operation.

6.3.2.2.4 Cooling water

6.3.2.2.4.1 Component cooling system

During the recirculation mode, the Component Cooling System is used to cool the recirculation fluid as it passes through the residual heat exchanger. One of the three component cooling pumps and one of the two component cooling heat exchangers provide the cooling function during recirculation.

6.3.2.2.4.2 Service water system

The service water system is provided with redundant and independent loop headers and valves such that the two component cooling heat exchangers which are supplied with service water for cooling can have flow directed to them from the two independent headers. Two of the four service water pumps are required to operate during the recirculation phase.

6.3.2.2.5 Changeover from injection phase to recirculation phase

The sequence, from the time of the SI signal, for the changeover from the injection to the recirculation is as follows:

1. First, sufficient water is delivered into the containment during the injection phase to provide the required NPSH of the RHR pumps to allow the change to recirculation.

2. Second, the first low level alarm on the RWST sounds. At this point, the operator takes appropriate action to assure that sufficient NPSH exists for the operating pumps to run until the RWST is nearly empty. Between the first RWST low level alarm and the second low level alarm the operator performs the system alignment for recirculation. The change-over from injection to recirculation is effected by the operator

in the Control Room via a series of manual switching operations according to written procedures. Valves SI-856A and SI-856B are manually blocked closed at the valves. Valve SI-870A or SI-870B is manually closed if the motor operator is inoperable. At the first alarm the operator stops pumps to achieve the following configuration.

RHR pumps	-None running
SI pumps	-One running
Spray pumps	-One running (if required)
Charging pumps	-None running

The suction of the RHR pumps is aligned to the containment sump. If the RCS pressure is below the discharge pressure of the RHR pumps, a RHR pump is restarted when there is sufficient water level in containment. RHR will recirculate flow from the containment sump to the RCS.

3. Finally the second low level alarm on the RWST sounds. At this time, the operator will stop the remaining pumps taking a suction from the RWST.

   If RCS pressure is greater that the discharge pressure of the RHR pumps or if Containment Spray is required for pressure control, then the RHR system is aligned for the RHR pumps to recirculate flow from the containment sump to the SI pump suction header. ECCS flow to the RCS is interrupted for a period of time while this alignment is performed.

Remotely operated valves for the injection phase of the SIS (Figures 6.3.2-1 and 6.3.2-2) which are under manual control, (this is, valves which normally are in their ready position and do not receive a SI signal) have their positions indicated on a common portion of the control board. At any time during operation, when one of these valves is not in the ready position for injection, it is shown visually on the board. Table 6.3.2-1 is a listing of the instrumentation readouts on the control board which the operator can monitor during recirculation.

6.3.2.2.5.1 Location of the major components required for recirculation

The RHR pumps are located in the RHR pump room (Elevation 203 ft 0 in.) which is below the basement floor of the Auxiliary Building (Elevation 226 ft 0 in). The RHR pump room is located between the Containment Building and the Auxiliary Building. The residual heat exchangers are located on the first floor of the Auxiliary Building.

The high head SI pumps, component cooling pumps and component cooling heat exchangers are located in the Auxiliary Building (Elevation 226 ft 0 in).

The service water pumps are located in the intake structure, and the redundant piping to the component cooling heat exchangers is run underground.

6.3.2.2.6 Accumulators

The accumulators are pressure vessels filled with borated water and pressurized with nitrogen gas. During normal plant operation, each accumulator is isolated from the RCS by two check valves in series.

Should the RCS pressure fall below the accumulator pressure, the check valves open and borated water is forced into the RCS. Mechanical operation of the swing-disc 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 (ESF) because the gas forces 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 RCS.

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. The flow from the remaining accumulators provides sufficient water to fill the volume outside of the core barrel below the nozzles, the bottom plenum, and one-half the core.

The accumulators are carbon steel, clad with stainless steel and designed to American Society of Mechanical Engineers (ASME) Section III, Class C. Connections are provided for remotely draining or filling the fluid space during normal plant operation.

The minimum boron concentration of 1950 ppm during refueling, together with the control rods, maintains ≥ 6% Δk/k shutdown margin in the core for these operations. The boron concentration is also sufficient to maintain the core in a shutdown condition without any rod cluster control (RCC) rods during refueling. For cold shutdown, at the beginning of core life, a lower concentration is sufficient for one percent shutdown with all but one stuck rod inserted. The boron concentration for refueling is well within solubility limits at ambient temperature.

The minimum boron concentration required in the accumulators is 1950 ppm as specified in the HBR 2 Technical Specifications, Section 3.5. Thus the boron concentration in the accumulators is more than adequate to maintain the core subcritical following a LOCA.

The level of borated water in each accumulator tank is adjusted remotely as required during normal plant operations. Refueling water is added using a SI pump. Water level is reduced by draining to the reactor coolant drain tank. Samples of the solution in the tanks are taken at the sampling station for periodic checks of boron concentration. Redundant level and pressure indicators are provided with readouts on the control board. Each indicator is equipped with high and low level alarms.

The accumulator design parameters are given in Table 6.3.2-2.

6.3.2.2.7 Boron Injection Tank (BIT)

The tank is vertical with the outlet nozzle on top. A level alarm is provided from a stand pipe/vent arrangement on the outlet pipe at an elevation higher than the top of the tank. This alarm assures that the tank is maintained full of solution at all times.

Design parameters are given in Table 6.3.2-3.

6.3.2.2.8 Refueling water storage tank

In addition to its usual duty of supplying borated water to the refueling canal for refueling operations, this tank provides borated water to the SI pumps, the RHR pumps, and the containment spray pumps for mitigation of a LOCA. During plant operation, it is aligned to the suction of the pumps. It is constructed of stainless steel.

The capacity of the RWST is based on the requirement for filling the refueling canal, with a minimum of 300,000 gal being available for delivery. This capacity provides an amount of borated water to assure:

1. A volume sufficient to refill the reactor vessel above the nozzles

2. The volume of borated refueling water needed to increase the concentration of initially spilled reactor coolant to a point that assures no return to criticality with the reactor at cold shutdown and all control rods except the most reactive RCC assembly inserted into the core.

3. A sufficient volume of water on the containment floor to permit the initiation of recirculation during a LOCA.

The water in the tank is borated to a concentration which assures reactor shutdown margin of $\geq$ 6% Δk/k when all RCC assemblies are inserted and when the reactor is cooled down for refueling. The maximum boric acid concentration in the tank is approximately 1.4 weight percent boric acid. At 32ºF, the solubility limit of boric acid is 2.2 percent. Therefore, the concentration of boric acid in the RWST is well below the solubility limit at 32ºF.

The RWST is thermally insulated and provided with a temperature monitoring system capable of measuring and displaying the RWST fluid average temperature.

Two level indications with low level alarms are provided

A dynamic response analysis similar to that performed for the Containment Structure has been performed to determine the horizontal loads to be applied to the RWST for the hypothetical earthquake. Vertical seismic loads equal to 0.133g have been applied simultaneously. Wave generation in the tank has been taken into account. A membrane stress analysis of the vertical cylindrical tank was performed considering the discontinuities at the base and top.

The allowable stress criteria are 95 percent of yield for tension, 90 percent for compression and shear.

The RWST design parameters are given in Table 6.3.2-4.

6.3.2.2.9 Safety injection pumps

The three high head SI pumps for supplying borated water to the RCS are horizontal, centrifugal pumps driven by electric motors. Parts of the pumps in contact with borated water are stainless steel or equivalent corrosion resistant material. A minimum flow bypass line is provided on each pump discharge to recirculate flow to the

RWST in the event the pumps are started with the normal flow paths blocked. The design parameters are presented in Table 6.3.2-5, and Figures 6.3.2-4A, 6.3.2-4B, and 6.3.2-4C gives the performance characteristics of the high head SI pumps. The LOCA delivery data is based on the composite minimum pump performance data degraded by 5%.

The two RHR (low head) pumps of the Auxiliary Coolant System are used to inject borated water at low pressure into the RCS. They are also used to recirculate fluid from the containment sump back to the RCS, to the suction of the spray pumps, or to the suction of the high head SI pumps. These pumps are of the in-line, centrifugal type, driven by electric motors. Parts of the pumps which contact the borated water and sodium hydroxide solution during recirculation are stainless steel or equivalent corrosion resistant material. A minimum flow bypass line is provided on the discharge of each residual heat exchanger to recirculate cooled fluid to the suction of its RHR pump, should these pumps be started with their normal flow paths blocked. The design parameters for the RHR pumps are presented in Table 6.3.2-5, and the characteristics are shown in Figure 6.3.2-5.

The pressure-containing parts of the pumps are castings conforming to American Society for Testing and Materials (ASTM) A-351 Grade CF8 or CF8M. Stainless steel forgings were procured per ASTM A-182 Grade F304 or F316 or ASTM A-336, Class F8 or F8M, and stainless plate was constructed to ASTM A-240, Type 304 or 316. All bolting material conforms to ASTM A-193. Materials such as weld-deposited Stellite or Colmonoy are used at points of close running clearances in the pumps to prevent galling and to assure 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 checked to ensure conformance with the applicable ASTM specification. In addition, all pressure-containing parts of the pump were liquid penetrant inspected in accordance with Appendix VIII of Section VIII of the ASME Boiler and Pressure Vessel (B&PV) 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 were made for shaft deflection and clearances between stationary parts.

Where welding of pressure containing parts 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 also applies to any repair welding performed on pressure containing parts.

The pressure-containing parts of the pump were assembled and hydrostatically tested to 1.5 times the design pressure for 30 minutes.

Each pump was given a complete shop performance test in accordance with Hydraulic Institute Standards. The pumps were run at design flow and head, shut-off head, and three additional points to verify performance characteristics. Where NPSH is critical, this value was established at design flow by means of adjusting suction pressure.

Details of the component cooling and service water pumps which serve the SIS are presented in Section 9.2.

6.3.2.2.10 Heat Exchangers

The two residual heat exchangers of the Auxiliary Coolant System cool the recirculated sump water. These heat exchangers are sized for the cooldown of the RCS. Table 6.3.2-6 gives the design parameters of the heat exchangers.

The ASME Code has strict rules regarding the wall thicknesses of all pressure-containing parts, material quality assurance provisions, weld joint design, radiographic and liquid penetrant examination of materials and joints, and hydrostatic testing of the unit, as well as requiring final inspection and stamping of the vessel by an ASME Code inspector.

The design of the heat exchangers also conforms to the requirements of Tubular Exchanger Manufacturer's Association (TEMA) for Class R heat exchangers. Class R is the most rugged class of TEMA heat exchangers and is intended for units where safety and durability are required under severe service conditions. Items such as tube spacing, flange design, nozzle location, baffle thickness and spacing, and impingement plate requirements are set forth by TEMA Standards.

In addition to the above, additional design and inspection requirements were imposed to ensure rugged, high quality heat exchangers such as: confined-type gaskets, main flange studs with two nuts on each end to ensure reliable leak tightness, general construction and mounting brackets suitable for the plant seismic design requirements, tubes and tube sheet capable of withstanding full shell side pressure and temperature with atmospheric pressure on the tube side, ultrasonic inspection in accordance with Paragraph N-324.3 of Section III of the ASME Code of all tubes before bending, penetrant inspection in accordance with Paragraph N-627 of Section III of the ASME Code of all welds and all hot or cold formed parts, a hydrostatic test duration of not less than thirty minutes, the witnessing of hydro and penetrant tests by a qualified inspector, a thorough, final inspection of the unit for good workmanship of any gouge marks or other scars that could act as stress concentration points, a review of the radiographs and of the certified chemical and physical test reports for all materials used in the unit.

The residual heat exchangers are conventional vertical shell and U-tube type units. The tubes are seal welded to the tube sheet. The shell connections are flanged to facilitate shell removal for inspection and cleaning of the tube handle. Each unit has an SA-212-B carbon steel shell, an SA-212-B carbon steel shell end cap, SA-213 TP-304 stainless steel tubes, an SA-240 Type 304 stainless steel channel, an SA-240 Type 304 stainless steel channel cover and an SA-240 Type 304 stainless steel tube sheet.

6.3.2.2.11 Valves

All parts of valves used in the SIS in contact with borated water are austenitic stainless steel or equivalent corrosion-resistant material. The motor operators on the injection line isolation valves are capable of rapid operation. All valves required for initiation of SI or isolation of the system have remote position indication in the Control Room.

Valving is specified for exceptional tightness and, where possible, such as instrument valves, 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 cc/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. Control and motor-operated valves, 2 ½ in. and above, which are exposed to recirculation flow, are provided with double-packed stuffing boxes and stem leak off connections or have had their leak off line removed and various packing arrangements including live loading and standard bolting.

The check valves which isolate the SIS from the RCS are installed immediately adjacent to the reactor coolant piping to reduce the probability of an injection line rupture causing a LOCA.

Two relief valves are associated with the post loss-of-coolant recirculation. One is located outside the containment at the BIT discharge to prevent overpressure in the header and in the BIT. The high head SI piping leading to the hot legs is protected by a relief valve inside the containment in the test line.

The relieving capacity of these valves is based on a flow several times greater than the expected leakage rate through the check and isolation valves. They will also prevent overpressurization due to thermal expansion.

The SI Cold Leg Injection Lines between the SI-870 and SI-868 valves are protected from overpressurization by a relief valve (SI-857B) located downstream of SI-868B. The relieving capacity of this valve is greater than the expected check valve leakage from the RCS. The relief valve discharges to the pressurizer relief tank.

The RHR loop is protected by a relief valve in the common header leading to the accumulator pipes. The valve is located inside the containment and is relieved to the pressurizer relief tank. Apart from relieving possible leakage from the RCS, the valve is sized to relieve flow from one charging pump.

The gas relief valves on the accumulator protect them from pressures in excess of the design value.

6.3.2.2.12 Motor-operated valves

The pressure-containing parts (body, bonnet, and discs) of the valves employed in the SIS are designed per criteria established by the ANSI B.16.34, USAS B16.5 or MSS SP66 specifications. The materials of construction for these parts are procured to applicable ASME or ASTM specifications. All material in contact with the primary fluid, except the packing, is austenitic stainless steel or equivalent corrosion resisting material. The pressure-containing cast components were radiographically inspected as outlined in ASTM E-71, Class 1 or Class 2. The body, bonnet, and discs were liquid penetrant inspected in accordance with the ASME Code, Section VIII, Appendix VIII with acceptance standards as outlined in USAS B31.1 Case N-10, or equivalent.

When a gasket is employed, the body-to-bonnet joint was designed to meet or exceed ASME B&PV Code Section VIII or USAS B16.5 with a fully trapped, controlled compression, spiral wound, asbestos or graphite-filled gasket with provisions for seal welding, or of the pressure seal design with provisions for seal welding. RHR-759A and B were evaluated to use Flexpro style gaskets. The body-to-bonnet bolting and nut materials are procured per ASTM A193 and A194 or equivalent, 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 per inch of wall thickness or in accordance with ASME Section III. Any leakage was cause for rejection. The seating design for gate valves is of the Darling parallel disc 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 unbalance on the disc and the packing box friction. The discs are guided throughout the full disc 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. Nickel-chrome-boron may be used as an alternate hard-surfacing material.

The stem material is ASTM A276 Type 316 condition B, or precipitation hardened 17-4 PH stainless procured and heat treated to Westinghouse or DEP approved Specifications. These materials were selected because of their corrosion resistance, high tensile properties, and their resistance to surface scoring by the packing. With the exception of valves which have had their leak off lines removed and various packing arrangements including live loading and standard bolting, the valve stuffing box is designed with a lantern ring leak-off connection.

The motor operator is extremely rugged and is noted throughout the power industry for its reliability. The unit incorporates a "hammer blow" feature that allows the motor to impact the discs away from the fore or backseat upon opening or closing. This "hammer blow" feature not only impacts the disc but allows the motor to attain its operational speed.

The valve was assembled, hydrostatically tested, seat-leakage tested (fore and back), operationally tested, cleaned, and packaged per specifications. During original construction, all manufacturing procedures employed by the valve supplier such as hard facing, welding, repair welding and testing were submitted to Westinghouse for approval.

For those valves which must function on the SI signal, operators are provided to support valve operations that are consistent with stroke times necessary to support their safety function.

Valves which must function against system pressure were designed such that they function with differential pressure postulated to occur under design basis conditions including normal and accident.

RHR-750 and RHR-751 are interlocked with SI-862 A & B and SI-863 A & B. RHR-750 and RHR-751 will open from the RTGB only if SI-862 A & B and SI-863 A & B are closed, their breakers are closed with power available, and the Normal/Defeat switches in the back of the RTGB are in Normal. If SI-862A/B or SI-863A/B are de-energized, RHR-750/751 will detect these valves as being open so RHR 750/751 will not open. There is no interlock to close RHR-750/751, so under these conditions, the valves will close, but not reopen. This interlock will help avoid loss of RCS inventory to the RWST and/or over-pressurizing the low pressure portions of Safety Injection system.

6.3.2.2.13 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 meet or exceed with USAS B16.5. The materials of construction of the body, bonnet, and disc conformed to the requirements of ASTM A105 Grade II, A181 Grade II, or A216 Grade WCB or WCC or equivalent. The carbon steel valves pass only non-radioactive fluids and were subjected to hydrostatic test as outlined in MSS SP-61, except that the test pressure was maintained for at least 30 minutes per inch of wall thickness or in accordance with ASME Section III. Since the fluid controlled by the carbon steel valves is not radioactive, the double packing and seal weld provisions included in the stainless steel valve design were not provided.

6.3.2.2.14 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 Westinghouse Atomic Power Division (WAPD) specifications. The cast pressure-containing parts were radiographed in accordance with ASTM E-94 and the acceptance standard 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 the 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 minutes. The seat leakage test was conducted in accordance with the manner prescribed in MSS SP-61, except that the acceptable leakage was 2 cc/hr/in. nominal pipe diameter.

The valve was designed with a low presure drop configuration with all operating parts contained 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 or nickel-chrome-boron materials. The various working parts were selected for their corrosion resistant, tensile, and bearing properties.

The disc and seat rings are manufactured from a forging. The mating surfaces are hard faced with Stellite No. 6 or nickel-chrome-boron to improve the valve seating life. The disc is permitted to rotate, providing a new seating surface after each valve opening.

The valves are intended to be operated in the closed position, with a normal differential pressure across the disc of approximately 1550 psi. The valves remain in this position except for testing and SI. Since the valve is not required to normally operate in the open condition, it will not be subjected to impact loads caused by sudden flow reversal, and it is expected that this equipment will not have difficulties performing its required functions.

When the valve is required to function, a differential pressure of less than 25 psig will shear any particles that may attempt to prevent the valve from functioning. Although the working parts are exposed to the boric acid solution contained within the reactor coolant system, a boric acid "freeze up" is not expected with this low a concentration.

The experience derived from the check valves employed in the Emergency Injection System of the Carolina - Virginia Tube Reactor (CVTR) in a similar system indicates that the system is reliable and workable. The CVTR Emergency Injection System, normally maintained at containment ambient conditions, was separated from the main coolant piping by a single six inch check valve. A leak detector was provided at a proper elevation to accumulate any leakage coming back through the check valve. A level alarm provided a signal on excessive leakage. The pressure differential was 1500 psi and the system was

stagnant. The valve was located 2 to 3 ft from the main coolant piping, which resulted in some heatup and cooldown cycling. The CVTR went critical late in 1963 and operated until 1967. During that time, the level sensor in the detection never alarmed due to check valve leakage.

6.3.2.2.15 Relief Valves

The accumulator relief valves are sized to pass nitrogen gas at a rate in excess of the accumulator gas fill line delivery rate. The relief valves will also pass water in excess of the expected leak rate, but this is not necessary because the time required to fill the gas space gives the operator ample opportunity to correct the situation. For an inleakage rate 15 times the manufacturing test rate, there will be about 1000 days before water will reach the relief valves. Prior to this, level and pressure alarms would have been actuated.

The SI test line relief valve is provided to relieve any pressure above design that might build up in the high head SI piping. The valve will pass 50 gpm, which is far in excess of the manufacturing design leak rate of 24 cc/hr.

6.3.2.2.16 Leakage Limitations

Valving was specified for exceptional tightness. Small, normally open valves have backseats which can limit leakage to less than one cubic centimeter per hour per inch 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.

Valves that perform a modulating function are equipped with two sets of packing and standard bolting, or are configured with a live load packing arrangement. The valves include an intermediate leak off connection or have had their leak off lines removed and capped.

6.3.2.2.17 Piping

All SIS piping in contact with borated water is austenitic stainless steel. Piping joints are welded, except for the flanged connections at the SI and containment spray pumps. The leak off lines for RHR-759A, 759B, 757A, 757B, 757C, and 757D are capped via a threaded joint.

The piping beyond the accumulator stop valves was designed for RCS conditions (2485 psig, 650ºF). All other piping connected to the accumulator tanks was designed for 900 psig and 650ºF.

The SI pump suction piping (150 psig at 300ºF) from the RWST was designed for low pressure losses to meet NPSH requirements of the pumps.

The SI high pressure branch lines (1750 psig at 300ºF) to the hot legs were designed for high pressure losses to limit the flow rate out of a potential rupture of a branch line at the connection to the reactor coolant loop.

The piping was designed to meet the minimum requirements set forth in the USAS B31.1 Code for Pressure Piping, Nuclear Code Case N-7, USAS B36.10 and B36.19, ASTM Standards, and supplementary standards plus additional quality control measures.

Minimum wall thicknesses were determined by the USAS Code formula in Power Piping, Section 1, USAS Code for Pressure Piping. This minimum thickness was increased to account for the manufacturer's permissible tolerance of minus 12-1/2 percent on the nominal wall. 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, and manufacturing tolerance.

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. Piping is supported to accommodate expansion due to temperature changes during the accident.

Pipe and fitting materials were procured in conformance with all requirements of the applicable ASTM and USAS specifications. All materials were verified for conformance to specification 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 conformed to ASTM A376 and met the supplementary requirements S6 ultrasonic testing (UT).

3. Fittings 2 1/2 inches and above conformed to the requirements of ASTM A403. Fittings 3 inches and above had requirements for UT inspections similar to S6 of ASTM A376. The 6 inch diameter end caps used in fabricating strainers for the 3/4 inches diameter piping branching off of the 3 inch discharge lines of the safety injection pumps are an exception.

Shop fabrication of piping subassemblies was performed by reputable suppliers in accordance with specifications which defined and governed material procurement, detailed design, shop fabrication, cleaning, inspection, identification, packaging and shipment.

Welds for pipes sized 2-½ in. and larger are butt welded. Reducing tees are used where the branch size exceeds ½ of the header size. Branch connections of sizes that are equal to or less than ½ of the header size are of a design that conforms 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. For new piping installations, it is acceptable to use reinforced branch connections exceeding ½ of the header size, as long as the design conforms to the requirements of ANSI/USAS B31.1.

All welding was performed by welders and welding procedures qualified 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 were required to have prior approval.

All high pressure piping butt welds containing radioactive fluid at greater than 600ºF temperature and 600 psig pressure or equivalent were radiographed. The remaining piping butt welds were randomly radiographed. The technique and acceptance standards were those outlined in UW-51 of the ASME Code, Section VIII. In addition, butt welds were liquid penetrant examined in accordance with the procedure of the 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 permits, on the inside root surfaces.

A post-bending 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 clean-up 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.

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.

6.3.2.2.18 Pump and Valve Motors

6.3.2.2.18.1 Motors Outside the Containment

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.

Although the motors which were provided only to drive ESF equipment are normally run only for test, the design loading and temperature rise limits were based on accident conditions. Normal design margins were specified for these motors to make sure the expected lifetime included allowance for the occurrence of accident conditions.

Criteria for motors of the SIS required that under any anticipated mode of operation, the motor name plate rating not be exceeded. Design and test criteria ensured that motor loading does not exceed the application criteria.

6.3.2.2.18.2 Motors Inside the Containment

The motor operators for the valves inside containment were designed to withstand containment environmental conditions following a LOCA so that the valves can perform their required function during the recovery period.

Periodic operation of the motors and testing of the insulation ensure that the motors remain in a reliable condition.

6.3.2.3 Applicable Code and Classifications

The ECCS has been designed to conform with the codes and classifications applicable at the time of construction. These are discussed in Section 3.2.

6.3.2.4 Material Specifications and Compatibility

Material specifications for each component are given in the component descriptions in Subsection 6.3.2.2.

Emergency core cooling system components are austenitic stainless steel, and hence are quite compatible with the spray solution over the full range of exposure in the post-accident regime. While this material is subject to crevice corrosion by hot concentrated caustic solution, the NaOH additive cannot enter the containment or ECCS without first being diluted and partially neutralized with boric acid to a mild solution. Corrosion tests performed with simulated spray showed negligible attack, both generally and locally, in stressed and unstressed stainless steel at containment and ECCS conditions. These tests are discussed in Reference 6.3.2-1.

6.3.2.5 System Reliability

To provide protection for large area ruptures in the RCS, the SIS must respond to rapidly reflood the core following the depressurization and core voiding that is characteristic of large area ruptures. The accumulators act passively to perform the rapid reflooding function with no dependence on the normal or emergency power sources.

Operation of this system with two of the three available accumulators delivering their contents to the reactor vessel (one accumulator spilling through the break) prevents fuel cladding melting and limits the metal-water reaction to an insignificant amount (<1 percent).

The function of the SI or RHR pumps is to complete the refill of the reactor vessel and ultimately return the core to a subcooled state. As discussed in Section 15.6.5, the flow from one SI pump and one RHR pump is sufficient to complete this refill function. Moreover, there is sufficient excess water delivered by the accumulators to tolerate a delay in starting the pumps.

Initial response of the injection systems is automatic, with appropriate allowances for delays in actuation of circuitry and active components. The active portions of the injection systems are automatically actuated by the SI signal (Section 7.3). In addition, manual actuation of the entire injection system and individual components can be accomplished from the Control Room. In analysis of system performance, delays in reaching the programmed trip points and in actuating components are conservatively established on the basis that only onsite emergency power is available.

The starting sequence of the SI, RHR pumps and the related emergency power equipment is designed so that delivery of the full rated flow is available within the times specified in Chapter 15.0 after the process parameters reach the setpoints for the injection signal.

No credit is taken in the Chapter 15.0 analysis for the partial flow which occurs before the full rated flow is reached.

For the small break LOCA analysis, an additional delay time is allowed to account for the receipt of SIS, either from low pressurizer pressure or from high containment pressure.

6.3.2.5.1 Single Failure Analysis

A qualitative single active failure analysis of the Safety Injection System is presented in Table 6.3.2-8. The analysis of the LOCA is consistent with the single failure analysis. It is based on the worst single failure (generally a pump failure) in both the SI and RHR pumping systems. The analysis shows that the failure of any single active component will not prevent fulfilling the design function. In addition, an alternative flow path is available to maintain core cooling if any part of the recirculation flow path becomes unavailable. This is evaluated in Table 6.3.2-9.

During the ECCS injection phase, the single failure is limited to a failure of an active component to complete its function, as required. During the ECCS recirculation phase, the failure definition is expanded to consider either an active failure or a passive fluid system failure without the loss of the protective function.

Reference 6.3.2-2 provides a detailed description of the H. B. Robinson Unit 2 single failure criteria.

6.3.2.5.2 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.

6.3.2.5.3 Passive Systems

The accumulators are a passive safety feature in that they can 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 two check valves.

The working parts of the check valves are exposed to fluid of relatively low boric acid concentration contained within the reactor coolant loop. Even if some unforeseen deposition accumulated, calculations have shown that a differential pressure of about 25 psi will shear any particles in the bearing that may attempt to prevent the valve from functioning.

The isolation valve at each accumulator is only closed momentarily for testing, or when the reactor is intentionally depressurized. The isolation valve is normally opened. It receives a signal to open when SIS is initiated.

The check valves operate in the closed position with a nominal differential pressure across the disc of approximately 1550 psi. They remain in this position except for testing or when called upon to function. Since the 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.

When the RCS is being pressurized during the normal plant heatup operation, the check valves are tested for leakage. This test confirms the seating of the disc and whether or not there has been an increase in the leakage since the last test. When this test is completed, the RCS pressure increase continues. 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.

The accumulators can accept leakage back from the RCS without effect on their availability. Table 6.3.2-10 indicates that inleakage rates, over a given time period, require readjusting the level at the end of the time period. In addition, these rates are compared to the maximum allowed leak rates for manufacturing acceptance tests (20 cc/hr; i.e., 2 cc/hr/in.).

In-leakage at a rate of 5 cc/hr/in., 2-½ times test, would require that the accumulator water volume be adjusted approximately once every 30 months. This would indicate that level adjustments can be scheduled for normal refueling shutdowns and that this work can be done at the operator's convenience. At a leakrate of 30 cc/hr/in. (15 times the acceptance leak rate), the water level will have to be readjusted approximately once every 5 or 6 months. This readjustment will take about 2 hr maximum.

The accumulators are located inside the reactor containment and are protected from the RCS piping and components by a missile barrier. Accidental release of the gas charge in the three accumulators would cause an increase in the containment pressure of approximately 0.1 psi. This release of gas has been included in the containment pressure analysis for the LOCA.

During normal operation, 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 operation of the accumulator is not restricting.

6.3.2.5.4 Emergency Flow to the Core

Special attention is given in the analysis to factors that could adversely affect the accumulator and SI flow to the core. These factors are as follows:

a) Steam binding in the core, including flow blockage due to loop sealing

b) Carryover of accumulator water during blowdown

c) Short circuiting of the accumulator from the core to another part of the RCS, and

d) Loss of accumulator water through the break.

The analysis model incorporates a detailed thermal-hydraulic representation of the accumulators and injection water sources, including the valves and piping connecting them to the reactor coolant system.

6.3.2.5.5 Recirculation Loop Leakage

An input to Chapter 15 LOCA dose consequence analyses is the assumed leakage of highly radioactive containment sump water, outside containment, from the post-accident recirculation heat removal systems. These are the systems that are required to recirculate sump water in order to provide cooling to the core and the containment. For RNP, these are the portions of the Residual Heat Removal (RHR), Safety Injection (SI), and Containment Spray (CS) Systems that are outside containment and would contain sump water during the recirculation mode. The leakage of concern is leakage that can become airborne and be released to the environment. Also included is the sump suction line penetration, as leakage past that penetration could also result in sump water becoming airborne outside containment. The leak rate assumed is plant specific. The dose analysis assumes a leak rate of two times the specified limit. For RNP, the Alternative Source Term LOCA dose analysis assumes a leak rate of two gallons per hour. Therefore, the limit, per TRM Section 3.23, is one gallon per hour.

A leak rate limit for these systems existed in the Technical Specifications prior to the Three Mile Island (TMI) accident. The TMI accident releases were primarily from leakage from systems outside containment that aren't part of post-accident recirculation heat removal. This included letdown, liquid radwaste, and sampling systems. Therefore, one of the TMI lessons learned commitments was to establish a program to minimize leakage from systems that could contain highly radioactive fluids post-accident. The requirements for such a leakage control program were incorporated into Technical Specification 5.5.2, which lists the systems involved and provides the requirements for inspections and preventative maintenance. There is no quantified limit for this program, just a requirement to maintain leakage as low as practicable. RHR, SI, CS, and the sump suction line penetration are a subset of this broader leakage control program. They are the only ones required to meet the one gallon per hour quantified limit, and therefore the only ones listed in TRMS 3.23. The other systems only need to be maintained with leakage as low as practicable. The basis for this distinction is that RHR, SI and CS must operate for extended periods to maintain core and/or containment cooling. The other systems are typically not used, or used for short periods, and could be isolated if leakage is excessive.

Leakage detection exterior to containment is achieved through use of sump level detection. The Auxiliary Building sump pumps start automatically in the event that liquid accumulates in the sump, and an alarm in the Control Room indicates that water has accumulated in the sump. Valving is provided to permit the operator to individually isolate each RHR pump. Radiation monitors could also provide an indication of system leakage.

6.3.2.5.6 Guard Pipe Protection for Sump Suction Line

In the unlikely event that the sump suction line should fail, the guard pipe and bellows are capable of containing fluid at 60 psig at 365ºF, which is in excess of the required 42 psig at 263ºF. This failure would be identified during the performance of 10 CFR 50, Appendix J testing.

The containment pipe penetration assemblies consist of an expansion joint element welded to a pipe and sleeve going through the containment wall. The expansion joint elements were hydraulically formed from a stainless steel cylinder having a single longitudinal weld. Each longitudinal weld was radiographed. One end of the element is welded to a closure plate of the same material as the corresponding process line and the other end of the element is welded to a carbon steel closure plate. The latter plate is welded to a sleeve.

The bellows expansion joints meet the requirements of Section III of the ASME Code. These expansion joints each contain a butt joint which is radiographed. All shop and field welds were examined using the liquid penetrant test. Each welder and welding procedure used has been qualified in accordance with the requirements of Section IX of the ASME Code.

The following is a list of documents required to assure that all phases of fabrication of the expansion joints and their attachment to the containment are performed:

a) Mill test reports of the element and plate

b) Welding and welder procedures and qualifications

c) Nondestructive test reports, and

d) Certified copies of the Charpy V-notch Impact Test on carbon.

The post-operational inspection program consists of a visual inspection during each refueling interval of the pipe and the valve at the containment penetration. Inspection of the piping in the penetration sleeve (guard pipe) requires no post-operational inspection because the penetration is periodically tested in accordance with 10 CFR 50, Appendix J, providing assurance that the integration of the sleeve (guard pipe) bellows and suction line is maintained.

6.3.2.6 <u>Protection Provisions</u>

All associated components, piping, structures, and power supplies, of the SIS are designed to Seismic Class I criteria. This is discussed further in Section 3.7.

All components inside the containment are capable of withstanding or are protected from differential pressure which may occur during the rapid pressure rise to 42 psig in 10 sec.

Motors which operate only during or after the postulated accident were designed as if used in continuous service. Periodic operation of the motors and the tests of the insulation ensures that the motors remain in a reliable operating condition.

All motors, instruments, transmitters, and their associated cables located inside the containment which are required to operate following the accident are designed to function under the post-accident temperature, pressure, and humidity conditions. This is discussed further in Section 3.11.

Protection against pipe whip is discussed in Section 3.6. Missile protection is discussed in Section 3.5.

6.3.2.7 Provisions for Performance Testing

The design provides for periodic testing of active components of the SIS for operability and functional performance.

Power sources are arranged to permit individual actuation of each active component of the SIS.

The SI pumps can be tested periodically during plant operation using the minimum flow recirculation lines provided. The RHR pumps are used every time the RHR loop is put into operation. All remotely operated valves can be exercised, and actuation circuits can be tested during routine plant operation.

The design provides for capability to test initially, to the extent practical, the full operational sequence up to the design conditions for the SIS to demonstrate the state of readiness and capability of the system.

An integrated system test can be performed during the late stages of plant cooldown when the RHR loop is in service. This test would not introduce flow into the RCS, but would demonstrate the operation of the valves, pump circuit breakers, and automatic circuitry upon initiation of SI.

The accumulator tank pressure and level are continuously monitored during plant operation, and flow from the tanks can be checked at any time using test lines.

Flow in each of the hot leg injection lines and in the main flow line for the RHR pumps is monitored by a flow indicator. Pressure instrumentation is also provided for the main flow paths of the high head and the RHR pumps. Level and pressure instrumentation are provided for each accumulator tank.

TABLE 6.3.2-1

INSTRUMENTATION READOUTS ON THE CONTROL BOARD FOR OPERATOR MONITORING DURING RECIRCULATION

VALVES

SYSTEM	VALVE NUMBER
SIS	MOV 759 A, B
SIS	MOV 844 A, B
SIS	MOV 860 A, B
SIS	MOV 861 A, B
SIS	MOV 862 A, B
SIS	MOV 863 A, B
SIS	MOV 864 A, B
SIS	MOV 866 A, B
SIS	MOV 867 A, B
SIS	MOV 869
SIS	MOV 870A, B
SIS	MOV 880 A, B, C, D
SIS	AOV 856 A, B
ACS	AOV 758
ACS	AOV 605
ACS	MOV 744 A, B
ACS	MOV 749 A, B

INSTRUMENTATION

SYSTEM	CHANNEL NUMBER
SIS	FI 940
SIS	FI 943
SIS	FI 932
SIS	FI 933
SIS	FI 958 A, B
SIS	LI 1925 A, B
SIS	PI 934
SIS	PI 940
SIS	PI 943
ACS	FI 605
ACS	LI 614
ACS	TR 604, A, B
RCS	LRCA 459
RCS	LICA 460
RCS	LICA 461
RCS	LI 462

HBR 2
UPDATED FSAR

TABLE 6.3.2-1 (Continued)

PUMPS

SYSTEM IDENTIFICATION	SYSTEM
SIS	Safety Injection
SIS	Containment Spray
RHR	Residual Heat Removal
ACS	Component Cooling
SWS	Service Water
SWS	Service Water Booster Pump

TABLE 6.3.2-2

ACCUMULATOR DESIGN PARAMETERS

Number	3
Type	Stainless steel clad/ carbon steel
Design pressure, psig	700
Design temperature, ºF	300
Operating temperature, ºF	70 - 120
Nominal operating pressure, psig	660
Minimum operating pressure, psig	600
Total volume, ft^3	1200
Minimum water volume at operating conditions, ft^3	825
Boron concentration (as boric acid), ppm	1950-2400
Relief valve setpoint, psig*	700

*The relief valves have soft seats and are designed and tested to ensure zero leakage at normal operating pressure

TABLE 6.3.2-3

BORON INJECTION TANK

Number	1
Type	Vertical
Total volume	900 gal
Design pressure	2735 psig
Design temperature	300ºF max., 32ºF min.
Operating pressure, psig	0 - to 1500
Operating temperature	Ambient
Fluid (Minimum)	1950 ppm Boron concentration (as boric acid)
Material	Stainless Steel
Code	ASME Section VIII, Division 2

TABLE 6.3.2-4

REFUELING WATER STORAGE TANK DESIGN PARAMETERS

Number	1
Material	Stainless Steel
Total volume, gal	350,000
Minimum volume, (solution) gal	300,000
Normal pressure, psig	Atmospheric
Operating temperature, ºF	Ambient
Design pressure, psig	Head Height
Design temperature, ºF	200
Minimum Boron concentration (as boric acid), ppm	1950

TABLE 6.3.2-5

PUMP PARAMETERS

Safety Injection Pump Design Parameters	
Number	3
Design pressure, discharge, psig	1,750
Design temperature, ºF	300
Design flow rate, gpm	300 (Note 2)
Max. flow rate, gpm	650 (Note 2)
Design head, ft	2,500 (Note 1 & 2)
Shutoff head, ft	3,400 (Note 1)
Material	11 - 14 Chrome
Motor H.P.	350
Type	Horizontal centrifugal
Residual Heat Removal Pump Design Parameters	
Number of pumps	2
Type	Inline centrifugal
Design pressure, discharge, psig	600
Design temperature, ºF	400
Design flow, gpm	3,750
Design head, ft	240
Material	Austenitic stainless steel
Motor H.P	300

Note 1: See figures 6.3.2-4A, 6.3.2-4B, and 6.3.2-4C.
Note 2: Refer to References 6.3.2-3 and 6.3.2-4 for pump parameter.

TABLE 6.3.2-6

RESIDUAL HEAT EXCHANGERS DESIGN PARAMETERS

Number	2
Design heat duty, Btu/hr (normal)	29.4×10^6
Design UA, Btu/hr/ºF	1.55×10^6
Design cycles (85ºF - 350ºF)	200
Type	Vertical Shell & U-tube

	Tube-Side	Shell-side
Design pressure, psig	600	150
Design flow, lb/hr	1.88×10^6	4.31×10^6
Inlet temperature, ºF	140	108
Outlet temperature, ºF	124	115
Design temperature, ºF	400	200
Material	Stainless steel	Carbon steel

TABLE 6.3.2-7

Deleted in Revision No. 21

TABLE 6.3.2-8

SINGLE FAILURE ANALYSIS-SAFETY INJECTION SYSTEM

COMPONENT	MALFUNCTION	COMMENTS
A. Accumulator (injection phase)	Delivery to broken loop	Totally passive system with one accumulator per loop. Evaluation based on two accumulators delivering to the core and one spilling from ruptured loop
B. Pump: (injection phase)		
1. High head safety injection	Fails to start	Two provided. Evaluation based on operation of one
2. Residual heat removal	Fails to start	Two provided. Evaluation based on operation of one
3. Component cooling*	Fails to start	Three provided. One required for recirculation cooling
4. Service water	Fails to start	Four provided. Evaluation based on operation of two.(See also Section 6.2-3)
C. Automatically Operated Valves: Open on SIS - Injection phase		
1. High head injection line isolation valve at the loops	Fails to open	**See Note
2. Deleted		

TABLE 6.3.2-8 (Continued)

COMPONENT	MALFUNCTION	COMMENTS
3. High head cold leg isolation injection header valve (located at BIT outlet)	Fails to open	Two parallel valves. One required to open
4. Residual heat removal pump isolation valve at injection line	Fails to open	Two parallel valves. One required to open
D. Valves operated from Control Room for recirculation: (recirculation phase)		
1. Containment sump recirculation isolation	Fails to open	Two lines in parallel with two valves in series in each line, one pair of valves in either line is required to open
2. Safety injection pump suction valve at residual heat exchanger discharge	Fails to open	Two parallel valves. One required to open
3. Isolation valve on the test line returning to the refueling water storage tank	Fails to close	Two valve in series. One required to close. Both valves manually closed during switchover from injection to recirculation mode.
4. Isolation valve at suction header from refueling water storage tank	Fails to close	Two valves in series. One required to close

TABLE 6.3.2-8 (Cont'd)

SINGLE FAILURE ANALYSIS-SAFETY INJECTION SYSTEM

COMPONENT	MALFUNCTION	COMMENTS
5. Isolation valves at residual heat removal pump suction line from refueling water storage tank.	Fails to close	Two valves in series. One required to close

The status of all active components of the Safety Injection System is indicated on the main control board.
Reference is made to Table 6.2-2

*Recirculation phase

** Note: In the original design, four out of five hot and cold leg injection isolation valves were assumed to open. After a reevaluation of the safety injection system was completed in 1971, an engineering change was completed which removed the power supplies from the motor operators for the three cold leg injection isolation valves and these valves were locked open. Later a modification was completed on the two hot leg injection isolation valves. These valves are administratively key locked shut and do not receive an automatic safeguards open signal.

TABLE 6.3.2-9

LOSS OF RECIRCULATION FLOW PATH

FLOW PATH	INDICATION OF LOSS OF FLOW PATH	ALTERNATIVE FLOW PATH
Low head recirculation		
From containment sump to low head injection header via the residual heat removal pumps and the residual heat exchangers	1. No flow in low head injection header. (flow monitor in main header)	From containment sump to high head injection header via the residual heat removal pumps, the residual heat exchangers, and the safety injection pumps
High head recirculation		
From containment sump to high head head injection header via the residual heat removal pumps, residual the residual heat exchangers, high the high head injection pumps header, suction header, and the high head injection pumps	1. No flow in high head injection headers (four flow monitors and two pressure monitors)	From containment sump to high injection header via the residual heat removal pumps, the heat exchangers, the redundant head recirculation suction and either or both of two of the three high head injection pumps
	2. Flow in only one of the two high head injection branch headers	As 1 except that flow from the safety injection pump(s) is only supplied to the unbroken branch header.

NOTE: As shown on Figure 6.3.2-1 and 6.3.2-2, there are valves at all locations where alternative flow paths are provided.

TABLE 6.3.2-10

ACCUMULATOR INLEAKAGE*

TIME PERIOD BETWEEN LEVEL ADJUSTMENTS	OBSERVED LEAK RATE cc/hr	(OBSERVED LEAK RATE) MAX. ALLOWED DESIGN)
1 month	3270	163.5
3 months	1090	54.4
6 months	545	27.2
9 months	363	18.1
1 year	273	13.7
10 years	27.3	1.37

* A total of 83.3 cubic ft, added to the initial amount, can be accepted in each accumulator before an alarm is sounded.

TABLE 6.3.2-11

RESIDUAL HEAT REMOVAL SYSTEM
DESIGN, OPERATION AND TEST CONDITIONS

	PUMPS	HEAT EXCHANGERS	VALVES	PIPES AND FITTINGS
Design Conditions				
Pressure, psig	600	600	665	700
Temperature, ºF	400	400	400	400
Operating Conditions (Max)*				
Pressure, psig	160	160	160	160
Temperature, ºF	180	180	180	180
Test Pressure, psig	1200	900	1100	900
Allowable pressure at operating temperature, psig	>600	>600	>690	>850

*During post loss-of-coolant recirculation

6.3.3 PERFORMANCE EVALUATION

The analyses specified by 10CFR50.46 are presented in Section 15.6.5. The results are shown to be in compliance with the acceptance criteria. The analytic techniques are in compliance with Appendix K of 10CFR50 and are described in Section 15.6.5.

6.3.3.1 Reliance on Interconnected Systems

During the injection phase, the high head SI pumps do not depend on any portion of other systems with the exception of the suction line from the refueling water storage tank. During the recirculation phase of the accident for small breaks, suction to the SI pumps is provided by a RHR pump.

The RHR (low head) pumps are normally used during reactor shutdown operations. Whenever the reactor is at power, the pumps are aligned for emergency duty.

The minimum size of debris that will be excluded from entry into the recirculation system will be 3/32 in. diameter. Debris larger than ¼ in. diameter could result in clogging of the containment spray nozzles.

Debris accumulation in the piping during construction was minimized by controlled cleanliness procedures. However, the system was flushed with clean water after construction was completed to remove any debris that may have entered the system inadvertently.

6.3.3.2 Shared Function Evaluation

Table 6.3.3-1 is an evaluation of the main components, which have been previously discussed, and a brief description of how each component functions during normal operation and during the accident.

TABLE 6.3.3-1

SHARED FUNCTIONS EVALUATION

COMPONENT ARRANGEMENT	NORMAL OPERATING FUNCTION	NORMAL OPERATING ARRANGEMENT	ACCIDENT FUNCTION	ACCIDENT
Refueling Water suction Storage Tank injection removal pumps	Storage tank for refueling operations	Lined up to suction of safety injection residual heat removal, and spray pumps	Source of borated water for core and spray nozzles	Lined up to of safety residual heat and spray
Accumulators (3)	None	Lined up to cold legs of reactor coolant piping	Supply borated water to core promptly	Lined up to cold legs of reactor coolant piping
Safety Injection legs Pumps (3) coolant	None	Lined up to cold legs of reactor coolant piping	Supply borated water to core	Lined up to cold of reactor piping
Residual Heat Removal Pumps (2)	Supply water to core to remove residual heat during shutdowns	Lined up to take suction from refueling water storage tank	Supply borated water to core	Lined up to take suction from refueling water
Service Water service Pumps (4)	Supply lake cooling water to component cooling heat exchangers	Two pumps in service	Supply lake cooling water to component cooling heat exchangers	Two pumps in

TABLE 6.3.3-1 (Cont'd)

SHARED FUNCTIONS EVALUATION

COMPONENT ARRANGEMENT	NORMAL OPERATING FUNCTION	NORMAL OPERATING ARRANGEMENT	ACCIDENT FUNCTION	ACCIDENT
Component Cooling service Pumps (3)	Supply cooling water to station nuclear components	One pump in service	Supply cooling water to residual heat exchangers and S.I. pump bearings	One pump in
Residual Heat Exchangers (2)	Remove residual heat from core during shutdown	Lined up for recirculation	Cool water in containment sump for core cooling and containment spray	Lined up for recirculation
Component Cooling exchanger Heat Exchangers (2)	Remove heat from component cooling water	One heat exchanger in service	Cool water for residual heat exchangers	One heat in service

6.3.4 TESTS AND INSPECTIONS

6.3.4.1 ECCS Performance Tests

The Preoperational Test Program, including ECCS performance tests, is described in Chapter 14.

6.3.4.2 Reliability Tests and Inspections

6.3.4.2.1 Inspection Capability

All components of the SIS can be inspected periodically to demonstrate system readiness.

The pressure containing systems can be inspected for leaks from pump seals, valve packing, flanged joints, and safety valves during system testing.

In addition, to the extent practical, the critical parts of the reactor vessel internals, injection nozzles, pipes, valves, and SI pumps can be inspected visually or by boroscopic examination for erosion, corrosion, and vibration wear evidence, and for nondestructive test inspection where such techniques are desirable and appropriate.

6.3.4.2.2 System Testing

Surveillance requirements are specified in the Technical Specifications.

Testing can be conducted during plant shutdown to demonstrate proper automatic operation of the SIS. A test signal is applied to initiate automatic action and verification made that the SI pumps attain required discharge heads. The test demonstrates the operation of the valves, pump circuit breakers, and automatic circuitry.

The test is considered satisfactory if control board indication and visual observations indicate all components have operated and sequenced properly.

The accumulator pressure and level are continuously monitored during plant operation and flow from the tanks can be checked using test lines, however, this function is no longer utilized.

The accumulators and the injection piping up to the final isolation valve are maintained full of borated water while the plant is in operation. The accumulators and injection lines are refilled with borated water as required by using the SI pumps to recirculate refueling water through the injection lines. A small test line is provided for this purpose in each injection header.

Flow in each of the hot leg injection lines and in the main flow line for the RHR pumps is monitored by flow indicators. Pressure instrumentation is also provided for the main flow paths of the SI and RHR pumps.

6.3.4.2.3 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. Thereafter, periodic tests demonstrate that components are functioning properly.

Each active component of the SIS can be individually actuated on the normal power source at any time during plant operation to demonstrate operability. The test of the SI pumps employs the minimum flow recirculation test line which connects back to the refueling water storage tank. Remote operated valves are exercised and actuation circuits tested. The automatic actuation circuitry, valves and pump breakers also may be checked during integrated system tests performed during a planned cooldown of the RCS.

The operation of the remote stop valves in the accumulator discharge line can be tested by opening the remote test valves in the test line connected just downstream of the stop valves. Flow through the test line may be measured, and the opening and closing of the discharge line stop valves 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. The piping and valves exist but are no longer utilized for testing.

The isolation valves are closed at any time that the RCS is depressurized. The SI actuation signal will cause this valve to open should it be in the closed position at the time of a LOCA.

The entire recirculation loop is pressurized during periodic testing of the ESF components. The recirculation piping is also leak tested at the time of the periodic re-tests of the containment.

Since the recirculation flow path is operated at a pressure in excess of the containment pressure, it is hydrotested during periodic re-tests at the recirculation operating pressures. This is accomplished by running each pump utilized during recirculation (safety injection, spray, and RHR pumps) in turn at near shut off head conditions and checking the discharge and recirculation test lines. The suction lines are tested by running the RHR pumps and opening the flow path to containment spray and SI pumps in the same manner as described above.

During the above test, system joints, valve packings, pump seals, leakoff connection, 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.

To verify the mechanical performance and assess operational readiness of components to fulfill their required safeguard functions and also to serve as post maintenance tests, ECCS systems and components tests are performed as follows:

1. Safety Injection System Component Test (Recirculation Quarterly)

2. Safety Injection System High Head Check Valve Test (as specified by the Inservice Testing Program)

3. Residual Heat Removal Component Test (Recirculation Quarterly)

4. RHR Pump Flow Test (Full Flow Biennially)

The requirements for Inservice Testing of Class 1, 2, and 3 components are described in Section 3.9.6.

The requirements for Inservice Inspection of Class 1 components are described in Section 5.2.4.

The requirements for Inservice Inspection (ISI) of Class 2 and 3 components are described in Section 6.6.

Additional surveillance tests for ECCS systems and components are as follows:

1. Safety Injection, Residual Heat Removal and Containment Spray Systems Flowpath Verification (Monthly Interval at Power) to verify valves are in the proper position.

2. Accumulator Check Valves test (during plant heatup) to verify check valves are closed prior to opening accumulator discharge valves.

3. Pressure Isolation Check Valve Back Leakage Test during periods of extended shutdowns as specified by the Inservice Testing Program and Technical Specifications to measure leak rate through SI pressurization isolation check valves.

4. Accumulator Isolation Valve Operability Test (as specified by the Inservice Testing Program) to verify operability of SI accumulator isolation valves.

5. Emergency Diesel Generator Auto Start on Loss of Power and Safety Injection and Emergency Diesel Trips Defeat (Refueling).

6. Safety Injection System Valve Position Indicator Verification (every two years) to verify remote position indication.

7. Safety Injection System High Head Component Test (as specified by the Inservice Testing Program) to verify readiness of component to meet its required safeguard function.

8. RHR Component Test (Quarterly) to verify readiness of component to meet its required safety function.

9. RHR and SI System Check Valve Test, as specified by the Inservice Testing Program, to verify readiness of components to meet their required safeguard function.

10. RHR Pump Pit Level Instrumentation's Check Valve Back Flow Testing to verify readiness of level instrumentation's check valves to meet their required safeguard function.

11. RHR Loop Valves Interlock Test (Refueling) to demonstrate SI-863A and SI-863B and SI-862A and SI-862B cannot be opened unless RHR loop pressure is less than 210 psig and to demonstrate that RHR-750 and RHR-751 cannot be opened unless RCS pressure is less than or equal to 474 psig.

12. RHR Valve Position Indicator Verification (every two years) to verify remote position indication.

Tests of ECCS leakage are required by Technical Requirements Manual Section 3.23.

6.3.5 INSTRUMENTATION REQUIREMENTS

Instrumentation provisions for the ECCS are discussed in Section 7.3.

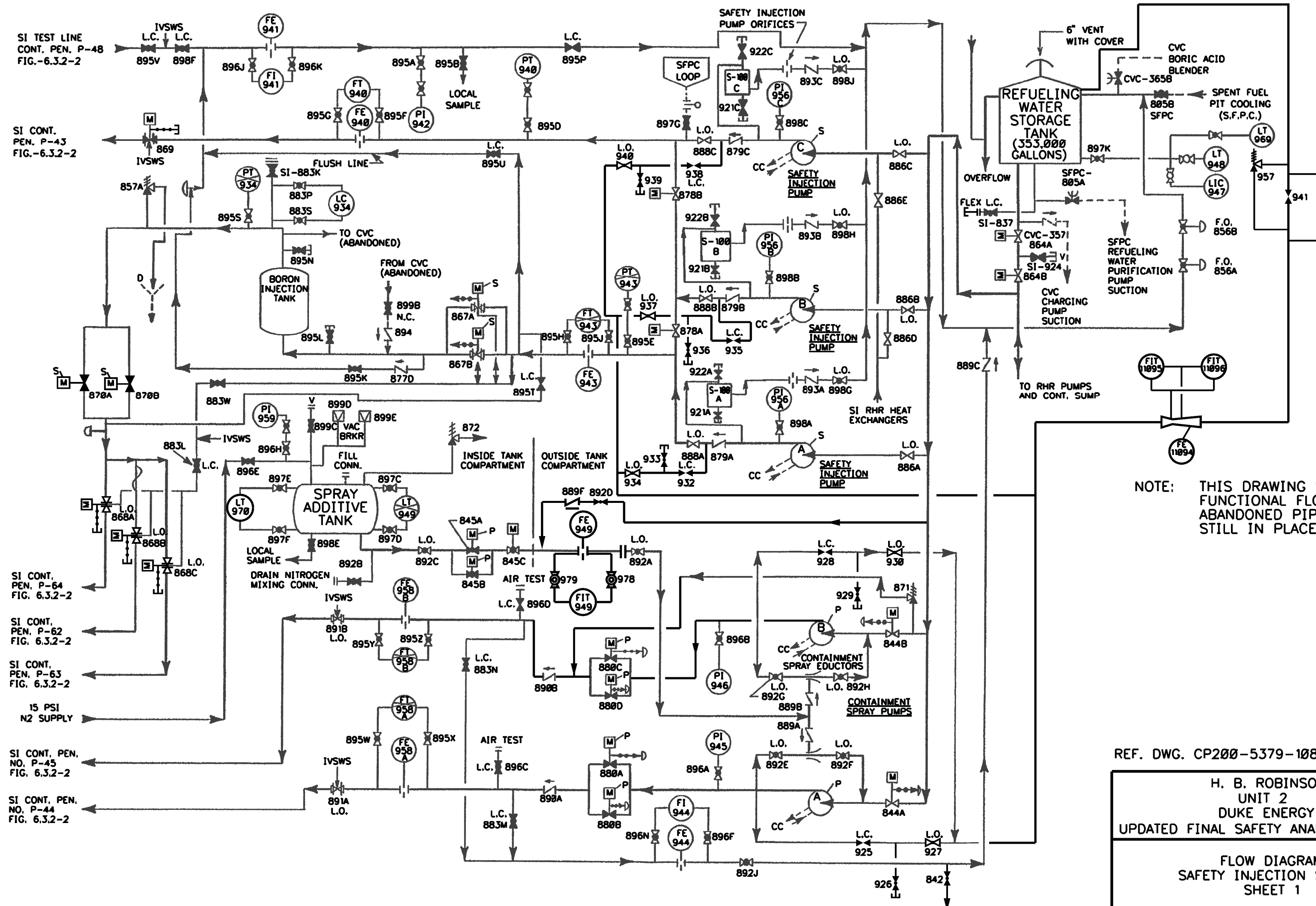
REFERENCES SECTION: 6.3

6.3.2-1 WCAP-7153, "Investigations of Chemical Additives for Reactor Containment Sprays," M. J. Bell, et al, March 1968 (W Confidential).

6.3.2-2 GID/R87038/0013, “Single Failure”

6.3.2-3 WCAP-12070, “Safety Injection System Design Summary Document,” R. K. Stirzel & K. J. King, December 1988 (W Confidential).

6.3.2-4 728-800-08, “Instructions For Installation of Safety Injection Pumps,” Worthington Corporation, October 1960



NOTE: THIS DRAWING SHOWS FUNCTIONAL FLOW PATHS. ABANDONED PIPING IS STILL IN PLACE.

REF. DWG. CP200-5379-1082

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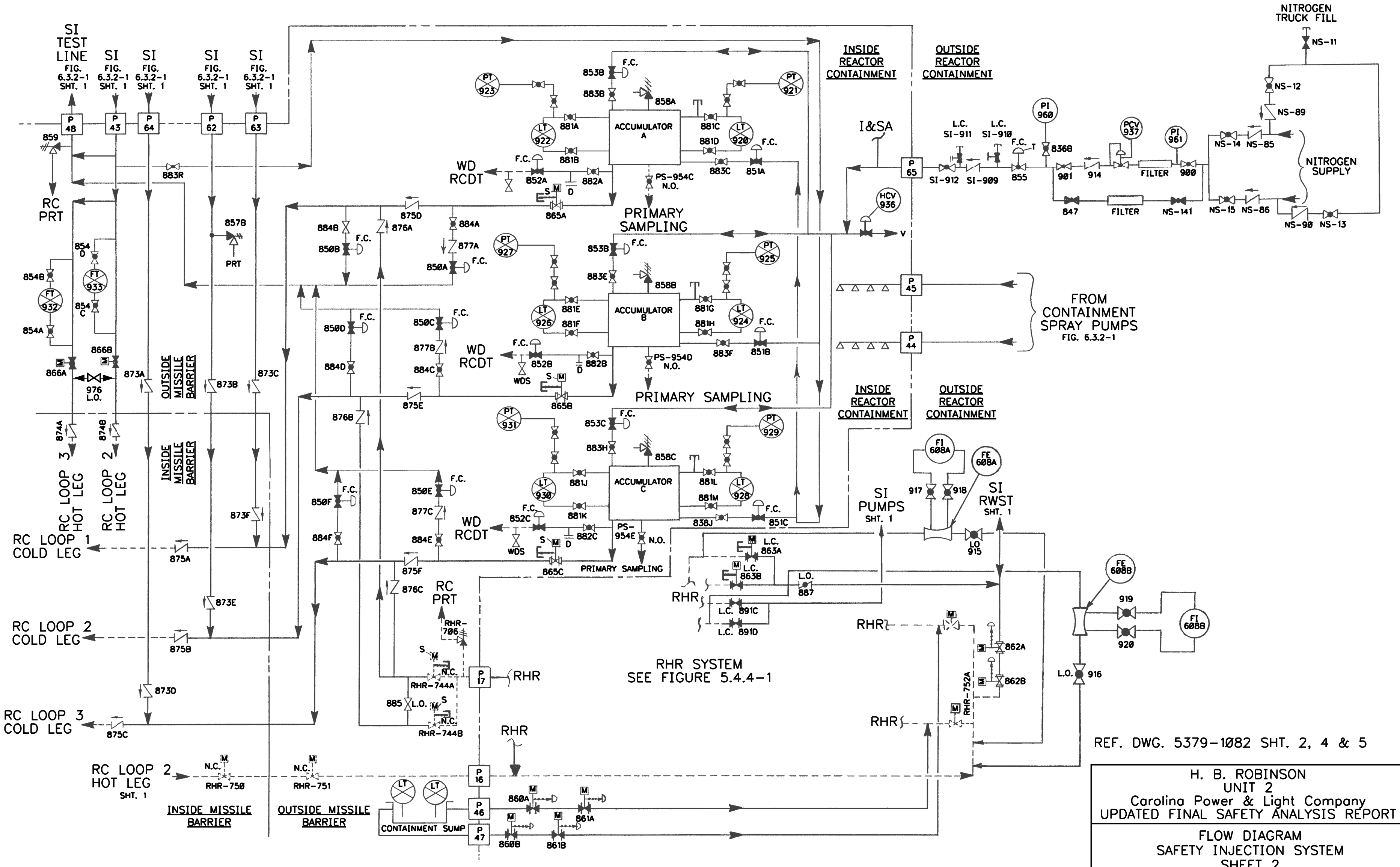
FLOW DIAGRAM
SAFETY INJECTION SYSTEM
SHEET 1

FIGURE 6.3.2-1

REVISION 26



REF. DWG. 5379-1082 SHT. 2, 4 & 5

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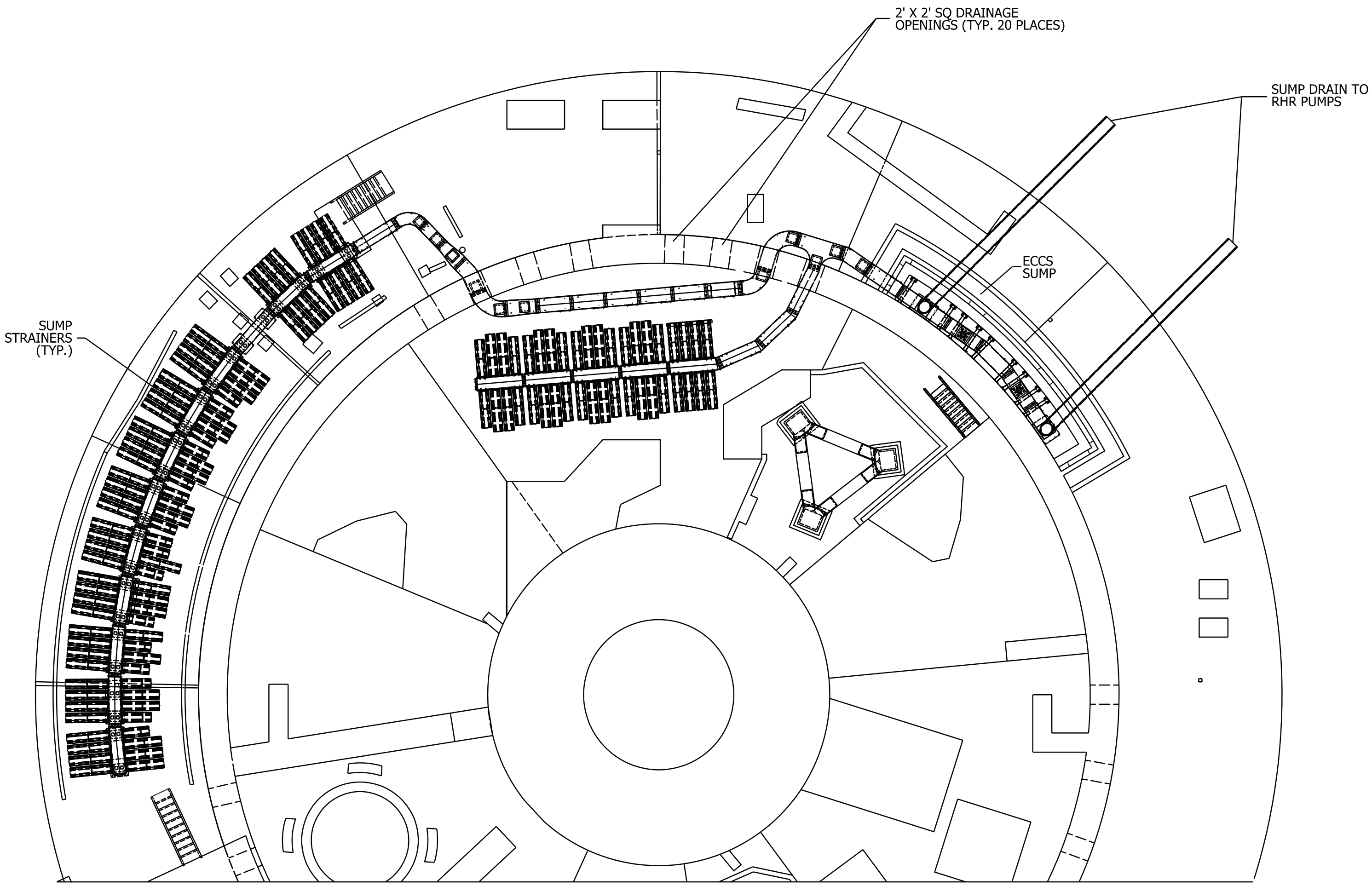
FLOW DIAGRAM
SAFETY INJECTION SYSTEM
SHEET 2

FIGURE 6.3.2-2 REVISION NO. 25



REVISION NO. 27

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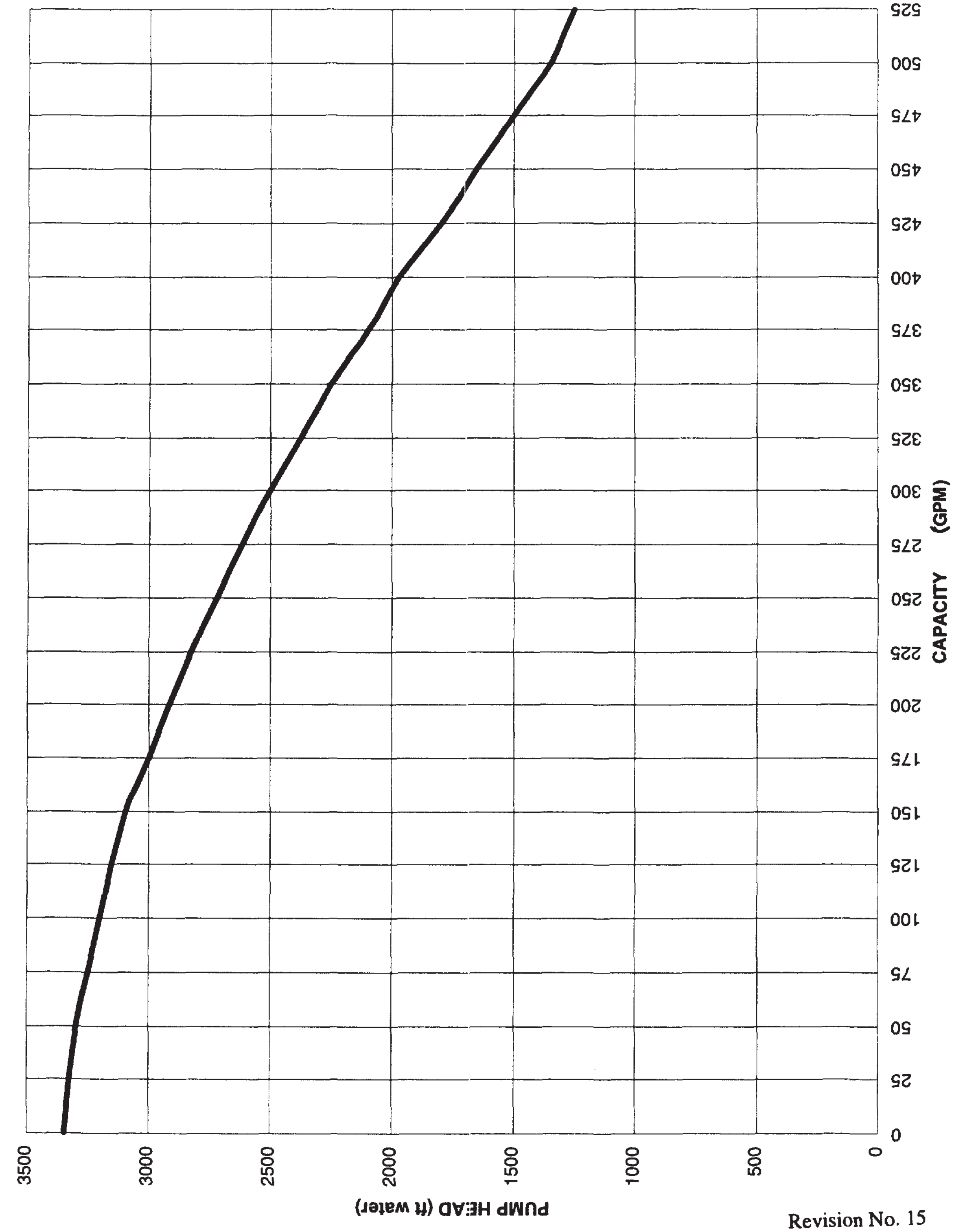
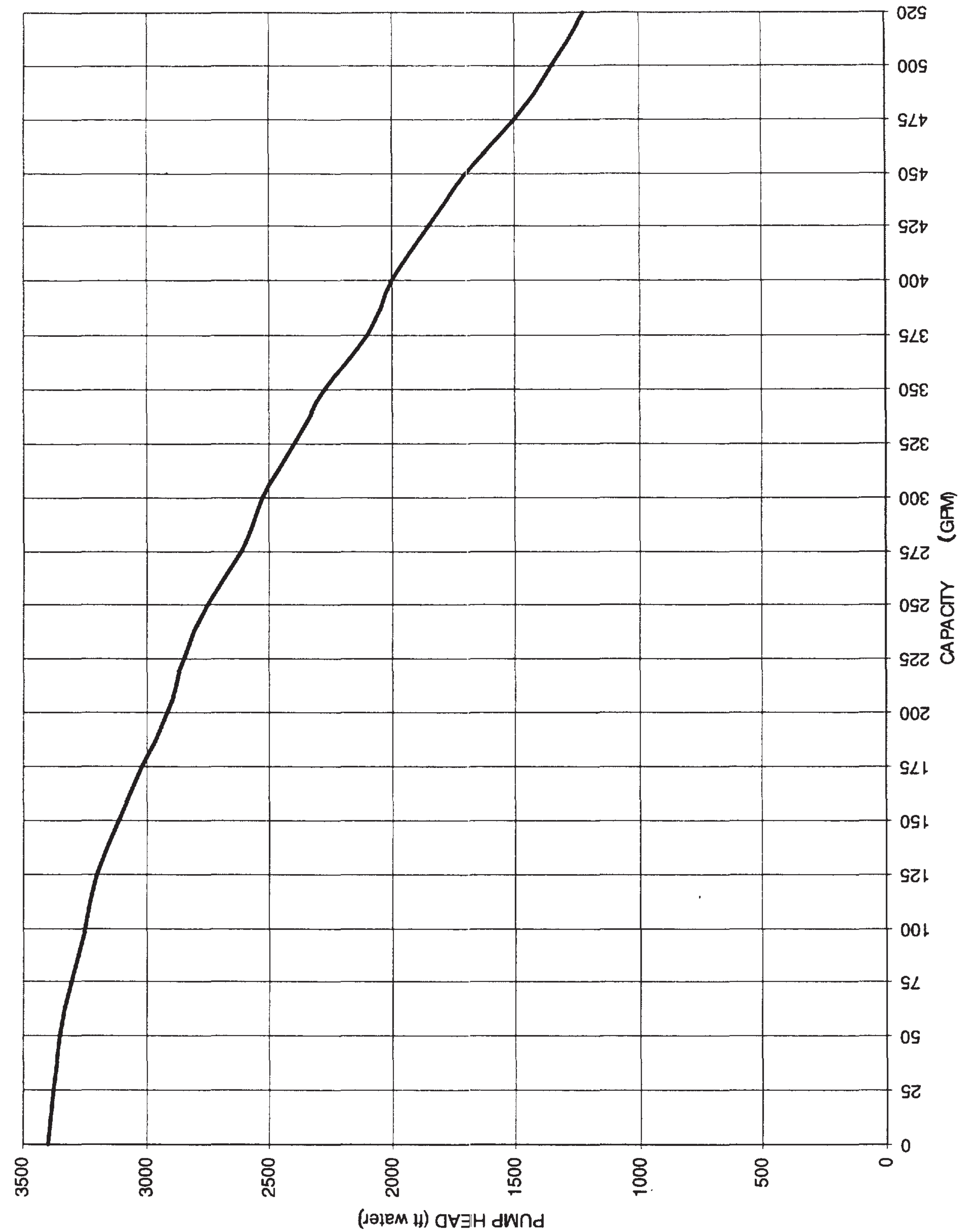
REACTOR BUILDING
COOLING WATER DRAINAGE
SCHEME
FIGURE 6.3.2-3

REF. DWG. HBR2-12301 SH00001



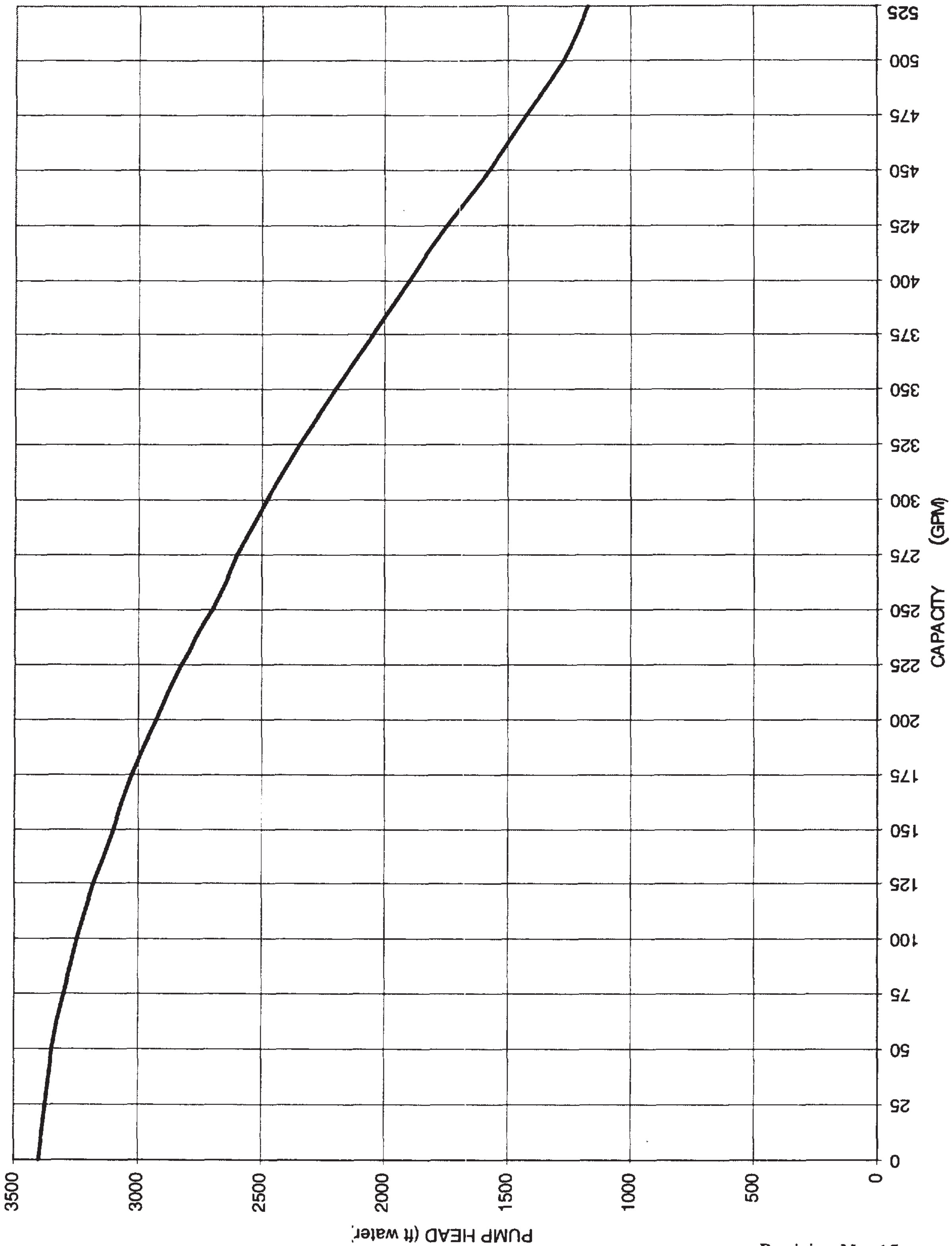
Revision No. 15

H. B. ROBINSON UNIT 2 Carolina Power & Light UPDATED FINAL SAFETY ANALYSIS REPORT	SAFETY INJECTION PUMP PARAMETERS 'A' SI PUMP	FIGURE 6.3.2-4a



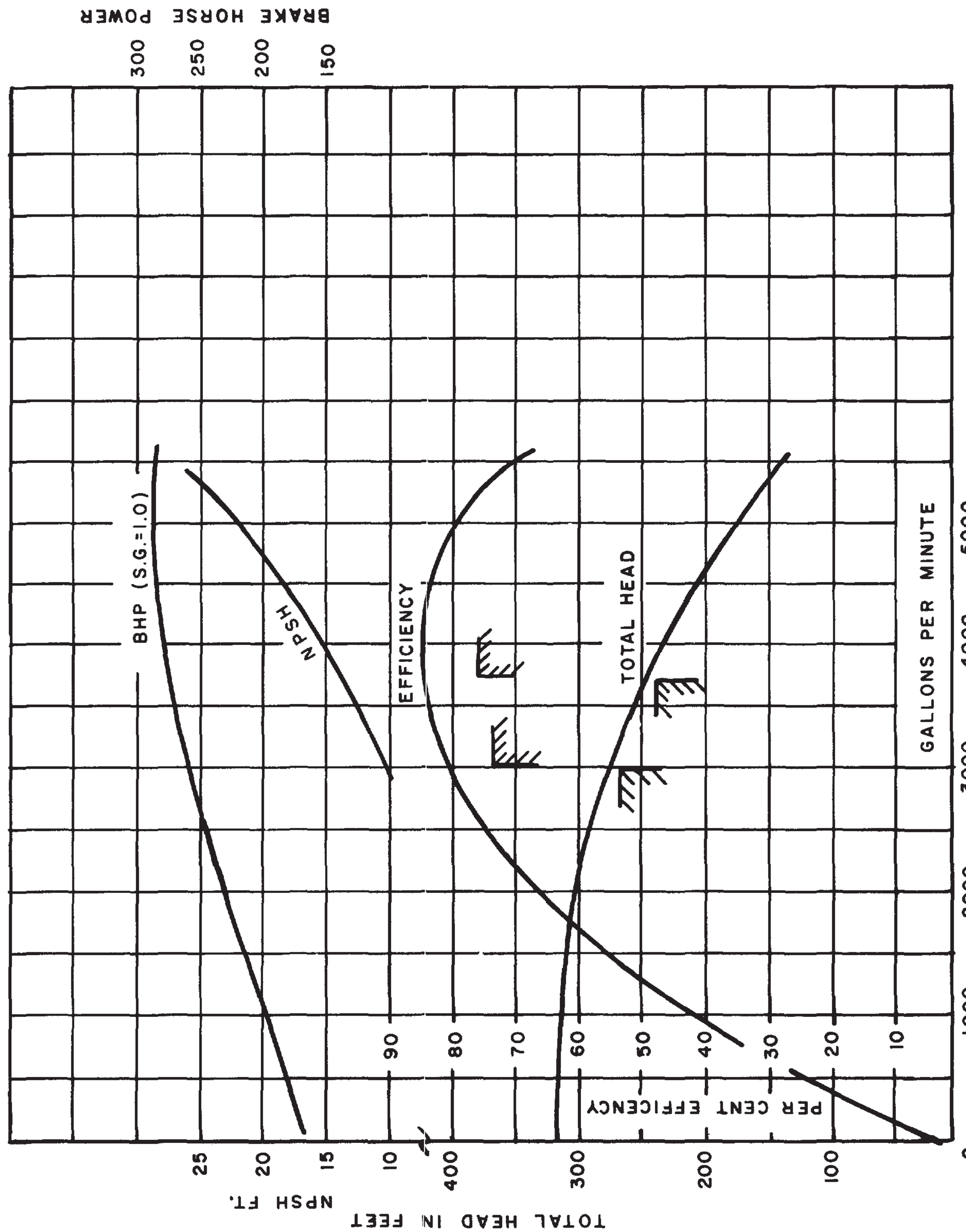
Revision No. 15

H. B. ROBINSON UNIT 2 Carolina Power & Light UPDATED FINAL SAFETY ANALYSIS REPORT	SAFETY INJECTION PUMP PARAMETERS 'B' SI PUMP	FIGURE 6.3.2-4b



Revision No. 15

H. B. ROBINSON UNIT 2 Carolina Power & Light UPDATED FINAL SAFETY ANALYSIS REPORT	SAFETY INJECTION PUMP PARAMETERS 'C' SI PUMP	FIGURE 6.3.2-4C



H. B. ROBINSON UNIT 2 Carolina Power & Light Company UPDATED FINAL SAFETY ANALYSIS REPORT	TYPICAL RHR PUMP CURVES	FIGURE 6.3.2 - 5

6.4 HABITABILITY SYSTEM

The Control Room Habitability Systems include equipment, supplies and procedures which give assurance that the Control Room operators can remain in the Control Room and take effective actions to operate the nuclear power plant safely under normal conditions and maintain the facility in a safe condition following a postulated accident as required by General Design Criterion 19 of Appendix A to 10 CFR 50 and by 10 CFR 50.67.

The H. B. Robinson Unit 2 (HBR 2) habitability systems were evaluated as described in References 6.4.0-1 and 6.4.0-4.

6.4.1 DESIGN BASIS

The habitability systems include systems and equipment to protect the Control Room operators and to allow them to remain in the Control Room for an extended period.

The bases upon which the functional design of the habitability systems were established include the following:

a) Control Room Envelope - The Control Room envelope contains all critical areas requiring access, such as the Control Room, kitchen, sanitary facility, and storage area. Those areas not requiring access are excluded from the envelope by means of closed doors.

b) Capacity - The minimum shift complement in the Control Room is defined by the HBR 2 Technical Specifications, Appendix A to Facility Operating License No. DPR-23. In an emergency, Duke Energy Progress, LLC will limit the number of people in the Control Room to the minimum required to cope with the emergency.

c) Food, Water, Medical Supplies, and Sanitary Facilities - There are no specific storage facilities in the Control Room for storage of emergency food or potable water. However, the zone serviced by the Control Room ventilation system contains all critical areas requiring access in case of emergency, including the kitchen, sanitary facility, and storage area.

d) Radiation Protection - Sufficient shielding, distance, and containment integrity are provided to assure that Control Room personnel shall not be subjected to doses under postulated accident conditions that would exceed the applicable limits specified in either GDC 19 of Appendix A to 10 CFR 50, or 10 CFR 50.67. The radiation exposure will not exceed 5 rem TEDE for the duration of any design basis accidents. The Control Room air conditioning consists of a system having a large percentage of recirculated air. During a design basis accident, the system is automatically configured to pressurize the Control Room with a limited amount of clean filtered outdoor air to control the intake of airborne activity.

e) Toxic or Noxious Gas Protection - Self-contained breathing apparatuses are available in the Control Room. No special protection against toxic gas intrusion and no toxic gas detectors are provided in the design of the HBR 2 Control Room. (See Section 6.4.4.2 for a more detailed discussion on toxic gas protection).

f) Respiratory, Eye and Skin Protection for Emergencies - Self-contained breathing apparatuses are kept in the Control Room for respiratory and eye protection during emergencies. Available for skin protection are full-face masks and protective clothing.

g) Habitability System Operation During Emergencies - The Control Room air conditioning and filter system is nuclear safety related and designed to Seismic Class I requirements. The system is capable of performing its safety-related functions assuming an active component single failure. During a postulated LOCA, the Control Room air conditioning system is automatically switched from the normal ventilation mode to the emergency pressurization mode of operation by a safety injection signal or Control Room radiation monitor alarm signal. This activates the air cleaning unit filter train and isolates the Control Room exhaust to the outdoors. The Control Room has been designed to protect the Control Room operators from all design basis natural phenomena and design basis accidents allowing safe shutdown of the plant.

h) Emergency Monitors and Control Equipment - Provisions have been made to detect radioactivity and smoke in the Control Room and smoke in the Control Room outside air intake. The Control Room filter system is automatically put into service by either a safety injection signal input or Control Room radiation monitor alarm input. Status of the air conditioning system is not changed due to smoke alarm.

6.4.2 SYSTEM DESIGN

6.4.2.1 Definition of Control Room Envelope

The Control Room safe shutdown controls are within the Control Room emergency zone. In addition, the washroom, the kitchen, and a small storage closet are included within the emergency zone and are accessible at all times.

6.4.2.2 Ventilation System Design

The Control Room envelope air conditioning process includes an environmental control operation and an emergency air cleanup operation. The system is required to be operable during all modes of plant operation, except cold shutdown.

The environmental control operation is the primary function of the air conditioning system and is normally in service at all times. This is accomplished by the use of redundant active air conditioning equipment. Passive features of the system are not redundant.

The emergency air cleanup operation is the secondary function of the air conditioning system. This is accomplished by the use of redundant active air cleanup equipment. Passive features of the system are not redundant. The emergency air cleanup equipment is normally on stand-by and is automatically put into service by either a safety injection signal or Control Room radiation monitor alarm input.

Only active components of the nuclear safety-related portions of the Control Room air conditioning system are redundant. Redundant safety-related active components include the air handling unit fans, refrigeration equipment, air cleaning unit fans, and control dampers. Passive nuclear safety-related portions include the air cleaning unit housing and filters, ductwork, and gravity dampers. Fire dampers are defined as passive components. The Control Room electric duct heater does not serve a safety function and is not redundant.

The Control Room air conditioning system is designated as nuclear safety-related and design to Seismic Class I requirements. Components contained in the system which do not perform a nuclear safety-related function but could adversely impact nuclear safety-related components are seismically designed to Seismic Class I requirements.

The Control Room air conditioning system is designed such that a single active failure concurrent with an initiating event will not render the system inoperable.

Two independent trains of active components are provided, each powered from a separate safety-related power-supply.

A more detailed description of the Control Room air conditioning system is contained in FSAR Section 9.4.1.

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6.4.2.3 Leak Tightness

The Control Room envelope is maintained under a positive differential pressure with respect to adjacent areas during the emergency pressurization mode of operation. The only exception to this occurs when there is a loss of Auxiliary Building exhaust fan HVE-7 concurrent with Control Room pressurization. In this case, testing has shown that adjacent areas served by HVS-1 and HVE-7 could be slightly positive with respect to the Control Room. In-leakage testing has shown that the dose to the Control Room operator would be satisfactory under this condition

During a postulated LOCA, a maximum makeup rate of 400 CFM is allowed for pressurizing the Control Room envelope. Periodic testing is required to demonstrate that at the beginning of each cycle, the Control Room is pressurized to a minimum of +1/8 inch water gage with respect to the outdoors with an outside air make-up rate of 400 CFM or less while in the emergency pressurization mode of operation. During normal operation, the system is periodically tested to demonstrate that a positive pressure with respect to the outdoors can be maintained in the Control Room.

All openings to the Control Room have a low leakage design. This includes doors, penetrations, and walls. Leak tightness is not required to be seismically qualified; a LOCA is not postulated concurrent with a seismic event. HVAC ductwork passing through the Control Room envelope belonging to the Unit 2 Hagan Room is a low leakage design and is seismically designed and supported.

A maximum of 400 CFM of makeup air will not result in the overall doses to the Control Room operators exceeding the radiation dose limit of General Design Criterion 19 of Appendix A to 10 CFR 50 under design basis accidents.

6.4.2.4 Interaction With Other Zones and Pressure-Containing Equipment

The following provisions are taken into consideration in the Control Room air conditioning system design to assure that there are no radioactive gases that would transfer into the Control Room:

1. The Control Room envelope is placed under positive pressure relative to the adjacent areas when in the emergency pressurization operating mode.

2. The Control Room air conditioning system is independent and completely separated from other adjacent ventilation zones except that the H and V Equipment Room containing most of the Control Room ventilation equipment is ventilated by the Reactor Auxiliary Building Ventilation system.

3. There is no other HVAC equipment within the Control Room envelope that serves other ventilation zones with the exception of a limited amount of duct passing through the Control Room envelope. This duct is specifically designed to limit infiltration and exfiltration.

4. All doors, duct, and cable penetrations are of low leakage design.

6.4.2.5 Shielding Design

The Control Room is separated from the radioactive sources in the Reactor Auxiliary Building by several floor levels, and is offset an appreciable distance to the south and east of the sources. Thus, the dose rate in the Control Room due to sources from the Auxiliary Building is negligible. However, the Control Room is exposed to post-LOCA direct radiation from the containment and the plume. The calculated dose from these two sources is presented in Chapter 15.

6.4.3 SYSTEM OPERATIONAL PROCEDURES

The normal operation of the Control Room air conditioning system is discussed in detail in Section 6.4.2.2 and Section 9.4.2.

The post accident operation of the Control Room air conditioning system is discussed in detail in Section 9.4.2.

Upon failure of the normal power supply, all electrically operated safety-related components of the system will be automatically switched to their respective emergency power source.

Upon receipt of a safety injection signal or high radiation signal from the Control Room area radiation monitor, the Control Room air conditioning system is automatically placed into the emergency pressurization mode of operation.

A Safety Injection signal will also shut down or prevent the WCCUs from operating until the last SI block has timed-out at 39.5 seconds. An SI interrupts power to the WCCUs. Upon a loss of power, the WCCUs are designed to go through a 3 minute start-up sequence before continued operation.

Manual isolation capability via the Emergency Recirculation mode of operation is provided for limiting the intake of hazardous chemicals or smoke. Hazardous chemicals are not stored or transported on or near the site in sufficient quantity as to require isolation capability as a regulatory requirement. However, isolation capability is beneficial and this operational mode is included in the system design to allow the Control Room operators to isolate outside air makeup from the Control Room envelope.

6.4.4 Design Evaluations

6.4.4.1 Radiological Protection

The evaluation of radiological dose to control room occupants from the design basis accidents is presented in Chapter 15.

6.4.4.2 Toxic Gas Protection

The buildup of toxic chemical concentrations at the Control Room air intake and within the Control Room volume was evaluated to determine the effect on Control Room habitability from postulated toxic chemical releases.

Table 6.4.4-1 summarizes the general input data used in the analysis. Table 6.4.4-3 presents the tier two chemicals in the plant vicinity that require analysis.

Toxic chemical concentrations at the Control Room air intake were analyzed using the guidance of Regulatory Guide 1.78. A Gaussian dispersion model was used to calculate the concentration dilution from the release point to the air intake. For those toxic chemicals that are liquefied compressed gases, the quantity of the puff release (flash fraction) is evaluated assuming an isenthalpic expansion. Based on this analysis, chemicals with concentrations at the Control Room air intake less than the toxic limit were eliminated from further study.

Chemicals for which the calculated concentrations at the Control Room air intake exceed the toxic limit were analyzed further to determine the buildup of chemical concentration in the Control Room, using conservation of mass equations for the Control Room HVAC system operation. For purposes of this analysis, the normal mode HBR 2 plant ventilation system was used.

Table 6.4.4-3 summarizes the numerical results of this HBR 2 plant toxic chemical habitability analysis and shows compliance with the appropriate limits. The Regulatory Guide 1.78 screening procedure eliminated most off-site chemicals stored in the vicinity as possible threats to Control Room habitability.

Amines, which are detectable by strong odor and are less toxic than ammonia, may be stored and used on-site.

6.4.4.3 Asphyxiants

The worst case release of propane 0.6 miles from the plant does not result in a reduction of the O_2 concentration in the control room to the level at which an oxygen deficient atmosphere would be created. Release of 150 lb of Refrigerant 22 in the control room complex could temporarily reduce the oxygen concentration in the control room to 18%, which is adequate to provide life support. Evaluation of the toxic hazards associated with a sudden release of R-22 in the HVAC Equipment Room have also been evaluated and found acceptable.

6.4.4.4 Control Room Design Review

A Detailed Control Room Design Review was completed and submitted to the NRC by letter dated September 23, 1986, Serial NLS-86-345.

TABLE 6.4.4-1
SUMMARY OF INPUT DATA

PARAMETER	DATA	UNITS
Meteorological:		
Pasquill stability	G	Classification
Average wind speed	1.0	m/sec
HVAC System		
Normal operation:		
Fresh air makeup	400	ft^3/min
Inleakage	170	ft^3/min
Outleakage and exhaust	570	ft^3/min
Filter removal, toxic chemical	None	None
Loop flow	5800	ft^3/min
Air exchange rate, outside air	1.7	Per hour
Volume of Control Room	20,124	ft^3

TABLE 6.4.4-2

This Table was deleted in Revision 19

TABLE 6.4.4-3

RESULTS OF TOXIC CHEMICAL ANALYSIS

STATIONARY SOURCES

LOCATION	DISTANCE (mi)	CHEMICAL	STORAGE CONDITION	QUANTITY STORED (lb)	QUANTITY ALLOWED PER REG GUIDE 1.78 (lb)	TOXIC LIMIT (mg/m^3)	CONCEN TRATION AT INTAKE (mg/m^3)	CONCENTRATI ON IN CONTROL ROOM AT 2 MIN AFTER HUMAN DETECTION (mg/m^3)
Sonoco	> 5 a							
Royster Clark	4.9 b							
Talley Metals	1.9 c	Propane	Liquid	2 x 100,000	365,000			
Darlington County Water Well Head - Cunningham	3.5 c	Chlorine	Liquid	2x 150	7800			
Darlington County Water Well Head – Ruby Road	4.0 c	Chlorine	Liquid	2x 150	7800			

TABLE 6.4.4-3
(Continued)

LOCATION	DISTANCE (Feet)	CHEMICAL	STORAGE CONDITION	QUANTITY STORED (Gallon)	QUANTITY ALLOWED PER REG GUIDE 1.78	TOXIC LIMIT (mg/m^3)	CONCENTRATION AT INTAKE (mg/m^3)	CONCENTRATION IN CONTROL ROOM AT 2 MIN AFTER HUMAN DETECTION (mg/m^3)
HBR 2	250	Ammonium Hydroxide	Liquid	250		210 mg/m^3 (as ammonia)	16 d	g
HBR 2	250	Hydrazine	Liquid	350		66.5	10 d	g
HBR 2	185	Liquid Nitrogen	Liquid	1000		77,700 (Asphyxiant)	1.24 d 14.5 e	g
HBR 2	250	Dimethylamine	Liquid	250		935	10 d	g
HBR 2		f						

a: Per Regulatory Guide 1.78, Rev. 1, chemicals stored or situated at distances greater than 5 miles from the plant need not be considered.

b: Reported hazardous chemical inventory does not have a credible transport mechanism to the HBR 2 site.

c: Further consideration of a hazardous chemical is not required if the weight stored in the largest container is less than the equivalent weight appearing in the Attachment A Table of RG 1.78, Rev. 1.

d: Ten minute release.

e: Puff release.

f: Other reported hazardous chemical inventory does not have a credible transport mechanism to the HBR 2 Control Room.

g: Since concentration at the intake is less than the toxic limit, no Control Room concentration calculation is required.

TABLE 6.4.4-3
(Continued)

RESULTS OF TOXIC CHEMICAL ANALYSIS
MOBILE SOURCES

POSTULATED RELEASE LOCATION	DISTANCE TO CONTROL ROOM INTAKE (mi)	CHEMICAL	TRANSPORTED CONDITION	TRANSPORTED QUANTITY (lb)	TOXIC LIMIT (mg/m^3)	RELEASE TYPE	CONCENTRATION AT INTAKE (mg/m^3)	CONCENTRATION IN CONTROL ROOM AT 2 MIN AFTER HUMAN DETECTION (mg/m^3)
Highway 151 and Old Camden Road	0.6 mi	Chlorine	Liquid	150	30	Puff	441.3	< 30
Highway 151 and Old Camden Road	0.6 mi	Chlorine	Liquid	150	30	Continuous (10 minute)	141	< 30
Highway 151 and Old Camden Road	0.6 mi	Propane	Liquid	47,560	19,215 (Asphyxiant)	Puff	179,300	< 19,215
Highway 151 and Old Camden Road	0.6 mi	Propane	Liquid	47,560	19,215 (Asphyxiant)	Continuous (10 minute)	44,620	< 19,215

6.4.5 TESTING AND INSPECTION

The tests to verify that the Control Room filter system will adequately remove radioactivity from the incoming ambient air, should there be an accidental radiation release to the atmosphere, are specified in the HBR 2 Technical Specifications, Appendix A to Facility Operating License No. DPR-23.

The inspection of the charcoal bed and charcoal filter housings of the filter system is performed each refueling outage as part of a refueling periodic test. This inspection includes a visual check of each system's filter banks. The inspection also includes a freon leak check which would immediately detect a system leak caused by insufficient charcoal in the bed and by deformation of the housing.

Testing and inspection is also conducted to demonstrate Control Room envelope leak tightness and satisfactory operation of air cleaning unit fans, air handling unit fans, and the refrigeration equipment.

6.4.6 INSTRUMENTATION REQUIREMENTS

The Control Room air conditioning system instrumentation is designed to assist the operator to monitor habitability conditions in the Control Room. System instrumentation, control switches and alarms on the Main Control Board provide the operator with the information concerning the status of the system and enables the operator to take the proper course of action.

Also, system instrumentation, control switches, and alarms are located in the Equipment Room and on a panel located on the mezzanine level of the Turbine Building immediately outside the Equipment Room and in close proximity of the Control Room. All controls and instrumentation required by the operators to maintain the Control Room air conditioning system in a safe and operable condition are located inside the Control Room envelope. Some controls and instrumentation normally located in the Control Room by good design practice are located in other areas due to human factors concerns.

The Control Room air conditioning system controls are designed such that failure of any safety-related fan or water cooled condensing unit will result in auto shutdown of the failed unit and auto start of the stand-by unit without requiring operator action.

All Control Room air conditioning system damper actuators are designed to fail to the position required for post-accident operation upon loss of electric or pneumatic power.

Smoke detectors are provided inside the outside air intake duct and throughout the Control Room area.

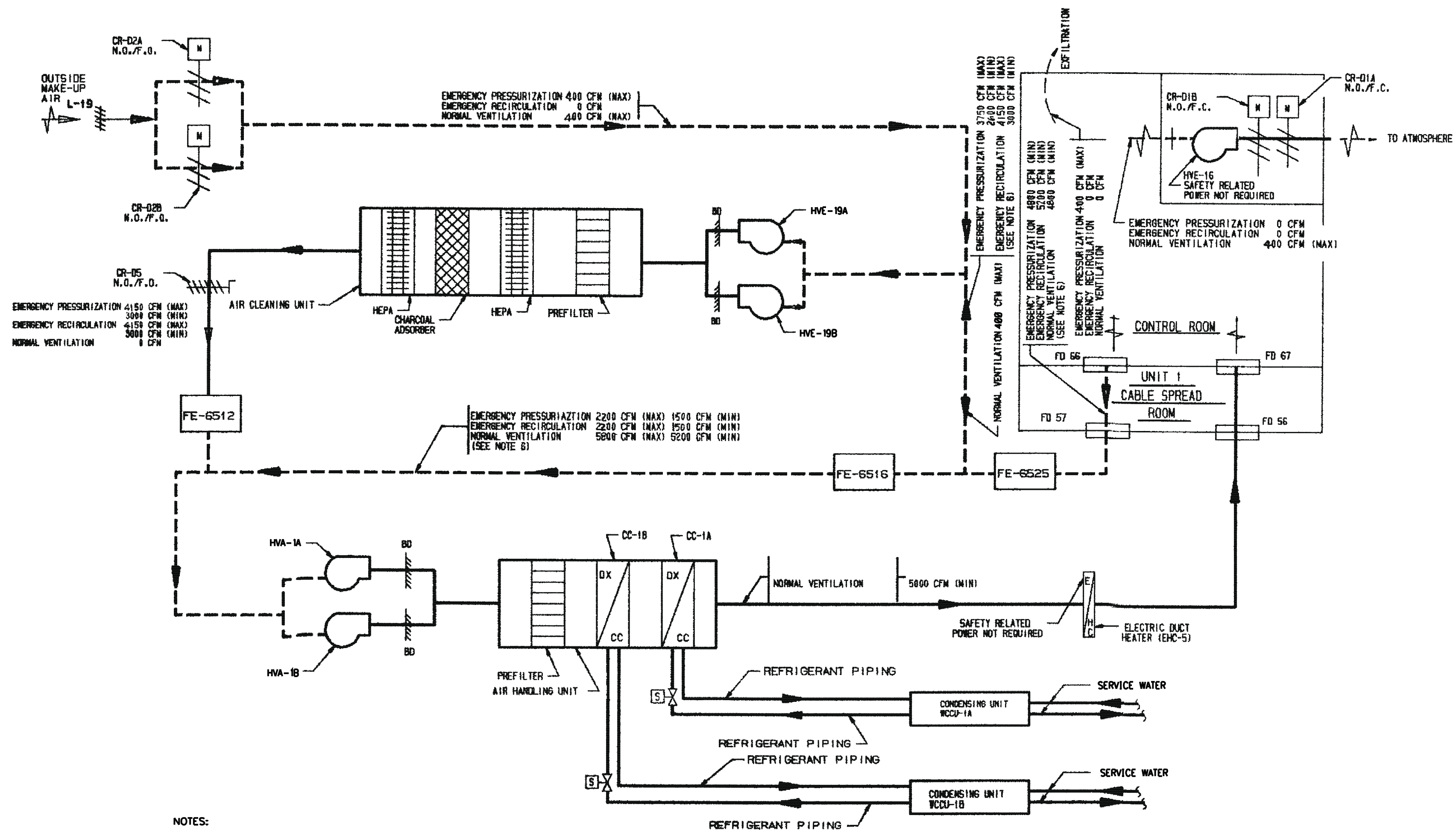
REFERENCES: SECTION 6.4

6.4.0-1 Letter NO-80-1947, from CP&L to NRC; "Control Room Habitability," December 31, 1980.

6.4.0-2 Letter NLS-85-495, from NRC to CP&L; "H. B. Robinson Unit 2 - NUREG-0737 Item No. III.D.3.4 Control Room Habitability."

6.4.0-3 Letter NRC-90-641, from NRC to CP&L; "Control Room Habitability System Modification, H. B. Robinson Steam Electric Plant, Unit No. 2."

6.4.0-4 NRC Letter dated September 24, 2004, “H. B. Robinson Steam Electric Plant, Unit No. 2 – Issuance of an Amendment on Full Implementation of the Alternative Source Term (TAC No. MB5105).”



NOTES:

1. THE MODES OF OPERATION ARE AS FOLLOWS:
   A. EMERGENCY PRESSURIZATION
   B. EMERGENCY RECIRCULATION
   C. NORMAL VENTILATION
2. A RADIATION MONITOR SIGNAL OR SAFETY INJECTION SIGNAL SHALL PLACE THE HVAC SYSTEM IN 'EMERGENCY PRESSURIZATION' MODE.
3. ALL FLOWS IDENTIFIED FOR THE 'EMERGENCY PRESSURIZATION' MODE OF OPERATION ARE BASED ON 'LOADED' AIR CLEANING UNIT AND AIR HANDLING UNIT FILTERS. OTHER MODES OF OPERATION AIR FLOWS ARE APPROXIMATE AND ARE BASED ON LOADED FILTERS.
4. THE AIR CLEANING UNIT IS RATED FOR 6000 CFM AIR FLOW THROUGH THE UNIT WITH CLEAN FILTERS WILL NOT EXCEED THIS VALUE.
5. ALL AIR FLOW RANGES DESIGNATE AIR FLOWS FOR CLEAN TO LOADED FILTER CONDITIONS.
6. MAXIMUM AND MINIMUM AIR FLOWS ARE BASED ON MAXIMUM OUTSIDE MAKE-UP AIR FLOWS AS SHOWN.

H&V EQUIPMENT ROOM

CONTROL ROOM HABITABILITY - HVAC

NUCLEAR SAFETY RELATED
(UNLESS OTHERWISE NOTED)
(REF. DWG. D-994-M-2100)

REVISION NO. 25

H.B. ROBINSON
UNIT 2
Carolina Power & Light Company
UPDATED FINAL SAFETY ANALYSIS REPORT

SCHEMATIC DIAGRAM OF THE CONTROL ROOM VENTILATION SYSTEM

Figure 6.4.2-1

6.5 FISSION PRODUCT REMOVAL AND CONTROL SYSTEMS

6.5.1 ENGINEERED SAFETY FEATURE (ESF) FILTER SYSTEMS

This section describes the ESF safety-related filter systems which are credited with reducing accidental release of fission products following a postulated design basis accident (DBA). The following ESF filter systems are included in this category:

a) The Containment Purge System, which will limit the release of fission products resulting from a fuel handling accident in containment (refer to Sections 9.4.3 and 15.7.4). (The containment purge system is expected to be available in case of a fuel handling accident in containment, but the dose analyses conservatively take no credit for the filtration system.)

b) The Spent Fuel Storage Area Subsystem of the Fuel Handling Building Ventilation System, which is required to limit the release of fission products resulting from a fuel handling accident in the Fuel Handling Building (refer to Sections 9.4.3 and 15.7.4).

c) The Control Room Ventilation System, which is required to maintain control room habitability (refer to Sections 6.4.1 and 9.4.2).

6.5.2 CONTAINMENT SPRAY SYSTEMS

6.5.2.1 Design Basis

In addition to its heat removal function, the Containment Spray System was designed to add sodium hydroxide (NaOH) during the initial period of spray operation to effectively remove iodine from the containment atmosphere and meet LOCA dose limits. The heat removal capability of the spray system is discussed in Section 6.2.2 (Containment Heat Removal).

Those portions of the spray systems located outside of the containment which were designed to circulate, under post-accident conditions, radioactively contaminated water collected in the containment were provided with closed systems for collection of discharges from pressure-relieving devices and adequate shielding to maintain radiation levels within the guidelines of 10 CFR 50.67.

The spray system was designed to operate over an extended period of time and to withstand, without loss of functional performance, the post-accident containment environment.

All associated components, piping, structures, and power supplies of the Containment Spray System were designed to Seismic Class I criteria.

Redundant active components were provided. System piping located within the containment was designed to be redundant with the redundant components separated in arrangement, unless it is fully protected by other means from damage which may follow any primary coolant system failure.

The starting sequence of the containment spray pumps and their related emergency power equipment was designed so that delivery of the minimum required flow is reached within 60 sec from receipt of the initiating signal, which is the delay assumed for the starting of containment cooling. The initiation of the addition of sodium hydroxide to the spray flow is automatic with no additional time delay.

The design bases for sizing of spray system components are discussed in Section 6.2.2 for spray pumps and piping, and in Section 6.1.1.2 for the spray additive eductor and spray additive tank. The pH characteristics, materials compatibility, and core spray stability are also discussed in Section 6.1.1.2.

Design basis, accident conditions and fission product releases are discussed in Section 15 for the LOCA.

6.5.2.2 System Design

Containment iodine removal capability is provided by the Containment Spray System shown in Figure 6.2.2-1. The components of this system are aligned into two subsystems. Each subsystem contains a pump, associated valving, and spray headers independently capable of delivering one-half of the total required flow of 2322 gpm. If one train is inoperable, the minimum delivered flow is, therefore, 1161 gpm. This system operates in two sequential modes:

a) Spray from the refueling water storage tank into the entire containment atmosphere using the containment spray pumps. During this mode, the contents of the spray additive tank (sodium hydroxide) are mixed into the spray stream to provide adequate iodine removal from the containment atmosphere.

b) Recirculation of water from the containment sump is provided by the diversion of a portion of the recirculation flow from the discharge of the residual heat removal heat exchangers to the suction of the spray pumps after injection from the refueling water storage tank has been terminated.

The principal components of the Containment Spray System are two pumps, one spray additive tank, spray ring headers and nozzles, and the necessary piping and valves. The containment spray pumps and the spray additive tank are located in the Auxiliary Building. The spray pumps take suction directly from the refueling water storage tank.

The Containment Spray System also utilizes the two residual heat removal pumps, two residual heat exchangers, and associated valves and piping of the Safety Injection System (SIS) for the long-term recirculation phase of containment cooling and iodine removal.

During spray injection, approximately 80 gpm of pump discharge flow is diverted from the spray pump discharge through the spray eductors. The liquid from the tank then mixes with the liquid entering the suction of the pumps via the eductors. The pH of the resulting solution is suitable for the removal of iodine from the containment atmosphere (refer to Section 6.1.1.2).

During spray recirculation operation, the water is screened through a 3/32 in. perforated plate before leaving the containment sump.

The spray nozzles are stainless steel and have a 3/8 in. diameter orifice. The spray nozzles, of the ramp bottom design, are not subject to clogging by particles less than 1/4 in. in maximum dimension. Since particles larger than 3/32 in. in dimension (plus 10% to account for deformable particles) are prevented from entering the spray recirculation flow, as indicated above, the spray nozzles are effectively protected against clogging 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 nozzles are connected to six ring headers located within the dome of the Containment Building. The lowest ring header is located at Elevation 372.3 ft and the highest ring header is located at Elevation 412.1 ft. There are 116 Spraco Model 1713 nozzles distributed on the six headers.

The nozzles and headers are so oriented as to ensure adequate spray coverage of the containment volume.

The procedure for the change-over from the injection mode to the recirculation mode of operation is described in Section 6.2.2.

All associated components, piping, structures, and power supplies of the Containment Spray System are described in Section 6.2.2, with the exception of the spray additive tank and eductor, which are described in Section 6.1.1.2.

6.5.2.3 Design Evaluation

The design parameters and iodine and particulate removal capabilities of the containment spray system used in the LOCA dose analysis are provided in Chapter 15.

6.5.2.4 Tests and Inspections

6.5.2.4.1 Inspection Capability

All components of the Containment Spray System can be 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. During the operational testing of the containment spray pumps, the portions of the system subjected to pump pressure are inspected for leaks.

6.5.2.4.2 Component Testing

All active components in the Containment Spray System are tested both in preoperational performance tests in the manufacturer's shop and during in-place testing after installation.

The containment spray pumps can be tested singly by opening the valves in the miniflow line. Each pump in turn can be started by operator action and checked for flow establishment. The spray injection valves can be tested with the pumps shut down.

The spray additive tank valves can be opened periodically for testing. The contents of the tank are periodically sampled to determine that the proper solution is present.

Initially the containment spray nozzle availability was tested by blowing smoke through the nozzles and observing the flow through the various nozzles in the containment.

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

6.5.2.4.3 System Testing

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

The air test lines, for checking initially the spray nozzles, connect downstream of the isolation valves. Air flow through the nozzles is monitored by the use of hot air and infrared thermography.

During the initial preoperational tests of the spray system, the flow bypass through the spray educators was checked. This initial test and all subsequent system tests are made with the spray additive tank isolation valves closed.

6.5.2.4.4 Operational Sequence Testing

The functional test of the SIS described in Section 6.3.4 demonstrates proper transfer to the emergency diesel generator power source in the event of loss of power. A test signal simulating the containment spray signal will be used to demonstrate operation of the spray system up to the isolation valves on the pump discharge.

6.5.2.5 Instrumentation Requirements

The spray system is actuated by the coincidence of two sets of two out of three (high-high) containment pressure signals. This starting signal starts the pumps and opens the discharge valves to the spray header. The valves associated with the spray additive tank open automatically upon receipt of the containment spray signal. After the containment spray signal is actuated, the system may be overridden by the operator to stop the sodium hydroxide addition and operate or stop equipment, or reset the initiating signal if he determines that the actuation was not warranted. The system also has the capability to allow the operator to manually reinitiate the sodium hydroxide addition if required. Emergency procedures set forth guidelines for these actions. If required, the operator can manually actuate the entire system from the Control Room.

Remotely operated valves of the Containment Spray System, which are under manual control (that is, valves which normally are in their ready position and do not receive a containment spray signal), have their positions indicated on a common portion of the control board. At any time during operation when one of these valves is not in the ready position for injection, it is shown visually on the board.

Containment spray additive tank level is indicated in the Control Room. A level indicating alarm is provided in the Control Room to alarm if, at any time, the spray additive tank contains less than the required amount of sodium hydroxide solution. Periodic sampling confirms that proper sodium hydroxide concentration exists in the tank.

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 have been set for the flow being sent to the core and for the flow being sent to the containment spray pump suction. Sufficient flow instrumentation is provided so that the operator can perform appropriate flow adjustments with the remote throttle valves in the flow path as shown in Figure 6.3.2-1 and 6.3.2-2.

6.5.2.6 Materials

A complete discussion of materials utilized in the Containment Spray System is presented in Section 6.1.1.1. The chemical composition and stability of spray additives in storage, in the spray solution, and in the sump are presented in Section 6.1.1.2.

6.5.3 FISSION PRODUCT CONTROL SYSTEMS

6.5.3.1 Primary Containment

For a discussion of the primary containment structural and functional design and other containment systems, refer to the following sections:

Concrete Containment	3.8.1
Containment Functional Design	6.2.1
Containment Heat Removal System	6.2.2
Containment Isolation System	6.2.4
Combustible Gas Control in Containment (Post-Accident Venting System)	6.2.5
Containment Leakage Testing	6.2.6
Isolation Valve Seal Water System	6.8.1
Penetration Pressurization System	6.9.1
Containment Ventilation System	9.4.3

Refer to Sections 6.2.2 and 6.5.2 for a discussion of the Containment Spray System. Credit is taken for the Containment Spray System as a safety-related fission product removal system. Assumptions related to the containment in the design basis LOCA dose analysis are provided in Chapter 15.

A non-nuclear safety airborne radioactivity removal system is provided for the Containment to maintain the fission product activity at low level for safe personnel entry during normal operation. The system is discussed in Section 9.4.3.

6.5.3.2 Secondary Containment

HBR 2 does not utilize a secondary containment system.

REFERENCES: SECTION 6.5

None

6.6 IN-SERVICE INSPECTION OF CLASS 2 AND 3 COMPONENTS

The Inservice Inspection (ISI) Program for HBR 2 is in accordance with the applicable rules and requirements of the ASME Section XI Code, Rules for Inservice Inspection of Nuclear Power Plant Components. This program is required by 10 CFR 50, Section 55a(g). The ISI program complies with the Edition and/or Addenda of the ASME Section XI Code specified in 10 CFR 50.55a(b)(2) or as approved by the Nuclear Regulatory Commission (NRC).

The Edition(s) and/or Addenda(s) of the ASME Section XI Code applicable to the ISI program are defined in site administrative procedures.

The boundary of Class 2 and 3 systems or portions thereof subject to examination and/or testing are illustrated on the applicable system's Piping and Instrument Diagram (P&ID). Inspection boundaries for major components are illustrated by the use of flags, which graphically define the system boundaries that are subject to the ASME Section XI Code rules and requirements.

In response to IE Bulletin 79-17, the nondestructive examination (NDE) program to be implemented for the Class 2 and Class 3 portions of systems containing stagnant borated water is presented in References 3.9.6-1 and 3.9.6-2. No evidence of intergranular stress corrosion cracking was found during any of the inspections (Reference 3.9.6-2).

Written relief requests are granted for deviations to the ASME Section XI Inservice Inspection requirements by the Commission pursuant to 10 CFR 50, Section 55a(g)(6)(i). These relief requests are identified in the ISI program.

The requirements for Inservice Inspection (ISI) of Class 1 components are described in Section 5.2.4.

The requirements for Inservice Testing (IST) of Class 1, 2, and 3 components are described in Section 3.9.6.

6.7 MAIN STEAM LINE ISOLATION VALVE LEAKAGE CONTROL SYSTEM (MSIV-LCS)

This section does not apply to HBR 2.

6.8 ISOLATION VALVE SEAL WATER SYSTEM

6.8.1 DESIGN BASIS

The Isolation Valve Seal Water System (IVSW) assures the effectiveness of certain containment isolation valves during any condition which requires containment isolation, by providing a water seal at the valves. These valves are located in lines that are connected to the Reactor Coolant System (RCS), or that could be exposed to the containment atmosphere in the event of a loss-of-coolant accident (LOCA). The system provides a simple and reliable means for injecting seal water between the seats and stem packing of the globe and double disc types of isolation valves, and into the piping between closed diaphragm type isolation valves. This system operates to limit the fission product release from the containment.

Although no credit was taken for operation of this system in the calculation of offsite accident doses, it does provide assurance that, should an accident occur, the containment leak rate is lower than that assumed in the accident analysis - as indicated by the results of the Unit Integrated Leak Rate tests (Section 6.2.6).

6.8.2 System Design

6.8.2.1 System Description

The IVSW system flow diagram is shown in Figure 6.8.2-1.

System operation is initiated either manually or by any automatic safety injection (SI) signal. When actuated, the IVSW System interposes water inside the penetrating line between two isolation points located outside the containment. The resulting water seal blocks leakage of the containment through valve seats and stem packing. The water is introduced at a pressure greater than the containment design basis accident pressure of 40.5 psig. The possibility of leakage from the containment or the RCS past the first isolation point is thus prevented by assuring that if leakage does exist, it will be from the seal water system into the containment. Service is discontinued after the manual reset buttons for PCV-1922 A and B are reset after a containment isolation Phase A reset.

The system includes one seal water tank capable of supplying the total requirements of the system. The design data for the tank are given in Table 6.8.2-1. The tank is pressurized with a nitrogen blanket supplied from two independent sources. Primary supply is from the plant nitrogen supply header through a pressure regulating control valve. Automatic backup supply is provided from two high pressure nitrogen bottles through separate high and low pressure regulating valves. Design pressure of the tank and piping is 150 psig. The injection piping runs and the piping from the nitrogen supply bottles are fabricated using 3/8 in. OD stainless steel tubing, which is capable of 2500 psig service. Relief valves are provided to prevent over-pressurization of the system if a pressure control valve fails, or if a seal water injection line communicates with a high pressure line due to a check valve failure in the seal water line. The seal water tank requires no external power source to maintain the required driving pressure.

Local instrumentation is also provided, as shown in Figure 6.8.2-1. The primary source of N_2 from the plant N_2 supply header is backed up by two, independent, high pressure N_2 bottles. If there should be a break or failure of the N_2 header, the N_2 blanket pressure is maintained by the tanks and blowdown through the N_2 header is prevented by check valves.

The tank supplies pressurized water to four distribution headers. Header "A" is the manual header, meaning an isolation valve on this header must be pressurized by opening a manual valve supplying the individual isolation valve. Headers "B", "C", and "D" are automatic headers that are pressurized through one or both of two redundant, fail-open, air-operated valves in parallel. These valves open on receipt of an SI signal. A loss of power will cause the automatic valves to open, since automatic initiation is a de-energized signal to vent air from the valve operators. System operation is initiated by a Phase A containment isolation signal which accompanies any SI signal. System operation is discontinued after the manual reset buttons for valves PCV-1922 A and B are reset after the Phase A reset.

Liquid carrying piping two inches and larger with design pressure or temperature exceeding 200 psig or 200°F is typically isolated by one manual or remote-operated, double disc gate valve. A drawing of this valve is presented in Figure 6.8.2-2. Redundant isolation barriers are provided when the valve is closed. The upstream and downstream discs are forced against their respective seats by the closing action of the valve. Seal water is injected through the valve bonnet or body and pressurizes the space between the two valve discs.

The seal water pressure in excess of the potential accident pressure eliminates any outleakage past the first isolation point.

For smaller lines, isolation is typically provided by two globe valves in series with the seal water injected into the pipe between the valves. The valves are oriented such that the seal water wets the stem packing. When the valves are closed for containment isolation, the first isolation point is the valve plug in the valve closest to containment, and the water seal is applied between the valve plug and stem packing. In a number of the smaller lines, isolation is provided by two diaphragm (Saunders Patent) valves in series, with the seal water injected into the pipe between the valves.

The original design of the IVSW System was based on the conservative assumption that all containment isolation valves serviced by the IVSW System are leaking at 50 cc/hr/inch of nominal pipe diameter.

In addition, should one of the isolation valves fail to close, flow through the failed valve will be limited by a restricting orifice to a maximum leakage value of 63,200 cc/hr. A water seal at the failed valve is assured by proper slope of the protected line, or a loop seal, or by additional valves on the side of the isolation valves away from the containment.

The sizing of the seal water tank was originally based on providing at least a 24 hr supply of seal water under the following adverse circumstances: isolation valves leaking at the design rate of 50 cc/hr/in. plus the failure of the largest containment isolation valve to seat, resulting in leakage at the maximum rate of 1000 cc/hr/in. The seal water tank is sized to satisfy these conditions. However, during the worst case scenario involving a containment phase ‘A’ isolation signal, several containment isolation valves will remain open until receipt of a containment phase ‘B’ isolation signal. In this condition, the IVSW tank inventory may be depleted in approximately 90 minutes following the onset of the event. Two separate, independent, seismically qualified sources of makeup water (primary water and service water) are provided to ensure that an adequate supply of seal water is available for long-term operation. Service water makeup is from two sources - the service water header, and from each of the service water booster pumps. This assures a redundant long-term supply of water from a source at greater than the 1.1 times the containment design pressure (approximately 46.2 psig). Based on maximum leakage and flows into the tank from makeup sources, use of the makeup source would be required for only minimal amounts of time each day at very low flows which will not affect other functions of the makeup system.

The IVSW tank water volume required by Technical Specification SR 3.6.8.2 will provide sufficient for a minimum of 24 hours provided the total IVSW header leakage meets the requirements of Technical Specification SR 3.6.8.6.

6.8.2.2 Isolation Valve Seal Water Actuation Criteria

Containment isolation and seal water injection are accomplished automatically for certain penetrating lines requiring early isolation, and manually for others, depending on the status of the system being isolated and the potential for leakage in each case.

The automatically operated containment isolation valves are actuated to the closed position by one of two separate containment isolation signals. The first of these signals is derived in conjunction with automatic SI actuation,

and trips the majority of the remotely operated isolation valves. These valves are in the so-called "non-essential" process lines penetrating the containment. This is defined as the Containment Isolation Phase “A” Signal (T-Signal).

This signal also initiates automatic seal water injection. The second, or "Phase B" containment isolation signal, is derived upon actuation of the Containment Spray System, and actuates the remotely operated containment isolation valves in the so-called "essential" process lines penetrating the containment. This signal is designated by the letter "P".

A manual containment isolation signal or SI signal can be generated from the Control Room. This signal performs the same functions as the automatically derived "T" signal, i.e., "Phase A" isolation and automatic seal water injection.

Generally, the following criterion determines whether the isolation and seal water injection is automatic or manual. Automatic containment isolation and automatic seal water injection are required for lines that could communicate with the containment atmosphere and be void of water following a LOCA.

These lines include:

1. Reactor coolant pump seal water return line (Phase B isolation)

2. Letdown line

3. RCS sample lines

4. Reactor coolant vent line

5. Reactor coolant drain tank gas analyzer line.

Automatic containment isolation and automatic seal water injection are also provided for the following lines, which are not connected directly to the RCS, but terminate inside the containment at certain components. These components can be exposed to the reactor coolant or to the containment atmosphere as the result of leakage or failure of a related line or component. The isolation lines are not required for post-accident service.

These lines include:

1. Pressurizer relief tank gas analyzer line

2. Pressurizer relief tank makeup line

3. SI System test line

4. Reactor coolant drain tank pump discharge line

5. Steam generator blowdown lines

6. Steam generator blowdown sample lines

7. Accumulator sample line, and

8. Containment sump pump discharge.

Manual containment isolation and manual seal water injection are provided for designated lines that are normally filled with water and will remain filled following the LOCA, and for lines that must remain in service for a time following the accident. The manual seal water injection assures a long-term seal. These lines include:

1. Reactor coolant pump seal water supply lines

2. Charging line

3. SI hot leg header, and

4. Containment spray headers.

Seal water injection is not necessary to ensure the integrity of isolated lines in the following categories:

1. Lines that are connected to non-radioactive systems outside the containment, and in which a pressure gradient exists that opposes leakage from the containment. These include nitrogen supply lines to the pressurizer relief tank, accumulators, the reactor coolant drain tank, the instrument air header, the pressurizer deadweight tester line, and the plant air header.

2. Lines that do not communicate with the containment atmosphere or RCS and are missile-protected throughout their length inside containment. These lines are not postulated to be severed or otherwise opened to the containment atmosphere as a result of a LOCA. These include the steam and feedwater headers, the containment ventilation system cooling water supply and return lines, and the excess letdown heat exchanger cooling water supply and return lines. The reactor coolant pump cooling water supply and return lines are also included in this category; however, seal water injection is provided. Reference 6.2.4-1 provides additional details.

3. Lines that are designed for long-term, post-accident service as part of the engineered safety features. The only lines in this category are the containment sump recirculation lines. These lines are connected to a closed system outside containment.

4. Special lines such as the fuel transfer tube, containment purge ducts, and the containment pressure and vacuum relief lines. These lines are tested as per 10 CFR 50, Appendix, J.

6.8.2.3 Components

A description of the materials and criteria for IVSW components, piping and structures may be found in Section 6.1.1.1.6.

TABLE 6.8.2-1

ISOLATION VALVE SEAL WATER TANK

Material	ASTM A-240
Design Pressure, psig	150
Design Temperature, °F	200
Operating Pressure, psig	50-100
Operating Temperature, °F	Ambient
Code	ASME Code, Section VIII

6.8.3 DESIGN EVALUATION

The IVSW System provides an extremely prompt and reliable method of limiting the fission product release from the containment isolation valves in the event of a LOCA.

The employment of the system during a LOCA, while not considered for analysis of the consequences of the accident, provides an additional means of conservatism in ensuring that leakage is minimized. No detrimental effect on any other safeguards system will occur should the seal water system fail to operate.

The IVSW System can operate and meet its design function without reliance on any other system. Electric power is not required for system operation, although instrument power is required to provide indication in the Control Room of seal water tank pressure and level.

6.8.3.1 System Response

Automatic containment isolation will be completed within approximately two seconds following generation of the phase A containment isolation signal. This is the estimated closing time of the air operated containment isolation valves. Since the IVSW System is actuated by this signal, automatic seal water injection will be in effect within this time period.

Subsequent generation of the phase B isolation signal on containment high pressure (spray actuation signal) will close a number of motor operated isolation valves with typical closing time of 10 sec. Automatic seal water injection flow will have been initiated in advance of this signal by the phase A signal.

The operator has the ability to override containment isolation valves as necessary; for example, the isolation valves in the steam generator blowdown lines and valves in those systems required for post-accident operation. (Refer to Section 6.3).

6.8.3.2 Single Failure Analysis

A single failure analysis is presented in Table 6.8.3-1. The analysis shows that the failure of any single active component will not prevent fulfilling the design function of the system.

TABLE 6.8.3-1

SINGLE FAILURE ANALYSIS-ISOLATION VALVE SEAL WATER SYSTEM

COMPONENT		MALFUNCTION	COMMENTS
A.	Automatically Operated Isolation Valves for Injection Headers (Open on Phase A Containment Isolation Signal)	Fails to open	Two provided. Operation of one required.
B.	Instrumentation		
	1. Level Transmitter	Fails	Local level indicator at tank also provided
	2. Pressure Transmitter	Fails	Local pressure indicator at tank also provided.
	3. Pressure Regulator	Fails to open	
	a) In Plant N_2 Supply Header		a) Automatic backup supply from two high pressure N_2 bottles through separate high and low pressure regulating valves.
	b) In N_2 Header between N_2 Bottles and Seal Water Injection Tank		b) N_2 header manually cross-connected to separate regulator to bypass failed regulator.
C.	Plant N_2 Supply	Loss of main header	Backup supply from two N_2 bottles through separate regulators as discussed in Item B. N_2 bottles manually cross-connected with N_2 header pressure regulator.
D.	Isolation Valve Supplied by Automatic Seal Water Injection	Fails to close	Restricting orifice limits flow. System capacity is designed for the largest isolation valve failing to close with no loss of system function.

6.8.4 TESTS AND INSPECTION

The IVSW System is required to be operable by the HBR 2 Technical Specifications, Appendix A to Facility Operating License No. DPR-23, and is functionally tested at each refueling outage. Section 3.9.6, In-Service Testing of Pumps and Valves, contains additional information regarding testing and inspection of the IVSW System.

6.8.5 Instrumentation Requirements

The sections below provide information regarding instrumentation indicators, setpoints, and operation.

6.8.5.1 Instrumentation Indicators and Setpoints

Remote indications are:

1. IVSW tank level indicated on RTGB from LT-1912
2. IVSW tank pressure indicated on RTGB from PT-1911
3. IVSW valves PCV-1922A and B indication on RTGB containment isolation Phase "A" Panel open or closed.

Local indications are:

1. IVSW tank level LI-1912
2. IVSW tank pressure PI-1910
3. IVSW tank sight glass LG-1913
4. IVSW header "A" pressure PI-1915
5. IVSW header "B" pressure PI-1916
6. IVSW header "C" pressure PI-1917
7. IVSW header "D" pressure PI-1918
8. IVSW header "A" flow indicator FI-1914
9. IVSW header "B" flow indicator FI-1919
10. IVSW header "C" flow indicator FI-1920, and
11. IVSW header "D" flow indicator FI-1921

The following is a list of instrumentation that supply alarms; their setpoints will be found in the annunciator procedure.

CONTROLLER NO.	WINDOW NAME	WINDOW NO.
LIT-1912	Seal Water Injection Tank Low Level	APP-007-E6
PT-1911	Seal Water Injection Tank Low Pressure	APP-007-D6
PC-1059/PC-1060	N_2 Header Pressure	APP-036-C8

6.8.5.2 Instrumentation Operation

IVSW System operation modes are:

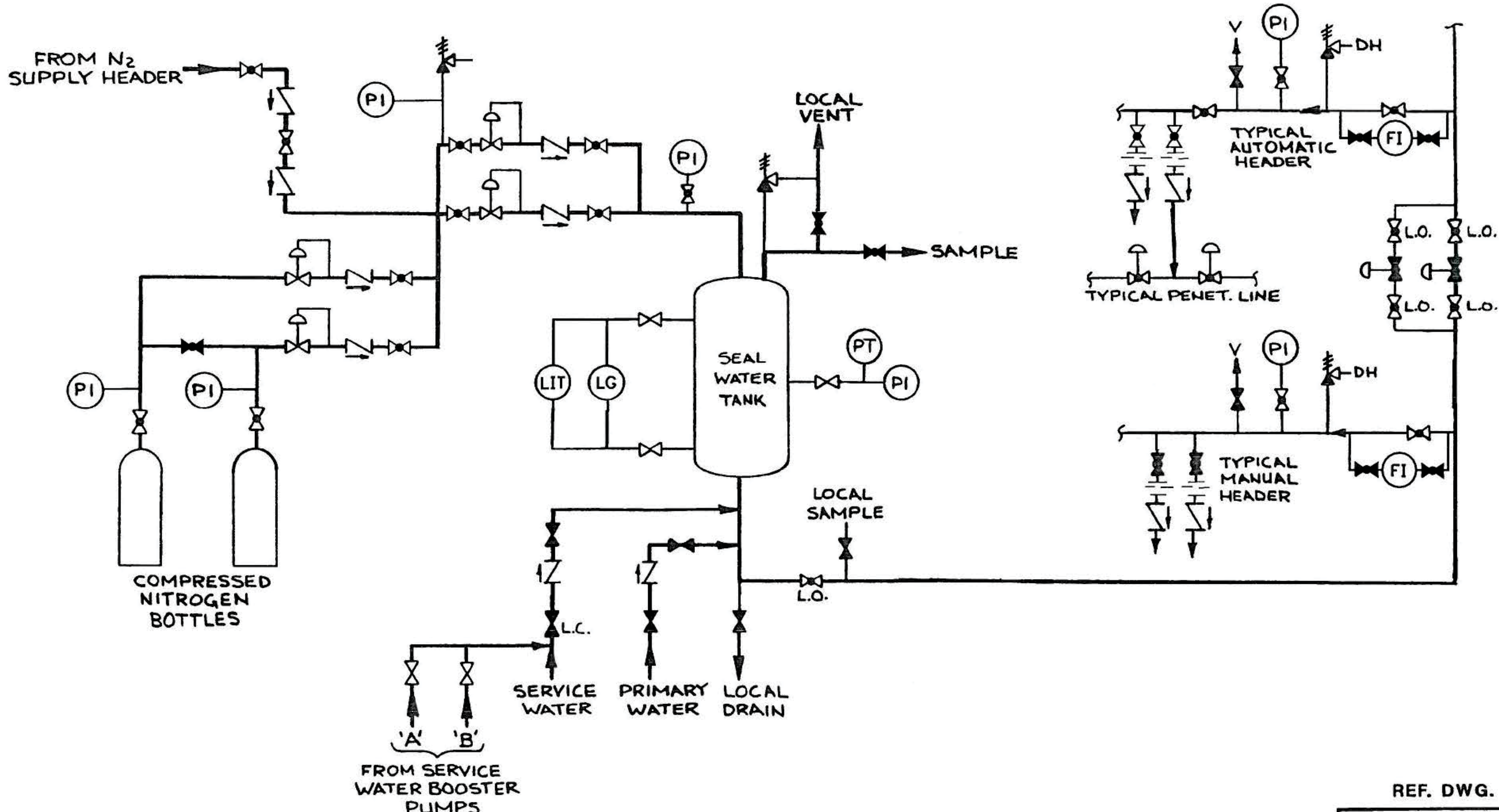
1. Automatic Operation - The isolation valve seal water system is normally in a static condition, with the seal water injection tank pressurized to 54 psig. A low pressure alarm at 51 psig and a low level alarm at 70 percent full are provided in the Control Room on the RTGB.

   A SI or containment phase "A" signal will de-energize EV-1922 A and B which opens PCV-1922 A and B and injects seal water at 54 psig to distribution manifolds 1919, 1920, and 1921. For the list of systems and piping supplied by each manifold, refer to System Description SD-035.

2. Manual Operation - The isolation valve seal water system may be initiated manually by pushing the SI or containment isolation buttons on the RTGB. This action will put manifolds 1919, 1920, and 1921 in service. Manifold 1914 may be put in service anytime the seal water injection tank is pressurized. To inject seal water via manifold 1914, the isolation valves must be opened manually. Normally the only time manifold 1914 would be used is when post-accident equipment is secured.

3. Terminating System Operation - If the isolation valve seal water system was actuated by SI or containment isolation phase "A" signals, its operation may be terminated at the discretion of the operator by pressing the reset buttons for valves PCV-1922 A and B after a containment isolation Phase "A" signal reset.

To terminate service from Manifold 1914, the isolation valves must be closed locally at the manifold.



Revision No. 22

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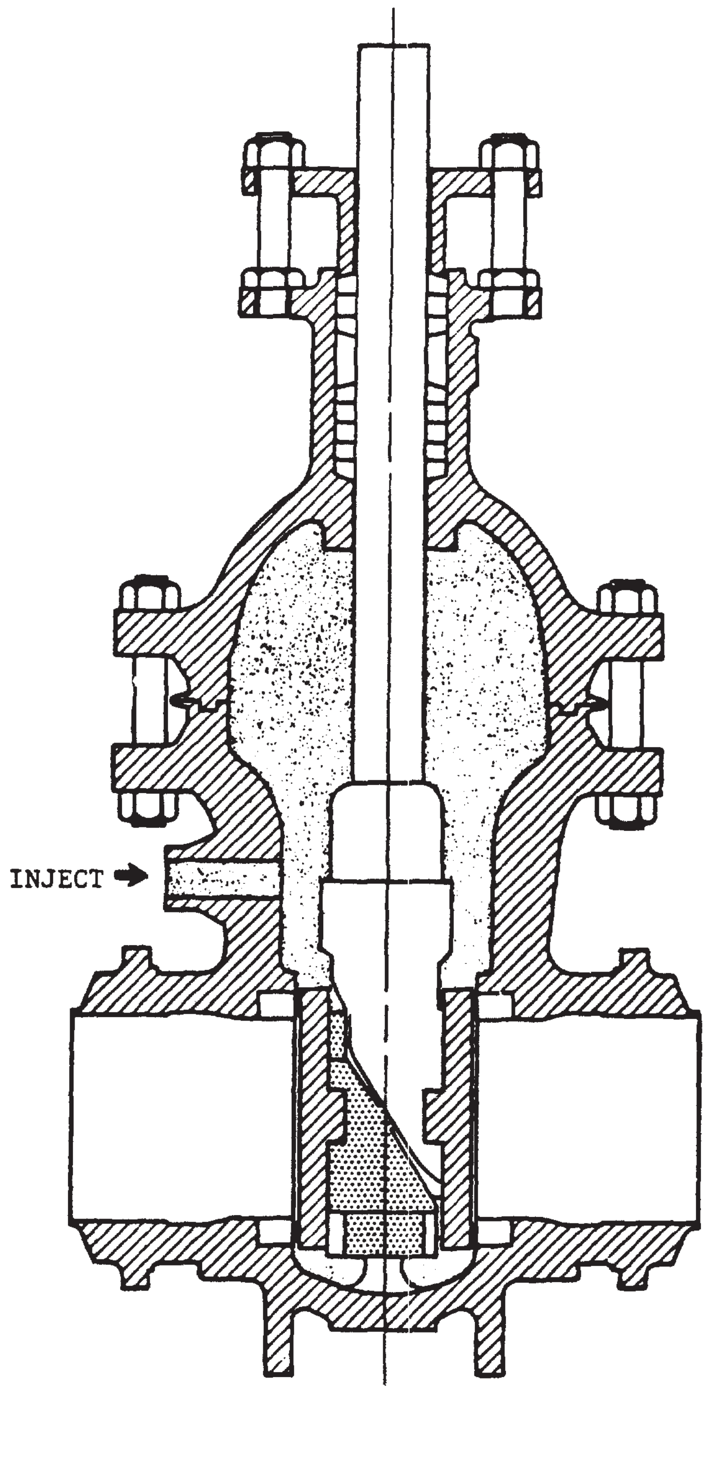
Carolina Power & Light Company
UPDATED FINAL SAFETY ANALYSIS REPORT

FLOW DIAGRAM
ISOLATION VALVE SEAL WATER SYSTEM
FIGURE 6.8.2-1



H. B. ROBINSON UNIT 2 Carolina Power & Light Company UPDATED FINAL SAFETY ANALYSIS REPORT	DOUBLE DISK ISOLATION VALVE WITH SEAL WATER INJECTION	FIGURE 6.8.2 - 2

6.9 Containment Penetration Pressurization System

6.9.1 Design Basis

The Containment Penetration Pressurization System (PPS) provides a means of testing pressure zones incorporated into the containment penetrations. It was originally designed to provide a means of continuously pressurizing the positive pressure zones in order to maintain these zones above the maximum containment post-accident pressure and to provide a means for continuous or intermittent monitoring of the leakage status of the containment penetrations.

ESR/MOD 95-00888 removed the automatic continuous pressurization and monitoring features of this system. It is now only used during power operation to test the personnel hatch and during outages to test containment penetrations (local leak rate tests). The system is capable of providing continuous pressurization should the need arise.

In the cartridge type electrical penetrations, the entire cartridge is pressurized. In the capsule type electrical penetrations, only the sealing head assembly is pressurized.

No credit is taken for system operation in calculation of off-site doses. It is designed as a Seismic Class I system.

6.9.2 System Design

6.9.2.1 System Description

The Containment Penetration Pressurization System utilizes a regulated supply of clean and dry compressed air from the instrument air system, which is backed up by the service air system, to test all containment penetrations (only the sealing head assembly is pressurized in the CAPSULE type electrical penetrations). The system is capable of demonstrating compliance with Local Leak Rate Surveillance testing requirements in accordance with 10 CFR 50, Appendix J. Typical piping and electrical penetrations are described in Section 3.8.1.

The primary source of air for this system is the 100 psig instrument air system (Section 9.3). Two instrument air compressors are used, although only one is required to maintain pressurization at the maximum allowable leakage rate of the pressurization system. The service air compressor acts as a backup to the instrument air compressors (Section 9.3).

A standby source of gas pressure for the system is provided by a bank of nitrogen cylinders. These will deliver nitrogen at a slightly lower pressure (approximately 44 psi) than the normal regulated air supply pressure of approximately 46 psig.

Leakage from the system and potential leakage from penetrations are determined by measurement of the air flow.

During Appendix J testing, pressurization of each penetration can be verified by closing off its air supply line, and opening a test connection at the penetration to observe the escape of the pressurizing medium.

6.9.2.2 Containment Inleakage

Assuming a continuous inleakage to the containment from the penetration pressurization system of 0.02 percent of the containment free volume per day, the calculated time for the containment pressure to rise by 0.3 psig is approximately 25 days. Therefore inleakage is not considered to be an operating or safety problem. From the standpoint of allowable pressure, a much greater inleakage would be permitted. The activity of the air in the containment is limited during normal operation through the use of two containment charcoal auxiliary filter units. Each unit contains high efficiency particulate air (HEPA) and charcoal filters, and permits containment overpressure relief, as required, through the pressure relief line to the plant vent. The containment pressure relief line is also equipped with HEPA and charcoal filters.

6.9.2.3 Components

All associated components, piping, and structures, of the Containment Penetration Pressurization System were designed to Seismic Class I criteria. Refer to Section 6.1.1.7.

For a description of the instrument air compressors and the service air compressors, refer to Service Air System, Section 9.

The nitrogen cylinders used are designed in accordance with the requirements of Section VIII (Unfired Pressure Vessels) of the ASME Boiler and Pressure Vessel Code. The cylinders are designed for 2200 psig maximum pressure. A total of 17,350 scf of nitrogen is required to provide a 24-hour backup supply based on a PPS leakage rate of 0.2% of the containment volume per day, at the containment design pressure, if PPS is providing a continuous pressurization function.

TABLE 6.9.2-1

CONTAINMENT PENETRATION
PRESSURIZATION AIR RECEIVERS

Material	ASTM A-285-C
Design pressure, psig	140
Design temperature, ºF	200
Operating pressure, psig	100
Operating temperature, ºF	100
Code	ASME Code, Section VIII

6.9.3 Design Evaluation

The Penetration Pressurization system is not considered in the analysis of the consequences of the accident.

6.9.3.1 System Response

Each plenum can be isolated and its leak tightness measured by the absolute pressure-volume method. A leakage sensitivity of the order of 0.2 ft^3/day/penetration at design pressure can be readily obtained in this way. The absolute accuracy of this test method is highly variable, especially if tests are made during reactor operation, due to non-isothermal conditions in the penetrations. On the average, an accuracy of 40 ft^3/day is a reasonable estimate, which would correspond to 0.002 percent of the net containment volume per day in this plant.

6.9.3.2 Reliance on Interconnected Systems

The Containment Penetration Pressurization System can operate and meet its design function without reliance on any other system, except as limited by air compressor availability following depletion of all reserves in the air receivers and backup nitrogen cylinders. Electric power is not necessary for operation of the system.

6.9.4 Inspections and Tests

6.9.4.1 Inspections

The system components located outside the containment can be visually inspected at any time. Components inside the containment can be inspected during shutdown. All pressurized zones have provisions for either local pressure indication outside the containment or remote low pressure alarms in the Control Room.

6.9.4.2 Testing

No special testing of system operation or components is necessary. The PPS system supports 10 CFR 50, Appendix J Testing. The exception to this is the CAPSULE type electrical penetrations in which only the sealing head assembly is pressurized. The welded interface between the containment and the penetration is inspected in accordance with LLRT and ILRT requirements.