CHAPTER 6
ENGINEERED SAFETY FEATURES

6.0 INTRODUCTION

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

1. Core design to preclude release of fission products from the fuel (Chapter 3).

2. Retention of fission products in the reactor coolant for whatever leakage occurs (Chapters 4 and 6).

3. Retention of fission products by the containment for operational and accidental releases beyond the reactor coolant boundary (Chapters 5 and 6).

4. Optimizing fission product dispersal to minimize population exposure (Chapters 2 and 11).

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

The engineered safety features systems in this plant are the containment system, detailed in Chapter 5, the safety injection system, detailed in Section 6.2, the containment spray system, detailed in Section 6.3, the containment air recirculation cooling system, detailed in Section 6.4, the isolation valve seal-water system, detailed in Section 6.5, and the containment penetration and weld channel pressurization system, detailed in Section 6.6.

Evaluations of techniques and equipment used to accomplish the central objective including accident cases are detailed in Chapters 5, 6, and 14.

6.1 GENERAL DESIGN CRITERIA

Criteria applying in common to all engineered safety features are given in Section 6.1.1. Thereafter, criteria that are related to engineered safety features, but which are more specific to other plant features or systems, are listed and cross referenced in Section 6.1.2.

6.1.1 Engineered Safety Features Criteria

6.1.1.1 Engineered Safety Features Basis for Design

Criterion: Engineered safety features shall be provided in the facility to back up the safety provided by the core design, the reactor coolant pressure boundary, and their protection systems. Such engineered safety features shall be designed to cope with any size reactor coolant piping break up to and including the equivalent of a circumferential rupture of any pipe in that boundary, assuming unobstructed discharge from both ends. (GDC 37)

The design, fabrication, testing, and inspection of the core, reactor coolant pressure boundary, and their protection systems give assurance of safe and reliable operation under all anticipated

normal, transient, and accident conditions. However, engineered safety features are provided in the facility to back up the safety provided by these components. These engineered safety features have been designed to cope with any size reactor coolant pipe break up to and including the circumferential rupture of any pipe assuming unobstructed discharge from both ends as discussed in Section 14.3.3.3. They are also designed to cope with any steam or feedwater line break up to and including the main steam or feedwater lines as discussed in Section 14.2.5.

Limiting the release of fission products from the reactor fuel is accomplished by the safety injection system, which by cooling the core, keeps the fuel in place and substantially intact and limits the metal water reaction to an insignificant amount.

The safety injection system consists of high- and low-head centrifugal pumps driven by electric motors and of passive accumulator tanks that are self-energized and which act independently of any actuation signal or power source.

The release of fission products from the containment is limited in three ways:

1. Blocking the potential leakage paths from the containment. This is accomplished by:

    a. A steel-lined, reinforced-concrete reactor containment with testable, double-sealed penetrations, and liner weld channels, the spaces of which are continuously pressurized above accident pressure and which form a virtually leaktight barrier to the escape of fission products should a loss-of-coolant accident occur. (Section 6.6.2 lists those portions of the Weld Channel Pressurization System that have been disconnected because repairs have been determined not to be practical.)

    b. Isolation of process lines by the containment isolation system, which imposes double barriers in each line that penetrates the containment except for lines used during the accident. Pipes penetrating the containment are sealed as shown in Table 5.2-1. This table presents the sealing method for all containment piping penetrations and valving.

2. Reducing the fission product concentration in the containment atmosphere. This is accomplished by containment spray, which removes elemental iodine vapor and particulates from the containment atmosphere by washing action. The spray is chemically treated during the recirculation phase to enhance iodine retention.

3. Reducing the containment pressure and thereby limiting the driving potential for fission product leakage. This is accomplished by cooling the containment atmosphere by the following independent systems of approximately equal heat removal capacity that together also function to ensure the containment design criteria is maintained even with an assumed single failure:

    a. Containment spray system.
    b. Containment air recirculation cooling system.

6.1.1.2 Reliability and Testability of Engineered Safety Features

Criterion: All engineered safety features shall be designed to provide such functional reliability and ready testability as is necessary to avoid undue risk to the health and safety of the public. (GDC 38)

A comprehensive program of plant testing was formulated for all equipment, systems, and system control vital to the functioning of engineered safety features. The program consisted of performance tests of individual pieces of equipment in the manufacturer's shop, and integrated tests of the systems as a whole. Periodic tests of the actuation circuitry and mechanical components ensure reliable performance, upon demand, throughout the life of the plant.

The initial tests of individual components and the integrated test of the system as a whole complemented each other to ensure the performance of the system as designed and to prove proper operation of the actuation circuitry.

Routine periodic testing of the engineered safety features components is performed, in accordance with the Technical Specifications.

6.1.1.3 Missile Protection

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

A loss-of-coolant accident or other plant equipment failure might result in dynamic effects or missiles. For engineered safety features that are required to ensure safety in the event of such an accident or equipment failure, protection is provided primarily by the provisions that are taken in the design to prevent the generation of missiles. In addition, protection is also provided by the layout of plant equipment or by missile barriers in certain cases. (Refer to Section 5.1.2 for a discussion of missile protection.)

Injection paths leading to unbroken reactor coolant loops are protected against damage resulting from the maximum reactor coolant pipe rupture by layout and structural design considerations. Injection lines penetrate the main missile barrier, which is the crane wall, and the injection headers are located in the missile protected area between the crane wall and the containment wall. Individual injection lines, connected to the injection header, pass through the barrier and then connect to the loops. The separation of the individual injection lines is provided to the maximum extent practicable. The movement of the injection line, associated with a rupture of a reactor coolant loop, is accommodated by line flexibility and by the design of the pipe supports such that no damage outside the missile barrier is possible.

The containment structure is capable of withstanding the effects of missiles originating outside the containment and which might be directed toward it so that no loss-of-coolant accident can result.

All hangers and anchors are designed in accordance with USAS B31.1, Code for Pressure Piping. This code provides minimum requirements on material, design, and fabrication with ample safety margins for both dead and dynamic loads over the life of the equipment.

Concrete missile barriers, bumpers, walls and other concrete structures are designed in accordance with ACI 318-63, Building Code Requirements for Reinforced Concrete.

In 1989, the NRC approved changes to the design basis with respect to dynamic effects of postulated primary loop ruptures, as discussed in Section 4.1.2.4

6.1.1.4 Engineered Safety Features Performance Capability

Criterion: Engineered safety features, such as the emergency core cooling system and the containment heat removal system, shall provide sufficient performance capability to accommodate the failure of any single active component without resulting in undue risk to the health and safety of the public. (GDC 41)

Each engineered safety feature provides sufficient performance capability to accommodate any single failure of an active component and still function in a manner to avoid undue risk to the health and safety of the public.

The extreme upper limit of public exposure is taken as the levels and time periods presently outlined in 10 CFR 50.67 (i.e., 25 rem total effective dose equivalent (TEDE) at the exclusion radius for the worst two hour interval, 25 rem TEDE over the duration of the accident at the low-population-zone distance and 5 rem TEDE for the operators in the control room for the duration of the accident). The accident condition considered is the hypothetical case of a release of fission products per NUREG-1465. Also, the total loss of all outside power is assumed concurrently with this accident. With minimum engineered safety features systems functioning, the offsite exposure would be within 10 CFR 50.67 limits as discussed in Section 14.3.6.

Under these accident conditions, the containment air recirculation cooling system and the containment spray system are designed and sized so that both systems, each operating with partial effectiveness, are able to supply the necessary postaccident cooling capacity to ensure the maintenance of containment integrity, that is, keeping the pressure below design pressure at all times assuming that the core residual heat is released to the containment as steam. Partial effectiveness is defined as the operation of a system with at least one active component failure. Containment spray relies on a sufficient amount of passive sodium tetraborate stored in containment to raise the pH of the recirculating solution for continued iodine removal following an accident. The containment spray system alone is able to supply the post accident iodine removal required to restrict the offsite exposure to within 10 CFR 50.67 limits.

6.1.1.5 Engineered Safety Features Components Capability

Criterion: Engineered safety features shall be designed so that the capability of these features to perform their required function is not impaired by the effects of a loss-of-coolant accident to the extent of causing undue risk to the health and safety of the public. (GDC 42)

Instrumentation, pumps, fans, cooling units, valves, motors, cables, and penetrations located inside the containment are selected to meet the most adverse accident conditions to which they may be subjected. These items are either protected from containment accident conditions or are designed to withstand, without failure, exposure to the worst combination of temperature, pressure, and humidity expected during the required operational period.

In response to NRC Generic Letter 95-07, safety-related power-operated gate valves have been evaluated for susceptibility to pressure locking and thermal binding. The results of this evaluation identified that those potential conditions will not prevent the plant from achieving a safe shutdown, as all valves evaluated remain operable. This conclusion is based upon valve design, plant configuration during normal, accident, and post accident operating modes and sufficient actuator thrust to open the valve. The details of the system and valve evaluations are documented in Reference 1. Subsequent to the original evaluations and Reference 1, since the position of valve MOV-744 during the recirculation phase of a LOCA could be open or closed depending on the location of a postulated single passive failure, its position during recirculation was changed from normally closed/open when required to normally open/close when required. The option of using procedural controls to avoid opening a valve under potential pressure locking conditions is permitted by the Generic Letter.

In response to NRC Generic Letter 96-06, isolated pipe line segments that penetrate containment have been analyzed to evaluate their susceptibility to overpressurization caused by thermal expansion of the contained fluid in the event of a design basis accident. The results show that potential overpressurization will not cause lines to fail; all remain operable. Those that are protected by safety relief devices were further evaluated for the effects of stuck-open relief valves under accident conditions. No failure modes were identified that would adversely affect the ability for safety-related systems to perform their intended functions during accidents.

The safety injection system pipes serving each loop are restrained in such a manner as to restrict potential accident damage to the portion of piping downstream of the crane wall that constitutes the missile barrier in each loop area. The restraints are designed to withstand, without failure, the thrust force of any branch line severed from the broken reactor coolant pipe and discharging fluid to the atmosphere and to withstand a bending moment equivalent to that which produces a failure of the piping under the action of free-end discharge to the atmosphere or a motion of the broken reactor coolant pipe to which the injection pipes are connected. This prevents possible failure at any point upstream from the support point, including the branch line connection into the piping header.

In 1989, the NRC approved changes to the design basis with respect to dynamic effects of postulated primary loop ruptures, as discussed in Section 4.1.2.4.

Designated valves that are located in areas that would have excessive radiation levels in the event of a release of fission products from the core are provided with capability for remote operation.

6.1.1.6 Accident Aggravation Prevention

Criterion: Protection against any action of the engineered safety features, which would accentuate significantly the adverse after effects of a loss of normal cooling shall be provided. (GDC 43)

The introduction of borated cooling water into the core results in a negative reactivity addition. The control rods insert and remain inserted.

The supply of water by the safety injection system to cool the core cladding does not produce significant metal-water reaction (<1.0-percent).

The delivery of cold safety injection water to the reactor vessel following accidental expulsion of reactor coolant does not cause further loss of integrity of the reactor coolant system boundary.

6.1.1.7 Sharing of Systems

Criterion: Reactor facilities may share systems or components if it can be shown that such sharing will not result in undue risk to the health and safety of the public. (GDC 4)

The engineered safety features at Indian Point 2 do not share systems or components with other units.

6.1.2 Related Criteria

The following are criteria, which although related to all engineered safety features, are more specific to other plant features or systems, and, therefore, are discussed in other chapters, as listed.

Name	Discussion
Quality Standards (GDC 1)	Chapter 4
Performance Standards (GDC 2)	Chapter 4
Records Requirements (GDC 5)	Chapter 4
Instrumentation and Control Systems (GDC 12)	Chapter 7
Engineered Safety Features Protection Systems (GDC 15)	Chapter 7
Emergency Power (GDC 39)	Chapter 8

REFERENCES FOR SECTION 6.1

1. NRC Letter dated May 20, 1999, Jeffrey F. Harold to A. Alan Blind, Subject: Safety Evaluation of Generic Letter 95-07, "Pressure Locking and Thermal Binding of Safety-Related Power-Operated Gate Valves," for Indian Point Nuclear Generating Unit No.2 (TAC M93473).

6.2 SAFETY INJECTION SYSTEM

At Indian Point Unit 2 the emergency core cooling function is performed by the safety injection system. Therefore, whenever the term "emergency core cooling system" or ECCS is referenced in the document it is synonymous with the safety injection system.

6.2.1 Design Basis

6.2.1.1 Emergency Core Cooling System Capability

Criterion: An emergency core cooling system with the capability for accomplishing adequate emergency core cooling shall be provided. This core cooling system and the core shall be designed to prevent fuel and clad damage that would interfere with the emergency core cooling function and to limit the clad metal-water reaction to acceptable amounts for all sizes of breaks in the reactor coolant piping up to the equivalent of a double-ended rupture of the largest pipe.

The performance of such emergency core cooling system shall be evaluated conservatively in each area of uncertainty. (GDC 44)

Adequate emergency core cooling is provided by the safety injection system (which constitutes the emergency core cooling system) whose components operate in three modes. These modes are delineated as passive accumulator injection, active safety injection, and residual heat removal recirculation.

The primary purpose of the safety injection system is the automatic delivery of cooling water to the reactor core in the event of a loss-of-coolant accident. This limits the fuel clad temperature and thereby ensures that the core will remain intact and in place, with its essential heat transfer geometry preserved. This protection is afforded for:

1. All pipe break sizes up to and including the hypothetical instantaneous circumferential rupture of a reactor coolant loop, assuming unobstructed discharge from both ends.

2. A loss of coolant associated with the rod ejection accident.

3. A steam-generator tube rupture.

The basic design criteria for loss-of-coolant accident evaluations prior to codification under 10 CFR 50.46 were as follows:

1. The cladding temperature is to be less than:

    a. The melting temperature of Zircaloy-4.
    b. The temperature at which gross core geometry distortion, including clad fragmentation, may be expected.

2. The total core metal-water reaction will be limited to less than 1-percent.

These criteria ensure that the core geometry remains in place and substantially intact to such an extent that effective cooling of the core is not impaired.

Subsequently, the basic design criteria for loss-of-coolant accident calculations have been revised to those required under 10 CFR 50.46.

For any rupture of a steam pipe and the associated uncontrolled heat removal from the core, the safety injection system adds shutdown reactivity so that with a stuck rod, no offsite power, and minimum engineered safety features, there is no consequential damage to the reactor coolant system and the core remains in place and coolable as discussed in Section 14.2.5.

Redundancy and segregation of instrumentation and components are incorporated to ensure 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 station auxiliary power coincident with the loss of coolant, and is tolerant of failures of any single component or instrument channel to respond actively in the system. During the recirculation phase, the system is tolerant of a loss of any part of the flow path since backup alternative flow path capability is provided as described in Section 6.2.3.3.

The ability of the safety injection system to meet its capability objectives is presented in Section 6.2.3. The analysis of the accidents is presented in Chapter 14.

6.2.1.2 Inspection of Emergency Core Cooling System

Criterion: Design provisions shall, where practical, be made to facilitate inspection of physical parts of the emergency core cooling system, including reactor vessel internals and water injection nozzles. (GDC 45)

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

6.2.1.3 Testing of Emergency Core Cooling System Component

Criterion: Design provisions shall be made so that components of the emergency core cooling system can be tested periodically for operability and functional performance. (GDC 46)

The design provides for periodic testing of active components of the safety injection system for operability and functional performance.

Power sources are arranged to permit individual actuation of each active component of the safety injection system.

The safety injection pumps and residual heat removal pumps can be tested periodically during plant operation using the minimum flow recirculation lines provided. The residual heat removal pumps are used every time the residual heat removal loop is put into operation. All remote-operated valves can be exercised, and actuation circuits can be tested either during normal operation or routine plant maintenance.

6.2.1.4 Testing of Emergency Core Cooling System

Criterion: Capability shall be provided to test periodically the operability of the emergency core cooling system up to a location as close to the core as is practical. (GDC 47)

An integrated system test can be performed when the plant is cooled down and the residual heat removal loop is in operation. This test would not introduce flow into the reactor coolant system, but would demonstrate the operation of the valves, pump circuit breakers, and automatic circuitry upon the initiation of safety injection.

Level and pressure instrumentation is provided for each accumulator tank, and accumulator tank pressure and level are continuously monitored during plant operation. Flow from the tanks can be checked at any time using test lines as described in Section 6.2.5.3.1.

6.2.1.5 Testing of Operational Sequence of Emergency Core Cooling System

Criterion: Capability shall be provided to test initially, under conditions as close as practical to design, the full operational sequence that would bring the emergency core cooling system into action, including the transfer to alternate power sources. (GDC 48)

The design provides for the capability to test initially, to the extent practical, the full operational sequence up to the design conditions for the safety injection system to demonstrate the state of readiness and capability of the system. Details of the operational sequence testing are presented in Section 6.2.5.

6.2.1.6 Codes and Classifications

Table 6.2-1 lists the codes and standards to which the safety injection system components are designed.

6.2.1.7 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. Per the 12/06/04 NRC generic SER on NEI 04-07 (Reference 5), evaluations of PWR post accident emergency recirculation performance to address the potential impact of debris blockage per GSI-191 (Reference 4) and Generic Letter 2004-02 (Reference 3) will use a mission time of 30 days.

6.2.2 System Design And Operation

6.2.2.1 System Description

Adequate emergency core cooling following a loss-of-coolant accident is provided by the safety injection system shown in Plant Drawing 9321-2735 [Formerly UFSAR Figure 6.2-1]. Plant Drawing 235296 [Formerly UFSAR Figures 6.2-2] and Figures 6.2-2 through 6.2-5 depict how this system concept is translated into plant layout design. The system components operate in the following possible modes:

1. Injection of borated water by the passive accumulators.

2. Injection by the safety injection pumps drawing borated water from the refueling water storage tank.

3. Injection by the residual heat removal pumps also drawing borated water from the refueling water storage tank.

4. Recirculation of spilled reactor coolant, injected water, and containment spray system drainage back to the reactor from the recirculation sump by the recirculation pumps. The residual heat removal pumps provide backup recirculation capability through the independent containment sump as described in Section 6.2.3.3.

The initiation signal for core cooling by the safety injection pumps and the residual heat removal pumps is the safety injection signal, which is described in Section 7.2.3.2.3.

6.2.2.1.1 Injection Phase

The principal components of the safety injection system, which provide emergency core cooling immediately following a loss of coolant are the accumulators (one for each loop), the three safety injection (high-head) pumps, and the two residual heat removal (low-head) pumps. The safety injection and residual heat removal pumps are located in the primary auxiliary building.

The accumulators, which are passive components, discharge into the cold legs of the reactor coolant piping when pressure decreases to the minimum Technical Specification value, thus rapidly ensuring core cooling for large breaks. They are located inside the containment, but outside the crane wall, therefore each is protected against possible missiles.

The safety injection signal opens certain of the safety injection system isolation valves, provides confirmatory open signals to system isolation valves that are normally open, and starts the safety injection pumps and residual heat removal pumps.

The three safety injection pumps (high-head) deliver borated water to two separate discharge headers. The flow from the discharge headers can be injected into the four cold legs and two hot legs of the reactor coolant system. The motor-operated isolation valves in the four cold-leg injection lines are open during normal plant operation. The motor-operated isolation valves in the two hot-leg injection lines are closed during normal plant operation. The hot-leg injection lines are provided for later use during hot-leg recirculation following a reactor coolant pressure boundary break. The high-head safety injection system is configured with two cold leg injection lines physically connected to the reactor coolant pressure boundary and the other two lines connected to the accumulator discharge lines upstream of the pressure boundary. Since a small break in the reactor coolant pressure boundary can include a cold leg injection line, safety injection flow capability can be limited by the resulting flow from only three intact cold leg injection lines. Depending on the assumed single failure, either two or three safety injection pumps can be operating. To maximize the fraction of safety injection flow delivered to the reactor coolant system with a broken cold leg injection line, the four cold leg injection lines are flow balanced to within an allowable range. The resulting system flow capability is sufficient for the makeup of coolant following a small break that does not immediately depressurize the reactor coolant system to the accumulator discharge pressure. Credit is not taken for operator action to isolate a broken cold leg injection line.

For large breaks, the reactor coolant system would be depressurized and voided of coolant rapidly (about 26 sec for the largest break as shown in Figure 14.3-12) and a high flow rate is required to recover quickly the exposed fuel rods and limit possible core damage as discussed in Section 14.3.3.3.1. To achieve this objective, one residual heat removal pump and two safety injection pumps are required to deliver borated water to the cold legs of the reactor coolant loops. Two residual heat removal and three safety injection pumps are available to provide for an active component failure. Delivery from these pumps supplements the accumulator discharge. Since the reactor coolant system backpressure is relatively low (rapid depressurization for large breaks), a broken injection line would not appreciably change the flows in the other injection line's delivery to the core.

The residual heat removal pumps take suction from the refueling water storage tank. In addition, the charging pumps of the chemical and volume control system are available but are not required to augment the flow of the safety injection system.

Because the injection phase of the accident is terminated before the refueling water storage tank is completely emptied, all pipes are kept sufficiently filled with water before recirculation is initiated to ensure the systems remain operable and perform properly. Water level indication and alarms on the refueling water storage tank give the operator ample warning to terminate the injection phase. Additional level indicators and alarms are provided in the recirculation and containment sumps, which also give backup indication when injection can be terminated and recirculation initiated.

6.2.2.1.2 Recirculation Phase

After the injection operation, coolant spilled from the break and water collected from the containment spray are cooled and returned to the reactor coolant system by the recirculation system.

When the break is large, depressurization occurs due to the large rate of mass and energy loss through the break to containment. In the event of a large break, the recirculation flow path is within the containment. The system is arranged so that the recirculation pumps take suction from the recirculation 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 also arranged to allow either of the residual heat removal pumps to take over the recirculation function. The residual heat removal pumps would only be used if backup capacity to the internal recirculation loop is required as described in Section 6.2.3.3. Water is delivered from the containment to the residual heat removal pumps from the separate containment sump inside the containment.

For small breaks, the depressurization of the reactor coolant system is augmented by steam dump from and auxiliary feedwater addition to the steam generators. For the smaller breaks in the reactor coolant system where recirculated water must be injected against higher pressures for long-term cooling, the system is arranged to deliver the water from residual heat removal heat exchanger 21 to the high-head safety injection pump suction and by this external recirculation route to the reactor coolant loops. If this flow path is unavailable, an alternate flow path is provided as indicated in Table 6.2-11. Thus, if depressurization of the reactor coolant system proceeds slowly, the safety injection pumps may be used to augment the flow-pressure capacity of the recirculation pumps in returning the spilled coolant to the reactor. In this system configuration, the recirculation pump (or residual heat removal pump) provides flow and net positive suction head to the operating safety injection pumps. To prevent safety injection pump flow in excess of its maximum allowable (i.e., runout) limit, variable flow orifices are installed at the discharge of the safety injection pumps and the hot and cold leg motor-operated isolation valves are preset with mechanical stops based on data from operational flow testing to limit system maximum flow capability.

The recirculation pumps, the residual heat removal heat exchangers, piping, and valves vital to the function of the recirculation loop are located in a missile-shielded space inside the polar crane support wall on the west side of the reactor primary shield.

There are two sumps within the containment, the recirculation sump and the containment sump. Both sumps collect liquids discharged into the containment during the injection phase of the design-basis accident.

As part of the resolution of GSI-191 and Generic Letter 2004-02, various flow channeling barriers are installed in the Vapor Containment, EL 46’-0” to force the recirculation flow into the Reactor Cavity Sump area, up and out the Incore Instrumentation Tunnel, through the Crane Wall via the three nominal 20 inch square openings and into the annulus area outside the Crane Wall. The recirculation flow will migrate towards the Recirculation Sump Strainer or the Containment Sump Strainer depending on which pump(s) are operating. Flow channeling barriers are installed on the Reactor Cavity Platform, EL 29’-4”, around the Incore Instrumentation Tunnel, on the Recirculation Sump Trenches, at the Containment Sump, and on Crane Wall penetrations up to the flood level. Flow channeling barrier doors are installed in the Northeast and Northwest quadrant openings of the Crane Wall. In addition, flow channeling barrier doors are installed in the North and South entrances to the Recirculation Sump area. Perforated plate is installed on the RHR Heat Exchanger Platform, EL 66’-0” to preclude debris from washing through the existing grating and into the Recirculation Sump area. Forcing the recirculation flow path into the Reactor Cavity Sump area (a low velocity zone) allows the larger debris an opportunity to settle.

The Recirculation Sump and Containment Sump strainers consist of a matrix of multi-tube top-hat modules, which are fabricated from perforated stainless steel plate and mounted in the horizontal position. The perforated plate has 3/32” diameter holes sized to limit downstream affects. The top-hat modules have four (4) layers of perforated surfaces for straining debris from the sump fluid. Typical Recirculation Sump and Containment Sump strainer top-hat modules consist of a 12-1/2” diameter outer perforated tube with a respective 10-1/2” diameter inner perforated tube and a second set of tubes, which consist of a 7-1/2” diameter outer perforated tube with a respective 5-1/2” diameter inner perforated tube. The top-hat modules feature an internal vortex suppressor, which prevents air ingestion into the piping system. Stainless steel mesh has been installed between each pair of perforated plate tubes to minimize fiber bypass through the strainers. The top-hat modules are attached to strainer water boxes. Frame structures supporting sections of grating are installed above the Internal Recirculation and Containment Sump strainers including the sump strainer extension in the containment annulus providing for additional vortex suppression function. The Containment Sump Level Detection System is discussed in Section 6.7.2.13.

The Recirculation Sump relies on two, connected water boxes with 249 top-hat modules in the sump pit for the purpose of preventing particles greater than 3/32” in diameter from entering the suction of the recirculation pumps. The recirculation sump strainer has an effective surface area of ~3,156 square feet and an effective interstitial volume of ~476 cubic feet. Water will enter the top-hat modules through the perforated plates and flow through the stainless steel mesh inside either of the two (2) annuli flow paths within each top-hat module. Upon exiting the top-hat modules, water will flow into either of the two connected strainer water boxes, flow over the Recirculation Sump weir wall and into the Recirculation Pump Bay towards the pumps.

The Containment Sump relies on a water box with 23 top-hat modules in the Containment Sump pit and a plenum extension out into the annulus with a water box with an additional 40 top-hat modules for the purpose of preventing particles greater than 3/32” in diameter from entering the Containment Sump suction line to the RHR Pumps. The Containment Sump strainer has a combined effective surface area of ~1182 square feet (~412 sq. ft. for sump pit

and ~770 sq. ft. for the annulus extension) and an effective combined interstitial volume of ~161 cubic feet. Water will enter the top-hat modules through the perforated plates and flow through the stainless steel mesh inside either of the two (2) annuli flow paths within each top-hat module. Upon exiting the top-hat modules, water will flow into the strainer water box, which is connected to the Containment Sump suction line and the RHR System. The containment sump level detection system is discussed in Section 6.7.1.2.13.

Each sump strainer is qualified to handle the post LOCA design basis accident debris loads predicted by the mechanistic evaluations required by GL 2004-02. There are two classifications for the debris generated by an RCS break: 1) conventional debris (e.g., insulation, tags, coatings, dust and dirt), 2) chemical debris (principally the precipitation of Aluminum based compounds Sodium Aluminum Silicate and Aluminum Oxy Hydroxide) which are conservatively predicted by use of a model detailed in WCAP-16530-NP-A (Reference 6). An Argonne National Laboratory (ANL) formula was used to predict the post-LOCA chemical precipitation temperature. The precipitation temperature is determined from the post-accident containment sump pool conditions (Aluminum concentration, temperature and pH). Chemical precipitants are not predicted to develop prior to the required switchover to hot-leg recirculation. Consequently, the internal recirculation sump strainer qualification uses predicted head losses associated with conventional debris loads up to the switchover to hot-leg recirculation and then conventional and chemical debris loads after the transfer to hot-leg recirculation occurs when reduced sump flow rates are expected to be less than two HHSI pumps at runout (2 x 675 GPM).

Per the 12/06/04 NRC generic SER on NEI 04-07 (Reference 5), evaluations of PWR post accident emergency recirculation performance to address the potential impact of debris blockage per GSI-191 (Reference 4) and Generic Letter 2004-02 (Reference 3) will use a mission time of 30 days. The internal recirculation sump strainers are qualified for a GL 2004-02 defined 30-day mission time. The containment sump strainers were qualified using the same methodology but are qualified from 24 hours post large break LOCA until the end of the 30 day mission time. The containment sump is not required to handle the same full debris loads at the start of the recirculation phase as the recirculation sump since the only postulated failure that would require its use is a passive failure, which is only postulated after 24 hours into the event (reference Technical Specification Amendment #257). However, to maintain redundancy for the more probable small break LOCAs, the containment sump strainers have been qualified for 6 inch diameter breaks and smaller from the start of the recirculation phase. A condition of Amendment #257 was that the Emergency Operating Procedures continue to utilize the containment sump as an alternative path should both the recirculation sump trains become unavailable.

As identified above in the strainer descriptions, both sump strainers are constructed of concentric cylindrical tubes perforated with 3/32 inch diameter holes and have stainless steel mesh behind the perforations to reduce the quantity of fine fibers able to pass through the strainer perforations, although some fine fibers and particulates may still pass through the strainer. The passive sump strainers provide adequate protection to the downstream components from the majority of the accident generated debris. As part of the resolution of GL 2004-02, an analysis of the components downstream of the strainers required for accident mitigation was performed to ensure satisfactory operation for the defined 30-day mission time. Pumps, isolation and throttle valves, orifices, instrument connections, and piping were examined using the guidance provided by revision 1 of WCAP-16406 (Reference 7) with justification provided for any methodology deviations. All equipment was found to have

sufficient clearance: to allow passage of debris, to limit blockage to an acceptable level, and / or to have sufficient resistance to wear as to not affect there function for the defined 30-day mission time. Chemical effects on the fuel elements were also examined and not predicted to interfere with heat transfer per analysis based on WCAP-16793-NP (Reference 8).

The low-head external recirculation loop via the containment sump line and the residual heat removal pumps provides backup recirculation capability to the low-head internal recirculation loop. The containment sump line has two remote motor-operated normally closed valves located outside the containment and a remote motor-operated butterfly valve inside containment. The high-head external recirculation flow path via the high-head safety injection pumps is required for the range of small-break sizes for which the reactor coolant system pressure remains in excess of the shutoff head of the recirculation pumps at the end of the injection phase. The recirculation pumps, or residual heat removal pumps if backup capability is required, are also used to provide flow to the high-head safety injection pumps during hot leg recirculation.

The external recirculation flow paths within the primary auxiliary building are designed so that external recirculation can be initiated immediately after the accident. Those portions of the safety injection system outside of the containment, which are designed to circulate, under postaccident conditions, radioactivity contaminated water collected in the containment meet the following requirements:

1. Shielding to limit radiation levels.
2. Collection of discharges from pressure-relieving devices into closed systems.
3. Means to detect and control radioactivity leakage into the environs.

These criteria are met by minimizing leakage from the system. External recirculation loop leakage is discussed in Section 6.2.3.8. The radiological consequences of external recirculation loop leakage following a design basis accident are presented in Section 14.3.6.6. Detection and control of leakage from external recirculation loop components is also discussed in Section 6.7.

One recirculation pump and one residual heat exchanger of the recirculation system provide sufficient cooled recirculated water to keep the core flooded with water by injection through the cold-leg connections while simultaneously providing, sufficient containment recirculation spray flow to reduce containment airborne activity. These systems are kept sufficiently filled with water to ensure the systems remain operable and perform properly. Three of the five fan cooler units prevent the containment pressure from rising above design limit (the minimum containment safeguards case which assumes the single failure of one emergency diesel generator is discussed in Section 14.3.5). Analysis demonstrates that flow will be determined by system resistance provided by the physical configuration of the recirculation piping and components, and will be hydraulically balanced such that sufficient flow is established to the core and the spray header. Only one pump and one heat exchanger are required to operate for this capability at the earliest time recirculation spray is initiated. With both recirculation pumps in operation and both spray header valves open, a recirculation spray flow rate can be established such that no containment cooling fans (Section 6.4) are required. Likewise with five containment cooling units in operation, no containment spray is required to maintain containment pressure below its design limit. The system is also arranged to allow either of the residual heat removal pumps to take over the recirculation function following a passive failure

as defined in Section 6.2.3.3. This design ensures that heat removal from the core and containment is effective in the event of a pipe or valve body rupture.

6.2.2.1.3 Cooling Water

The service water system (Section 9.6) provides cooling water to the component cooling loop, which in turn cools the residual heat exchangers, both of which are part of the auxiliary coolant systems (Section 9.3). Three non-essential service water pumps are available to take suction from the river and discharge to the two component cooling heat exchangers. Three component cooling pumps are available to discharge through their heat exchangers and deliver to the two residual heat exchangers. During the recirculation phase following a loss-of-coolant-accident, only one residual heat removal heat exchanger, one recirculation or residual heat removal pump, one non-essential service water pump, one component cooling water pump, one auxiliary component cooling water pump, and one component cooling water heat exchanger are required to meet the core-cooling function. The auxiliary component cooling water pump is required only to support the function of the recirculation pump. With the exception of the residual heat removal heat exchangers and the recirculation pumps, all of the cited equipment is located outside of the containment.

6.2.2.1.4 Changeover From Injection Phase to Recirculation Phase

Assuming that the three high-head safety injection pumps, the two residual heat removal pumps, and the two containment spray pumps (Section 6.3) are running at their maximum capacity, the time sequence, from the time of the safety injection signal, for the changeover from injection to recirculation in the core of a large rupture is as follows:

1. In approximately 15 min, sufficient water has been delivered to provide the required net positive suction head to start the recirculation pumps.

2. In approximately 20 min, (a) low-level alarms on the refueling water storage tank sound, and (b) the redundant containment and recirculation sump level indicators show the sump water level. The alarm(s) serve to alert the operator to start the switchover to the recirculation mode. The redundant containment and recirculation sump level indicators provide verification that the refueling water storage tank water has been delivered during the injection phase, as well as giving consideration to the case of a spurious (i.e., early) refueling water storage tank low-level alarm. The operator would see on the control board that the redundant sump level indications are at the appropriate points; switchover via the eight-switch sequence is performed at that time.

3. With the initiation of the eight-switch sequence (i.e., switch No. 1), only one spray pump will continue in operation. This spray pump will continue to draw from the refueling water storage tank until the level drops below 2 feet.

Recirculation pump motors are 2-ft 2-in. above the highest water level after the addition of the injected water to the spilled coolant.

The changeover from injection to recirculation takes place when the level indicator or level alarms on the refueling water storage tank indicate that the fluid has been injected. The level indicators in the containment sump will also verify that the level is sufficient within the

containment. The sequence is followed regardless of which power supply is available. All switches are grouped together on the safeguard control panel. The component position lights verify when the function of a given switch has been completed. Should an individual component fail to respond, the operator can take corrective action to secure appropriate response from the backup component. The manual switchover by the operator:

1. Terminates safety injection signal in order that the control logic permits manipulation of the system (at any time following completion of the auto-start sequence).

2. Closes switches one and three (removes and isolates unnecessary loads from the diesels).

   Switch One:

   a. Trips one (i.e., pump 22) of three high-head safety injection pumps if all three are operating (no action if two are operating), and isolates the pump suction to the refueling water storage tank if the tripped pump is the middle safety injection pump (i.e. pump 22).

   b. Trips one containment spray pump if both are operating (no action if one is operating).

   c. Closes isolation valves at the inoperative spray pump discharge.

   Switch Three:

   a. Trips both residual heat removal pumps.

   b. Sends close signals to isolation valves in the residual heat removal pump suction and discharge headers, which are administratively reenergized later in the sequence. (Technical Specifications require the motor operators for these valves to be deenergized.)

3. Closes switch two (establishes cooling flow for residual heat removal heat exchangers)

   a. Starts one service water pump, non-essential header (the second or third pump is given a start signal if the first or second pump fails to start).

   b. Starts one component cooling pump (the second or third pump is given a start signal if the first or second pump fails to start).

4. Isolates one RHR heat exchanger flow path. (if both are open)

5. Manually initiates internal recirculation flow

   a. Manually starts recirculation pump 21 (if pump 21 fails to start, uses manual start on pump 22). (Pump 22 control switch is adjacent to switch four).

   b. Closes switch four to open valves on discharge of recirculation pumps. Starting a recirculation pump prior to closing switch four minimizes the potential pressure differential across these motor operated valves.

6. Checks flow to reactor coolant system via the low-head injection lines to ensure minimum flow requirements are established. If minimum flow requirements are established, the closes switch seven and switch eight to establish low-head recirculation.

   If minimum flow requirements are not established, then closes switch six and switch eight to establish high-head recirculation.

7. Close switch six (supplies recirculation for reactor coolant system pressures greater than 150 psig, which impedes flow via the low-head injection lines).

   a. Opens valves 888A and 888B to provide a flow path from the recirculation pump discharge to the high-head safety injection pump suction.

   b. Activates the low-pressure alarm circuit off of PT-947. If for some reason PT-947 alarm is not activated by this switch (RS-6), the operator can switch "HI HEAD PUMP LO SUCTION PRESS ALARM" to activate this alarm. This latter switch is on the safeguards panel.

   c. Closes valves 842 and 843 (high-head pump test line) (if their control feed interlock switches were first placed to the "OFF" position), and sends a close signal to valves 746 and 747 (residual heat removal heat exchanger discharge).

8. Closes switch seven (removes the two running safety injection pumps from service since they are no longer needed).

9. Closes switch eight (completes the isolation of the safety injection system and containment spray system lines from the refueling water storage tank).

   a. Closes the valve on the spray test line.

   b. Sends a close signal to the valve in the safety injection pumps suction line from the refueling water storage tank, which is administratively reenergized later in the sequence. (Control power for this valve is deenergized in accordance with Technical Specifications requirements).

10. Close switch five (Establishes additional cooling capability if adequate power is available i.e. all diesel breakers are either open or racked out, or at least one breaker from each of the three diesels is racked in and closed).

a. Starts second service water pump, non-essential header (the third pump is given the start signal if the second pump fails to start).

b. If (a) completed, starts second component cooling pump (the third pump is given the start signal if the second pump fails to start).

c. If (b) completed, starts recirculation pump 22 (unless already running). *[**Note**: running two (2) recirculation pumps is restricted to Low Head Recirculation. If High Head Recirculation is required, operator action is taken to prevent two recirculation pumps from operating simultaneously.]*

Although the listed switches are manual, each automatically causes the operations listed. An indicating lamp is provided to show the operator when the operations of a given switch have been performed. In addition, lamps indicating completion of the individual functions for a given switch are provided. These lamps are adjacent to the switches. Should an individual component fail to respond, the operator can take corrective action to secure appropriate response from controls within the control room.

Remote-operated valves for the injection phase of the safety injection system (Plant Drawings 9321-2735 and 235296 [Formerly Figure 6.2-1]), which are under manual control (i.e., valves, which normally are in their ready position and do not receive a safety injection signal) have their positions indicated on a common portion of the control board. 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 6.2-2 lists the instrumentation readouts on the control board and assessment panel, which the operator can monitor during recirculation. In addition, an audible annunciation alerts the operator to the condition.

6.2.2.1.5 Location of the Major Components Required for Recirculation

The residual heat removal pumps are located in the residual heat removal pump room, which is on the basement floor of the primary auxiliary building (elevation 15-ft). The residual heat exchangers are on a platform above the basement floor in the containment building (elevation 66-ft).

The recirculation pumps are directly above the recirculation sump in the containment building (elevation 46-ft).

The component cooling pumps and heat exchangers are located in the primary auxiliary building (elevations 68-ft and 80-ft, respectively).

The auxiliary component cooling water pumps are located in the Mezzanine area of the piping penetration area at elevation 68-ft.

The service water pumps are located at the river water intake structure, and the redundant piping to the component cooling heat exchangers is run underground, until it surfaces just prior to its penetrating the Primary Auxiliary Building exterior wall.

6.2.2.2 Steam Line Break Protection

A large break of a steam system pipe rapidly cools the reactor coolant causing insertion of reactivity into the core and the depressurization of the system. Compensation is provided by the injection of boric acid from the refueling water storage tank. The analysis of the steam line rupture accident is presented in section 14.2.5.

6.2.2.3 Components

All associated components, piping, structures, and power supplies of the safety injection system are designed to seismic Class I criteria.

All components inside the containment are capable of withstanding or are protected from differential pressure that may occur during the rapid pressure rise to 47 psig in 10 sec.

Electrical equipment that has been determined to be important to safety and located in potentially harsh environments are environmentally qualified to ensure performance of their safety function under post-accident temperature, pressure, and humidity conditions.

Emergency core cooling components are either austenitic or an equivalent corrosion-resistant stainless steel, and hence, are compatible with the spray solution over the full range of exposure in the post-accident regime. Corrosion tests performed with simulated spray indicated negligible attack, both generally and locally, in stressed and unstressed stainless steel at containment and emergency core cooling system conditions. These tests are discussed in Reference 1.

The quality standards of all safety injection system components are given in summary form in Table 6.2-3.

6.2.2.3.1 Accumulators

The accumulators are pressure vessels filled with borated water and pressurized with nitrogen gas. During normal plant operation, each of the four accumulators is isolated from the reactor coolant system by two check valves in series. Should the reactor coolant system pressure fall below the accumulator pressure, the check valves open and borated water is forced into the cold legs of the reactor coolant system. Mechanical operation of the swing-disk check valves is the only action required to open the injection path from the accumulators to the core via the cold leg.

The level of borated water in each accumulator tank is adjusted remotely as required during normal plant operations. Refueling water is added using the accumulator topping pump (or safety injection pump 22 or 23). Water level is reduced by draining to the reactor coolant drain tank or through the chemistry sampling panel. Samples of the solution in the tanks are taken at the sampling station for periodic checks of boron concentration. Pressure is adjusted by adding nitrogen as required.

The accumulators are passive engineered safety features since the gas forces injection and no external source of power or signal transmission is needed to obtain fast-acting, high-flow capability when injection is required. One accumulator is attached to each of the four cold legs of the reactor coolant system.

The design capacity of the accumulators is based on the assumption that flow from one of the accumulators spills onto the containment floor through the ruptured loop. The flow from the three remaining accumulators provides sufficient water to fill the volume outside of the core barrel below the nozzles, the bottom plenum, and approximately one-half the core.

To assure the independence of the accumulators from each other, operating procedures require that only one liquid fill valve and only one nitrogen stop valve can be open at a time when reactor temperature is equal to or greater than 350°F.

The accumulators are carbon steel, internally clad with stainless steel and designed to ASME Section III, Class C. Connections for remotely draining or filling the fluid space, during normal plant operation, are provided.

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 6.2-4.

6.2.2.3.2 Boron Injection Tank

The boron injection tank has been removed.

6.2.2.3.3 Refueling Water Storage Tank

In addition to its normal duty to supply borated water to the refueling canal for refueling operations, this tank provides borated water to the safety injection pumps, the residual heat removal pumps, and the containment spray pumps for the Loss-of-Coolant Accident. These systems are kept sufficiently filled with water to ensure the systems remain operable and perform properly. During plant operation, this tank is aligned to these pumps.

The capacity of the refueling water storage tank is based on the requirement for filling the refueling canal; a minimum of 345,000 gal is required by the Technical Specifications to be maintained in the refueling water storage tank. This capacity provides an amount of borated water to assure:

1. A sufficient volume of water on the floor to permit the initiation of recirculation (246,000 gal).

2. A volume water sufficient to allow time for completing the switchover to recirculation and securing High Head Injection and Containment Spray Flow from the RWST (60,000 gal).

3. A sufficient volume of water to allow for instrument inaccuracies, additional margin, and for water that is physically unavailable from the bottom of the tank (39,000 gal).

4. The RWST water volume injected into containment, when added to accumulator discharge to the reactor coolant system, assures no return to criticality with the reactor at cold shutdown and no control rods inserted into the core.

The water in the tank is borated to ensure a minimum shutdown margin as discussed in Section 14.1.5.2.1. The maximum boric acid concentration is approximately 1.4 wt percent boric acid. At 32°F, the solubility limit of boric acid is 2.2-percent. Therefore, the concentration of boric acid in the refueling water storage tank is well below the solubility limit at 32°F. Steam heating is provided for the tank, and the outside lines are heat traced to maintain the temperature above freezing.

Each of two redundant channels of refueling water storage tank level instrumentation provide level indication and low-level alarms in the central control room. In addition, a third instrument provides local level indication.

The design parameters are presented in Table 6.2-6.

6.2.2.3.4 Pumps

The three high-head safety injection pumps for supplying borated water to the reactor coolant system are horizontal centrifugal pumps driven by electrical motors. Parts of the pump in contact with borated water are stainless steel or an equivalent corrosion-resistant material. Each safety injection pump is sized at 50-percent of the capacity required to meet the design criteria outlined in Section 6.2.1. The design parameters are presented in Table 6.2-7; Figure 6.2-6 gives the performance characteristics of these pumps.

A minimum flow bypass line is provided on each pump discharge to recirculate flow to the refueling water storage tank in the event the pumps are started with the normal flow paths blocked. Valves in the minimum flow bypass line (which are normally open) are equipped with motor operators. If either valve closes, an alarm annunciates in the control room. Power is de-energized to prevent spurious valve closure.

The safety injection pump bearing oil is cooled by CCW circulating water pumps using component cooling water as a heat sink. The CCW circulating water pumps are directly connected to the injection pump motor shaft. The pump seals are designed to operate during the injection phase without forced component cooling water flow. During the recirculation phase, cooling water is supplied by the component cooling system or alternately from the primary water system. Emergency backup is available via connections to the city water system. The two residual heat removal (low-head) pumps of the auxiliary coolant system are used to inject borated water at low pressure into the reactor coolant system. The two recirculation pumps are used to recirculate fluid from the recirculation sump back to the reactor, to the spray headers, or to suction of the safety injection pumps. The recirculation pumps will only be required to operate during the recirculation phase. In addition, the recirculation pumps are required to be operable for a period of one year. Per the 12/06/04 NRC generic SER on NEI 04-07 (Reference 5), evaluations of PWR post accident emergency recirculation performance to address the potential impact of debris blockage per GSI-191 (Reference 4) and Generic Letter 2004-02 (Reference 3) will use a mission time of 30 days. All four of these pumps are of the vertical centrifugal type driven by electric motors. The recirculation pumps are open suction, well-type pumps. Parts of the pumps, which contact the borated water solution during recirculation are stainless steel or an equivalent corrosion-resistant material. A minimum flow bypass line is provided on the discharge of the residual heat removal heat exchangers to recirculate cooled fluid to the suction of the residual heat removal pumps should these pumps be started with their normal flow paths blocked. There are two normally open motor-operated

valves in this line. The control power to the two normally open motor operated valves is locked open. The emergency procedures ensure that the RHR pumps are not run in parallel for extended time periods with RCS pressure at or above their shutoff head. A minimum flow bypass, discharging back into the recirculation sump, is provided to protect the recirculation pumps should these flow paths be blocked. Valves in these lines are manually operated and are in the open position during normal plant operation. Figures 6.2-7 and 6.2-8 give the performance characteristics of these pumps. The design parameters are presented in Table 6.2-7.

The recirculation pump motors are air-to-water cooled in a similar manner as the containment cooling fan motors described in section 6.4.2.2.5, item 2. The motor fans are integral to the recirculation pump motor shafts. Cooling water to the motor heat exchanger is component cooling water. The sump water cools the pump bearings. The two auxiliary component cooling water pumps are started during the injection phase. However, their function during this phase is not required to protect the recirculation pump motors from the containment atmosphere. Since the recirculation pumps do not operate during injection, their motors do not experience any self-heating. Without this self-induced heat up, the motor's functional capabilities and EQ characteristics have been shown in motor qualification testing to be unaffected by the post-LOCA environment. Even with an auxiliary component cooling water pump running, effectively no motor cooling occurs during the injection phase because the air circulating fans integral to the motor that drive cooling air through the heat exchanger and motor are not operating.

Details of the component cooling pumps and service water pumps, which serve the safety injection system, are presented in Section 9.3 and 9.6, respectively.

The pressure-retaining parts of the high-head safety injection pumps are castings conforming to ASTM A-296, Grade CA-15 or ASME SA-487, Grade CA-6NM. The pressure-retaining parts of the residual heat removal pumps and the recirculation pumps are castings conforming to ASTM A-351, Grade CF-8A (chromium content 21.0 to 22.5%) and ASTM A-351, Grade CF-8, respectively. Stainless steel forgings are procured per ASTM A-182, Grade F304 or F316, or ASTM A-336, Class F8 or F8M, and stainless steel plate is constructed to ASTM A-240, Type 304 or 316. All bolting material conforms to ASTM A-193. Material such as Monel is used at points of close running clearances in the pumps to prevent galling and to ensure continued performance ability in high-velocity areas subject to erosion.

All pressure-retaining parts of the pumps were chemically and physically analyzed, and the results were checked to ensure conformance with the applicable ASTM specification. In addition, all pressure-retaining parts of the pump were liquid-penetrant inspected in accordance with Appendix VIII of Section VIII of the ASME Boiler and Pressure Vessel Code. The acceptance standard for the liquid-penetrant test is 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 include the evaluation of the shaft seal and bearing design to determine whether adequate allowances had been 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 Boiler and

Pressure Vessel Code Welding Qualifications. This requirement also applied 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 min.

Each pump was given a complete shop performance test in accordance with Hydraulic Institute Standards. The pumps were run at design flow and head, shutoff head, and three additional points to verify performance characteristics. Where net positive suction head is critical, this value was established at design flow by means of adjusting suction pressure.

An accumulator topping pump is provided to fill the accumulators rather than using safety injection pump 22 or 23. The pump is a double-diaphragm type with a capacity of approximately 5 gpm (293 gph). It is located in the safety injection system pump area and is operated from a local key-locked push button switch. The topping pump is capable of withstanding the safe shutdown earthquake but does not operate following safety injection actuation.

6.2.2.3.5 Heat Exchangers

The two residual heat exchangers of the auxiliary coolant system are sized for the cooldown of the reactor coolant system. Table 6.2-8 gives the design parameters of the heat exchangers. During the recirculation phase following a loss-of-coolant-accident, only one residual heat removal heat exchanger is required to ensure that heat removal requirements from the core and containment are met.

The ASME Boiler and Pressure Vessel 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 designs of the heat exchangers also conform to the requirements of the Tubular Exchanger Manufacturers 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 the following:

1. Confined-type gaskets, general construction and mounting brackets suitable for the plant seismic design requirements.

2. 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 hot- or cold-formed parts.

3. A hydrostatic test duration of not less than 30 min, the witnessing of hydro and penetrant tests by a qualified inspector, a thorough final inspection of the unit for good workmanship and the absence of any gouge marks or other scars that could act as stress concentration points, and 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 bundle. Each unit has an SA-285, Grade C, carbon steel shell; an SA-234 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.

6.2.2.3.6 Valves

All parts of valves used in the safety injection system in contact with borated water are austenitic stainless steel or an equivalent corrosion-resistant material. The motor operators on the injection line isolation valves are capable of rapid operation. All valves required for the initiation of safety injection or isolation of the system have remote position indication in the control room.

Valving is specified for exceptional tightness. All valves, except those which perform a control function, are provided with backseats that are capable of limiting leakage. The estimated leakage of backseated valves outside containment is provided in Table 6.2-9. Those valves, which are normally open are backseated, except when operational considerations do not allow. [***Note*** *- The following valves may not be backseated based on operational requirements: 744, 850A, 850B, 851A, 851B, 883, 885A, 885B, 887A, 887B, 888A, 888B and 958.*] Normally closed globe valves are installed with recirculation flow under the seat to prevent the leakage of recirculated water through the valve stem packing. Relief valves are totally enclosed. Control and motor-operated valves, 2.5-in. and above, which are exposed to recirculation flow, are generally provided with double-packed stuffing boxes and stem leakoff-connections that are piped to the waste disposal system.

The check valves that isolate the safety injection system from the reactor coolant system are installed immediately adjacent to the reactor coolant piping to reduce the probability of a safety injection line rupture causing a loss-of-coolant accident.

A relief valve is installed in the safety injection pump discharge header discharging to the pressurizer relief tank to prevent overpressure in the lines that have a lower design pressure than the reactor coolant system. RV-855 is a thermal relief valve which protects the Safety Injection System piping and components from overpressurization due to thermal expansion of fluid in the system or from in-leakage of reactor coolant. The setpoint of RV-855 was changed to 1670 psig to ensure that the valve does not lift when operating the SI system at a pressure near the shutoff head of the SI pumps.

The gas relief valves on the accumulators protect them from pressures in excess of the design value.

6.2.2.3.7 Motor-Operated Butterfly Valve (Containment Sump Valve)

The pressure-containing parts (body, disks) of the valves employed in the safety injection system are designed per criteria established by USAS B16.5 or MSS SP-67 specifications. The materials of construction for these parts are procured per ASTM A182, F316 or A351, Grade CF8M or CF8. All material in contact with the primary fluid, except the packing and the liner, is austenitic stainless steel or an equivalent corrosion-resistant material. The liner is EPT-NORDEL (Du Pont). The pressure-containing cast components are radiographically inspected as outlined in ASTM E-71, Class 1 or Class 2. The body and disk are liquid-penetrant inspected in accordance with ASME Boiler and Pressure Vessel Code, Section VIII, Appendix VIII. The liquid-penetrant acceptable standard is as outlined in USAS B31.1, Case N-10.

The entire assembled unit is hydrotested as outlined in MSS SP-67, with the exception that the test is maintained for a minimum period of 30 min. The motor operator is evaluated in accordance with the GL 89-10 Motor Operated Valve Program to assure its capability to meet the required stem torque for opening and closing.

The shaft material is ASTM A276, Type 316, condition B, or precipitation hardened 17-4 pH stainless steel procured and heat treated to Westinghouse specifications. These materials are selected because of their corrosion-resistant, high-tensile properties, and their resistance to surface scoring by the packing.

The motor operator is located above the maximum sump fluid level and therefore is never submerged. 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 disks away from the fore or backseat upon opening or closing. This hammer-blow feature not only impacts the disk but allows the motor to attain its operational speed.

The valve is assembled, hydrostatically tested, seat-leakage tested (fore and back), operationally tested, cleaned, and packaged per specifications. All manufacturing procedures employed by the valve supplier such as welding, repair welding, and testing are submitted to Westinghouse for approval.

The valve operator completes its cycle from one position to the other in approximately 120 sec.

Valves that must function against system pressure are designed such that they function with a pressure drop equal to full system pressure across the valve disk.

6.2.2.3.8 Motor-Operated Gate Valves

The pressure-containing parts (body, bonnet, and disks) of the valves employed in the safety injection system are designed per criteria established by USAS B16.5 or MSS SP-66 specifications. The materials of construction for these parts are procured per ASTM A182, F316 or A351, Grade CF8M or CF8. All material in contact with the primary fluid, except the packing, is austenitic stainless steel or an equivalent corrosion-resistant material. The pressure-containing cast components are radiographically inspected as outlined in ASTM E-71, Class 1 or Class 2. The body, bonnet, and disks are liquid-penetrant inspected in accordance with ASME Boiler and Pressure Vessel Code, Section VIII, Appendix VIII. The liquid-penetrant acceptable standard is as outlined in USAS B31.1, Case N-10.

When a gasket is employed, the body-to-bonnet joint is designed per ASME Boiler and Pressure Vessel Code, Section VIII, or USAS B16.5 with a fully trapped, controlled compression, spiral wound gasket with provisions for seal welding, or of the pressure seal design with provisions for seal welding. The body-to-bonnet bolting and nut materials are procured per ASTM A193 and A194, respectively.

The entire assembled unit is hydrotested as outlined in MSS SP-61, with the exception that the test is maintained for a minimum period of 30 min. The seating design is of the Darling parallel disk design, the Crane flexible wedge design, or the equivalent. These designs have the feature of releasing the mechanical holding force during the first increment of travel. Thus, the motor operator is evaluated in accordance with the GL 89-10 Motor Operated Valve Program to assure its capability to meet the required stem thrust for opening and closing. The disks are guided throughout the full disk travel to prevent chattering and to provide ease of gate movement. The seating surfaces are hard faced (Stellite No. 6 or equivalent) to prevent galling and reduce wear.

The stem material is ASTM A276, Type 316, condition B, or precipitation hardened 17-4 pH stainless steel procured and heat treated to Westinghouse specifications. These materials are selected because of their corrosion-resistant, high-tensile properties, and their resistance to surface scoring by the packing. The valve stuffing box was originally designed with a lantern ring leakoff 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. An alternate packing arrangement may be installed in these valves upon approval for substitution. Experience with designs utilizing live load and graphite packing has been favorable.

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 disks away from the fore or backseat upon opening or closing. This hammer-blow feature not only impacts the disk but allows the motor to attain its operational speed.

The valve is assembled, hydrostatically tested, seat-leakage tested (fore and back), operationally tested, cleaned, and packaged per specifications. All manufacturing procedures employed by the valve supplier such as hard facing, welding, repair welding, and testing are submitted to Westinghouse for approval.

For those valves that must function on the safety injection signal, approximately 10-sec operation is required. For all other valves in the system, the valve operator completes its cycle from one position to the other in approximately 120 sec. Operating times greater than these values are permitted on a case by case basis if properly justified by an individual safety evaluation.

Valves that must function against system pressure are designed such that they function with a pressure drop equal to full system pressure across the valve disk.

6.2.2.3.9 Manual Valves

The stainless steel manual globe, gate, and check valves are designed and built in accordance with the requirements outlined in the motor-operated valve description above with the following exceptions:

1. Alternate materials, evaluated to be equivalent, have been used in some replacement valves.

2. Liquid-penetrant inspection of the body, bonnet, and disks to ASME V Article 6 with acceptance per ASME III has been used on some replacement valves.

The carbon steel valves are built to conform with USAS B16.5. The materials of construction of the body, bonnet, and disk conform to the requirements of ASTM A105, Grade II; A181, Grade II; or A216, Grade WCB or WCC. Alternate materials, evaluated to be equivalent, have been used in some replacement valves. The carbon steel valves pass only nonradioactive fluids and are subjected to hydrostatic test as outlined in MSS SP-61, except that the test pressure is maintained for at least 30 min/in. of wall thickness. Since the fluid controlled by the carbon steel valves is not radioactive, the double packing and seal weld provisions are not provided.

6.2.2.3.10 Accumulator Check Valves

The pressure-containing parts of this valve assembly are designed in accordance with MSS SP-66. All parts in contact with the operating fluid are of austenitic stainless steel or of equivalent corrosion-resistant materials procured to applicable ASTM or WAPD specifications. The cast pressure-containing parts are radiographed in accordance with ASTM E-94 and the acceptance standard as outlined in ASTM E-71, Class 1 or Class 2. The cast pressure-containing parts, machined surfaces finished hard facings, and gasket bearing surfaces are liquid-penetrant inspected per the ASME Boiler and Pressure Vessel Code, Section VIII, and the acceptance standard is as outlined in USAS B31.1, Code Case N-10. The final valve is hydrotested per MSS SP-61, except that the test pressure is maintained for at least 30 min. The seat leakage is conducted in accordance with the manner prescribed in MSS SP-61, except that the acceptable leakage is 2 cm^3/hr-in. nominal pipe diameter.

The valve is 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 is manufactured from 17-4 pH stainless steel heat treated to Westinghouse specifications. The clapper arm shaft bushings are manufactured from Stellite No. 6 material. The various working parts are selected for their corrosion-resistant, tensile, and bearing properties.

The disk and seat rings are manufactured from forgings. The mating surfaces are hard faced with Stellite No. 6 to improve the valve seating life. The flexible disc-hinge connection permits the disc to completely contact the seat even if there is minor seat movement.

The valves are intended to be operated in the closed position with a normal differential pressure across the disk of approximately 1650 psi. The valves remain in the closed position except for testing and safety injection. Since the valve will normally not be required to operate in the open condition, hence be subjected to impact loads caused by sudden flow reversal, 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 loop, a boric acid "freeze up” is not expected with this low a concentration.

The experience derived from the check valves employed in the safety injection system of the Carolina-Virginia Tube Reactor (CVTR) in a similar system has indicated that the system is reliable and workable. The CVTR emergency injection system, maintained at atmospheric conditions, was separated from the main coolant piping by one 6-in. check valve. A leak detection pit was provided in the CVTR to accumulate any leakage coming back through the check valve. A level alarm provided a signal on excessive leakage. There was a gas volume in the upper space of the loop. 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. Since that time and up to initial operation of Indian Point Unit 2, the level alarm in the detection pit had never gone off due to check valve leakage.

6.2.2.3.11 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 maximum expected leak rate of 1.0 gpm identified in Technical Specifications with leakage being from the reactor coolant system into an accumulator through an accumulator discharge line. The accumulators are provided with level and pressure alarms. Operator response to inleakage causing these alarms to actuate would preclude the need for the relief valves to perform in a water relief capacity.

The safety injection test line relief valve is provided to relieve any overpressure, that might build up in the high-head safety injection piping due to thermal expansion of fluid in the system or from leakage from the reactor coolant system past the SI header check valves. The valve will pass a nominal 15 gpm ($2.25 \times 10^5 cm^3/hr$), which is far in excess of the manufacturing design in leakage rate from the reactor coolant system of 24 cm^3/hr.

6.2.2.3.12 Leakage Limitations of Valves

Valving is specified for exceptional tightness.

Normally open valves have backseats that limit leakage as shown in Table 6.2-9. 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.

Motor-operated valves, which are exposed to recirculation flow, are generally provided with double-packed stuffing boxes and stem leakoff connections that are piped to the waste disposal system.

The specified leakage across the valve disk required to meet the equipment specification and hydrotest requirements is as follows:

1. Conventional globe - 3 cm^3/hr-in. of nominal pipe size.

2. Gate valves - 3 cm^3/hr-in. of nominal pipe size; 10 cm^3/hr-in. for 300- and 150-lb USA Standard.

3. Motor-operated gate valves - 3 cm^3/hr-in. of nominal pipe sizes; 10 cm^3/hr-in. for 300- and 150-lb USA Standard.

4. Check valves - 3 cm^3/hr-in. of nominal pipe size; 10 cm^3/hr-in. for 300- and 150-lb USA Standard.

5. Accumulator check valves - 2 cm^3/hr-in. of nominal pipe size; relief valves are totally enclosed.

Leakage from components of the recirculation loop including valves, is given in Table 6.2-9.

6.2.2.3.13 Piping

All safety injection system piping in contact with borated water is austenitic stainless steel. Piping joints are welded except for the flanged connections at the safety injection pumps, the recirculation pumps, and valve 741A.

The piping beyond the accumulator stop valves is designed for reactor coolant system conditions (2485 psig, 650°F). All other piping connected to the accumulator tanks is designed for 700 psig and 400°F.

The safety injection pump and residual heat removal pump suction piping (210 psig at 300°F) from the refueling water storage meets net positive suction head requirements of the pumps.

The safety injection high-pressure branch lines (1500 psig at 300°F) are designed for high-pressure losses to limit the flow rate out of the branch line, which may have ruptured at the connection to the reactor coolant loop. The system design incorporates the ability to isolate the safety injection pumps on separate headers such that full flow from at least one pump is ensured should a branch line break.

The piping is designed to meet the minimum requirements set forth in (1) the USAS B31.1 Code (1955) for the Pressure Piping, (2) Nuclear Code Case N-7, (3) USAS Standards B36.10 and B36.19, and (4) ASTM Standards with supplementary standards plus additional quality control measures.

Minimum wall thicknesses are determined by the USAS Code (1955) formula found in the power piping Section 1 of the USAS Code (1955) for Pressure Piping. This minimum thickness has been increased to account for the manufacturer's permissible tolerance of −12.5-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 have been performed. Special attention is 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.

Piping between valves 730 and 731 (Line 10) has 6" thick insulation to assure operability of these valves during design basis accident conditions.

Pipe and fittings materials are procured in conformance with all requirements of ASTM and USAS specifications. All materials are verified for conformance to specification and documented by certification of compliance to ASTM material requirements. Specifications impose additional quality control upon the suppliers of pipes and fittings as listed below.

1. Check analyses are performed on both the purchased pipe and fittings.

2. Pipe branch lines 2.5-in. and larger between the reactor coolant pipes and the isolation stop valves conform to ASTM A376 and meet the supplementary requirement S6 for ultrasonic testing. Fittings conform to the requirements of ASTM A403. Fittings 2.5-in. and above have requirements for ultrasonic testing inspection similar to S6 of A376.

Shop fabrication of piping subassemblies is performed by reputable suppliers in accordance with specifications that define and govern material procurement, detailed design, shop fabrication, cleaning, inspection, identification, packaging, and shipment.

Welds for pipes sized 2.5-in. and larger are butt welded. Reducing tees are used where the branch size exceeds one-half of the header size. Branch connections of sizes that are equal to or less than one-half 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.

All welding is performed by welders and welding procedures qualified in accordance with the ASME Boiler and Pressure Vessel Code, Section IX, Welding Qualifications. The shop fabricator is 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 must have prior approval.

All high-pressure piping butt welds containing radioactive fluid at greater than 600°F temperature and 600 psig pressure or equivalent are radiographed. The remaining piping butt welds are randomly radiographed. The technique and acceptance standards are those outlined in UW-51 of the ASME Boiler and Pressure Vessel Code, Section VIII. In addition butt welds are liquid-penetrant examined in accordance with the procedure of ASME Boiler and Pressure Vessel Code, Section VIII, Appendix VIII, and the acceptance standard as defined in the USAS Nuclear Code Case N-10. Finished branch welds are liquid-penetrant examined on the outside, and where size permits, on the inside root surfaces. Effective with the implementation of the EN Welding Program, in lieu of the above, the in-process quality assurance examinations for safety related piping welds and the acceptance criteria will be in accordance with ASME Section III 1992 Edition.

A postbending solution anneal heat treatment is performed on hot-formed stainless steel pipe bends. Completed bends are then completely cleaned of oxidation from all affected surfaces. The shop fabricator is required to submit the bending, heat treatment, and cleanup procedures for review and approval-prior to release for fabrication.

General cleaning of completed piping subassemblies (inside and outside surfaces) is 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.

The packaging of the piping subassemblies for shipment is done so as to preclude damage during transit and storage. Openings are closed and sealed with tight fitting covers to prevent the entry of moisture and foreign material. Flange facings and weld end preparations are protected from damage by means of wooden cover plates and securely fastened in position. The packing arrangement proposed by the shop fabricator is subject to approval.

6.2.2.3.14 Pump and Valve Motors Outside Containment

Motor electrical insulation systems are supplied in accordance with IEEE, and NEMA standards and are tested as required by standards.

Temperature rise design selection is such that normal long life is achieved even under accident loading conditions.

Criteria for motors of the safety injection system require that under any anticipated mode of operation the motor nameplate rating is not exceeded. The pump motors have a 1.15 service factor for normal operation. Design and test criteria ensure that motor loading does not exceed the application criteria.

6.2.2.3.15 Pump and Valve Motors Inside Containment

Motors for the recirculation pumps were originally specified to operate in an ambient condition of saturated steam of 270°F and 47 psig pressure for 1 day, followed by indefinite operation at 155°F and 5 psig in a steam atmosphere. These ambient conditions and operating times have been updated and are maintained by the ongoing Environmental Qualification Program discussed in Section 7.1.4. As part of this program, the recirculation pump motors are qualified to withstand containment environmental conditions following the loss of coolant accident so that the pumps can perform their required function during the recovery period (one year). These motors are of a similar design as the containment fan cooler motors. Refer to Section 6.4.2.2.5 for a description and evaluation of the motor design. Per the 12/06/04 NRC generic SER on NEI 04-07 (Reference 5), evaluations of PWR post accident emergency recirculation performance to address the potential impact of debris blockage per GSI-191 (Reference 4) and Generic Letter 2004-02 (Reference 3) will use a mission time of 30 days.

The motors for the valves inside containment are designed to withstand containment environment conditions following the loss-of-coolant accident so that the valves can perform the required function during the recovery period.

Periodic operation of the motors and tests of the insulation ensure that the motors remain in a reliable operating condition.

Although the motors that are provided only to drive engineered safety features equipment are normally run only for tests, the design loading and temperature rise limits are based on accident conditions. Normal design margins are specified for these motors to make sure the expected lifetime includes allowance for the occurrence of accident conditions.

6.2.2.3.16 Valve Motor Operators

Environmental Qualification

As part of the original plant design, a program of environmental qualifications was performed on valve motor operators important to plant safety. Tests to demonstrate the adequacy of valve motor operators to be functional after exposure to temperature, pressure, and radiation were conducted in two groups.

The first group test was the exposure of valve motor operators to both temperature and pressure. Two suppliers, Philadelphia Gear Corporation Limitorque Division and Crane Company Teledyne Division, conducted simulated containment pressure and temperature tests as follows with pressure and temperature similar to that predicted for the incident:

1. Operator located inside a pressure vessel with the operator exposed to approximately 330°F at 90 psig.

2. Operator cycled approximately 3 times under simulated valve operating loads.

3. Pressures and temperatures reduced in step change to 285°F at 60 psig, 219°F at 20 psig, and 152°F at atmosphere or less.

4. Operator cycled approximately 3 times at each of the levels of change. Full recordings of pertinent data were taken throughout the tests.

5. Unit was examined after completion of test and operating data compared to data prior to exposure.

The second group test was the radiation test on a motor from the valve operator.

1. Two production line motors were used for this test; one exposed to 1.5×10^8 rads of gamma radiation for an approximate period of 1 month, the other motor used for the final comparative analysis.

2. Both units were tested for coil resistance, insulation meggering both before and after motor vibration, and reversing operations.

More recently, a program of environmental requalifications of items important to plant safety has been initiated using the "Division of Operating Reactors" or NUREG-0588 guidelines. See Section 7.1.4 for a discussion of this ongoing program.

In response to IE Information Notice 86-03, all limitorque motor operators on the EQ Master List (see Section 7.1.4) were inspected and serviced to assure that wiring, limit switches and torque switches have been environmentally qualified.

In response to the IE Bulletin 85-03, the operability of key safety Motor Operated Valves was verified with associated full differential pressure.

6.2.2.4 Electrical Supply

Details of the normal and emergency power sources for the safety injection system are presented in Chapter 8.

6.2.2.5 Protection Against Dynamic Effects

The injection lines penetrate the containment adjacent to the primary auxiliary building. For most of the routing, these lines are outside the crane wall, and hence are protected from missiles originating within these areas. Each line penetrates the crane wall near the injection point to the reactor coolant pipe. In this manner, maximum separation and hence protection it provided in the coolant loop area.

Coolant loop supports are designed to restrict the motion in one loop due to rupture in another loop to about one-tenth of an inch, whereas the attached safety injection piping can sustain a 3-in. displacement without exceeding the working stress range. The analysis assumes that the injection flow to the ruptured loop is spilled on the containment floor.

In 1989, the NRC approved changes to the design basis with respect to dynamic effects of postulated primary loop ruptures, as discussed in Section 4.1.2.4

All hangers and anchors are designed in accordance with USAS B31.1, Code for Pressure Piping. This code provides minimum requirements on materials, design, and fabrication with ample safety margins for both dead and dynamic loads over the life of the equipment. Concrete missile barriers, bumpers, walls, and other concrete structures are designed in accordance with ACI 318-63, Building Code Requirements for Reinforced Concrete. Specifically, these standards require the following:

1. All materials used are in accordance with ASTM specifications that establish quality levels for the manufacturing process, minimum strength properties, and for test requirements that ensure compliance with the specifications.

2. Welding processes and welders must be qualified for each class of material welded and for types and positions of welds.

3. Maximum allowable stress values are established, which provide an ample safety margin on both yield strength and ultimate strength.

6.2.3 Design Evaluation

6.2.3.1 Range of Core Protection

The measure of effectiveness of the safety injection system is the ability of the pumps and accumulators to keep the core flooded or to reflood the core rapidly when the core has been uncovered for postulated large area ruptures. The result of this performance is to limit sufficiently any increase in clad temperature below a value where emergency core cooling objectives are met (Section 6.2.1). The sequence of events involving safety injection actuation for small and large breaks of a reactor coolant pipe are presented in Section 14.3.2.

6.2.3.2 System Response

To provide protection for large area ruptures in the reactor coolant system, the safety injection system must respond rapidly to reflood the core following the depressurization and core voiding that is characteristic of large area ruptures. The accumulators act to perform the rapid reflooding function with no dependence on the normal or emergency power sources and also with no dependence on the receipt of an actuation signal.

The operation of this system with three of the four available accumulators delivering their contents to the reactor vessel (one accumulator spilling through the break) prevents fuel clad melting and limits metal-water reaction to an insignificant amount (<1-percent).

The function of the safety injection and residual heat removal pumps is to complete the refill of the vessel and ultimately return the core to a subcooled state. 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 the actuation of circuitry and active components. The active portions of the injection systems are automatically actuated by the safety injection signal (Chapter 7). In addition, manual actuation of the entire injection system and individual components can be accomplished from the control room. In the analysis of system performance, delays in reaching the programmed trip points and in the actuation of components are conservatively established on the basis that only emergency onsite power is available.

The starting sequence of the safety injection and residual heat removal pumps and the related emergency power equipment is discussed in sections 7.2 and 8.2.3.4 and their analyzed performance is discussed in the various Chapter 14 safety analyses.

6.2.3.3 Single-Failure Analysis

A single active failure analysis is presented in Table 6.2-10. All credible active system failures are considered. This analysis is based on the worst single failure (generally a pump failure) in both the safety injection and residual heat removal pumping systems. The analysis shows that the failure of any single active component will not prevent fulfilling the design function. The analysis of the loss-of-coolant accident presented in Section 14.3 is consistent with this single-failure analysis.

In addition to active failures, an alternative flow path is available to maintain core cooling if any part of the recirculation flow path becomes unavailable due to a single passive failure. This is evaluated in Table 6.2-11. The procedure followed to establish the alternative flow path also isolates the spilling line. A valve is provided in the containment recirculation line to the residual heat removal pumps to isolate this line should it be required.

Therefore, the ECCS design incorporates redundancy of components such that neither a single active component failure during the injection phase nor an active or passive failure during the recirculation phase will degrade the ECCS function. Only active failures are assumed to occur within the first 24 hours following the initiating event.

Failure analyses of the component cooling and service water system under loss-of-coolant accident conditions are described in Sections 9.3 and 9.6, respectively.

6.2.3.4 Reliance on Interconnected Systems

During the injection phase, the high-head safety injection pumps do not depend on any portion of other systems, with the exception of the suction line from the refueling water storage tank and the component cooling loop as a heat sink for bearing and lube oil cooling. During the recirculation phase of the accident for small breaks, suction to the high-head safety injection pumps is provided by the recirculation pumps or, should backup capability be required, the residual heat removal pumps. The residual heat removal (low-head) pumps are normally used during reactor shutdown operations. Whenever the reactor is at power, the pumps are aligned for emergency duty.

6.2.3.5 Shared Function Evaluation

Table 6.2-12 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.

6.2.3.6 Passive Systems

The accumulators are a passive safety feature in that they perform their design function in the total absence of an actuation signal or power source. The only moving parts in the accumulator injection train are in the two check valves.

The working parts of the check valves are exposed to fluid of relatively low boric acid concentration. Even if some unforeseen deposition accumulated, a reversed differential pressure of about 25 psi can shear any particles in the bearing that may tend to prevent valve functioning. This is demonstrated by calculation.

The isolation valve at each accumulator is only closed when the reactor is intentionally depressurized or momentarily for testing when pressurized. The isolation valve is normally open and an alarm in the control room sounds if the valve is inadvertently closed. It is not expected that the isolation valve will have to be closed due to excessive leakage through the check valves.

The check valves operate in the closed position with a nominal differential pressure across the disk of approximately 1650 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, and therefore function as required.

When the reactor coolant system is being pressurized during the normal plant heatup operation, the check valves can be tested for leakage as soon as there is about 150 psi differential across the valve. This test confirms the seating of the disk and whether or not there has been an increase in the leakage since the last test. When this test is completed, the discharge line test valves are opened and the reactor coolant system pressure increase is continued. There should be no increase in leakage from this point on since increasing reactor coolant pressure increases the seating force and decreases the probability of leakage.

The accumulators can accept leakage from the reactor coolant system without effect on their availability. Table 6.2-13 indicates what 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 cm^3/hr i.e., 2 cm^3/hr/in).

Inleakage at a rate of 5 cm^3/hr-in., 2.5 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 leak rate of 30 cm^3/hr-in. (15 times the acceptance leak rate), the water level will have to be readjusted approximately once every 5 to 6 months. This readjustment will take about 2 hr maximum.

The accumulators are located inside the reactor containment and protected from the reactor coolant system piping and components by a missile barrier. Accidental release of the gas charge in the accumulator would cause an increase in the containment pressure. This release of gas has been included in the containment pressure analysis for the large break loss-of-coolant accidents, (Section 14.3.3.3 and 14.3.5.1.1).

During normal operation, the flow rate through the reactor coolant piping is approximately 5 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.

6.2.3.7 Emergency Flow to the Core

Special attention is given to factors that could adversely affect the accumulator and safety injection flow to the core. These factors are as follows:

1. Steam binding in the core, including flow blockage due to loop sealing.
2. Loss of accumulator water during blowdown.
3. Short circuiting of the accumulator from the core to another part of the reactor coolant system.
4. Loss of accumulator water through the breaks.

All of the above are considered in the analysis of the Loss of Coolant Accident which is discussed in Section 14.3.

6.2.3.8 External Recirculation Loop Leakage

Table 6.2-9 summarizes the maximum potential leakage from the leak sources of the external recirculation loop, which goes through the residual heat removal pumps, a residual neat exchanger, and the high-head safety injection pumps. In the analysis, a maximum leakage is assumed from each leak source. For conservatism, 3 times the maximum expected leak rate from the pump seals was assumed, even though the seals are acceptance tested to essentially zero leakage, and a leakage of 10 drops/min was assumed from each flange although each flange would be adjusted to essentially zero leakage. The total maximum potential leakage resulting from all sources is 999 cm^3/hr to the auxiliary building atmosphere and 21 cm^3/hr to the drain tank.

During external recirculation, significant margin exists between the design and operating conditions of the residual heat removal system components, as shown in Table 6.2-14. In addition, during normal plant cooldown, operation of the residual heat removal system is initiated when the primary system pressure and temperature have been reduced to below 365 psig (the upper limit to prevent RHR system overpressurization) and 350°F, respectively. Even assuming a conservative maximum RHR System pressure of 232 psig and a conservative maximum RHR System temperature of 277°F during recirculation as shown in Table 6.2-14, significant margin also exists between normal operating and accident conditions. In view of the above margins, it is considered that the leakage rates tabulated in Table 6.2-9 are conservative. The radiological consequences of external recirculation loop leakage following a design basis accident are presented in Section 14.3.6.6.

6.2.3.9 Pump Net Positive Suction Head Requirements

6.2.3.9.1 Residual Heat Removal Pumps

The net positive suction head (NPSH) of the residual heat removal pumps is evaluated for normal plant shutdown operation and the operation of both the injection and recirculation phases of the design-basis accident.

The residual heat removal pumps are used as backup to the internal recirculation pumps in the event of failures to the normal recirculation path; this duty provides the pumps with the minimum NPSH condition. For the design case of one pump recirculating through one heat exchanger path, the available NPSH exceeds the NPSH required, assuming saturated fluid and no operator action to throttle back the flow. There is no postulated failure that requires both RHR pumps to operate in the recirculation phase.

6.2.3.9.2 Safety Injection Pumps

The NPSH for the safety injection pumps is evaluated for both the injection and recirculation phase operation of the design-basis accident. The end of injection phase operation gives the limiting NPSH requirement; the NPSH available is determined from the elevation head and vapor pressure of the water in the refueling water storage tank and the pressure drop in the suction piping from the tank to the pumps. At the end of the injection phase, greater than 20-percent NPSH margin is available assuming all three pumps running together with two residual heat removal pumps at maximum flow conditions permitted by the system alignment.

6.2.3.9.3 Recirculation Pumps

The NPSH for the recirculation pumps is evaluated for recirculation operation. The NPSH available is determined considering the elevation head of the water above the pump NPSH reference line (eye of the 1st stage impeller) in the sump, level drawdown due to the flow path in the containment, fluid temperature adjustments, and strainer head losses. The NPSH determination met the requirements of GL 2004-02. The containment water level is confirmed to be above the minimum level required for NPSH, prior to starting the recirculation pumps during the changeover from the injection phase to the recirculation phase. The RWST level is confirmed to be less than 2 feet prior to stopping the remaining operating containment spray pump and establishing simultaneous recirculation flow to the core and the spray headers. This maximizes the available NPSH to the recirculation pumps in this mode of operation.

The internal recirculation pumps are conventional vertical condensate pumps and are of double suction design, requiring less NPSH. At the initiation of the recirculation phase, the NPSH requirement is met with one or both pumps operating at the pump design flow. When simultaneous recirculation flow to the core and spray headers is established, the available NPSH requirement will be met at expected pump flows.

6.2.4 Minimum Operating Conditions

The Technical Specifications establish limiting conditions regarding the operability of the system when the reactor is critical.

6.2.5 Inspections and Tests

6.2.5.1 Inspection

All components of the safety injection system are inspected periodically to demonstrate system readiness.

The pressure-containing components are inspected for leaks from pump seals, valve packing, flanged joints, and safety valves during system testing.

Current requirements for safety injection system surveillance are discussed in Sections 3.5 and 5.5.6 of the facility Technical Specifications and in UFSAR Section 1.12, "Inservice Inspection and Testing Programs".

6.2.5.2 Preoperational Testing

6.2.5.2.1 Component Testing

Preoperational performance tests of the components were performed in the manufacturer's shop. The pressure-containing parts of the pump were hydrostatically tested in accordance with Paragraph UG-99 of Section VIII of the ASME Code. Each pump was given a complete shop performance test in accordance with Hydraulic Institute Standards. The pumps were run at design flow and head, shutoff head, and at additional points to verify performance characteristics. Net positive suction head was established at design by means of adjusting suction pressure for a representative pump. This test was witnessed by qualified Westinghouse personnel.

The remote-operated valves in the safety injection system are motor-operated. Shop tests for each valve included a hydrostatic pressure test, leakage tests, a check of opening and closing time, and verification of torque switch and limit switch settings. The ability of the motor operator to move the valve with the design differential pressure across the gate was demonstrated by opening the valve with an appropriate hydrostatic pressure on one side of the valve.

The recirculation piping and accumulators were initially hydrostatically tested at 150-percent of design pressure.

The service water and component cooling water pumps were tested prior to initial operation.

6.2.5.2.2 System Testing

An initial functional test of the core cooling portion of the safety injection system was conducted during the hot-functional testing of the reactor coolant system before initial plant startup. The purpose of the initial systems test was to demonstrate the proper functioning of instrumentation and actuation circuits and to evaluate the dynamics of placing the system in operation. This test was performed following the flushing and hydrostatic testing of the system.

The functional test was performed with the water level below the safety injection setpoint in the pressurizer and with the reactor coolant system initially cold and at low pressure. The safety injection system valving was set initially to simulate the system alignment for plant power operation.

To initiate the test, the safety injection block switch was moved to the unblock position to provide control power allowing the automatic actuation of the safety injection relays from low-pressure signals from the pressurizer instrumentation. Simultaneously, the breakers supplying outside power to the 480-V buses were tripped manually and operation of the emergency diesel system automatically commenced. The high-head safety injection pumps and the residual heat removal pumps were started automatically following the prescribed diesel loading sequence. The valves were operated automatically to align the flow path for injection into the reactor coolant system.

The rising water level in the pressurizer provided indication of system delivery. Flow into the reactor coolant system terminated with the filling of the pressurizer, and the operation of the safety injection systems was terminated manually in the control room.

This functional test provided information to confirm valve operating times, pump motor starting times, the proper automatic sequencing of load addition to the emergency diesels, and delivery rates of injection water to the reactor coolant system.

The functional test was repeated for the various modes of operation needed to demonstrate performance at partial effectiveness, that is, to demonstrate the proper loading sequence with two of the three emergency diesels and to demonstrate the correct automatic starting of a second pump should the first pump fail to respond. These latter cases were performed without delivery of water to the reactor coolant system, but included starting of all pumping equipment involved in each test.

The systems were accepted only after the demonstration of proper actuation and after the demonstration of flow delivery and shutoff head within design requirements.

Flow was introduced into the reactor coolant loops through the accumulator discharge line to demonstrate the operability of the check valves and remotely actuated stop valve, and to confirm length to diameter (L/D) ratios of accumulator discharge lines used in the calculation.

6.2.5.3 Post-operational Testing

6.2.5.3.1 Component Testing

Routine periodic testing of the safety injection system components and all necessary support systems at power is done. No inflow to the reactor coolant system will occur whenever the

reactor coolant pressure is above the safety injection pump shutoff head. If such testing indicates a need for corrective maintenance, the redundancy of equipment in these systems permits such maintenance to be performed without shutting down or reducing load under conditions defined in the Technical Specifications. These conditions include such matters as the period within which the component is to be restored to service and the capability of the remaining equipment to meet safety limits within such a period.

Test Circuits are provided to examine periodically the leakage back through the check valves and to ascertain that these valves seat whenever the reactor system pressure is increased. The recirculation pumps are normally in a dry sump. These pumps can only be started and allowed to reach full speed with the plant at cold shutdown. Flow testing of these pumps is performed during refueling operations by filling the recirculation sump and directing the flow back to the sump through the valve on the discharge of the pump. The service water and component cooling pumps not running during normal operation may be tested by alternating with the operating pumps.

The contents of the accumulators and the refueling water storage tank are sampled periodically to determine the boron concentration.

6.2.5.3.2 System Testing

System testing is conducted during plant shutdown to demonstrate proper automatic operation of the safety injection system. A test signal is applied to initiate automatic action and verification made that the safety injection and residual heat removal pumps receive start signals. The test demonstrates the operation of the valves, pump circuit breakers and automatic circuitry. Isolation valves in the injection line will be blocked closed so that flow is not introduced into the reactor coolant system. The test is considered satisfactory if control board indication and visual observations indicate all components have operated and sequenced properly.

The safety injection piping up to the final isolation valve is maintained sufficiently full of borated water at refueling water concentration while the plant is in operation to ensure the system remain operable and perform properly. The safety injection pumps recirculate refueling water through the injection lines via a small test line provided for this purpose.

Flow in each of the safety injection headers and in the main flow line for the residual heat removal pumps is monitored by a local flow indicator. Pressure instrumentation is also provided for the main flow paths of the safety injection and residual heat removal pumps. Accumulator isolation valves are blocked closed for this test.

The high pressure safety injection pumps are run and the variable orifices and injection line valves are adjusted to balance flowrates within the specified range.

The eight-switch sequence for recirculation operation is tested to demonstrate proper sequencing of valves and pumps. The recirculation pumps are blocked from starting during this test.

The external recirculation flow paths are hydrotested during periodic retests at the operating pressures. This is accomplished by running each pump, which could be used during external recirculation (safety injection and residual heat removal pumps) and checking the discharge

and recirculation test lines. The suction lines are tested by running the residual heat removal pumps and opening the flow path to the safety injection pumps in the same manner as described above.

During the above test, all system joints, valve packings, pump seals, leakoff connections, or other potential points of leakage can be visually examined. Valve gland packing, pump seals, and flanges are adjusted or replaced as required to reduce the leakage to acceptable proportions. For power-operated valves, final packing adjustments are made, and the valves are put through an operating cycle before a final leakage examination is made.

The entire recirculation loop, except the recirculation line to the residual heat removal pumps, is pressurized during periodic testing of the engineered safety features components. The recirculation line to the residual heat removal pump is capable of being hydrotested during plant shutdown, and it is also leak-tested at the time of the periodic retests of the containment.

REFERENCES FOR SECTION 6.2

1. M. J. Bell, et al., Investigations of Chemical Additives for Reactor Containment Spray Systems, WCAP-7153, Westinghouse Electric Corporation, March 1968.

2. Deleted

3. NRC Generic Letter 2004-02, "Potential Impact of Debris Blockage on Emergency Recirculation During Design Basis Accidents at Pressurized-Water Reactors", dated September 13, 2004.

4. NRC Generic Safety Issue (GSI) 191, "Assessment of Debris Accumulation on PWR Sump Performance."

5. NRC Generic SER dated 12/06/04 on the NEI 04-07 Guidance Report entitled, "Pressurized Water Reactor Sump Performance Evaluation Methodology."

6. A. E. Lane, et al., Evaluation of Post Accident Chemical Effects in Containment Sump Fluids to Support GSI-191, WCAP-16530-NP-A, Westinghouse Electric Corporation LLC, March 2008.

7. T. S. Andreychek, et al., Evaluation of Downstream Sump Debris Effects in Support of GSI-191, WCAP-16406-P-A, Revision 1; Westinghouse Electric Corporation LLC, March 2008.

8. T. S. Andreychek, et al., Evaluation of Long-Term Cooling Considering Particulate, Fibrous and Chemical Debris in the Recirculating Fliud, WCAP-16793-NP, Revision 0; Westinghouse Electric Corporation LLC, May 2007.

TABLE 6.2-1
Safety Injection System – Code Requirements

Component	Code
Refueling Water Storage Tank	AWWA D100-65
Residual Heat Exchanger	
Tube Side	ASME Section III Class C
Shell Side	ASME Section VIII
Accumulators	ASME Section III Class C
Valves	USAS B16.5 (1955)
Piping	USAS B31.1 (1955)

TABLE 6.2-2 (Sheet 1 of 2)
Instrumentation Readouts On The Control Board
For Operator Monitoring During Recirculation

Valves

System	Valve No.
SIS	MOV 1802 A, B
SIS	MOV 885 A, B
SIS	MOV 889 A, B
SIS	MOV 888 A,B
SIS	MOV 866 A, B, C, D
SIS	MOV 851 A, B
SIS	MOV 856 A, B, C, D,E,F
SIS	MOV 882
SIS	MOV 842
SIS	MOV 843
ACS	MOV 744
ACS	MOV 745 A,B
ACS	MOV 746
ACS	MOV 747
ACS	MOV 1810
ACS	HCV 638
ACS	HCV 640

Instruments

System	Channel No
SIS	FI 945 A, B
SIS	FI 946 A, B, C,D
SIS	FI 924
SIS	FI 925
SIS	FI 926
SIS	FI 927
SIS	LI 938
SIS	LI 939
SIS	LI 941
SIS	LT 3300
SIS	LT 3301
SIS	LT 3302
SIS	LT 3304
SIS	PI 922
SIS	PI 923
SIS	PI 947
ACS	FI 640
ACS	LI 628
ACS	TR 636

TABLE 6.2-2 (Sheet 2 of 2)
Instrumentation Readouts On The Control Board
For Operator Monitoring During Recirculation

Instruments (continued)

System	Channel No.
RCS	LI 459
RCS	LI 460
RCS	LI 461
RCS	LI 462

Pumps

System	Pumps
SIS	Safety Injection
SWS	Service Water
ACS	Component Cooling
CS	Containment Spray
RS	Recirculation
ACS	Residual Heat Removal

Key:

ACS - Auxiliary Coolant System
CS - Containment Spray System
RCS - Reactor Coolant System
RS - Recirculation
SIS - Safety Injection System
SWS - Service Water System

TABLE 6.2-3 (Sheet 1 of 3)
Quality Standards Of Safety Injection System Components

Residual Heat Exchanger

A. Tests and inspections
   1. Hydrostatic test
   2. Radiograph of longitudinal and girth welds (tube side only)
   3. Ultrasonic testing of tubing or eddy current tests
   4. Dye penetrant test of welds
   5. Dye penetrant test of tube to tube sheet welds
   6. Gas leak test of tube to tube sheet welds before hydro and expanding of tubes

B. Special manufacturing process control
   1. Tube to tube sheet weld qualifications procedure
   2. Welding and NDT and procedure review
   3. Surveillance of supplier quality control and product

Component Cooling Heat Exchanger

A. Tests and inspections
   1. Hydrostatic Test
   2. Dye penetrant test of welds

B. Special Manufacturing Process Control
   1. Welding and NDT and procedure review
   2. Surveillance of supplier quality control and product

Safety Injection, Recirculation, and Residual Heat Removal Pumps

A. Test and inspections
   1 Performance test
   2 Dye penetrant of pressure retaining parts (except Internal Recirculation Pump)
   3. Hydrostatic test

B. Special manufacturing process control
   1. Weld, NDT, and inspection procedures for review
   2. Surveillance of supplier quality control system and product

Accumulators

A. Tests and inspection
   1. Hydrostatic test
   2. Radiography of longitudinal and girth welds
   3. Dye penetrant/magnetic particle of weld

B. Special manufacturing process control
   1. Weld, fabrication, NDT, and inspection procedure review
   2. Surveillance of supplier quality control and product

TABLE 6.2-3 (Sheet 2 of 3)
Quality Standards Of Safety Injection System Components

Valves

A. Tests and inspections
   1. 200 psi and 200°F or below (cast or bar stock)
      a. Dye penetrant test
      b. Hydrostatic test
      c. Seat leakage test
   2. Above 200 psi and 200°F
      a. Forged valves (2-1/2-in. and larger)
         (1) Ultrasonic tests of billet prior to forging
         (2) Dye penetrant 100-percent of accessible areas after forging
         (3) Hydrostatic test
         (4) Seat leakage test

      b. Case valves
         (1) Radiograph 100-percent (radioactive service only)
         (2) Dye penetrant all accessible areas (radioactive service only)
         (3) Hydrostatic test
         (4) Seat leakage

   3. Functional tests required for:
      a. Motor operated valves
      b. Auxiliary relief valves

B. Special manufacturing process control
   1. Weld, NDT, performance testing, assembly and inspection procedure review
   2. Surveillance of supplier quality control and product
   3. Special weld process procedure qualification (e.g., hard facing)

Piping

A. Tests and inspections
   Class 1501 and below
   Seamless or welded. If welded 100-percent radiography is required, shop-fabricated and field-fabricated pipe weld joints are inspected as follows:

   2501R – 610R: 100-percent radiographic inspection and penetrant examination
   301R – 302R: 20-percent random radiographic inspection
   151R – 152R: 100-percent liquid penetrant examination

B. Special manufacturing process control
   Surveillance of supplier quality control and product

TABLE 6.2-3 (Sheet 3 of 3)
Quality Standards Of Safety Injection System Components

Refueling Water Storage Tank

A. Tests and inspections
   1. Vacuum box test of tank bottom seams
   2. Hydrostatic test of tank
   3. Hydrostatic test of tank heater coil
   4. Spot radiography of longitudinal and girth welds

B. Special manufacturing process control
   1. Weld, fabrication, NDT, and inspection procedure review
   2. Surveillance of suppliers quality control and product
   3. Material chemical and physical properties certification

[Deleted]

TABLE 6.2-4
Accumulator Design Parameters

Number	4
Type	Stainless steel lined/ carbon steel
Design pressure, psig	700
Design temperature, °F	300
Operating temperature, °F	100-150
Normal operating pressure, psig	Note 2
Total volume, ft^3 (each)	1100
Water volume at operating conditions, ft^3 (each)	Note 2
Minimum boron concentration (as boric acid), ppm	2000
Relief valve setpoint, psig[1]	700

Notes:
1. The relief valves have soft seats and are designed and tested to ensure exceptional tightness

2. Minimum and maximum operating pressure and volume are controlled by Technical Specifications

TABLE 6.2-5
DELETED

TABLE 6.2-6
Refueling Water Storage Tank Design Parameters

Number:	1
Material:	Stainless Steel
Nominal Capacity, gal.	350,000
Volume Required by Technical Specifications (solution), gal.	345,000
Normal pressure, psig	Atmospheric
Operating temperature, ^{o}F	40-110
Design pressure, psig	Atmospheric
Design temperature, ^{o}F	120
Boron concentration (as boric acid), ppm	2400 (minimum)
Type of heating	Steam

TABLE 6.2-7
Pump Design Parameters

Safety injection pump

Number	3
Design pressure, discharge, psig	1750
Design pressure, suction, psig	200
Design temperature, °F	285
Design flow rate, gpm	400
Maximum flow rate, gpm	650
Design head, ft	2500
Shutoff head, ft	3550
Material	Martensitic stainless steel
Motor, hp	400
Type	Horizontal centrifugal

Recirculation pump

Number of pumps	2
Type	Vertical centrifugal
Design pressure, discharge, psig	250
Design temperature, °F	300
Design flow, gpm	3000
Design head, ft	360
Material	Austenitic stainless steel
Maximum flow rate, gpm	4428
Shutoff head, ft	476
Motor, hp	350

Residual heat removal pump

Number of pumps	2
Type	Vertical centrifugal
Design pressure, discharge, psig	600
Design temperature, °F	400
Design flow, gpm	3000
Design head, ft	350
Material	Austenitic stainless steel
Maximum flow rate, gpm	5500
Shutoff head, ft	390
Motor, hp	400

TABLE 6.2-8
Residual Heat Exchangers Design Parameters

Heat Exchangers

Number		2
Design heat duty, Btu/hr (Normal)		30.8×10^6
Design UA_1, Btu/hr-°F		1.2×10^6
Design cycles (85°F – 350°F)		200
Type		Vertical shell and U-tube
Normal condition	Tube side	Shell side
Design pressure, psi	600	150
Design flow, lb/hr	1.44×10^6	2.46×10^6
Inlet temperature, °F	135	88.3
Outlet temperature, °F	113.5	100.8

Notes:
1. Total heat transfer coefficient x Area

TABLE 6.2-9
Estimated External Recirculation Loop Leakage[3]

Items	Number of Units	Type of Leakage Control and Unit Leakage Rate	Leakage to Atmosphere (cm^3/hr)	Leakage to Tank (cm^3/hr)
Residual heat removal pumps (low-head safety injection)	2	Mechanical seal with leakoff-drop/min	0_1	6
High-head safety injection pumps	3	Same as residual heat removal	0_1	9
Flanges:		Gasket-adjusted to zero leakage following any test - 10 drops/min per flange		
a. Pump	15		450_1	0
b. Valves - Bonnet to Body (larger than 2-in.)	16		480_2	0
Valves - stem leakoffs	6	Backseated, double packing with leakoff 1 cm^3/hr-in.stem diameter	0_2	6
Misc. small valves	23	Flanged body packed stems - 1 drop/min	69_2	0
		Totals	999	21

Notes:
1. Total estimated leakage from RHR and SI pump mechanical seals and flanges is 450 cc/hr
2. The total leakage estimated from all sources including valve stem leakage, packing leakoffs, flanged body packed stems and other potential sources (pumps and flanges) of External Recirculation Loop is 999 cc/hr
3. Actual measured leakage is limited by Technical Specifications. The radiological consequences of external recirculation loop leakage following a design basis accident are presented in Section 14.3.6.6

TABLE 6.2-10 (Sheet 1 of 3)
Single Active Failure Analysis – Safety Injection System

Component[1]	Malfunction	Comments
A. Accumulator (injection phase)	Deliver to broken loop	Totally passive system with one accumulator per loop. Evaluation based on three accumulators delivering to the core and one spilling from ruptured loop.
B. Pump: (injection phase)		
1. Safety injection	Fails to start	Three provided. Evaluation based on operation of two.
2. Residual heat removal	Fails to start	Two provided. Evaluation based on operation of one.
3. Essential service water	Fails to start	Three provided. Evaluation based on operation of two.
4. Component cooling[2]	Fails to start	A total of 1 of 3 required during recirculation
5. Nonessential service water[2]	Fails to start	A total of 1 of 3 required during recirculation
6. Recirculation[2]	Fails to start	Two provided. One required to operate during recirculation.
7. Auxiliary component cooling pump	Fails to start	Two provided. One required to operate during recirculation.

TABLE 6.2-10 (Sheet 2 of 3)
Single Active Failure Analysis – Safety Injection System

Component	Malfunction	Comments
C. Automatically operated valves: (open on safety injection signal) (injection phase)		
1. Safety injection line isolation valve at the loops	Fails to open (if closed)	Active failure to open is not credible since the injection valves are maintained in the open position when the reactor is critical
2. Residual heat removal line isolation valve at residual heat exchanger discharge	Fails to open	Two parallel lines, one valve in either line is required to open
3. Isolation valve on component cooling water line from residual heat exchangers	Fails to open	Two parallel lines, one valve in either line is required to open
D. Valves operated from control room for recirculation: (recirculation phase)		
1. Recirculation sump internal recirculation isolation	Fails to open	Two lines in parallel, one valve in either line is required to open
2. Safety injection pump suction valve at residual heat exchanger discharge	Fails to open	Two parallel lines, one valve in either line required to open
3. Isolation valve on the mini-flow line returning to the refueling water storage tank	Fails to close	Two valves in series, one required to close
4. Isolation at suction header from refueling water storage tank to safety injection pumps	Fails to close	Two valves in series, one required to close (one valve is a check valve)

TABLE 6.2-10 (Sheet 3 of 3)
Single Active Failure Analysis – Safety Injection System

Component	Malfunction	Comments
5. Residual heat removal pump recirculation line	Fails to close	Two valves in series, one required to close
6. Residual heat removal pump discharge line	Fails to close	Two valves in series, one required to close (one valve is a check valve) (Valve 744 operated from Control Room once AC power restored to valve controls)

Notes:

1. 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.

2. Recirculation phase

TABLE 6.2-11 (Sheet 1 of 3)
Single Passive Failure Analysis
(Loss Of Recirculation Flow Path)[5]

Flow Path	Indication Of Loss Of Flow Path	Alternative Flow Path[1]
Low head recirculation		
From recirculation sump to low-head injection header via the recirculation pumps and the residual heat exchangers	1. Insufficient flow in low-head injection lines (one flow monitor in each of the four low-head injection lines[2])	From recirculation sump to high-head injection header via the recirculation pumps one of the two residual hea exchangers and the safety injection pump[3]
	2. As 1 above	a. From containment sump to discharge header of the residual heat exchangers via the residual heat removal pumps b. If flow not established in low-head injection lines, as (a), except path is from discharge of one residual heat exchanger to the high-head injection header via the safety injection pumps

TABLE 6.2-11 (Sheet 2 of 3)
Single Passive Failure Analysis
(Loss Of Recirculation Flow Path)[5]

Flow Path	Indication Of Loss Of Flow Path	Alternative Flow Path[1]
High-head recirculation		
From recirculation sump to high-head injection header via the recirculation pumps, one of the two residual exchangers and the high-head injection pumps	1. No flow in high-head injection header (four flow monitors, one in each cold leg injection line and one pressure monitor)	a. From containment sump to high head injection header via the residual heat removal pumps, one of the residual heat exchangers and the high-head injection pumps
		b. If flow is not established in high-head injection header – as (a), except path is from discharge of the residual heat removal pumps to the high-head injection pumps via the middle safety injection pump (by-passing the residual heat exchangers[4]
	2. Flow in only one of the two high-head injection branch headers (two flow monitors per branch header)	a. As 1(b), except that flow from the middle safety injection pump is only supplied to the unbroken branch header

TABLE 6.2-11 (Sheet 3 of 3)
Single Passive Failure Analysis
(Loss Of Recirculation Flow Path)[5]

Notes:

1. As shown in Plant Drawings 9321-2735 & 235296 [Formerly UFSAR Figure 6.2-1], there are valves at all locations where alternative flow paths are provided

2. If minimum flow requirements have been established, the supply of recirculated water using low-head recirculation will maintain the core flooded even in the event of a low-head spilling line and one failed flow meter or other single failure

3. Manual start

4. In this recirculation mode, water is returned to the core without being cooled by the residual heat exchangers. Heat is removed from the core by boiloff of the water to the containment; heat is then removed from the containment by either the containment fan coolers and/or the containment spray system (using cooled water from the recirculation sump via the recirculation pumps and one residual heat exchanger).

5. Loss of the recirculation flow path due to a passive failure is not postulated until 24 hours into the accident (Reference Technical Specification Amendment #257)

TABLE 6.2-12 (Sheet 1 of 2)
Shared Functions Evaluation

Component	Normal Operating Function	Normal Operating Arrangement	Accident Function	Accident Arrangement
Refueling water storage tank	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 suction of safety injection, residual heat removal, and spray pumps
Accumulators (4)	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 pumps (3)	None	Lined up to hot and cold legs of reactor coolant piping	Supply borated water to core	Lined up to hot and cold legs of reactor coolant piping
Residual heat removal pumps (2)	Supply water to core to remove residual heat during shutdowns	Lined up to cold legs of reactor coolant piping	Supply borated water to core	Lined up to cold legs of reactor coolant piping
Recirculation pumps (2)	None	Lined up to cold legs of reactor coolant piping, spray headers and suction of safety injection pumps	Supply borated water to core and spray nozzles from recirculation sump	Lined up to cold legs of reactor coolant piping, spray headers and suction of safety injection pumps
Service water pumps (non-essential header) (3)	Supply river cooling water to component cooling heat exchangers and non-nuclear components	One or two pumps in service	Supply river cooling water to component cooling heat exchangers	Lined up to non-essential service water header[1]
Component cooling pumps (3)	Supply cooling water to station nuclear components	Up to three pumps in service	Supply cooling water to residual heat exchangers, S.I. pump bearings and recirculation pump motor coolers	Lined up to CCW header[1]

TABLE 6.2-12 (Sheet 2 of 2)
Shared Functions Evaluation

Residual heat exchangers (2)	Remove residual heat from core during shutdown	Lined up for residual heat removal pump operation	Cool recirculated water in containment for core cooling and containment spray	Lined up to discharge of recirculation pumps or RHR pumps
Component cooling heat exchangers (2)	Remove heat from component cooling water	One or two heat exchangers in service	Cool water for residual heat exchangers and other services	Both heat exchangers in service
Auxiliary component cooling pumps (2)	None	Lined up for pump operation	Provide component cooling water to recirculation pump motor coolers$_2$	Lined up for pump operation$_2$
Service water pumps (essential header) (3)	Supply river water to station safeguards	Up to three pumps in service	Supply river cooling water to safeguards components	Lined up to essential service water header

Notes:

1. Recirculation Phase

2. These pumps start on a Safety Injection Signal and operate in the injection and recirculation phases of the accident. However, their function is not required during the injection phase. The supply of adequate cooling water to the recirculation pump motor coolers is required only during the recirculation phase.

TABLE 6.2-13
Accumulator Inleakage[1]

Time Period Between Level Adjustments	Observed Leak Rate ($cm^{3/}hr$)	Observed Leak Rate Maximum Allowed Design
1 month	1955	99.8
3 months	665	33.3
6 months	333	16.7
9 months	221	11.1
1 year	167	8
10 years	16.7	0.8

Notes:
1. A total of 83.3-ft^3, added to the initial amount, can be accepted in each accumulator before an alarm is sounded

TABLE 6.2-14
Residual Heat Removal System
Design, Operation, And Preoperational Test Conditions

	Pumps	Heat Exchangers	Valves	Pipes And Fittings
Design conditions				
Pressure, psig	600	600	665	700
Temperature, °F	400	400	400	400
Operating conditions (max.) - During recirculation[2]				
Pressure, psig	232	232	232	232
Temperature, °F	277	277	277	277
Preoperational Hydrostatic Test pressure, psig	1200	900	1100	900

Notes:
1. Located inside containment
2. These maximum values have been conservatively calculated assuming saturated conditions at the containment design pressure of 47 psig to demonstrate that significant margin exists between design, normal operating and hypothetical accident conditions. These values are used to support the conservatism of Table 6.2-9 and are not used in the Chapter 14 Accident Analysis or for any other purpose.

6.2 FIGURES

6.3 CONTAINMENT SPRAY SYSTEM

6.3.1 Design Bases

6.3.1.1 Containment Heat Removal Systems

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

Adequate containment heat removal capability for the containment is provided by two separate, full capacity, engineered safety feature systems. The containment spray system, whose components operate in the sequential modes described in Section 6.3.2, and the containment air recirculation cooling system, which is discussed in Section 6.4.

The primary purpose of the containment spray system is to spray cool water into the containment atmosphere when appropriate in the event of a loss-of-coolant accident and thereby ensure that containment pressure does not exceed its design value, which is 47 psig at 271°F. (100-percent relative humidity) This protection is afforded for all pipe break sizes up to and including the hypothetical instantaneous circumferential rupture of a reactor coolant loop as discussed in UFSAR Section 14.3.5.1.1. Pressure and temperature transients for a loss-of-coolant accident are presented in Section 14.3. Although the water in the core after a loss-of-coolant accident is quickly subcooled by the safety injection system, the containment spray system design is based on the conservative assumption that the core residual heat is released to the containment as steam.

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

1. Containment Spray alone as follows:
   - Both containment spray pumps operating up to the time the transfer to core recirculation flow begins (during injection phase).
   - One spray pump continuing to take suction from the RWST until the level in the RWST decreases to 2 feet.
   - Both recirculation pumps, both residual heat exchangers and both containment recirculation spray headers in operation when the level in the RWST decreases below 2 feet.

2. All five containment cooling fans (to be discussed in Section 6.4).

3. One containment spray pump and three of the five containment cooling fans (the minimum containment safeguards case which assumes the single failure of one emergency diesel generator is discussed in Section 14.3.5).

6.3.1.2 Inspection of Containment Pressure-Reducing Systems

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

Where practicable, all active components and passive components of the containment spray system are inspected periodically to demonstrate system readiness. The pressure-containing components are inspected for leaks from pump seals, valve packing, flanged joints, and safety valves. During operational testing of the containment spray pumps, the portions of the system subjected to pump pressure can be inspected for leaks. Design provisions for inspection of the safety injection system, which also function as part of the containment spray system, are described in Section 6.2.5.

6.3.1.3 Testing of the Containment Pressure-Reducing Systems Components

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

All active components in the containment spray system are adequately tested both in preoperational performance tests in the manufacturer's shop and inplace testing after installation. Thereafter, periodic tests are also performed as required after any component maintenance. Testing of the components of the safety injection system that are used for containment spray purposes is described in Section 6.2.5.

The component cooling water pumps and the non-essential service water pumps that supply cooling water to the residual heat exchangers are in operation on a relatively continuous schedule during plant operation. Those pumps not running during normal operation may be tested by changing the operating pump(s).

6.3.1.4 Testing of Containment Spray Systems

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

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

Each spray pump is provided with a recirculation line from the pump discharge line back to the pump suction line with a globe valve to allow greater flow through the pump during surveillance testing of the pump. An ultrasonic flow instrument is installed on each recirculation line during testing to allow resetting of the globe valve, after testing to the original flow value of the eductor (112 gpm). Note the globe valve replaced the eductor.

Temporary test connections, downstream of the isolation valves, are provided to verify that spray nozzles are not obstructed. Air flow through the nozzles will be monitored by means of the helium-filled balloon method, or by using an infrared scanning technique.

6.3.1.5 Testing of Operational Sequence of Containment Pressure-Reducing Systems

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

Capability is provided to test initially to the extent practical the operational startup sequence of the containment spray system including the transfer to alternative power sources.

6.3.1.6 Performance Objectives

The containment spray system is designed to spray at least 5000 gpm of borated water into the containment whenever the coincidence of two sets of two out of three (Hi-Hi) containment pressure (approximately 50-percent of design value) signals occur or when a manual signal is initiated. Either of two subsystems containing a pump and associated valving and spray header is independently capable of delivering one-half of the designed flow, or 2500 gpm, which exceeds the minimum containment spray flow of 2180 gpm assumed in the Containment Analysis as described in Table 14.3-40.

The design basis for the containment spray system is, full capacity flow will provide sufficient heat removal capability to maintain the post accident containment pressure below 47 psig, assuming that the core residual heat is released to the containment as steam.

A second purpose served by the containment spray system is to remove elemental iodine and particulates from the containment atmosphere should they be released in the event of a loss-of-coolant accident. The analysis, indicating the system's ability to limit the offsite dose to within applicable limits after a hypothetical loss-of-coolant accident is presented in Section 14.3.6.

To meet the above bases, the following design requirements were established:

1. All components of the system have to meet Class I seismic criteria.
2. The system's initial response has to be fully automatic.
3. Total redundancy of equipment, flow paths, and power supply.
4. Provisions for periodic testing have to be provided.
5. Equipment is to be arranged to provide maximum protection from missiles.

The spray system, including recirculation spray, is designed to operate over an extended time period following a reactor coolant system failure, as required to restore and maintain containment conditions at or near atmospheric pressure. It has the capability of reducing the containment postaccident pressure and subsequent containment leakage. A tertiary function of the system is to provide an alternative means of filling the reactor refueling cavity during reactor vessel head removal.

Portions of other systems that share functions and become part of the containment spray system, when required, are designed to meet the criteria of the containment cooling function. Neither a single active component failure in such systems during the injection phase nor an active/passive failure during the recirculation phase will degrade the design heat removal capability of containment cooling (See section 6.2.3.3).

System piping located within the containment is redundant and separable in arrangement unless fully protected from damage that may follow any reactor coolant system loop failure.

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

6.3.1.7 Service Life

All portions of the system located within containment are designed to withstand, without loss of functional performance, the post-accident containment environment and to operate without benefit of maintenance for the duration of time required to restore and maintain containment conditions at near atmospheric pressure. The recirculation pumps are designed to be operable for 1 yr following a loss-of-coolant accident. Per the 12/06/04 NRC generic SER on NEI 04-07 (Reference 7), evaluations of PWR post accident emergency recirculation performance to address the potential impact of debris blockage per GSI-191 (Reference 6) and Generic Letter 2004-02 (Reference 5) will use a mission time of 30 days.

6.3.1.8 Codes and Classifications

Table 6.3-1 tabulates the codes and standards to which the containment spray system components are designed.

6.3.2 System Design And Operation

6.3.2.1 System Description

Adequate containment cooling and iodine removal by the containment spray system are provided by system components operating in sequential modes. These modes are:

1. Spray a portion of the contents of the refueling water storage tank into the entire containment atmosphere using the containment spray pumps.

2. Recirculation of water from the containment sump by the diversion of a portion of the recirculation flow from the safety injection system to the spray headers inside the containment after injection from the refueling water storage tank has been terminated.

The bases for the selection of the various conditions requiring system actuation are presented in Section 14.3.

The system diagram for the containment spray system is shown in Plant Drawings 9321-2735 & 235296 [Formerly UFSAR Figure 6.2-1].

The principal components of the containment spray system that provide containment cooling and iodine removal following a loss-of-coolant accident consist of two spray pumps, Sodium Tetraborate baskets located in containment, spray ring headers and nozzles, and the necessary piping and valves. The containment spray pumps are located in the primary auxiliary building and the spray pumps take suction directly from the refueling water storage tank.

The containment spray system also uses the two 100-percent capacity recirculation pumps, two residual heat removal heat exchangers and associated valves and piping of the safety injection system for the long-term recirculation phase of containment cooling and iodine removal after the refueling water storage tank has been exhausted.

The Containment Spray System suction piping and the Containment Spray pumps up to the first closed discharge line isolation valve will be maintained sufficiently full of water to ensure the system remains operable and performs properly.

The spray water is injected into the containment through spray nozzles connected to four 360 degree ring headers located in the containment dome area. Each of the spray pumps supplies two of the ring headers.

6.3.2.1.1 Injection Phase

Containment spray will be actuated by two sets of coincidence logic circuits each requiring two-out-of-three, high-high containment pressure signals. This starting signal will start the pumps and open the discharge valves to the spray header.

6.3.2.1.2 Recirculation Phase

When the refueling water storage tank level drops below 2 feet and its contents have been added to the containment floor, recirculation spray flow will be initiated since the NPSH requirements for the recirculation pumps are met for the additional flow needed for combined core injection and recirculation spray. The operator can remotely open the stop valves on either of the two spray recirculation lines. With this split flow, decay heat can be removed and containment airborne activity reduced. This mode of operation will be continued for a period of at least 3.4 hr following the accident in order to continue removal of airborne activity from the containment atmosphere.

After the 3.4 hr containment scrubbing operation it is expected that spray flow would be discontinued while maintaining containment pressure 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 removal heat exchanger once the sump water becomes heated. The heat removal capacity of three of the five fan coolers is sufficient to remove the corresponding energy addition to the vapor space as a result of steam boiloff from the core, assuming flow into the core from one recirculation pump at the termination of injection spray without exceeding containment design pressure (the minimum containment safeguards case which assumes the single failure of one emergency diesel generator is discussed in Section 14.3.5). Hence, it is not expected that continued spray operation for containment heat removal would be required. Spray flow termination is also assumed in the chemical generation analyses for GL 2004-02 compliance. Longer spray times increase exposure time of Aluminum components in the containment to the spray solution and may result in additional chemicals (precipitants) being generated than accounted for in sump strainer head loss calculations.

Sodium Tetraborate is stored at elevation 46’ inside the containment building. During the injection phase the level of the boric acid solution from the containment spray and the coolant lost from the reactor coolant system will rise above the Sodium Tetraborate bins. The Sodium

Tetraborate will dissolve into the solution, providing a solution with pH in the range of 7 to 7.6 to enhance long-term iodine retention in the solution and to minimize corrosion.

6.3.2.1.3 Cooling Water

The cooling water for the residual heat exchangers has been described in Section 6.2.

6.3.2.1.4 Changeover

The sequence for the changeover from injection to recirculation has been described in Section 6.2.

Remote-operated valves of the containment spray system that are under manual control (that is, valves that 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, an audible and visual annunciation is provided on the board to alert the operator of this condition.

6.3.2.2 Components

All associated components, piping, structures, and power supplies of the containment spray system are designed to Class I seismic criteria.

All components inside containment are capable of withstanding or are protected from differential pressures that may occur during the rapid pressure rise to 47 psig in 10 sec. Section 14.3.5.1.1 discusses the analyses that show that the calculated postaccident containment pressures are less severe than this. The lines of the system are protected from missile damage by the concrete crane wall and operating floor.

Parts of the system in contact with the spray solution are stainless steel or an equivalent corrosion-resistant material.

The containment spray system shares the refueling water storage tank capacity with the safety injection system. For a detailed description of this tank, see Section 6.2.

6.3.2.2.1 Pumps

The two containment spray pumps are of the horizontal centrifugal type driven by electric motors.

The design head of the pump is sufficient to continue at rated capacity, with a minimum level in the refueling water storage tank, against a head equivalent to the sum of the design pressure of the containment, the head to the uppermost nozzles, and the line and nozzle pressure losses. Pump motors are direct-coupled and large enough for maximum power requirement of the pump. The materials of construction, which are suitable for use with the spray solutions, are stainless steel or equivalent corrosion-resistant material. Design parameters are presented in Table 6.3-2 and the containment spray pump characteristics are shown on Figure 6.3-1.

The containment spray pumps are designed in accordance to the specifications discussed for the pumps in the safety injection system, Section 6.2.

The recirculation pumps of the safety injection system, which provide flow to the containment spray system during the recirculation phase, are described in Section 6.2.

Details of the component cooling pumps and service water pumps, which serve the safety injection system, are presented in Sections 9.3 and 9.6.

6.3.2.2.2 Heat Exchangers

The two residual heat removal heat exchangers of the safety injection system, which are used during the recirculation phase, are described in Section 6.2.

6.3.2.2.3 Spray Nozzles

The spray nozzles, which are of the hollow cone, ramp bottom design, are not subject to clogging by particles 0.25-in. in maximum dimension, and are capable of producing a surface area averaged drop diameter of approximately 1000 microns at 15 gpm and 40 psi differential pressure. With the spray pump operating at design conditions and the containment at design pressure the pressure drop across the nozzles will exceed 40 psi.

During recirculation spray operation, the water is screened through the 3/32" diameter holes of the perforated plate strainer modules before leaving the recirculation or containment sump. The spray nozzles are stainless steel and have a 0.375-in. diameter orifice. The nozzles are connected to four 360 degree ring headers (alternating headers connected) of radii 7-ft 1.75-in. (El. 228.5-ft), 25-ft 3.438-in. (El. 223.5-ft), 42-ft 3-in. (El. 218.5-ft) and 59-ft 6-in. (El. 213.5-ft). There are 315 nozzles distributed on the four headers. This nozzle and header arrangement results in maximum area coverage with either branch of the system operating alone, while ensuring minimum overlap of spray trajectories in the minimum flow case (Section 14.3).

6.3.2.2.4 Spray Additive Tank

The spray additive tank was removed based on the use of Trisodium Phosphate baskets stored in the Containment building. In response to NRC Generic Letter 2004-02 (Generic Safety Issue 191), the pH buffer material was changed from Trisodium Phosphate to Sodium Tetraborate to minimize the potential for sump screen blockage due to the formation of chemical products. The Sodium Tetraborate baskets are described in Section 6.3.2.2.12.

6.3.2.2.5 Spray Pump Recirculation Line

Each spray pump is provided with a recirculation line from the pump discharge line back to the pump suction line. A globe valve is installed in this line to allow setting of the desired flow rate for both on line and testing configurations.

6.3.2.2.6 Valves

The valves for the containment spray system are designed in accordance with the specifications discussed for the valves in the safety injection system (Section 6.2).

6.3.2.2.7 Piping

The piping for the containment spray system is designed in accordance to the specifications discussed for the piping in the safety injection system (Section 6.2).

The system is designed for 150 psig at 300°F on the suction side and 300 psig at 300°F on the discharge side of the spray pumps.

6.3.2.2.8 Motors for Pumps and Valves

The motors inside and outside containment for the containment spray system are designed in accordance with the specifications discussed for motors in the safety injection system (see Section 6.2).

6.3.2.2.9 Electrical Supply

Details of the normal and emergency power sources are presented in the discussion of the electrical system, Chapter 8.

6.3.2.2.10 Missile Protection

The spray headers are located outside and above the reactor and steam generator concrete shields. A shield, which is removable for refueling also provides missile protection for the area immediately above the reactor vessel. The spray headers are therefore protected from missiles originating within the reactor coolant system.

6.3.2.2.11 Material Compatibility

Parts of the system in contact with the spray solutions are stainless steel or an equivalent corrosion-resistant material. An analysis of materials compatibility with the long-term storage conditions of concentrated sodium hydroxide is presented in Appendix 6D. Appendix 6D is being retained for historical purposes.

All exposed surfaces within the containment have coatings that will not be affected by short term exposure to low pH containment spray solution or to long term exposure to high pH solution. An analysis of the materials exposed to the Post-accident Containment Environment using the original containment spray additive (NaOH) solution is presented in Appendix 6C.

Post-accident chemistry changes due to the elimination of the spray additive tank and the installation of Trisodium Phosphate Baskets were evaluated and it was determined that this change has little effect on the compatibility of materials located in containment, which will come in contact with the initial spray and recirculation spray solution. This evaluation is documented in Reference 2. This evaluation supersedes the information contained in Appendix 6C, with the exception of sections 6C.4.1, and 6C.7. Therefore, with these exceptions, Appendix 6C is being retained for historical purposes.

To improve post-accident ECCS performance, specifically in order to meet the requirements of Generic Letter 2004-02 (Generic Safety Issue 191), the Trisodium Phosphate pH buffer was replaced with Sodium Tetraborate (Reference 3). This buffer material replacement has also been evaluated with respect to post-accident chemistry and material interaction. The evaluation

is documented in Reference 4 and concluded that the pH buffer replacement is acceptable and does not detrimentally affect material compatibility. Appendix 6C has been updated where appropriate to include post accident buffer change to Sodium Tetraborate.

Maintaining the long-term pH of the recirculated ECC solution no less than 7.0, prevents chloride-induced stress corrosion cracking of austenitic stainless steel components, and minimizes hydrogen produced by the corrosion of galvanized surfaces and zinc-based paints as discussed in Reference 1. These chemistry changes using Sodium Tetraborate also do not affect the environmental qualification of equipment located within the containment required to mitigate the consequences of design basis accidents as discussed in Section 7.1.4.

6.3.2.2.12 Sodium Tetraborate Baskets

Sodium Tetraborate (STB) is stored in four baskets at elevation 46' in the containment building. During the injection phase the baskets will be flooded, allowing the STB to dissolve into the fluid for pH control. The four baskets are constructed of stainless steel and are seismically qualified and mounted.

6.3.3 Design Evaluation

6.3.3.1 Range of Containment Protection

For the first 15 to 20 min following the maximum loss-of-coolant accident (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, one spray pump continues to take suction from the RWST and spray into the containment until RWST level drops below 2 feet. This continued spray injection is sufficient to maintain the containment pressure below the design value even if no containment fans were operating.

With the completion of containment spray injection the operator sets up recirculation to one spray header and to the core; the systems are aligned so that sufficient cooled recirculated water is delivered to keep the core flooded as well as to provide flow to one spray header. Flow is maintained to the spray header at this stage primarily to continue scrubbing of airborne activity, i.e., for at least 3.4 hr after the accident; the flow, however, is also sufficient to maintain the containment pressure below the design value. Spray flow termination is also assumed in the chemical generation analyses for GL 2004-02 compliance. Longer spray times increase exposure time of Aluminum components in the containment to the spray solution and may result in additional chemicals (precipitants) being generated than accounted for in sump strainer head loss calculations.

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.

1. Containment Spray alone as follows:
   - Both containment spray pumps operating up to the time the transfer to core recirculation flow begins (during injection phase).
   - One spray pump continuing to take suction from the RWST until the level in the RWST decreases to 2 feet.

- Both recirculation pumps, both residual heat exchangers and both containment recirculation spray headers in operation when the level in the RWST decreases below 2 feet.

2. All five containment cooling fans (discussed in Section 6.4)

3. One containment spray pump and three of the five containment cooling fans (the minimum containment safeguards case which assumes the single failure of one emergency diesel generator is discussed in Section 14.3.5).

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 118-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. Drops of approximately 1000 micron average size will reach temperature equilibrium with the steam-air containment atmosphere after falling through less than half the available spray fall height as discussed in UFSAR Section 14.3.5.2.1.

At containment design temperature, 271^{o}F, the total design heat absorption capability of one spray pump is 218 x 10^{6} Btu/hr based on the assumption of 100^{o}F refueling water and design flow of 2500 gpm.

When the refueling water storage tank level drops below 2 feet, injection spray is terminated and the recirculation pumps supply the flow to the containment recirculation spray headers. Recirculation spray can be established at a flow rate that will maintain containment pressure below the design pressure of 47 psig even if no containment fan coolers are operating.

Elemental iodine and aerosols are removed by the containment spray system. Removal coefficients and the limitations on removal are discussed in Appendix 6A. A discussion of the effectiveness of containment spray as a fission product trapping process is contained in Reference 1.

A single train of containment spray will provide sufficient iodine removal capability to ensure postaccident fission product leakage that would not result in exceeding the applicable dose limits. This is evaluated in Section 14.3.6.

6.3.3.2 System Response

The starting sequence of the containment spray pumps and their related emergency power equipment is discussed in sections 7.2 and 8.2.3.4 and their analyzed performance is discussed in the various Chapter 14 safety analyses.

6.3.3.3 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 6.3-4.

In addition, each spray header is supplied from the discharge from one of the two residual heat removal heat exchangers. As described in Section 6.2.2.1.2, these two heat exchangers are

redundant and can be supplied with recirculated water via separate and redundant flow paths. The analysis of the loss-of-coolant accident presented in Section 14.3 reflects the single failure analysis.

6.3.3.4 Reliance on Interconnected Systems

For the injection phase, the containment spray system operates independently of other engineered safety features following a loss-of-coolant accident except that it shares the source of water in the refueling water storage tank with the safety injection system. The system acts as a backup for the cooling function of the containment air recirculation cooling system. For extended operation in the recirculation mode, water is supplied through recirculation pumps.

During the recirculation phase, some of the flow leaving the residual heat removal heat exchangers may be diverted to the containment spray headers or the high-head safety injection pumps. Minimum flow requirements are established for the flow being sent to the core and for the flow being sent to the containment spray. Sufficient flow instrumentation is provided so that the operator can monitor each flow path as shown in Plant Drawings 9321-2735 & 235296 [Formerly UFSAR Figure 6.2-1].

Normal and emergency power supply requirements are discussed in Chapter 8.

6.3.3.5 Shared Functions Evaluation

Table 6.3-5 contains an evaluation of the main components that have been discussed previously and a brief description of how each component functions during normal operation and during the accident.

6.3.3.6 Containment Spray Pump Net Positive Suction Head Requirements

The net positive suction head for the containment spray pumps is evaluated for injection operation. The end of the injection phase gives the limiting net positive suction head requirement. The net positive suction head available is determined from the elevation head and vapor pressure of the water in the refueling water storage tank, and the pressure drop in the piping to the pump. At the end of the injection phase, the net positive suction head available exceeds the net positive suction head required.

6.3.4 Minimum Operating Conditions

The Technical Specifications establish limiting conditions regarding the operability of the system.

6.3.5 Inspections And Tests

6.3.5.1 Inspections

All components of the containment spray system may 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. During the operational testing of the

containment spray pumps, the portions of the system subjected to pump pressure can be inspected for leaks.

6.3.5.2 Preoperational Testing

The principal components of the containment spray system are two pumps, spray ring headers and nozzles, and the necessary piping and valves.

In discussing preoperational testing and generally proving that the system meets the design specification, it is necessary to consider both individual component testing and onsite testing.

6.3.5.2.1 Offsite Test Work

Three components in the system were subjected to offsite test work:

1. Spray pumps - The spray pumps were subjected to conventional acceptance tests and the performance characteristic was plotted to illustrate the pumps met the design specification.

2. Spray nozzles – As part of the development work in support of Westinghouse Plants, a nozzle of the type used in the spray system was subjected to a performance test to demonstrate and prove the nozzle characteristic, e.g., flow/pressure drop, droplet size, spread of spray, etc.

   As part of the quality assurance program, a random 25-percent of the nozzles installed at the Indian Point Unit 2 site were given a general performance test.

6.3.5.2.2 Onsite Test Work

The aim of onsite preoperational testing was to:

1. Demonstrate and prove that the system is adequate to meet the design pressure conditions. Outside the containment, this involved partial radiographic inspection and partial hydro-testing; inside the containment, the spray headers were subjected to 100-percent radiographic inspection.

2. Demonstrate that the spray nozzles in the containment spray header are clear of obstructions by passing air through the test connections.

3. Verify that the proper sequencing of valves and pumps occurs on initiation of the containment spray signal and demonstrate the proper operation of all remotely operated valves

4. Verify the operation of the spray pumps; each pump was run at shutoff and the mini-flow directed through the normal path back to the refueling water storage tank. During this time, the mini-flow was adjusted to that required for routine testing.

5. Demonstrate the operation of the spray eductors. The eductor and spray additive system was checked by running, in turn, each spray pump on mini-flow with the spray additive tank filled with water and open to the spray eductor suction. During draindown of the spray additive tank, the tank level and corresponding eductor suction flow was recorded via the system instrumentation. Finally, the system performance with water was extrapolated to that with sodium hydroxide, and the adequacy of the system thus verified.

   In order to establish a reference eductor suction test flow for routine testing of the system, the above test was made with the spray additive tank isolated and the eductor drawing water through the refueling water storage tank/eductor suction test line.

6.3.5.2.3 System Testing

The functional test of the injection system described in Section 6.2.5 demonstrates proper transfer to the emergency diesel generator power source in the event of a loss of power. A test signal simulating the containment spray signal is used to demonstrate the operation of the spray system up to the isolation valves on the pump discharge. The isolation valves are blocked closed for the test. These isolation valves are checked separately.

6.3.5.3 <u>Post-operational Testing</u>

6.3.5.3.1 Component Testing

Routine periodic testing of the containment spray system components and all necessary support systems at power is performed. When testing indicates a need for corrective maintenance, the redundancy of equipment in these systems permits such maintenance to be performed without shutting down or reducing load under conditions defined in the Technical Specifications. These conditions would include such matters as the period within which the component should be restored to service and the capability of the remaining equipment to meet safety limits within such a period.

6.3.5.3.2 Routine Inservice Testing

The aim of the periodic test is:

1. To verify that the proper sequencing of valves and pumps occurs on initiation of the containment spray signal and demonstrate the proper operation of all remotely operated valves.

2. To verify the operation of the spray pumps; each pump will be run at shutoff and the mini-flow directed through the normal path back to the refueling water storage tank.

During these tests the equipment can be visually inspected for leaks, leaking seals, or packing; or flanges are tightened to eliminate the leak, and valves and pumps operated and inspected after any maintenance to ensure proper operation.

6.3.5.3.3 System Testing

The post operational testing of the safety injection system is described in Section 6.2.5. Section 8.5 describes the testing required to demonstrate proper transfer to the emergency diesel-generator power source in the event of a loss of power.

REFERENCES FOR SECTION 6.3

1. Letter from Jefferey F. Harold, NRC, to Stephen E. Quinn, Con Edison, Subject: Issuance of Amendment No. 191 for Indian Point Nuclear Generating Unit No.2, dated April 23,1997

2. “Elimination of the Emergency Containment Filtration and Spray Additive Systems from Indian Point Nuclear Generating Station Unit No.2” WCAP-14542 (Westinghouse Non-Proprietary Class 3)

3. Letter from John P. Boska, NRC, to Michael A. Balduzzi, Entergy Nuclear Operations, Subject: Indian Point Nuclear Generating Unit No. 2 – Issuance of Amendment 253, dated February 7, 2008

4. “Evaluation of Alternative Emergency Core Cooling System Buffering Agents,” WCAP-16596-NP Revision 0, July 2006

5. NRC Generic Letter 2004-02, “Potential Impact of Debris Blockage on Emergency Recirculation During Design Basis Accidents at Pressurized-Water Reactors”, dated September 13, 2004.

6. NRC Generic Safety Issue (GSI) 191, “Assessment of Debris Accumulation on PWR Sump Performance.”

7. NRC Generic SER dated 12/06/04 on the NEI 04-07 Guidance Report entitled, “Pressurized Water Reactor Sump Performance Evaluation Methodology.”

TABLE 6.3-1
Containment Spray System - Code Requirements

Component	Code
Valves	USAS B16-5
Piping (including headers and spray nozzles)	USAS B31.1

TABLE 6.3-2
Containment Spray System Design Parameters

Pumps	
Quantity	2
Design pressure, discharge, psig	300
Design pressure, suction, psig	150
Design temperature, ^{o}F	150
Design flow rate, gpm	2600
Design head, ft	450
Maximum pump flow rate, gpm	3450
Shutoff head, ft	490
Motor, hp	400
Type	Horizontal-centrifugal

TABLE 6.3-3
DELETED

TABLE 6.3-4 (Sheet 1 of 2)
Single Failure Analysis - Containment Spray System

Component	Malfunction	Comments and Consequences
A. Spray nozzles	Clogged	Large number of nozzles (315) renders clogging of a significant number of nozzles incredible.
B. Pumps 1) Containment spray pump	Fails to start	Two provided. Limiting containment integrity evaluation based on operation of one pump in addition to four out of five containment cooling fans operating during injection phase.
2) Recirculation pump	Fails to start	Two provided. Limiting containment integrity evaluation based on operation of one pump in addition to four out of five containment cooling fans operating during recirculation phase.
3) Non-essential service water	Fails to start	Three provided. Operation of one pump during recirculation required.
4) Component Cooling	Fails to start	Three provided. Operation of one pump during recirculation required.
5) Auxiliary Component Cooling Pump	Fails to start	Two provided. Operation of one pump during recirculation required.

TABLE 6.3-4 (Sheet 2 of 2)
Single Failure Analysis - Containment Spray System

Component	Malfunction	Comments and Consequences
C. Automatically Operated Valves: (Open on coincidence of two -2/3 high containment pressure signals)		
1) Containment spray pump discharge isolation valve	Fails to open	Two provided. Operation of one required.
2) Isolation valve on component cooling water lines from residual heat exchangers	Fails to open	Two parallel lines, one valve in either line is required to open.
D. Valves operated from control room for recirculation		
1) Containment sump recirculation isolation	Fails to open	Two lines in parallel, one valve in either line is required to open.
2) Containment spray header 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.
4) Residual heat removal pump discharge line	Fails to close	Two valves in series, one required to close (one valve is a check valve). (Valve 744 operated from Control Room once AC power restored to valve controls.)
E. Automatically operated valves (Close from control room on injection to recirculation changeover)		
1) Isolation valves at spray pump discharge	Fails to close	Check valve in series with two parallel valves provided. Operation of one of the two valve arrangement series required.

TABLE 6.3-5
Shared Functions Evaluation

Component	Normal Operating Function	Normal Operating Arrangement	Accident Function	Accident Arrangement
Containment Spray Pumps (2)	None	Lined up to spray headers	Supply spray water to containment atmosphere	Lined up to spray headers

NOTE: Refer to Section 6.2 for a brief description of the refueling water storage tank, recirculation pumps, non-essential service water pumps, component cooling pumps, residual heat exchangers, component cooling heat exchangers and the auxiliary component cooling pumps, which are also associated either directly or indirectly with the containment spray system.

6.3 FIGURES

Figure No.	Title
Figure 6.3-1	Containment Spray Pump Performance Characteristics

6.4 CONTAINMENT AIR RECIRCULATION COOLING SYSTEM

6.4.1 Design Basis

6.4.1.1 Containment Heat Removal Systems

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

Adequate heat removal capability for the containment is provided by two separate, full capacity, engineered safety features systems. These are the containment spray system, whose components are described in Sections 6.2 and 6.3 and the containment air recirculation cooling and filtration system, whose components operate as described in Section 6.4.2. These systems are of different engineering principles and serve as independent backups for each other. Together these two systems provide the single failure protection for the containment cooling function as analyzed in Chapter 14.

The containment air recirculation cooling system is designed to recirculate and cool the containment atmosphere in the event of a loss-of-coolant accident and thereby ensure that the containment pressure will not exceed its design value of 47 psig at 271°F (100-percent relative humidity). Although the water in the core after a loss-of-coolant accident is quickly subcooled

by the safety injection system, the containment air recirculation cooling system is designed on a conservative assumption that the core residual heat is released to the containment as steam.

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

1. All five containment cooling fans.

2. Containment Spray alone as follows:
   - Both containment spray pumps operating up to the time the transfer to core recirculation flow begins (during injection phase).
   - One spray pump continuing to take suction from the RWST until the level in the RWST decreases to 2 feet.
   - Both recirculation pumps, both residual heat exchangers and both containment recirculation spray headers in operation when the level in the RWST decreases below 2 feet.

3. One containment spray pump and three of the five containment cooling fans (the minimum containment safeguards case which assumes the single failure of one emergency diesel generator is discussed in Section 14.3.5).

6.4.1.2 Inspection of Containment Pressure-Reducing Systems

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

Design provisions are made to the extent practical to facilitate access for periodic visual inspection of all important components of the containment air recirculation cooling system.

6.4.1.3 Testing of Containment Pressure-Reducing Systems Components

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

The containment air recirculation cooling system is designed to the extent practical so that the components can be tested periodically, and after any component maintenance, for operability and functional performance.

A number of air recirculation and cooling units are normally in operation and no additional periodic tests are required. The service water pumps that supply the cooling units can be part flow-tested during plant operation via the installed bypass test loop.

6.4.1.4 Testing of Operational Sequence of Containment Pressure-Reducing Systems

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

Means are provided to test initially to the extent practical the full operational sequence of the air recirculation system including transfer to alternative power sources.

6.4.1.5 Inspection of Air Cleanup Systems

Criterion: Design provisions shall be made to the extent practical to facilitate physical inspection of all critical parts of containment air cleanup system, such as, ducts, filters, fans, and dampers. (GDC 62)

Access is available for periodic visual inspection of the containment of recirculation cooling system components.

6.4.1.6 Testing of Air Cleanup Systems Components

Criterion: Design provisions shall be made to the extent practical so that active components of the air cleanup systems, such as fans and dampers, can be tested periodically for operability and required functional performances. (GDC 63)

The valves in a non-operating unit can be periodically tested by actuating the controls and verifying deflection by instruments in the control room. A number of fans are normally in operation; no additional periodic fan tests are necessary.

6.4.1.7 Deleted

6.4.1.8 Deleted

6.4.1.9 Performance Objectives

The containment ventilation system, discussed in Section 5.3, of which all of the components of the containment air recirculation cooling system are a part, is designed to remove the normal heat loss from equipment and piping in the reactor containment during plant operation and to remove sufficient heat from the reactor containment, following the initial loss-of-coolant accident containment pressure transient, to keep the containment pressure from exceeding the design pressure as discussed in Section 14.3.5. The fans and cooling units continue to remove heat after the loss-of-coolant accident and reduce the containment pressure close to atmospheric within the first 24 hr as discussed in Section 14.3.5.1.3. The fan-cooler units could operate continuously after the loss-of-coolant accident and are designed to be operable for 1 year.

In addition to the design bases specified above, the following objectives are met to provide the engineered safety features functions:

1. The heat transfer rate that is assigned to the currently installed fan-cooler units under accident conditions is shown in Figures 14.3-104B and 14.3-104D. The establishment of basic heat transfer design parameters for the cooling coils of fan cooler units are discussed in Section 14.3.5.2.2. 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.

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

3. Each of the five air-handling units is equipped with moisture separators rated for full unit flow.

In addition to the above design bases, the equipment was originally specified to be capable of withstanding, without impairing operability, a pressure of 70.5 psig and 298°F for a period of one hour. The motors were further specified to be capable of running for 48 hours at required fan load in an atmosphere consisting of an air water vapor mixture initially at 47 psig and 271°F, and of continuous operation at 10 psig and 175°F. These ambient conditions and operating times have been updated and are maintained by the ongoing Environmental Qualification Program discussed in Section 7.1.4. As part of this program, the fan motors are qualified to withstand containment environment conditions following the loss of coolant accident so that the fans can perform their required function during the recovery period (1 year).

All components are capable of withstanding or are protected from differential pressures that may occur during the rapid pressure rise to 47 psig in 10 sec. Section 14.3.5.1.1 discusses the analyses that show that the calculated post-accident containment pressures are less severe than this.

Portions of other systems that share functions and become part of this containment cooling system when required are designed to meet the criteria of the containment cooling function. Neither a single active component failure in such systems during the injection phase nor an active/passive failure during the recirculation phase will degrade the heat removal capability of containment cooling (See Section 6.2.3.3).

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

1. Means for isolation of any section.
2. Means to detect and control radioactivity leakage into the environs.

6.4.2 System Design And Operation

The flow diagram of the containment air recirculation cooling system is shown in Plant Drawing 9321-4022 [Formerly UFSAR Figure 5.3-1].

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

6.4.2.1 Containment Cooling System Characteristics

The air recirculation system consists of five 20-percent capacity air-handling units, each including motor, fan, cooling coils, moisture separators, roughing filters, duct distribution system, instrumentation, and controls. The units are located on the intermediate floor between the containment wall and the primary compartment shield walls. The air recirculation system has a total heat removal capability of at least 308.5 MBtu/hr under conditions following a loss-of-coolant accident and at a service water temperature of 95°F.

Each fan is designed to supply 65,000 cfm at approximately 22.8-in. static pressure, 271°F, 0.175 lb/ft^3 density. The fans are direct-driven, centrifugal type, and the coils are plate fin-tube type.

Air-operated, tight-closing, 125 psi USAS butterfly valves isolate any inactive air-handling unit from the duct distribution system. Ductwork distributes the cooled air to the various containment compartments and areas. During normal and accident operation, the flow sequence through each air-handling unit is as follows: cooling coils, moisture separators, fan, discharge header.

Roughing filters are installed up-stream of the cooling coils during plant cleanup and any time the reactor is down. These roughing filters are not in place during power operation.

Plant Drawing 9321-4026 [Formerly UFSAR Figure 6.4-3] is an engineering layout drawing of an air-handling unit showing the arrangement of the above components in the unit. Plant Drawing 9321-2502 [Formerly UFSAR Figure 5.1-3] shows the location of the five units on the intermediate floor (elevation 68-ft-0-in.).

6.4.2.1.1 Actuation Provisions

The butterfly valves have only two positions, full open and full closed. These valves are air operated and spring loaded. Upon loss of control signal or control air, the spring actuates the valve to the accident position (fail-safe operation).

Upon either manual or automatic actuation of the safety injection safe-guards sequence, the butterfly valves are tripped to the accident position. Accident position is also the fail-safe position.

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

The containment pressure is sensed by eight separate pressure transmitters (six of which are used to generate automatic actuation signals) located outside the containment. Containment pressure is communicated to the transmitters through three ¾-in. stainless steel lines penetrating the containment vessel. A high containment pressure signal automatically actuates the safety injection safeguard sequence (reference is made to Section 6.2.2), which trips the valves to the accident position.

Two high-range containment pressure transmitters have been installed in response to NUREG-0737. These transmitters allow for the indication and recording of pressure up to 150 psig. These signals are recorded on two independent recorders located on the accident assessment panel in the common Unit 1/Unit 2 central control room. The fans are part of the engineered safety features and either all five, or at least three of five fans will be started after an accident, depending on the availability of emergency power. (Reference is made to Section 8.2.)

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 ductwork system, operating both normally and after an accident, indicate whether air is circulating in accordance with the design arrangement. Abnormal flow alarms are provided in the control room.

6.4.2.1.2 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 arrangement is shown in Plant Drawing 9321-4022 [Formerly UFSAR Figure 5.3-1].

In addition to ventilating areas inside the periphery of the shield wall, the distribution system also includes two branch ducts located at opposite extremes of the containment wall for ventilating the upper portion of the containment. These ducts are provided with nozzles and extend upward along the containment wall as required to permit the throw of air from nozzles to reach the dome area and ensure that the discharge air will mix with the atmosphere.

The air discharge inside the periphery of the shield wall will circulate and rise above the operating floor through openings around the steam generators where it will mix with air displaced from the dome area. This mixture will return to the air-handling units through floor gratings located at the operating floor directly above each air-handling unit inlet. The temperature of this air will be essentially the ambient existing in the containment vessel. The steam-air mixture from the containment entering the cooling coils during the accident will be at approximately 271^{o}F and have a density of 0.175 lb/ft^{3}. Part of the water vapor condenses on the cooling coil and is collected by the condensate trays. The condensate from the trays is directed below to the floor tray by means of an individual piping system. The condensate collection and drain system is important to the proper functioning of the cooling coils. The air leaving the unit thus will be saturated at a temperature slightly below approximately 265^{o}F. The fluid will leave the cooling coils and enter the moisture separators at approximately 265^{o}F and saturated (100-percent relative humidity) condition. The purpose of the moisture separators is to remove the entrained moisture.

The fluid will remain in this condition as it flows through 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 265°F and will decrease the relative humidity slightly below 100-percent.

With a flow rate of approximately 65,000 cfm from each fan under accident conditions and the containment free volume of 2,610,000-ft^3, the recirculation rate with five fans operating is approximately 7.5 containment volumes per hour.

6.4.2.1.3 [Deleted]

6.4.2.1.4 Cooling Water for the Fan-Cooler Units

The cooling water requirements for all five fan-cooling units during a major loss-of-primary-coolant accident and recovery are supplied by two of the three nuclear service water pumps. The service water system is described in Section 9.6.

The cooling water discharge from the fan and motor cooling coils flows to the discharge canal. As a protective measure a sample of the effluent from these coils is monitored for radioactivity. The sample from the coils is monitored by two redundant radiation monitors. Upon indication of radioactivity in the effluent, each cooler discharge line is monitored individually to locate the defective cooling coil, which when identified would be isolated and operation would continue with the remaining units. The service water system pressure at locations inside the containment in the incident mode system alignment (which the system automatically assumes following a safety injection signal) could be below the containment design pressure of 47 psig. However, since the cooling coils and service water lines are a completely closed system inside the containment, no contaminated leakage is expected into these units.

Local indication of service water discharge temperature from each fan-motor heat exchanger, as well as a fan cooler unit combined outlet header temperature indicator, are provided in the pipe pen outside containment. A fan-motor heat exchanger combined discharge header flow indicator is also located in the pipe pen outside containment. Flow for each fan cooler unit is indicated in the control room. Abnormal flow alarms are provided in the control room. A permanent differential pressure indicator has been installed across the 10-in. inlet and outlet water headers of the fan cooling units for pressure drop measurements.

During normal plant operation, flow through the cooling units can be throttled for containment temperature control purposes by a valve on the common discharge header from the cooling units. Two independent, full flow, isolation valves open automatically to bypass the control valve in the event of a safety injection signal. Both valves fail in the open position upon loss of air pressure and either valve is capable of passing the full flow required for all five fan cooling units.

6.4.2.1.5 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.

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

All of the air-handling units are located on the intermediate floor between the containment building and the primary containment shield wall. The distribution header and service water cooling piping are also located outside the shield wall. This arrangement provides missile protection for all components.

6.4.2.2 Components

6.4.2.2.1 Moisture Separators

The moisture separators are designed to remove a minimum of 99.9-percent of the entrained water in the air-steam mixture entering the air-handling units following a loss-of-coolant accident. With an air entrained moisture content of 0.35 lb $H_2O/1000\text{-}ft^3$, the water flow rate entering the moisture separator section is approximately 23 lb/min and the moisture separator effluent has essentially zero moisture content.

Each bank is designed for horizontal air flow and is composed of 40 elements. Each element or separator is 24-in. x 25-in. x 2-in. (minimum) thick and is mounted in a steel support frame.

A steel drain trough is incorporated for each horizontal tier of separators to collect and remove the water that is recovered from the air steam. Further, the design enables the separators to be removed from the upstream side of the support frame.

In order to prevent the bypass of air around the bank, airtight seals are provided between the floor, walls, plenum, and around the perimeter of each moisture separator. The tight seal is accomplished by gaskets, adhesive, and pressure-sealing tape, all of which can withstand a temperature of 300^o. The thickness of the gaskets is 0.25-in. for the separator elements and 0.375-in. for the perimeter sealing of the support frame; they do not extend into the media area when installed.

The moisture separator elements are of fire-resistant construction and consist of mats of fiberglass pads reinforced with stainless steel wire mesh. Nonstainless steel parts used in the construction are protected against corrosion by painting with one 3-mil shop coat of Carbo Zinc No. 11 or the equivalent. The separator frames are fabricated of type 304L stainless steel with welded joints.

6.4.2.2.2 Roughing Filters

The roughing filters remove the large particles from the air stream before contact is made with the cooling coils. The roughing filters are in operation during plant cleanup and any time the reactor is down. These are efficient for removing large particles and under normal conditions they offer a resistance to air flow of 0.2-in. of water.

As in the case for all components of the air-handling recirculating system, the bank is designed for horizontal air flow. The bank contains 40 filters, each of which has dimensions of 22.875-in. wide x 23.5-in. high x 2-in. thick.

All other details of the mounting frame, sealing and materials of construction, other than the filters themselves, are the same as described for the moisture separators.

The filter is of fire-resistant construction and the medium is fiber glassmat.

6.4.2.2.3 Humidity Detectors

Located just upstream of each fan cooling unit is a humidity detector used to determine the dewpoint in containment. See Section 6.7 for further details.

6.4.2.2.4 [Deleted]

6.4.2.2.5 Fan Motor Units

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

Each fan can provide a minimum flow rate of 65,000 cfm when operating against the system resistance of approximately 22.8-in. static pressure existing during the accident condition (0.175 lb/ft^3 density, containment pressure of 47 psig, and temperature of 271°F).

The reactor containment fan cooler motors are Westinghouse or Schulz Supplied Re-wound, totally enclosed water-cooled, 350 hp, induction type, three-phase, 60 cycles, 1200 rpm, 440-V with ample insulation margin. Significant motor details are as follows:

1. Insulation

   Class F (NEMA rated total temperature 155°C) Westinghouse Thermalastic or Class H (NEMA rated total temperature 180°C) Schulz Epoxilite. It is impregnated and coated to give a homogeneous insulation system that is highly impervious to moisture. Internal leads and the terminal box-motor interconnection are given special design consideration to ensure that the level of insulation matches or exceeds that of the basic motor system. The Fan Cooler Motors and their lubrication are environmentally qualified for use inside the containment building as documented in their respective EQ files.

2. 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 is returned to the motor. A flapper type vent relief valve permits incident ambient (increasing containment pressure) to enter the motor air system so the bearings will not be subjected to differential pressure. The cooling coil condensate drain line will enable pressure equalization by the motor heat exchanger as the containment pressure is reduced. Water connections are welded throughout and supply and discharge are common with the containment cooler water system, i.e., supplied from the nuclear service water header. The drain is piped to the containment cooler drain system.

3. 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. Continuous bearing monitoring that will alarm in the control room is provided.

4. Conduit (Connection) Box

The motor leads are brought out of the frame through a seal and into a sealed conduit box.

5. [Deleted]

6.4.2.2.6 Charcoal Filter Housing Pressure Equalization

The charcoal filter housing includes a hole on the external wall of the fan cooler unit. The hole remained after the spring-loaded damper and the charcoal filters were removed and the charcoal filter inlet and outlet dampers were blocked closed during the 2002 refueling outage. When containment pressure rapidly rises as a result of an accident condition, the hole will relieve air into the charcoal filter compartment so as to minimize the negative pressure differential across the walls of the charcoal filter unit. The hole will help minimize positive differential pressure (between containment and the charcoal filter compartment) that can occur as a result of relieving containment pressure during normal plant operation.

6.4.2.2.7 [Deleted]

6.4.2.2.8 Cooling Coils - Original Plant Design

This section describes the cooling coils provided as part of the original plant operation.

The heat removal capability of the cooling coils was 76.32 x 10^6 Btu/hr per air-handling unit at saturation conditions (271°F, 47 psig). The design internal pressure of the coil was 150 psig at 300°F and the coils could withstand an external pressure of 70.5 psig at a temperature of 300°F without damage.

Each recirculating unit consisted of 10 coil units mounted in two banks of five coils high. These banks were located one behind the other for horizontal series air flow and the tubes of the coil were horizontal.

Each coil assembly consisted of the first bank having six rows of coils and the second bank having four rows of coils. Each bank contained four Westinghouse Sturtevant designation WC-36208 (36-in. high by 108-in. long) coil, and one Westinghouse Sturtevant WC-30108 (30-in. high by 108-in. long) coil. This latter coil was at the top and had 16.7-percent fewer tubes. The coils were stacked five high to a bank. The total coil assembly (two banks of coils) was 42-in. wide. There were 10 rows of tubes in the horizontal flow direction and a total of 116 rows of tubes in the vertical direction. Cooling water flow was 1/3 velocity through the first coil bank (six rows of tubes in the horizontal). Tube supports were provided on 15-in. center lines to permit free expansion and contraction of the tubes. Supply and return manifolds at each coil were 4-in. and 3-in. in diameter 90-10 copper-nickel schedule 40 pipe for each bank, respectively.

Tubes were 0.625-in. in diameter with 0.035-in. wall thickness. Each "U" tube contained six brazed joints and was expanded to copper plate fins, each .008-in. thick along its straight lengths. The original coils were fabricated of copper plate fins vertically oriented on cupro-nickel tubes. For normal operation, 8 fins/in. were required to remove 2,000,000 Btu/hr using 108 tubes.

The original fan-motor heat exchangers had similar construction. Each tube had 8 passes and 18 brazed joints, was 0.625-in. in diameter and 0.049-in. wall thickness, and had manifold headers 2-in. in diameter.

In 1981, however, these coils were replaced by those of another manufacturer (CVI Corporation). These replacement coils are described in Section 6.4.2.2.9.

Local indication of service water discharge temperature from each fan-motor heat exchanger, as well as a fan cooler unit combined outlet header temperature indicator, are provided in the pipe pen outside containment. A fan-motor heat exchanger combined discharge header flow indicator is also located in the pipe pen outside containment. Flow for each fan cooler unit is indicated in the control room. Alarms indicating abnormal service water flow and radioactivity are provided in the control room.

Pressure taps to which a ΔP meter can be attached or provided to allow measurements of service water pressure drops through the individual fan cooler units. This instrumentation is intended to indicate fouling in the river water side of the cooling coil.

The coils are provided with drain pans and drain piping to prevent flooding during accident conditions. This condensate is drained to the containment sump. Reference is made to Section 6.7.

Drain flow is measured by a level transmitter located in a standpipe containing a slotted weir. A level alarm and indication is provided in the control room. Actual discharge flow rate is determined by referring the level to a calibration curve for the weir or by use of the weir meter.

If the drainage rate for all five units is nearly the same, it may be concluded that this water is condensate from the containment atmosphere. A particular unit with a high drainage rate with respect to the other units could be an indication of a leak in one of the cooling coils.

6.4.2.2.9 Cooling Coils – Modified

Fan-cooler cooling coils and fan-motor heat exchangers were replaced during the 1980/1981 refueling outage. Two of these coils and two exchangers were replaced during the 1986 refueling outage, and the remaining three were replaced during the 1987 refueling outage.

The modified (1980/1981) design for all cooling coils (fan cooler coils and motor heat exchangers) was changed to 90-10 copper-nickel (CuNi) water box headers with removable cover plates to allow for inservice inspection and maintenance. Water box headers were of bolted construction and consist of tubesheets, spacer sections, coverplates, and flanged elbows with gaskets for all mating surfaces. Perforated baffle plates, also of 90-10 coppernickel (CuNi) were installed at the water box inlet of each fan-cooler/motor cooler unit to provide a more even flow distribution through the cooling coils. All "U" tubes are of 0.049-in. wall thickness, "hair-pin" construction, and are rolled into a tubesheet. All brazed joints (approximately 1800) are

eliminated in this modified design. The modified fan cooler cooling coil assemblies have two banks each with five coils. However, both banks now have six rows of tubes for each coil. The upper coils are still proportionally shorter than the rest.

The coils and exchangers installed in 1986-87 are similar, but utilize larger titanium water boxes, AL6X tubes, and fully-captured "O" ring gaskets.

The gasket material provided with the new cooling coils is ethylene propylene diene monomer. Evaluations performed on this type of material have included exposure to radiation levels of 2 x 10^8 rads over one year (postaccident). The results of these tests indicated that this gasket material retained its functional capability (no leaks under hydrostatic test pressure of 225 psig). Evaluation of the integrated radiation dose rate to the cooling coils is estimated to be 7.4 x 10^6 rads based on the guidelines set forth in NUREG-0588 at a power level of 2758 MWt.

6.4.2.2.10 Ducting

The ducts are designed to withstand the sudden release of reactor coolant system energy and energy from associated chemical reactions without failure due to shock or pressure waves by incorporation of dampers along the ducts, which open at slight overpressure of 5 psi or less. The ducts are designed and supported to withstand thermal expansion during an accident.

Where flanged joints are used, joints are provided with gaskets suitable for temperatures to 300°F.

Ducts are constructed of corrosion-resistant material.

6.4.2.2.11 [Deleted]

6.4.2.2.12 Electrical Supply

Details of the normal and emergency power sources are presented in Chapter 8.

Further information on the components of the containment air recirculation cooling system is given in Section 5.3.

6.4.3 Design Evaluation

6.4.3.1 Range of Containment Protection

The containment air recirculation cooling system provides the design heat removal capacity for the containment following a loss-of-coolant accident 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 cooling coils to transfer heat from containment to service water.

The performance of the containment recirculation cooling system for pressure reduction is discussed in Section 14.3.

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

1. All five containment cooling fans.

2. Containment Spray alone as follows:
   - Both containment spray pumps operating up to the time the transfer to core recirculation flow begins (during injection phase).
   - One spray pump continuing to take suction from the RWST until the level in the RWST decreases to 2 feet.
   - Both recirculation pumps, both residual heat exchangers and both containment recirculation spray headers in operation when the level in the RWST decreases below 2 feet.

3. One containment spray pump and three of the five containment cooling fans (the minimum containment safeguards case which assumes the single failure of one emergency diesel generator is discussed in Section 14.3.5).

Following a loss-of-coolant accident both the containment spray system and reactor containment fan cooler system are placed in operation for heat removal, and containment air recirculation. The containment spray system also provides fission product reduction.

During the injection phase of the accident, a minimum of one spray pump and three of five fan coolers are in operation.

The heat removal requirement for the design basis accident are met with these minimum requirements during both the injection and recirculation phase. Section 14.3.5 discusses the pressure transient and heat removal capability of using minimum safeguards.

Since the spray is effective in removal of inorganic iodine during the first 3.4 hr period following the accident, the spray flow could be terminated (subsequent to 3.4 hr) after the containment pressure is reduced and stabilized. For Generic Letter 2004-02 compliance, spray termination must be no later than 3.5 hours.

The fan cooling units would continue in operation alone during the long-term recirculation phase during which the containment pressure is continually reduced. In addition, effective recirculation is provided to all parts of the containment. Suction to the fan cooling units is taken from the upper portion of the containment and discharged from the fan coolers through a ring header to various compartments below the operating deck.

6.4.3.2 System Response

The starting sequence of the five containment cooling fans and the related emergency power equipment is discussed in sections 7.2 and 8.2.3.4 and their analyzed performance is discussed in the various Chapter 14 safety analyses.

6.4.3.3 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 6.4-1.

The analysis of the loss-of-coolant accident presented in Section 14.3 is consistent with the single-failure analysis.

6.4.3.4 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. Three nuclear service water pumps are provided, only two of which are required to operate during the post-accident period.

6.4.3.5 Shared Functions Evaluation

Table 6.4-2 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.

6.4.3.6 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 "cleanup" because the interior air continually recirculates through the heat exchanger.

The fan cooler motor is an environmentally qualified motor designed to operate during accidents in the accident environment that it is exposed to. The increase in service water temperature to 95°F will not impact the life expectancy of the motors.

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.

The bearings are designed to perform in the incident ambient temperature conditions. However, it will be noted that the interior bearing housing details are cooled by the heat exchanger. It is expected that bearing temperatures would be 125°C to 140°C 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 function of preventing moisture from reaching the winding keeps the winding in much more favorable conditions. In addition, it will 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, if present, out of the winding.

Following the incident rise in pressure, it is not expected that there will be significant mixing of the motor (closed system) environment and the containment ambient.

The heat exchanger has been designed using a very conservative fouling factor.

To prove the effectiveness of the heat exchanger in inhibiting large quantities of the steam-air mixture from impinging on the winding and bearings, a full-scale motor of the exact same type as described was subjected to prolonged exposure of accident conditions. The test exposed the motor to a steam-air mixture as well as boric acid and alkaline spray at 80 psig and saturated temperature conditions. Insulation resistance, winding and bearing temperature, relative humidity, voltage and current, as well as heat exchanger water temperature and flow were recorded periodically during the test.

Following the test the motor was disassembled and inspected to ensure that the unit performed as designed. The post-testing inspection showed no degradation of the motor components (Details are reported in Reference 5).

6.4.4 Minimum Operating Conditions

The Technical Specifications establish limiting conditions regarding the operability of the air recirculation units.

6.4.5 Inspections And Testing

6.4.5.1 Inspection

Access is available for visual inspection of the containment fan-cooler and recirculation components including fans, cooling coils, butterfly valves, and ductwork.

6.4.5.2 Component Testing

The butterfly valves on each air-handling unit can be operated periodically to ensure continued operability. The degree of leaktightness of the valves was established by test at the time of installation.

6.4.5.3 System Testing

Each fan cooling unit was tested after installation for proper flow and distribution through the duct distribution system. Four of the fan cooling units are expected to be used during normal operation. Five will only be required for normal operation when the service water inlet temperature is 85°F or higher. The fan not in use can be started from the control room to verify readiness. The associated butterfly valves will be tested only when the fan is running.

6.4.5.4 Operational Sequence Testing

The test described in Section 6.2.5 demonstrates proper transfer and sequencing of the fan motor supplies to the diesel generators in the event of loss of power. A test signal is used to demonstrate proper valve motion and fan starting. This test verifies proper functioning of the vane-switch flow indicators.

REFERENCES FOR SECTION 6.4

1. Deleted

2. Deleted

3. Nuclear Safety Quarterly Report - August, September, October, 1969, Engineered Safety System Studies, BNWL-1266.

4. Deleted

5. C. V. Fields, Fan Cooler Motor Unit Development and Test, WCAP-9003 (Proprietary), Westinghouse Electric Corporation.

6. Schulz Electric Report No. N4446EQFWCD, “Environmental Qualification Report Number N4446EQFWCD for Schulz Electric Company’s Form Wound, Continuous Duty Insulation System”.

7. Schulz Electric Report No. 45925-1, “Schulz Electric Company’s Environmentally Qualified Insulation System Supplement 1.”

TABLE 6.4-1
Single Failure Analysis – Containment Air Recirculation Cooling System

Component	Malfunction	Comments And Consequences
A. Containment cooling fan	Fails to start	Five provided. Limiting containment integrity evaluation based on four fans in operation and one containment spray pump operating during the injection phase.
B. Nuclear service water pumps	Fails to start	Three provided. Two required for operation.
C. Automatically operated valves: (Open on automatic safeguards sequence)		
1. Charcoal filter compartment butterfly valves	Fails to open	None, charcoal filters are no longer credited and have been removed.
2. Nuclear service water discharge line isolation valve	Fails to open	Two provided. Operation of one required.

TABLE 6.4-2
Shared Functions Evaluation

Component	Normal Operation Function	Normal Operating Arrangement	Accident Function	Accident Arrangement
Containment cooling fan units (5)	Circulate and cool containment atmosphere	Up to five fan units in service	Circulate and cool containment atmosphere	Five fan units in service
Nuclear service water pump (3)	Supply river cooling water to fan units	Up to three pumps in service	Supply river cooling water to fan units	Three pumps in service

6.4 FIGURES

Figure No.	Title
Figure 6.4-1	Deleted
Figure 6.4-2	Deleted
Figure 6.4-3	Containment Building Air Recirculation Fan Cooler Filter Unit - Plan and Section, Replaced with Plant Drawing 9321-4026
Figure 6.4-4	Deleted

6.5 ISOLATION VALVE SEAL-WATER SYSTEM

6.5.1 Design Bases

The isolation valve seal-water system ensures the effectiveness of those containment isolation valves that are located in lines connected to the reactor coolant system or that could be exposed to the containment atmosphere during any condition, which requires containment isolation, by providing a water seal (and in a few cases a gas seal) at the valves. The system provides a simple and reliable means for injecting seal water between the seats and stem packing of the globe and double-disk 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 the isolation valve seal-water system is designed to automatically initiate during any accident condition requiring containment isolation, the primary function of the system is to limit the fission product release associated with a large break LOCA.

Although no credit is taken for the operation of this system in the calculation of offsite accident doses as discussed in Section 14.3.6, it does provide assurance that the containment leak rate is lower than that assumed in the accident analysis should an accident occur.

Design provisions for inspection and testing of the isolation valve sealwater system are discussed in Section 6.5.5.

See Section 5.2, containment isolation system, for containment isolation diagrams (Figures 5.2-1 through 5.2-29), the tabulation of isolation valve parameters (Table 5.2-1), and a description of the derivation of "phase A" and "phase B" containment isolation signals. Section 5.2.2 discusses the containment isolation valves that are sealed, post-accident, by air from the penetration and weld channel pressurization system.

6.5.2 System Design And Operation

6.5.2.1 System Description

The isolation valve seal-water system flow diagram is shown in Plant Drawing 9321-2746 [Formerly UFSAR Figure 6.5-1]. System operation is initiated either manually or by a "phase A" containment isolation signal. When actuated, the isolation valve seal water system interposes water inside the penetrating line between two isolation points located outside the containment. The resulting water seal blocks the leak-age of the containment through valve seats and stem packing. The water is introduced at a pressure slightly higher (52 psig, minimum) than the containment design pressure of 47 psig. The high-pressure nitrogen supply used to maintain pressure in the seal-water tank does not require an external power source to maintain the required driving pressure. The possibility of leakage from the containment or reactor coolant system past the first isolation point is thus prevented by ensuring that if leakage does exist, it will be from the seal-water system into the containment.

The following lines would be subject to pressures greater than the operating pressure of the seal water portion of the isolation valve seal-water system. These lines are supplied with 250 psig nitrogen for the high pressure portion of the system which exceeds worst case internal post-accident process pressure:

1. Residual heat removal loop inlet line.
2. Residual heat removal loop outlet line.
3. Bypass line from residual heat exchanger outlet to safety injection pumps suction.
4. Residual heat removal pumps mini-flow line.
5. Residual heat removal loop sample line.
6. Recirculation pump discharge sample line.

The isolation valves for those lines can be sealed by nitrogen gas from the high-pressure nitrogen supply of the isolation valve seal-water system. A self-contained pressure regulator operates to maintain the nitrogen injection pressure slightly higher than the maximum expected line pressure. The nitrogen gas injection is manually initiated. The system includes one seal-water tank capable of supplying the total requirements of the system. The tank is normally pressurized from the Nitrogen System through pressure control valves. As a backup, the tank may be pressurized from the system's own supply of high-pressure nitrogen cylinders through pressure control valves. Design pressure of the tank and injection piping [***Note*** *- The injection piping runs and nitrogen supply piping are fabricated using 3/8-in.-OD tubing, which is capable of 2500-psig service.*] is 150 psig, and relief valves are provided to prevent overpressurization 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 valve failure in the seal-water line. The design parameters of the seal-water tank are presented in Table 6.5-1.

In lines approximately 3-in. and larger, double disk gate valves are generally used for isolation. A drawing of this valve is presented in Figure 6.5-2. Redundant isolation barriers are provided when

the valve is closed. The upstream and downstream disks are forced against their respective seats by the closing action of the valve. Seal-water is injected through the valve bonnet and pressurizes the space between the two valve disks. The seal-water pressure in excess of the potential accident pressure eliminates any outleakage past the first isolation point.

For smaller lines, isolation is generally 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 maximum acceptable leakage across both the seat and stem packing of any gate or globe valve is nominally 10 cm^3/hr-in. of nominal pipe diameter. Tests on these valves have indicated that much lower leakage rates can be expected. However, the design of the isolation valve seal-water system is based on the conservative assumption that all isolation valves are leaking at five times the acceptable value, or 50 cm^3/hr-in. of nominal pipe diameter. In addition, a screening criterion of 25 cm^3/min for an individual isolation valve leakage has been established, beyond which point a determination will have to be made whether the valve is still functionally acceptable. Should one of the isolation valves close, but fail to seal, it is conservatively assumed that flow through the failed valve will be limited to approximately 100 times the maximum acceptable leakage valve, or 1000 cm3/hr-in. of nominal pipe diameter, by the resistance of seal-water injection path through the valve.

If a containment isolation valve fails to close, a water seal at the failed valve is ensure by proper slope of the potential line, or a loop seal, or by additional valves on the side of the isolation valves away from the containment. Excessive seal-water flow to those motor operated isolation valves that could potentially fail to close in response to a containment isolation signal is limited by flow restrictive orifices installed in the seal-water injection lines.

The seal-water tank is sized to provide at least 24-hr supply of seal-water with all of the isolation valves leaking at the design rate of 50 cm3/hr-in., plus the failure of the largest containment isolation valve to seat and leaking at the maximum rate of 1000 m3/hr-in. The seal-water volume required to satisfy these conditions is approximately 144 gal. The 176-gal seal-water tank is provided with low level and low-low level alarms to signal the need for makeup during normal operation and accident conditions, respectively. If all of the isolation valves seat properly, as expected, the tank volume is sufficient for more than 2.5 days of operation at design seal-water flow rates before makeup is required. Two separate sources of makeup water are provided to ensure that an adequate supply of seal-water is available for long-term operation.

For an event resulting in a "phase A" containment isolation signal, but not a "phase B" containment isolation signal, the isolation valve seal-water system is automatically initiated. Flow to the four isolation valves associated with a "phase B" containment isolation signal only will be automatically isolated by solenoid operated valves, and remain isolated unless the containment isolation valves close. This design will prevent seal-water flow to the opened containment isolation valves. In the event that the solenoid operated valves fail to close, excessive flow would be limited to flow restrictive orifices installed in the seal-water injection lines. This design ensures sufficient time for operator action to provide make-up water if long-term system operation is required to limit fission product releases. If long-term isolation valve seal-water system operation is not required, the system may be isolated by operator action.

There are two separate seal-water lines supplying the potentially radioactive and nonradioactive systems, respectively. This prevents the contamination of nonradioactive systems by way of isolation valve seal-water manifolds.

6.5.2.2 Seal-Water Actuation Criteria

Containment isolation (Section 5.2) and seal-water injection are accomplished automatically on phase A isolation actuation 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. Generally, the following criteria determine whether the isolation and seal-water injection are 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 loss-of-coolant accident. For example, these lines include:

1. Reactor coolant pump cooling water supply and return lines (phase B isolation).
2. Reactor coolant pump seal-water return line (phase B isolation).
3. Excess letdown heat exchanger cooling water supply and return lines (phase A isolation).
4. Chemical and volume control system letdown line (phase A isolation).
5. Reactor coolant system sample lines and sample return line (phase A isolation).
6. Containment vent header (phase A isolation).
7. Reactor coolant drain tank gas analyzer line (phase A isolation).
8. Auxiliary steam supply and condensate return lines (Manual valves qualifying as automatic isolation valves per Section 5.2.2).
9. Service air and city water lines (Manual valves qualifying as automatic isolation valves per Section 5.2.2).

Automatic containment isolation and automatic seal-water injection are also provided for the following lines, which are not connected directly to the reactor coolant system, but terminate inside the containment at certain components. These components can be exposed to the reactor coolant or containment atmosphere as the result of leakage or failure of a related line or component. The isolated lines are not required for post-accident service. For example, these lines include:

1. Pressurizer relief tank gas analyzer line.
2. Pressurizer relief tank makeup line.
3. Safety injection system test line.
4. Reactor coolant drain tank pump discharge line.
5. Steam-generator blowdown/sample lines.
6. Accumulator sample line.

7. Containment sump pump discharge.

Remote manual containment isolation and remote manual seal-water injection are provided for lines that are normally sufficiently filled with water and will remain sufficiently filled following the loss-of-coolant accident and for lines that must remain in service for a time following the accident. The remote manual seal-water injection ensures a long-term seal. For example, these lines include:

1. Reactor coolant pump seal-water supply lines.
2. Chemical and volume control system charging line.
3. Safety injection headers.
4. Containment spray headers.

Manual containment isolation and remote manual nitrogen seal injection are provided for lines that are sufficiently filled with water during the accident, but which are at a pressure higher than that provided by the isolation valve seal-water system. These lines must remain in service for a period of time following the accident or may be placed in service on an intermittent basis following the accident. For example, these lines include:

1. Residual heat removal loop inlet line.
2. Bypass line from residual heat exchanger outlet to safety injection pumps suction.
3. Residual heat removal loop sample line.
4. Recirculation pump discharge sample line.
5. Residual heat removal loop outlet line.
6. Residual heat removal pumps mini-flow line.

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, which opposes leakage from the containment. These include nitrogen supply lines to the pressurizer relief tank, accumulators, and reactor coolant drain tank, the instrument air header and the weld channel pressurization air lines.

2. Lines that do not communicate with the containment or reactor coolant system 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 loss-of-coolant accident. These include the steam and feedwater headers and the containment ventilation system cooling water supply and return lines.

3. Lines that are designed for post-accident service as part of the engineered safety features, such as the containment sump recirculation line. This line is connected to a closed system outside containment.

4. Special lines, such as the fuel transfer tube, containment purge ducts, and the containment pressure relief line, which are pressurized by the containment penetration and weld channel pressurization system (see Section 6.6). The zone between the two gaskets sealing the blind flange to the inner end of the fuel transfer tube is pressurized to prevent leakage from the containment in the event of an accident. The zone between the two butterfly valves in each containment

purge duct is pressurized above incident pressure, while the valves are closed during power operation, as are the two spaces between the three butterfly valves in the containment pressure relief line.

6.5.2.3 Components

All associated components, piping, and structures of the isolation valve seal-water system are designed to seismic Class I criteria. There are no components of this system located inside containment.

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

6.5.3 Design Evaluation

The isolation valve seal-water system provides an extremely prompt and reliable method of limiting the fission product release from the containment isolation valves in the event of a loss-of-coolant accident.

The employment of the system during a loss-of-coolant accident, while not considered for the analysis of the consequences of the accident as discussed in Section 14.3.6, 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.

6.5.3.1 System Response

Automatic containment isolation will be completed within approximately 10 sec following the generation of the phase A containment isolation signal. This is the estimated closing time of the non-essential containment isolation valves (Section 5.2). Closing times of greater than 10 seconds are permitted on a case by case basis if properly justified by a safety evaluation. Since the isolation valve seal-water system is also actuated by this signal, automatic seal-water injection will be in effect within this time period, which is less than the 1 min credited in Section 14.3.6.1.

Subsequent generation of the phase B isolation signal on containment high-high pressure (spray actuation signal) will close the essential containment isolation valves with an estimated closing time of 10 sec. Closing times of greater than 10 seconds are permitted on a case by case basis if properly justified by a safety evaluation. Automatic seal-water injection flow will have been initiated in advance of this signal by the phase A signal.

6.5.3.2 Single-Failure Analysis

A single-failure analysis is presented in Table 6.5-2. The analysis shows that the failure of any single active component will not prevent fulfilling the design function of the system.

6.5.3.3 Reliance on Interconnected Systems

Normally the high-pressure nitrogen supply used to maintain pressure in the seal water tank is from the Nitrogen System. However, in the backup mode, when the tank is pressurized from the system's own supply of high-pressure nitrogen cylinders, the isolation valve seal-water 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 on the Waste Disposal Panel of seal-water tank pressure and level.

6.5.3.4 Shared-Function Evaluation

Table 6.5-3 is an evaluation of the main components discussed previously and a brief description of how each component functions during normal operation and during an accident.

6.5.4 Minimum Operating Conditions

The Technical Specifications establish limiting conditions regarding the operability of the system.

6.5.5 Inspections And Tests

6.5.5.1 Inspections

The system components are all located outside the containment and can be visually inspected at any time.

6.5.5.2 Component Testing

Each automatic isolation valve can be tested for operability at times when the penetrating line is not required for normal service. Lines supplying automatic seal-water injection can be similarly tested.

6.5.5.3 System Testing

Containment isolation valves and the isolation valve seal-water system can be tested periodically to verify capability for reliable operation. The seal-water tank pressure and water level can be observed locally on the Waste Disposal Panel, and these parameters are also monitored continuously via local alarms on the Waste Disposal Panel and a category alarm in the control room.

The system will not be in service during the containment leak rate test.

6.5.5.4 Operational Sequence Testing

The capacity of the system to deliver water at the required rate was verified initially during the pre-operational test period of plant construction and startup. Prior to plant operation, a containment isolation test signal was used to ensure proper sequence of isolation valve closure and seal-water addition.

TABLE 6.5-1
Isolation Valve Seal-Water Tank

Number	1
Total volume, ft^3	23.6
Minimum volume, gal	144
Material	ASTM A-240
Design pressure, psig	150
Design temperature, ^oF	200
Operating pressure, psig	52-62
Operating temperature, ^oF	Ambient
Code	ASME UPV (Section VIII)

TABLE 6.5-2
Single Failure Analysis – Isolation
Valve Seal-Water System

Component	Malfunction	Comments
A. Automatically operated valves (open on phase A containment isolation signal)		
1. Isolation valve for automatic injection headers	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

TABLE 6.5-3
Shared Functions Evaluation

Component	Normal Operating Function	Normal Operating Arrangement	Accident Function	Accident Arrangement
Isolation valve seal-water storage tank (1)	None	Lined up to seal-water injection piping	Source of water for sealing isolation valves	Lined up to seal-water injection piping
N_2 supply bottles N_2 injection piping	None	Lined up to seal-water tank and water tank pressure and N_2 to those valves sealed with isolation valve seal-water system	Source of N_2 to maintain seal injection piping	Lined up to seal-water tank and N_2

6.5 FIGURES

Figure No.	**Title**
Figure 6.5-1	Isolation Valve Seal - Water System - Flow Diagram, Replaced with Plant Drawing 9321-2746
Figure 6.5-2	Double Disk Isolation Valve With Seal-Water Injection

6.6 CONTAINMENT PENETRATION AND WELD CHANNEL PRESSURIZATION SYSTEM

6.6.1 Design Bases

The containment penetration and weld channel pressurization system provides a means for continuously pressurizing the positive pressure zones incorporated into the containment penetrations and the channels over the welds in the steel inner liner and certain containment isolation valves in the event of a loss-of-coolant accident. Although no credit is taken for system operation in the calculation of offsite accident doses as discussed in Section 14.3.6, it is designed as an engineered safety feature and does provide assurance that the containment leak-rate in the event of an accident is lower than that assumed in the accident analysis.

The system is designed to provide a means for determining the leaktightness of the containment during power operation, thereby reducing the frequency for performing post-operational integrated leakage rate tests.

6.6.2 System Design And Operation

The containment penetration and weld channel pressurization system is shown in Plant Drawing 9321-2726 [Formerly UFSAR Figure 6.6-1]. A regulated supply of clean and dry compressed air from either of the plant's 100-psig compressed air systems located outside the containment is supplied to all containment penetrations and inner liner weld channels. The system maintains a pressure in excess of containment design pressure continuously during all reactor operations, thereby ensuring that there will be no outleakage of the containment atmosphere through the penetrations and liner welds during an accident. Typical piping and electrical penetrations are described in Sections 5.1 and 5.2.

The primary source of air for this system is the instrument air system (Section 9.6). Two instrument and control air compressors are used, although only one is required to maintain pressurization at the maximum allowable leakage rate of the pressurization system. The station air compressor acts as a backup to the instrument and control air compressors (Section 9.6) for added reliability.

A standby source of gas pressure for the system is provided by the bank of nitrogen cylinders. The associated nitrogen backup system will actuate a low weld channel pressure of approximately 49 psig and deliver nitrogen to the main weld channel pressure regulator. This regulator controls downstream weld channel pressure to containment at approximately 52 psig. Thus, in the event of failure of the normal and backup air supply systems during periods when the system is in operation, the penetration and weld channel pressure requirements will be automatically maintained by the nitrogen supply. This ensures reliable pressurization under both normal and accident conditions.

Containment penetrations and liner weld channels are grouped into four independent zones to simplify the process of locating leaks during operation. Each such zone is served by its own air receiver. In the event that all normal and backup air supplies are lost, each of the four pressurization system zones continues to be supplied with air from its respective air receiver. Each of the air receivers, (see Table 6.6-1), is sized to supply air to its pressurized zone for a period of at least 4 hr, based on a leakage rate of 0.2-percent of the containment free volume per day (0.1-percent leakage into the containment and 0.1-percent leakage to the environment).

If the receivers become exhausted before normal and backup air supplies can be restored, nitrogen from the bank of pressurized cylinders can be supplied to the affected zones. The nitrogen bank is sized to provide a 24-hr supply of gas to the system, again based on a total leakage rate from the pressurization system of 0.2-percent of the containment free volume in 24 hr. There are three nitrogen cylinders in the bank, each approximately 24-in. OD by 20-ft 6½-in long, providing a total volume of 153 cu-ft (51 cu-ft/cylinder). The nitrogen supply will also automatically assume the pressurization gas load in the event an air receiver fails.

A pressure relief valve set at 150 psig (sized for 167 scfm at 10-percent accumulation) protects the system from failure of the pressure-reducing valve in the line to each zone from the bank of nitrogen cylinders. Each zone of piping is also protected by a rupture disk, designed to open at 175 psig. In addition, the electric penetration assemblies (Zone 1) are protected by a pressure relief valve set at 70 psig. Pressure control valves, isolation valves, and check valves are located outside of the containment for ease of inspection and maintenance. The failure of any of these components does not lead to a loss of pressure in the system since backup systems automatically augment the normal air supply.

The line to each of the four pressurized zones is equipped with a critical pressure drop orifice (installed in the pressure control valve body) to ensure that air consumption will be within the capacity of the system. High air consumption in one zone cannot affect the operation of the other zones under any circumstances.

Means for ensuring that all the weld channels and penetrations are pressurized is provided by flow-through test lines connected to the pressurized weld channel zones and penetrations at points as far away from the supply points as possible. The pressurization of the zone can be verified by closing off the air supply line and opening the flow-through test line valve to observe the escape of the pressurizing medium.

Certain portions of the Weld Channel Pressurization System may be disconnected if that portion has become inoperable and repairs to that portion of the system have been determined not to be practicable. Currently, sections W-10, W-11, D-2, B-2, B-5, B-6, B-7, and connection to penetration MP-“O-O” are disconnected. The method of disconnection is selected so as not to interfere with the ability to pressurize those portions of weld channel to containment atmospheric conditions during an integrated leak rate test (ILRT) in accordance with 10 CFR 50 Appendix J.

6.6.2.1 Pressure Indication

The following instrumentation is provided as described below to ensure that the station operators are aware at all times that all penetrations and liner weld seam channels are pressurized.

The following pressurized zones are equipped with local pressure gauges, mounted outside the containment for ready accessibility and available for regular reading.

1. Each piping penetration.
2. Each electrical penetration.
3. The spaces between the two isolation (butterfly) valves in the purge supply and exhaust ducts.
4. The two spaces between the three isolation (butterfly) valves in the containment pressure relief line.
5. The double-gasketed space on the outside hatch of each of the two personnel air locks.

The pressurized zones located entirely inside the containment and those zones located in inaccessible areas are equipped to actuate pressure switches to provide remote low-pressure alarms in the central control room. Examples of the zones so equipped are as follows:

1. Each liner seam weld channel, except for the disconnected zones listed in Section 6.6.2.
2. The double-gasketed space on each inside hatch of the personnel airlocks.
3. The double-gasketed space on the equipment door flange.
4. The pressurized zones in the spent fuel transfer tube.
5. Shroud rings over penetration-to-containment liner weld piping and electrical penetrations.

The control room low pressure alarm switches and the nitrogen backup actuation switches are set above incident pressure and below the pressure setting of the main weld channel pressure regulators. Should pressure in any of these zones fall below the alarm pressure switch setpoint, a light and an alarm in the control room will be activated. Each penetration and each section of liner weld joint channel so alarmed will be represented by a separate light and identified.

6.6.2.2 Flow Indication

The flow to each zone is measured by two meters mounted in series. In addition to indication, low and high flow alarms are provided in the control room.

6.6.2.3 Personnel Air Lock Interlock

Continuous pressurization of air lock door double-gasketed barriers and the protection of the pressurization header against air loss are ensured by a set of interlocks. One interlock on each airlock door prevents the opening of the door until the pressurization line is isolated and pressure in the double-gasketed closure is relieved to atmosphere. This prevents excessive leakage from the pressurization system. The pressurization line to this zone is also equipped with a restricting orifice to ensure that air consumption, even upon failure of the interlock, will be within the capacity of the pressurization system, and will not result in a loss of pressure in other zones connected to the same pressurization header. Another set of interlocks prevents opening of one air lock door until the double-gasketed zone on the other door is repressurized.

6.6.2.4 Containment Purge Line Interlock

The containment ventilation purge penetration butterfly valves inside containment are interlocked to prevent their opening until the pressurization line to each purge duct pressurization zone has been isolated and the space between been depressurized. The isolation of the pressurization line to each purge duct-pressurized zone is accomplished from the fan room. Alarm lights, prominently displayed on a panel indicating the isolation status of the containment, remain lit identifying an open purge duct isolation valve or a low pressurization zone pressure. Restricting orifices are installed in each pressurization line to the ventilation purge ducts to ensure that air consumption, even on the failure of an interlock, will not result in a loss of pressure to the other zones connected to the same pressurization header.

The containment pressure relief line isolation valve inside containment (PCV-1190), and the pressurized space formed between it and the next butterfly valve in series, are provided with an interlock to prevent the opening of PCV-1190 until the adjacent intervalve space has been depressurized. By procedure the outside containment pressure relief line isolation valves PCV-1191 and PCV-1192 are opened from the central control room after isolation valve PCV-1190 is verified open. A time delay allows the pressurized space between the two outside isolation valves to vent air through the associated solenoid valve to atmosphere before these valves will open. The pressurization lines to these spaces are also equipped with flow restricting orifices; alarm lights in the control room identify open valves or low intervalve space pressure.

6.6.2.5 Containment Inleakage

With a continuous in-leakage to the containment from the penetration and liner weld joint channel pressurization system of 0.1-percent of the containment volume per day, the calculated time for the containment pressure to rise by 1 psi is approximately 14 days and therefore is not considered to be an operating or safety problem. From the standpoint of allowable pressure, a much greater in-leakage would be permitted. With the ability to limit the activity of the air in the containment during normal operation with the use of the two containment auxiliary charcoal filter units, each complete with roughing filter, HEPA filters, and charcoal filters (Section 5.3), containment overpressure can be relieved as required through the pressure relief duct and exhaust fan, passing up the discharge duct along with the exhaust air from the primary auxiliary building.

6.6.2.6 Components

All associated components, piping, and structures of the containment penetration and weld channel pressurization system are designed to seismic Class I criteria.

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

For a description of the instrument and control air compressors and the plant air compressors, refer to the discussion on the service air system, Section 9.6.

The three nitrogen cylinders used are designed in accordance with Section VIII (Unfired Pressure Vessels) of the ASME Boiler and Pressure Vessel Code for 2200-psig maximum pressure and contain a total of 22,000 scf of nitrogen.

6.6.3 Design Evaluation

The employment of this system following a loss-of-coolant accident, while not considered in the analysis of the consequences of the accident as discussed in Section 14.3.6, provides an additional means for ensuring that leakage is minimized if not altogether eliminated. No detrimental effect on any other safety features system will be felt should the pressurization system fail to operate.

6.6.3.1 System Response

Since the containment penetration and weld channel pressurization system is continuously pressurized above the containment design pressure during all reactor operations, there is no response time required for the system to operate.

6.6.3.2 Single Failure Analysis

A single failure analysis is presented in Table 6.6-2. The analysis shows that the failure of any single active component will not prevent fulfilling the design function of the system.

6.6.3.3 Reliance on Interconnected Systems

The containment penetration and weld channel pressurization system can operate and meet its design function without reliance on any other system, except as limited by air compressor availability following the depletion of all reserves in the system's air receivers and backup nitrogen cylinders. Electric power is not necessary for the operation of the system, although instrument power is required in order to provide indications in the control room of system operation.

6.6.3.4 Shared Functions Evaluation

Table 6.6-3 is an evaluation of the main components discussed previously and a brief description of how each component functions during normal operation and during an accident.

6.6.4 Minimum Operating Conditions

The Technical Specifications establish limiting conditions regarding the operability of the system.

6.6.5 Inspections And Tests

6.6.5.1 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, except for low pressure alarm lights associated with the disconnected weld channel zones listed in Section 6.6.2.

6.6.5.2 Testing

Since the system is in operation continuously during all reactor operations to maintain the penetrations and liner weld channels pressurized above containment design pressure, no special testing of system operation or components is necessary.

Should one zone indicate a leak during operation, the specific penetration or weld channel containing the leak can be identified by isolating the individual air supply line to each component in the zone and injecting leak test gas through a capped tube connection installed in each line.

Total leakage from penetrations and weld channels is measured by summing the recorded flows in each of the four pressurization zones. A leak would be expected to build up slowly and would therefore be noted before design leakage limits are exceeded. Thus, remedial action can be taken before the limit is reached.

In order to provide facility for containment testing in accordance with Technical Specification 4.4 and 10 CFR 50 Appendix J, test connections are provided in each of the zones.

Flow Instrumentation is installed in the piping to the personnel air locks during testing to provide measurement of the airlock leakage rate independent of the other components served by the same zone.

The makeup air flow to the penetrations and liner weld joint channels during normal operation is recognized to be only an indication of the potential leakage from the containment. However, it does indicate the leakage from the pressurization system, and the degree of accuracy will be increased when correlated with the results of the full-scale containment leak rate tests. The criteria for the selection of operating limits for air consumption of the pressurization system are based upon the design integrated containment leak rate and upon the maintenance of suitable reserve air supplies in the static reserves consisting of the air receivers and nitrogen cylinders. A summary of these operating limits is as follows:

1. A baseline air consumption rate was established for each of the four pressurization headers at the time of successful completion of the pre-operational integrated containment leakage rate tests. Unexplained increases from this consumption rate shall be considered as reason for concern and normal practice will require routine investigation and location of the point of leakage.

2. The upper limit for long-term uncorrected air consumption for the pressurization system shall be 0.2-percent of the containment volume per day (sum of four zones) at the system operating pressure, contingent on the following:

   a. Pressure in all pressurization zones is maintained above incident pressure.
   b. Air supply is maintained from the compressed air systems with compressors running.
   c. The full complement of standby nitrogen cylinders (three) is charged. This is consistent with maintenance of a 24-hr supply.

A variable area flow sensing device is located in each of the headers supplying makeup air to the four pressurization zones. The flow sensing device for each zone has two flow transmitters (high and low). Signal output from each of the two flow transmitters is applied to an integrator which has an output to a flow indicator for each zone located in the control room. Output from each of the four integrators is also applied to a single summing amplifier that drives a single total-flow recorder which is also located in the control room. Two high-flow alarms (one short term and one long term) are also derived in the recording channel to alert the operator in the control room.

TABLE 6.6-1
Containment Penetration And Weld Channel
Pressurization Air Receivers

Number	4
Volume (each), ft^3	360
Material	ASTM A-285-C
Design pressure, psig	140
Design temperature, °F	200
Operating pressure, psig	100
Operating temperature, °F	100
Code	ASME UPV (Section VIII)

TABLE 6.6-2
Single Failure Analysis Containment Penetration
And Weld Channel Pressurization System

Component	Malfunction	Comments
Instrument and control air compressor	Fails to maintain pressure	One of two instrument and control air compressors required to operate.
Pressure-reducing valve for each zone	Fails to maintain pressure	On valve failure, flow is limited to acceptable value (75 scfm) by the critical pressure drop orifice. Under low-flow conditions, overpressurization of system downstream of valve is prevented by a rupture disk.

TABLE 6.6-3
Shared Functions Evaluation

Component	Normal Operating Function	Normal Operating Arrangement	Accident Function	Accident Arrangement
Instrument and control air compressors (2)	Supply air to plant's instruments and controls and to penetrations and weld channels	2 air compressors in operation	Supply air to penetrations and weld channels	1 air compressor$_1$ in operation
Station air compressors (1)	Supply air to station air headers	1 air compressor in operation	Supply air to penetrations and weld channels	1 air compressor$_1$ in operation
N_2 cylinders (3)	Backup source of N_2 to maintain penetration and channel pressure	Lined up to penetration and weld channel pressurization system	Backup source of N_2 to maintain penetration and weld channel pressure	Lined up to penetration and weld channel pressurization system
Air receivers (1) and dryers (3)	Primary source of air for penetrations and weld channels	Lined up to penetrations and weld channel pressurization system	Primary source of air for penetrations and weld channels	Lined up to penetration and weld channel pressurization system

Notes:
1. Assuming offsite power available.

6.6 FIGURES

Figure No.	Title
Figure 6.6-1	Weld Channel and Penetration Pressurization System - Flow Diagram, Replaced with Plant Drawing 9321-2726

6.7 LEAKAGE DETECTION AND PROVISIONS FOR THE PRIMARY AND AUXILIARY COOLANT LOOPS

6.7.1 Leakage Detection Systems

The leakage detection systems reveal the presence of significant leakage from the primary and auxiliary coolant loops.

6.7.1.1 Design Bases

6.7.1.1.1 Monitoring Reactor Coolant Leakage

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

Positive indications in the control room of the leakage of coolant from the reactor coolant system to the containment are provided by equipment that permits continuous monitoring of containment air activity and humidity and of runoff from the condensate collecting pans under the cooling coils of the containment air recirculation units. This equipment provides an indication of normal background, which is indicative of a basic level of leakage from primary systems and components. Any increase in the observed parameters is an indication of change within the containment, and the equipment provided is capable of monitoring this change. The basic design criterion is the detection of deviations from normal containment environmental conditions, including air particulate activity, radiogas activity, humidity, condensate runoff, and in addition, in the case of gross leakage, the liquid inventory in the process systems and containment sump.

6.7.1.1.2 Monitoring Radioactivity Releases

Criterion: Means shall be provided for monitoring the containment atmosphere and the facility effluent discharge paths for radioactivity released from normal operations, from anticipated transients, and from accident conditions. An environmental monitoring program shall be maintained to confirm that radioactivity releases to the environs of the plant have not been excessive. (GDC 17)

The containment atmosphere, the ventilation exhaust from the residual heat removal pump compartments, the containment fan cooler service water discharge, the component cooling loop liquid, the liquid phase of the secondary side of the steam generator, and the condenser air ejector exhaust are monitored for radioactivity concentration during normal operation, anticipated transients, and accident conditions.

6.7.1.1.3 Principles of Design

The original design of the RHR and HHSI Pump seals incorporated a disaster bushing that would limit the flow to 50 GPM if the seal faces were severely damaged. For Generic Letter 2004-02 compliance, an analysis determined the wear of these disaster bushings if debris laden fluid passed through a failed seal. The potentially abrasive nature of the fluid can wear non-metallic disaster bushings over time, whereby the flow out past the damaged seal could eventually exceed 50 GPM. However, this effect is not immediate and as before, actions would be taken to isolate the pump before the 50 GPM flow rate is reached. The Chesterton seal, an

alternate type to the original seal, was tested to demonstrate that severely damaged seal faces would result in a flow rate of less than 50 GPM past the seal. Both the original seal designs and later Chesterton model seals are acceptable and may be used in the HHSI and RHR pumps.

The principles for the design of the leakage detection systems can be summarized as follows:

1. Increased leakage could occur as the result of a failure of pump seals, valve packing glands, flange gaskets, or instrument connections. The maximum leakage rate calculated for these types of failures is 50 gpm, which would be the anticipated flow rate of water through the pump seal if the entire seal were wiped out and the area between the shaft and housing were completely open.

2. The leakage detection systems should not produce spurious annunciation from normal expected leakage rates, but should reliably annunciate increasing leakage.

3. Increasing leakage rate is to be annunciated in the control room. Operator action is required to isolate the leak in the offending system.

6.7.1.2 Systems Design and Operation

For Class 1 systems located outside the containment, leakage is determined by one or more of the following methods:

1. For systems containing radioactive fluids, leakage to the atmosphere would result in an increase in local atmospheric activity levels and would be detected by either the plant vent monitors or by one of the area radiation monitors. Similarly, leakage to other systems that do not normally contain radioactive fluids would result in an increase in the activity level in that system.

2. For closed systems such as the component cooling system, leakage would result in a reduction in fluid inventory.

3. All leakage would collect in specific areas of the building for subsequent handling by the building drainage systems, e.g., leakage in the vicinity of the residual heat removal pumps would collect in the sumps provided, and would result in the operation, or increased operation, of the associated sump pumps and increased inventory in the liquid waste processing system.

Details of how these methods are used to detect leakage from Class 1 systems other than the reactor coolant system are given in the following sections and summarized in Table 6.7-1.

The Class 1 fluid systems for which no special leak detection outside containment is provided include the following:

1. Residual heat removal.
2. Component cooling.
3. Service water.
4. Auxiliary feedwater.
5. Waste disposal.

Various methods are used to detect leakage from either the primary loop or the auxiliary loops. Although described to some extent under each system description, all methods are included here for completeness.

6.7.1.2.1 Reactor Coolant System

In considering potential leakage from the reactor coolant system containing primary coolant at high pressure, four categories should be considered:

1. Leakage to the reactor coolant drain tank.
2. Leakage to the pressurizer relief tank.
3. Leakage to the containment environment.
4. Leakage to the interconnecting systems.

For clarity, each of these paths are discussed in turn.

6.7.1.2.1.1 Paths Directed to the Reactor Coolant Drain Tank

The routes directed to the reactor coolant drain tank may be summarized as follow:

1. Reactor coolant system loop drains.
2. Accumulator drains.
3. Auxiliary system equipment drains.
4. Excess letdown.
5. Valve leakoffs.
6. Reactor coolant pump seal leakage.
7. Reactor flange leakoff.

Of these paths, (1) through (4) do not present a leakage load on the reactor coolant drain tank during normal operation; leakage from the high-pressure systems is not expected because of the use of double isolation valves. Leakage paths (5) through (7), above, are evaluated as follows:

Valve Leakoffs

Source - There are 19 valves in the containment provided with leakoff connections. Of these valves, only four valves in the safety injection system will normally have their valve stem packing subjected to pressure. These are:

894A,B,C,D Accumulator isolation valves that are normally open are provided with backseats. Leakage would only be of borated radioactive water.

Estimated Leakage – Total leakage of reactor coolant fluid during normal power operation is conservatively estimated to be 8 cm^3/hr per the following:

For valves 894A, B, C, and D, leakages are assumed to be 2 cm^3/hr per valve, a total leakage of 8 cm^3/hr is assumed.

Indication to Operator - The operator is alerted to abnormal conditions by an increase of the drain tank water temperature and eventually the change in tank level. Drain tank temperature,

pressure, and level are continuously indicated on the "waste disposal/boron recycle" panel in the auxiliary building; high pressure, high temperature, high level, and low level are annunciated on this panel. Any alarm on this panel causes the annunciation of a single window on the main control board in the central control room.

Reactor Coolant Pump Seals

Source - Charging flow is directed to the reactor coolant pumps via a seal-water injection filter and enters each pump at a point between the labyrinth seals and the No. 1 face seal. Here the flow splits and a portion (normally about 5 gpm) enters the reactor coolant system via the labyrinth seals and thermal barrier cooler cavity. The remainder of the flow (normally about 3 gpm) flows up the pump shaft (cooling the lower bearing) and leaves the pump via the No. 1 seal where its pressure is reduced to about 25 to 30 psig and its temperature is increased from 130^{o}F to about 136^{o}F. The labyrinth flows (20 gpm total for four reactor coolant pumps) are removed from the system as a portion of the letdown flow. The No. 1 seal discharges (12 gpm total for four reactor coolant pumps) flow to a common manifold and then via a filter (seal-water filter) through the seal-water heat exchanger (where the temperature is reduced to about 130^{o}F) to the volume control tank.

The leakoff system between the No. 2 and No. 3 seals is considered to be part of the reactor coolant system. The leakoff system collects leakage passed by the No. 2 seal, and provides a constant backpressure on the No. 2 seal and constant pressure on the No. 3 seal. A standpipe is provided to give a constant backpressure during normal operation. The first outlet from the standpipe is orificed to permit normal No. 2 seal leakage to flow to the reactor coolant drain tank; excessive No. 2 seal leakage will result in a rise in the standpipe level and eventual overflow to the reactor coolant drain tank via a second overflow connection.

Leakage – The normal No. 2 seal leakage is anticipated to be approximately 3 gph per pump. This is the value specified in the reactor coolant pump equipment specification.

Indication to Operator - Level instrumentation on the standpipes is provided to alert the operator to abnormal conditions. The standpipe consists of a pipe with an orificed overflow above the midpoint, a normally closed drain (for service) at the bottom, and a free-flowing overflow at the top. Normal No. 2 seal leakage will flow freely out the orificed overflow. Excessive leakage will "back up" in the standpipe until it overflows at the top. A level switch in the upper standpipe actuates an annunciator indicating excessive flow. A level switch in the lower standpipe causes the annunciation of the opposite condition, which could result in undesirable dry operation of the No. 3 seal.

Reactor Vessel Flange Leakoff

Source - The reactor vessel flange and head are sealed by two metallic O-rings. To facilitate leakage detection, a leakoff connection is placed between the two O-rings and a leakoff connection is placed beyond the outer O-ring. Piping and associated valving is provided to direct any leakage to the reactor coolant drain tank.

Leakage - During normal operation, it is anticipated that the leakage will be negligible since it is specified in the reactor vessel equipment specification that there is to be zero leakage past the outer O-ring under normal operating and transient conditions.

Indication to Operator - A temperature detector will indicate leakage by a high temperature alarm. The operator is further alerted by the associated increase in drain tank water temperature and eventually the change in tank level.

6.7.1.2.1.2 Paths Directed to the Pressurizer Relief Tank

These leakage routes are evaluated below.

Source - The pressurizer relief tank condenses and cools the discharge from the pressurizer safety and relief valves. Discharge from small relief valves located inside the containment is also piped to the relief tank. During normal operation, leakage could possibly occur from either the pressurizer safety valves, pressurizer relief valves, or the chemical and volume control system letdown station relief valve.

Leakage - During normal operation, the leakage to the pressurizer relief tank is expected to be negligible since the valves are designed for essentially zero leakage at the normal system operating pressure as specified in the respective valve equipment specifications.

Indication to Operator - For these valves, temperature detectors are provided as an indication of possible leakage. In addition, each pressurizer safety valve is provided with an acoustic monitor in the discharge piping to alert the operator to possible leakage.

The rate of increase of the water temperature in the pressurizer relief tank and the level change will indicate to the operator the magnitude of the leakage. In the event of excessive leakage into an interconnecting system causing lifting of the local relief valves, the operator would again be alerted to the situation by a rising tank water temperature (refer to Section 6.7.1.2.1.4). The acoustic monitors provide a gross indication of safety valve leakage via an alarm in the central control room.

6.7.1.2.1.3 Releases to the Containment Environment

Leakage to the containment environment is discussed as follows.

Source - The main contributors of leakage to the containment environment may be listed as follows:

1. Valve stem leakages
   As previously discussed, the modulating valves within the containment are provided with leakoff connections, which in turn are either piped to the reactor coolant drain tank or are capped. Of the remaining valves that serve lines and components containing reactor coolant, only two are not normally fully open or fully closed; i.e., the continuous spray bypass needle valves around the main spray valves. The remaining valves are of the backseated type, which prevent the valve stem packings from being subjected to high pressures when in the open position.

2. Reactor coolant pump No. 3 seal leakage
   A small continuous leakage is anticipated past the No. 3 seal to the containment environment; this fluid will be seal injection water. The No.3 seal leakoff is diverted to the local open drains and is thus released to the containment environment.

3. Weld flanges
   The welded joints throughout the system are subjected to extensive nondestructive testing; leakage through metal surfaces and welded joints is very unlikely.

4. Flange joints
   There are a number of flanged joints in the system, all of which are subjected to leak testing before power operation. Experience has shown that hydrostatic testing is successful in locating leaks in a pressure-containing system.

   Methods of leak location that can be used during plant shutdown include visual observation for escaping steam or water or for the presence of boric acid crystals near the leak. The boric acid crystals would be present near the leak as a result of the evaporation process of the leaking fluid.

Leakages - The main contributors to leakage to the containment environment are considered to be items 1 and 2 above; experience with operating reactors has shown that following the normal preoperational testing, leakage from these sources are negligible.

1. Valves stems
   Normally open valves have backseats that limit leakage to less than 1 cm^3/hr-in. of stem diameter assuming no credit for packing in the valve. Normally closed globe valves are installed with recirculation flow under the seat to prevent stem leakage from the more radioactive fluid side of the seat. On the basis of these pessimistic assumptions, the leakage from valves is estimated to be approximately 50 cm^3/hr.

2. Reactor coolant Pump No. 3 Seal Leakage
   The fluid will be seal injection water and is anticipated to be approximately 100 cm^3/hr per pump. This is the value specified in the reactor coolant pump equipment specification.

Conclusion - On the basis of the above, the analysis of the situation indicates a total leak rate to the containment environment of about 450 cm^3/hr. For design purposes, 50 lb/day (i.e., 1000 cm^3/hr) is assumed. Allowable reactor coolant leakage rate and leakage rate to the containment free volume are specified in the Technical Specifications.

6.7.1.2.1.4 Leakage to Interconnecting Systems

Leakage paths to interconnecting systems are evaluated below.

System	Discussion
Chemical and volume control system	This is a normally operating interconnecting system with redundancy for isolating purposes if required.
Sampling system	In the event of sample valves failing to close or seat, adequate redundancy is provided by containment isolation valves; the piping between the sets of valves is designed for reactor coolant system pressure.
Residual heat removal system hot leg connection	Two isolation valves are provided; in the unlikely event of leakage past the two valves, interconnecting piping is provided to enable pressure relief via the residual heat removal system loop relief valve to the pressurizer relief tank.
Residual heat removal system cold leg	In the unlikely event of leakage past the accumulator check valves, residual heat removal system loop check valves and the motorized isolation valves, pressure relief will take place via the residual heat removal system loop relief valves to the pressurizer relief tank.
Safety injection system high-head pump injection lines	In the event of leakage past the check valves and motorized gate valve in any one of the four cold leg injection lines or check valves and motorized gate valve in either of the two hot-leg injection lines, pressure relief will take place to the pressurizer relief tank via the relief valve in the safety injection system test line.
Safety injection system accumulator connections	Provisions have been made to check the leak-tightness of the accumulator check valves. Leakage past these valves is discussed in Section 6.2.

Leakage of primary fluid to the secondary system via the steam-generator primary/secondary boundary would result in an increase of activity level in the secondary system and would be detected by the condenser air ejector gas monitor or by the steam-generator liquid sample monitor. (Refer to Section 11.2.)

During normal operation and anticipated reactor transients, the following methods are employed to detect leakage from the reactor coolant system.

6.7.1.2.2 Containment Air Particulate Monitor

This channel takes continuous air samples from the containment atmosphere and measures the air particulate beta radioactivity. The samples, drawn outside the containment, are in a closed, sealed system and are monitored by a scintillation detector assembly. This assembly

collects particulate matter greater than 1 micron in size on its moving filter paper surface, which is viewed by a scintillation photomultiplier combination. After passing through the gas monitor, the samples are returned to the containment.

The filter paper has a 25-day minimum supply at normal speed. The filter paper mechanism, an electromagnetic assembly, which controls the filter paper movement, is provided as an integral part of the detector unit.

The detector assembly is in a completely closed housing. The detector output is amplified by a preamplifier, processed and transmitted to the radiation monitoring system console, safety related display console and a recorder in the control room. Lead shielding is provided to reduce interference with the detector's sensitivity caused by background radiation.

The activity is indicated on digital displays and recorded by a multipoint recorder. High-activity alarm indications are displayed on the control board annunciator and the safety related display console. Local alarms provide operational status of supporting equipment such as pumps, motors, and flow and pressure controllers.

The containment air particulate monitor is sensitive to low rates. The rates of reactor coolant leakage to which the instrument is sensitive are 0.1 gpm to greater than 10 gpm, assuming corrosion product activity and no fuel cladding leakage. Under these conditions, an increase in reactor coolant system leakage of 1 gpm is detectable within 1 hour after it occurs.

The sensitivity of the air particulate monitors to an increase in reactor coolant leak rate is dependent upon the magnitude of the normal baseline leakage into the containment. The sensitivity is greatest where baseline leakage is low as has been demonstrated by the experience of Indian Point Unit 1 (Appendix 6B), Yankee Rowe, and Dresden Unit 1. Assuming a low background of containment air particulate radioactivity, if we assume a reactor coolant corrosion product radioactivity (Fe, Mn, Co, Cr) of approximately 0.4 $\mu Ci/cm^3$ (a value consistent with little or no fuel cladding leakage), and complete dispersion of the leaking radioactive solids into the containment air, the air particulate monitors would be capable of detecting an increase in coolant leakage rate as small as approximately 0.1 gpm (400 cm^3/min) within 20 min after it occurs. If only 10-percent of the particulate activity were a assumed to be dispersed in the air, leakage rate increases of about 1.0 gpm (4000 cm^3/min) would be detectable within the same time period.

For cases where baseline reactor coolant leakage falls within the detectable limits of the air particulate monitor, the instruments can be adjusted to alarm on leakage increases of from 2 to 5 times the baseline volume.

The containment air particulate monitor together with the other radiation monitors mentioned in this section are further described in Section 11.2.

6.7.1.2.3 Containment Radioactive Gas Monitor

This channel measures the gaseous gamma radioactivity in the containment by taking the continuous air samples from the containment atmosphere, after they pass through the air particulate monitor, and drawing the samples through a closed, sealed system to the gas monitor assembly.

The samples are constantly drawn through the fixed, shielded volumes and are viewed by a scintillation detector. The samples are then returned to the containment.

The detector is in a completely enclosed housing containing a gamma-sensitive scintillation detector mounted in a constant gas volume container. Lead shielding is provided to reduce interference with the detector's sensitivity caused by background radiation.

The detector output is amplified by a preamplifier, processed and transmitted to the radiation monitoring system console, the safety related display console and a recorder in the control room and indicated on digital displays. High-activity alarm indications are displayed on the control board annunciator and the safety related display console. Local alarms annunciate the supporting equipment's operational status.

The containment radioactive gas monitor is inherently less sensitive (threshold at 10^{-6} μCi/cm^3) than the containment air particulate monitor and would function in the event that significant reactor coolant gaseous activity exists from fuel cladding defects. The measuring range is 10^{-6} to 10^{-3} μCi/cm^3.

The containment air particulate and radioactive gas monitors have assemblies that are common to both channels. They are described as follows:

1. Each flow assembly includes a pump unit and selector valves that provide a representative sample (or a "clean" sample) to the detector.

2. The pump unit consists of:

   a. A pump to obtain the air sample.
   b. A flowmeter to indicate the flow rate.
   c. A flow control valve to provide flow adjustment.
   d. A flow alarm assembly to provide a low flow alarm signal.

3. Selector valves are used to direct the desired sample to the detector for monitoring and to vent flow when the channel is in maintenance or "purging" condition.

4. A pressure sensor is used to protect the system from high pressure. This unit automatically closes an inlet and outlet valve upon a high-pressure condition.

5. Purging is accomplished with a valve control arrangement whereby the normal sample flow is blocked and the detector purged with a "clean" sample.

6. The safety related display console in the control room permits remote operation of the local radiation monitor skid assembly. By operating a pushbutton on the console to start the sample pump, the containment sample can be monitored.

7. A sample flowmeter is calibrated linearly (from 0 to 56.6 standard liters per minute).

In addition to a common CCR High Rad/Trouble Annunciator, the following alarm lights are provided locally:

1. NO CPM - No detector signal in the counting circuit.
2. LOW FLOW - Flowrate drops below low flow setpoint.
3. T. TEAR - Particulate tape filter tears or the supply reel is run to empty.
4. LO TEMP - Monitor sample temperature is less than 10°F above containment air temperature. This alarm has been disabled.
5. WARN - Particulate or gas warn setpoint exceeded.
6. ALARM - Particulate or gas alarm setpoint exceeded.

6.7.1.2.4 Humidity Detectors

The humidity detection instrumentation offers a means for the detection of leakage into the containment. The instrumentation is sensitive to vapor originating from all sources within containment, including the reactor coolant and steam and feedwater systems. Plots of changes in dewpoint of the containment atmosphere will be sensitive to incremental increases of water leakage to the containment atmosphere approximately 1.0 gpm/°F of dewpoint temperature increase (this sensitivity will vary with cooling water temperature, containment air temperature, and air recirculation rate). These detectors are located just upstream of each fan cooler unit.

The information provided by this element and the temperature detector is used to determine the dewpoint in containment. This calculation is done automatically, and the resulting dewpoint information is recorded in the control room. The containment building high humidity alarm on the supervisory panel is initiated by the information being received by the recorder.

6.7.1.2.5 Condensate Measuring System

This method of leak detection is based on the principle that under equilibrium conditions the condensate flow draining from the cooling coils of the containment air-handling units will equal the amount of water (and/or steam) evaporated from the leaking system. A reasonably accurate measurement of leakage from the reactor coolant system by this method is possible because containment air temperature and humidity promote complete evaporation of any leakage from hot systems. The ventilation system is designed to promote good mixing within the containment. During normal operation, the containment air conditions will be maintained below 120°F dry bulb and 92°F wet bulb (approximately 36-percent relative humidity) by the fan coolers.

When the water from a leaking system evaporates into this atmosphere, the humidity of the fan cooler intake air will begin to rise. The resulting increase in the condensate drainage rate is given by the equation

$$D = L\,[1 - \exp(-\frac{Q}{V}t)]$$

Where:

D = condensate drainage rate, gpm
L = evaporated leakage, gpm
Q = containment ventilation rate, cfm
V = containment free volume, ft^3
t = time after start of leak, min

Therefore, if four fan cooler units are operating (Q = 280,000 cfm), the condensation rate would be within 5-percent of a new equilibrium value in approximately 200 min after the start of the leak. The detection of the increasing condensation rate, however, would be possible within 5 to 10 min. The condensate measuring device consists essentially of a vertical 6-in. diameter standpipe with a notched weir cut into the upper portion of the pipe to serve as an overflow. Each fan cooler is provided with a standpipe, which is installed in the drain line from the fan cooler unit. A differential-pressure transmitter near the bottom of the standpipe is used to measure the water level. Each unit can be drained by a remote-operated valve.

A wide range of flow rates can be measured with this device. Flows less than 1 gpm are measured by draining the standpipe and observing the water level rise as a function of time. Condensate flows from 1 gpm to 15 gpm are indicated in the central control room and can also be measured by observing the height of the water level above the notch of the weir. This water head can be converted to a proportional flow rate by means of a calibration curve. Flow rates greater than 15 gpm can also be determined using the calibration curves. A high-level alarm, set above the established normal (baseline) flow, is provided for each unit to alert the operator.

All indicators, alarms, and controls are located in the control room.

During the period of plant hot functional testing, a reactor coolant leak of known magnitude was simulated inside the containment vessel, and the performance of the humidity detector/condensate measuring system was observed. The leak was simulated by introducing steam into one of the loop compartments during a period when containment atmospheric conditions were stable and the fan cooler units were operating. The increase in containment atmosphere moisture content, as indicated by the humidity detectors, was recorded as a function of time following the initiation of the simulated leak. As a check, the same information was determined independently using different instrumentation. Elapsed time until condensation on the fan cooler unit cooling coils begins, as indicated by the condensate measuring devices, was recorded and compared with the calculated value on the basis of the initial containment humidity. Steam flow continued, and the performance of the condensate measuring devices in indicating the magnitude of steady cooling coil runoff was observed.

6.7.1.2.6 Component Cooling Liquid Monitor

This channel continuously monitors the component cooling loop of the auxiliary coolant system for activity indicative of a leak of reactor coolant from either the reactor coolant system, the recirculation loop, or the residual heat removal loop of the auxiliary coolant system. A scintillation detector is installed in the local radiation monitor skid assembly. This assembly is located in the primary auxiliary building and receives sample flow from the component cooling pump discharge downstream of the component cooling heat exchangers. The detector assembly output is amplified by a preamplifier, processed and transmitted to the radiation monitoring system console, the safety related display console and a recorder in the control room. The activity is indicated on digital displays. High-activity alarm indications are displayed on the control board annunciator and the safety related display console.

The measuring range of this monitor is 10^{-5} to 10^{-2} μCi/cm^3.

6.7.1.2.7 Condenser Air Ejector Gas Monitor

This channel monitors the discharge from the air ejector exhaust header of the condensers for gaseous radiation that is indicative of a primary to secondary system leak. The gas discharge is routed to the turbine roof vent. On high-radiation-level alarm, the condenser exhaust gases are diverted from the normal turbine building vent to the containment.

The processed detector output is transmitted to the radiation monitoring system console, the safety related display console and a recorder in the control room. The activity is indicated on digital displays. High-activity alarm indications are displayed on the control board annunciator and the safety related display console.

This monitor is composed of an in-line spool piece, containing a normal range and an accident range detector. Each detector uses a beta sensitive scintillation detector to monitor the gaseous radiation level. Each detector includes adequate shielding to reduce interference with the detector's sensitivity caused by background radiation. The normal maximum channel output for this monitor is presented in Table 11.2-7.

6.7.1.2.8 Steam Generator Blowdown Liquid Sample Monitor

This channel monitors the liquid phase of the secondary side of the steam generator for radiation, which would indicate a primary-to-secondary system leak, providing backup information to that of the condenser air ejector gas monitor. Samples from the bottom of each of the four steam generators are piped to a common header, and the mixed sample is continuously monitored by a scintillation counter and holdup tank assembly. Upon indication of a high-radiation level, each steam generator is individually sampled to determine the source. This sampling sequence is achieved by manually selecting the desired unit to be monitored and allotting sufficient time for sample equilibrium to be established (approximately 1 min).

A remote indicator panel, mounted at the detector location, indicates the radiation level and high-radiation alarm.

The measuring range of this monitor is 10^{-5} to 10^{-2} μCi/cm^3.

A photomultiplier tube-scintillation crystal (NaI) combination, mounted in a hermetically sealed unit, is used to monitor liquid effluent activity. Lead shielding is provided to reduce the background to a level so it does not interfere with the maximum sensitivity of the detector. The in-line, fixed-volume container is an integral part of the detector unit.

Personnel can enter the containment and make a visual inspection for leaks. The location of any leak in the reactor coolant system would be determined by the presence of boric acid crystals near the leak. The leaking fluid transfers the boric acid outside the reactor coolant system, and the process of evaporation deposits the crystals.

If an accident involving gross leakage from the reactor coolant system occurred, it could be detected by charging pump operation, system and sump inventories, containment sump pump operation, containment radiation monitors, humidity detectors, and the condensate monitoring system.

During normal operation, only one charging pump is operating. If a gross loss of reactor coolant to another closed system occurred, which was not detected by the methods previously described, the speed of the charging pump would indicate the leakage.

The leakage from the reactor coolant system will cause a decrease in the pressurizer liquid level that is within the sensitivity range of the pressurizer level indicator. The speed of the charging pump will automatically increase to try to maintain the equivalence between the letdown flow and the combined charging line flow and flow across the reactor coolant pump seals. If the charging pump at maximum speed is unable to maintain the required charging flow rate, then a pressurizer low level alarm actuates and a second charging pump may be started manually to maintain pressurizer level.

A break in the primary system would result in reactor coolant flowing into the containment and/or recirculation sumps. Gross leakage to these sumps would be indicated by the frequency of operation of the containment sump pump(s) and containment water level indicators. Since the building floor drains preferentially to the containment sump, the activity of the containment sump pumps and/or containment sump level would be more likely to indicate the leak first.

Gross leaks might be detected by an unscheduled increase in the amount of reactor coolant makeup water that is required to maintain the normal level in the pressurizer.

A large tube-side to shell-side leak in the non-regenerative (letdown) heat exchanger would result in reactor coolant flowing into the component cooling water and a rise in the liquid level in the component cooling water surge tank. The operator would be alerted by a high-water alarm for the surge tank and a high-temperature alarm actuated by a monitor at the component cooling water pump suction header and a high-radiation alarm actuated by a monitor sampling the component cooling water pump discharge.

A high-level alarm for the component cooling water surge tank and high-radiation and high-temperature alarms could also indicate a thermal barrier cooling coil rupture in a reactor coolant pump. However, in addition to these alarms, high temperature and low flow on the component cooling outlet line from the pump would activate alarms. Low thermal barrier component cooling water header return flow may be due to closure of FCV-625 on high flow or excessive usage by other loads.

Gross leakage might also be indicated by a rise in the normal containment and/or recirculation sump levels. Level transmitters with control room indication are provided for each sump.

6.7.1.2.9 Residual Heat Removal Loop

The residual heat removal loop removes residual and sensible heat from the core and reduces the temperature of the reactor coolant system during the second phase of plant shutdown.

During normal operation, the containment air particulate and radioactive gas monitors, the condensate measuring system, and the containment sump inventory monitoring capability provide means for detecting leakage from the section of the residual heat removal loop inside the reactor containment. These systems have been described previously in this section (see the description of leak detection from the reactor coolant system). Leakage from the residual heat removal loop into the component cooling water loop during normal operation would be

detected outside the containment by the component cooling loop radiation monitor (see the analysis of detection of leakage from the reactor coolant system in this section).

The physical layout of the two residual heat removal pumps is within separate shielded and isolated rooms outside of the containment. This will permit the detection of a leaking residual heat removal pump by means of the plant vent gas monitoring system. Alarms in the control room will alert the operator when the activity exceeds a preset level. Small leaks to the environment could be detected with these systems within a short time after they occurred.

When the plant is shut down, personnel can enter the containment to check visually for leaks. The detection of the location of significant leaks would be aided by the presence of boric acid crystals near the leak. In case of an accident that involves gross leakage from the part of the residual heat removal loop inside the containment, this leakage would be indicated by a rise in the containment and/or recirculation sump levels. Both of these sumps have level transmitters that provide level indication in the control room.

Should a large tube-side to shell-side leak develop in a residual heat exchanger or the RHR pump seal heat exchanger, the water level in the component cooling surge tank would rise, and the operator would be alerted by a high-water alarm. A radiation monitor for the component cooling loop provides a high activity alarm. A temperature monitor at the component cooling water pump suction header will also signal an alarm.

Leakage from both of the residual heat removal pumps, including leakage resulting from a residual heat removal pump seal failure, is drained to a common sump equipped with a sump pump. In addition, a level monitor in this sump will actuate an alarm when the level exceeds a preset level.

6.7.1.2.10 Recirculation Loop

If a break occurs in the reactor coolant system, the recirculation loop provides long-term protection by recirculating spilled reactor coolant and injected refueling water.

The containment air particulate and radioactive gas monitors, the humidity detectors, the condensate measuring system, and the containment sump inventory monitoring capability (see the section discussing leak detection for the reactor coolant system) provide means of detecting small leaks in the part of the recirculation loop inside the reactor containment.

Leakage from the residual heat exchanger would be detected by a radiation monitor (discussed in the section on leak detection from the reactor coolant system) that receives sample flow from the component cooling water pump discharge downstream of the component cooling heat exchangers.

During a containment entry personnel could check for leaks evidenced by the presence of boric acid crystals.

Gross leakage from the recirculation loop inside the containment might be indicated by a rise in the level of the containment and/or recirculation sumps. Both of these sumps have level transmitters that provide level indication in the control room.

A rise in the liquid level in the component cooling surge tank would result if a large tube-side to shell-side leak developed in a residual heat exchanger. The operator would be alerted by a high-level alarm in the component cooling water surge tank and a high-temperature alarm actuated by a monitor at the component cooling water pump suction header and a high-radiation alarm actuated by a monitor sampling the component cooling water pump discharge.

Leakage outside containment depending on location will be diverted by floor drains to either the PAB sump tank or PAB sump where it will then be transferred to the Waste Holdup Tank.

6.7.1.2.11 Component Cooling Loop

Leakage from the component cooling loop inside the reactor containment could be detected by the humidity detectors and/or condensate measuring system and the containment water level indicators (see the section on reactor coolant system leak detection for a description of these systems).

Visual inspection inside the containment is possible for some locations during containment entries.

Gross leakage from the component cooling loop would be indicated inside the containment by a rise in the liquid level of the containment and/or recirculation sumps. Both of these sumps have level transmitters that provide level indication in the control room.

Leakage outside containment depending on location will be diverted by floor drains to either the PAB sump tank or PAB sump where it will then be transferred to the Waste Holdup Tank.

6.7.1.2.12 Service Water System

During a loss-of-coolant accident, the containment fan coolers service water monitors check the containment fan service water discharge line for radiation indicative of a leak from the containment atmosphere into the service water. A small bypass flow from each of the heat exchangers is mixed in a common header and monitored by redundant scintillation detectors mounted in separate holdup tank assemblies. Upon indication of a high-radiation level, each heat exchanger is individually sampled to determine which unit is leaking. This sampling sequence is achieved by manually selecting the desired unit to be monitored and allotting sufficient time for sample equilibrium to be established (approximately 1 min). The discharge line from the fan coolers motor coolers is also monitored for radioactivity and isolated in a like manner following the detection of a leak.

The measuring range of this monitor is 10^{-5} to 10^{-2} μCi/cm^3.

Gross leakage from the service water system due to a faulty cooling coil in the containment air recirculation cooling system can be detected by weir level transmitters. Any significant cooling water leakage would be seen as flow into a fan cooler unit weir.

Leakage outside containment depending on location will be diverted by floor drains to either the PAB sump tank or PAB sump where it will then be transferred to the Waste Holdup Tank.

6.7.1.2.13 Containment Sump Level and Discharge Flow

The sump flow detection system includes flow metering and totalizing that is indicated in the control room. It is capable of detecting a 1 GPM leak within 4 hours. (Reference 1)

In addition to the original plant level transmitter (LT-941), redundant level transmitter LT-3304 has been installed to provide a more accurate method of measuring water level in the sump. These levels are indicated in the control room on the accident assessment panel. Another level instrument qualified to Class 1E, IEEE-323-1974, and IEEE-344-1975 is provided for continuous level indication (LT-3300). Additionally, existing level monitor (LT-941) was upgraded to meet environmental qualification requirements to support minimum NPSH requirements for recirculation pump start. LT-941 has since re-evaluated and can no longer be used for leakage detection based on the spacing and wiring of its associated sensors. (Reference 1)

Sump level is maintained by the action of containment sump pumps. The pumps are independently powered and have separate power and control cables to each pump.

6.7.1.2.14 Recirculation Sump Level

Water may also collect in the recirculation sump although under most circumstances the containment sump will be filled before the recirculation sump. Water level monitoring of the recirculation sump is provided by two level instruments, which actuate control room lights at discrete sump/containment water levels and provide an audible alarm for certain discrete levels within the recirculation sump. Level monitor LT-939 is environmentally qualified to support minimum NPSH requirements for recirculation pump start. A continuous level monitor is also provided for wide-range level indication.

6.7.1.2.15 Reactor Cavity Pit Level

The level in the reactor cavity pit is controlled through the action of reactor cavity pit pumps that are used to pump to the containment sump any water that may have leaked into the pit. The system is designed for usage during refueling/maintenance outages. Four alarms are located in the central control room on panel SB-1:

Reactor Cavity Pit Water Level High (LC-7049)
Reactor Cavity Pit Water Level High-High (LC-7049)
Reactor Cavity Sump Pit Pump No. 1 Auto Run (LC-7043)
Reactor Cavity Sump Pit Water Level High (LC-7042)

LC-7042 and LC-7043 control Reactor Cavity Sump Pit Pump No. 1. LC-7049 controls Reactor Cavity Pit Pump No. 2. A continuous level monitor (LT-3302) is also provided for wide-range level indication within the reactor cavity pit.

The reactor cavity pit pumps are independently powered and have separate control wiring.

6.7.2 Leakage Provisions

Provisions are made for the isolation and containment of any leakage.

6.7.2.1 Design Basis

The provisions made for leakage are designed to prevent uncontrolled leaking of reactor coolant or auxiliary cooling water. This is accomplished by (1) isolating the leak by valves, (2) designing relief valves to accept the maximum flow rate of water from the worst possible leak, (3) supplying redundant equipment that allows a standby component to be placed in operation while the leaking component is repaired, and (4) routing the leakage to various sumps and holdup tanks.

6.7.2.2 Design and Operation

Various provisions for leakage avert uncontrolled leakage from the primary and auxiliary coolant loops.

6.7.2.2.1 Reactor Coolant System

When significant leakage from the reactor coolant system is detected, action is taken to prevent the release of radioactivity to the atmosphere outside the plant.

If either the containment air particulate gamma activity or the radioactive gas activity exceeds preset levels, the containment purge supply and exhaust duct valves and pressure relief line valves are closed, if open.

On high-radiation alarm signaled by the condenser air ejector monitor, the condenser exhaust gases are diverted from the turbine roof vent to the containment.

A high-radiation alarm actuated by the steam-generator liquid sample monitor initiates the closure of the isolation valves in the blowdown lines. A sample stop valve and a blowdown tank inlet stop valve located downstream of each pair of isolation valves will automatically close when either of their associated blowdown line isolation valves leaves its fully open position in order to isolate their respective lines.

If the component cooling loop radiation monitor signals a high-radiation alarm, the valve in the component cooling surge tank vent line automatically closes to prevent gaseous activity release.

If a leak from the reactor coolant system to the component cooling loop were a gross leak or if the leak could not be isolated from the component cooling loop before the inflow completely filled the surge tank, the relief valve on the surge tank would open. The discharge from this valve is routed to the waste holdup tank in the primary auxiliary building.

A large leak in the reactor coolant system pressure boundary, which does not flow into another closed loop, would result in reactor coolant flowing into the containment sump and/or the recirculation sump.

Experience with the detection of primary system leakage into the containment vessel of Indian Point Unit 1 is discussed in Appendix 6B.

6.7.2.2.2 Residual Heat Removal Loop

High containment air particulate gamma activity or high radioactive gas activity will result in an alarm being activated by either the containment air particulate or radioactive gas monitors. The containment purge supply and exhaust duct valves and pressure relief line valves are closed automatically, if open. This prevents the release of radioactivity to the atmosphere outside the nuclear plant.

If leakage from the residual heat removal loop into the component cooling loop occurs, the component cooling radiation monitor will actuate an alarm and the valve in the component cooling surge tank vent line is automatically closed to prevent gaseous radioactivity release. If the leaking component (i.e., a residual heat exchanger) could not be isolated from the component cooling loop before the inflow completely filled the surge tank, the relief valve on the surge tank would open and the effluent would be discharged to the primary auxiliary building waste holdup tank.

Gross leakage from the section of the residual heat removal loop inside the containment, which does not flow into another closed loop would result in reactor coolant flowing into the containment sump and/or the recirculation sump.

Other leakage provisions for the residual heat removal loop are discussed in Section 9.3.

6.7.2.2.3 Recirculation Loop

The containment purge supply and exhaust duct valves and pressure relief line valves are automatically closed, if open, when either the containment air particulate or the radioactive gas monitors read above a preset level. This prevents radioactivity from escaping to the outside atmosphere.

Leakage from the recirculation loop into the component cooling loop results in a radiation alarm and the automatic closing of the component cooling surge tank vent line to prevent gaseous radioactivity release. If the leak were gross and filled the surge tank before the leaking component could be isolated from the component cooling loop, the relief valve on the surge tank would open and the effluent would be discharged to the waste holdup tank in the primary auxiliary building.

Gross leakage from the internal recirculation loop, which does not flow into another closed loop will flow into the containment sump and/or the recirculation sump. Leakage outside containment depending on location will be diverted by floor drains to either the PAB sump tank or PAB sump where it will then be transferred to the Waste Holdup Tank.

6.7.2.2.4 Component Cooling Loop

Gross leakage from the section of the component cooling loop inside the containment, which does not flow into another closed loop will flow into the containment sump and/or the recirculation sump. Leakage outside containment depending on location will be diverted by

floor drains to either the PAB sump tank or PAB sump where it will then be transferred to the Waste Holdup Tank.

Other provisions made for leakage from the component cooling loop are discussed in Section 9.3.

6.7.2.2.5 Service Water System

Leakage outside containment depending on location will be diverted by floor drains to either the PAB sump tank or PAB sump where it will then be transferred to the Waste Holdup Tank.

6.7.3 Minimum Operating Conditions

The Technical Specifications establish limiting conditions regarding the operability of the leakage detection systems.

REFERENCES FOR SECTION 6.7

1. Attachment C of IP2's May 23, 1988 letter to the NRC requesting elimination of Postulated Primary Loop Pipe Rupture as a Design Basis, "Indian Point Unit 2 – Evaluation of RG-1.45 Compliance."

TABLE 6.7-1
Class 1 Fluid Systems For Which
No Special Leak Detection Is Provided

	System	Remarks On Leakage Detection
1.	Residual heat removal	Refer to items 1, 2, and 3 (Section 6.7.1.2) and Section 6.7.1.2.9.
2.	Component cooling	Refer to items 2 and 3 (Section 6.7.1.2) and Section 6.7.1.2.11
3.	Service water	Refer to item 3 (Section 6.7.1.2) and Section 6.7.1.2.12
4.	Auxiliary feedwater	Visual
5.	Waste disposal	Auxiliary building sump pump operation. Also refer to item 1 (Section 6.7.1.2)

6.8 POST-ACCIDENT HYDROGEN CONTROL SYSTEMS

On April 14, 2005, NRC issued IP2 License Amendment 243 which eliminated the requirement for hydrogen recombiners to provide any combustible gas control function. Therefore, the technical specification requirements for the hydrogen recombiners have been eliminated. However, the actual equipment remains in service until such time that an alternate disposition of this equipment is established and implemented.

6.8.1 Design Basis

The function of the hydrogen control system is to control the hydrogen generated within the containment following a loss-of-coolant accident.

A Hydrogen Recombiner System is provided to control the post-accident hydrogen concentration in containment. The Hydrogen Recombiner System uses two Passive Hydrogen Recombiners(PHR). Each recombiner unit is capable of maintaining the hydrogen concentration at or below 4 volume percent. The flame type recombiners system installed originally has been retired in place within the containment building and the associated containment isolation valves have been de-energized in closed position.

The post-accident containment venting system provides a backup method to containment recombiners for controlling the potential hydrogen accumulation in the containment. This is accomplished by the controlled venting of containment atmosphere to maintain the hydrogen concentration at a safe level. The venting system is designed to limit the hydrogen concentration below 4 volume percent.

A containment air sample is taken from each of the containment fan cooler units at a point located downstream from the fan. Sample analysis will determine the requirement for the post-accident containment venting system operation.

6.8.2 System Design and Operation

6.8.2.1 Passive Hydrogen Recombiners

Two 100% capacity independent hydrogen recombiners are provided. The recombiners are passive devices, which contain no moving parts and do not need electrical power or any other support system. Recombination is accomplished by the attraction of oxygen and hydrogen molecules to the surface of the catalyst, Reference 3.

The PHRs used at IP 2 are Passive Autocatalytic Recombiners (PARs) designed and manufactured by NIS Ingenieurgesellschaft mbH of Germany (NIS). The PHR consists of a stainless steel sheet metal box open at the bottom and at both sides on the top as shown on Figure 6.8-1. The approximate size of the box is 1m x 1m x 1m. There are 88 catalytic cartridges inserted into each box. Each cartridge fabricated from perforated steel plates holds catalyst pellets. The catalyst pellets are made from aluminum oxide spheres and are coated with palladium and hydrophobic polymers. The palladium coating acts as a catalyst and the hydrophobic coating provides water proofing. The catalyst cartridges are installed vertically, spaced 1 cm apart, in each box. The spaces between the cartridges serve as flow channels for the gases. Airflow enters at the bottom and the catalyst combines hydrogen and oxygen in the flow channels to form gaseous water.

PHRs are designed for self-starting and self-sustaining reaction. The exothermic reaction of the combination produces heat, which results in a convective flow that draws more gases from the containment atmosphere into the unit from below.

A single recombiner is capable of maintaining the hydrogen concentration in containment below the 4 volume percent flammability limit. The second recombiner is redundant and is installed to provide margin and increased containment coverage.

PHRs are seismic class I.

6.8.2.2 Containment Vent System

The post-accident containment venting system consists of a common penetration line that acts as a supply line through which outside air can be admitted to the containment, and an exhaust line, with parallel valving and piping, through which hydrogen-bearing gases from containment may be vented through a filter. The system is shown in Figure 6.8-3.

The supply mode makes use of instrument air to feed containment. The nominal flow rate from either of the two instrument air compressors is 225 scfm.

In the exhaust mode, the line penetrates the containment and then is divided into the parallel lines. Each parallel line contains a pressure sensor and all the valves necessary for controlling the venting operation. The two lines then rejoin and the exhaust passes through a flow sensor and a temperature sensor before passing through charcoal and HEPA filters. The exhaust is then directed to the plant vent.

The venting system requires a differential pressure between the containment and the outside atmosphere in order to permit venting. This is based on a pressure of 2.14 psig in the containment. If required, the containment is pressurized to 2.14 psig with instrument air when the hydrogen reaches 3 volume percent after the loss-of-coolant accident. The hydrogen concentration is reduced by this pressurization. Purging is then delayed until the hydrogen concentration in the containment has once again built up to 3 volume percent.

6.8.2.3 Containment Air Analyzer System

Two hydrogen/oxygen analyzers have been installed to continuously monitor the hydrogen and oxygen concentrations in the containment atmosphere. A new system of valves, controlling equipment, and a central control room display have also been installed to replace the old method of manual containment air sampling. This system functions independently of the original vacuum pump (which has been isolated) and provides the required sampling capability under NUREG-0737, Item II.B.3. The containment atmosphere sampling system was evaluated by the NRC against the NUREG-0737 requirements and found acceptable.[1,2] The requirements for the containment atmosphere sampling system were removed from the Technical Specifications by License Amendment No. 222 as discussed in Section 9.4.1.1. The Technical Specification requirements for these instruments were eliminated and replaced by a licensee commitment to maintain the monitors as reliable and functional through a preventive maintenance program. In addition, Regulatory Guide 1.97 categorization for these instruments was changed from Category 1 to Category 3. The system is shown in Plant Drawing 208479 [Formerly UFSAR Figure 6.8-4].

The system has a closed-loop flow path with the sampled air withdrawn from and discharged to the containment. The operation of the system is from remote control panels located in the motor control center room at elevation 98-ft of the primary auxiliary building. The new solenoid-operated valves function to both pass a sample for analyzing and to provide containment isolation when not sampling or during an accident.

The hydrogen/oxygen analyzers are located on elevation 80-ft of the primary auxiliary building. Direct indicator readout is provided in the 98-ft motor control center room in the primary auxiliary building. The hydrogen/oxygen values are indicated and recorded on two recorders located on the control room accident assessment panel. One recorder is used for each channel. Nitrogen purging capability of the sample lines is also provided. The purging reduces line activity and radiation field in the primary auxiliary building after a sample is drawn. The purge is so arranged that it is sent back to the containment.

All equipment is procured and installed Class A, seismic Class I. The recorders, solenoid-operated valves and other electrical equipment are Class 1E. The eight solenoid-operated valves provided with this system are powered from four separate dc power supplies and configured so that the system meets single-failure criteria both for the path opening (i.e., sampling) and for the path closing (i.e., containment isolation) safety functions.

6.8.2.4 System Operation

PHRs are located inside the containment, they are totally passive, and are self-starting self-feeding devices. No operator action is required to initiate recombination of accident hydrogen with containment oxygen.

6.8.3 Post-accident Hydrogen Generation

During the post-accident period, hydrogen is generated in the containment from the following sources:

1. Zirconium-water reaction.
2. Chemical reaction of materials subject to corrosive attack.
3. Radiolytic decomposition of coolant in the core.
4. Radiolytic decomposition of coolant in the sump.

These results are shown in Plant Drawings 9321-2568 & -2569 [Formerly UFSAR Figure 6.8-2] for the first 30 days of the post-accident period and have been obtained on the following bases.

6.8.3.1 Zirconium-Water Reaction

1. Five percent of the core cladding reacts immediately with core cooling solution according to the reaction

$$Zr + H_2O \rightarrow ZrO_2 + H_2 + heat$$

2. There are 44,197 lbm of zirconium cladding in the core.

6.8.3.2 Corrosion of Materials of Construction

1. Corrosion of aluminum and zinc according to the reactions

   $2Al + 3H_2O = Al_2O_3 + 3H_2$

   $Zn + H_2O = H_2 + ZnO$

2. Aluminum and zinc corrosion rates versus time postaccident

Time (days)	Al Corrosion Rate (mil/yr)	Zn Corrosion Rate (mil/yr)
0.0	5,500	180
0.0035	1,700	180
0.0116	600	160
0.0232	200	110
0.0464	200	20
20	200	20

3. Aluminum available for reaction as follows:

Item	Mass (lb)	Area (ft^{2})	Thickness (inch)
Control rod drive mechanism connectors	25	14	0.207
Reactor vessel insulation foil	269	10,000	0.0019
Area monitors	6	4	0.107
Source, intermediate, and power range detectors	140	40	0.245
Process instrumentation and controls	420	84	0.356
Lighting fixtures and equipment	1061	380	0.199
Paint on steam generator, pressurizer, and reactor vessel	140	10,000	0.001
Contingency	250	85	0.209

4. A contingency for zinc available for reaction was made by assuming a 20,000 square-foot surface, thick enough (0.065 inch) to not corrode all the way through in 30 days.

6.8.3.3 Core Radiolysis

Regulatory Guide 1.7, Rev.2, “*Control of Combustible Gas Concentrations in Containment Following a Loss-of-Coolant Accident”* describes methods acceptable to the NRC staff for implementing the requirements of 10 CFR 50.44. For Indian Point 2, the core and sump radiolysis analyses have been done in accordance with that Guide.

In summary, the effects of core radiolysis are as follows:

1. 50-percent of the halogens, 100-percent of the noble gases, and 99-percent of all other fission products are retained in the core following the accident.

2. 0.50 molecules of hydrogen are generated per 100 eV of energy absorbed by water in the core.

3. 10.0-percent of the core fission product gamma energy is absorbed by the solution in the core.

4. Beta energy is absorbed by the fuel and cladding and does not contribute to hydrogen generation in the core.

5. Core fission product decay energy is calculated in accordance with Branch Technical Position ASB 9-2, as originally implemented in the COGAP computer code and transferred to STARGAP.

6. The plant is operational for 830 days at 3216 MWt before the accident.

6.8.3.4 Sump Radiolysis

1. 50-percent of the halogens, none of the noble gases, and 1-percent of all other fission products are released from the core to the sump during the accident.

2. 0.50 molecules of hydrogen are generated per 100 eV of energy absorbed by the sump solution.

3. All beta and gamma energy emitted by fission products in the sump solution are absorbed and contribute to hydrogen generation.

4. The plant is operational for 830 days at 3216 MWt before the accident.

6.8.4 Evaluation

To evaluate the potential for hydrogen accumulation in containment following a LOCA, the hydrogen generation as a function of time following the initiation of the accident is calculated. Assumptions recommended by Reference 6 are used to maximize the amount of hydrogen calculated. The hydrogen release to containment is graphically presented in UFSAR Figure 6.8-2. It will take more than 20 days after a LOCA for the hydrogen concentration in the containment to reach 4.0 volume percent, if no recombiner was functioning.

The PHRs are designed such that, a single recombiner is capable of limiting the peak hydrogen concentration in containment to less than 4.0 volume percent.

Two PHR are located on the operating deck at an approximate elevation of 29 m (95 feet) out side the missile shield wall. This location is away from the reactor coolant piping and possible impingement from high-energy line breaks.

Mixing of the hydrogen in containment is ensured through the use of the containment fan coolers. Three to five fan coolers operate in the post-accident environment. The capability of the fan coolers is described in Section 6.4. The flow distribution from fan coolers is directed to various locations including locations high in containment. Both PHRs are mounted on the open grating to allow free airflow through the PHRs. The housing extends above the catalyst elevation to provide a chimney to yield additional lift to enhance the efficiency of the device. One condition that is to be avoided for proper PHR functioning is submersion in water. The PHR are located above the containment flood level and they are designed with a spray hood to minimize the direct contact with the post-accident containment sprays.

Since PHRs are credited for post-LOCA hydrogen control the effects of potential catalyst inhibitors and poisons present during the accident conditions were accounted for. The Electric Power Research Institute (see Reference 5) has evaluated the effects of potential catalyst inhibitors and poisons. Limited decreases in catalyst effectiveness were noted. Any decrease in effectiveness is more than compensated for by the combination of conservative hydrogen generation assumptions and substantial excess installed PHR capacity.

6.8.4.1 Qualification Testing

As a prerequisite for final installation of PHRs at Indian Point 2, environmental qualification tests were performed at Wyle Laboratories in accordance with the applicable requirements of 10CFR21, 10CFR50 Appendix B, and ANSI N45.2. Sample cartridges from equipment device designed and built by NIS Ingenieurgesellschaft mbH (NIS) supplied for Indian Point 2 were subjected to the qualification testing. The results of these tests were submitted to NRC and NRC approved (Reference 4) the use of NIS PARs at Indian Point 2.

The details of these tests, the acceptance criteria and the results are documented in the Wyle test report, which is filed in Indian Point EQ files.

6.8.5 Inspections And Tests

A visual examination and an operating check of the system is performed periodically. Visual examination and cleaning if necessary is performed at each refueling outage to verify that there is no significant fouling by foreign material. Performance of a test on a sample plate removed from each hydrogen recombiner at each refueling outage ensures the recombiners are operational. The sample plate removed from each recombiner is inserted into a test device and a fixed flow mixture of gas that is 1% to 1.5% hydrogen in air is supplied to the device. The plate is judged to be degraded if the temperature developed is not within the acceptance criteria. In this case the neighboring plate will be tested. Any plates found to be degraded will be evaluated or replaced with new plates.

6.8.6 Minimum Operating Conditions

Operability requirements for the PHRs were removed from the Technical Specifications by license amendment #243.

REFERENCES FOR SECTION 6.8

1. Letter from S. A. Varga, NRC, to J. D. O'Toole, Con Edison, Subject: Postaccident Sampling at the Indian Point Unit 2, Safety Evaluation Report, dated June 28, 1984.

2. Letter from S. A. Varga, NRC, to J. D. O'Toole, Con Edison, Subject: Postaccident Sampling at the Indian Point Unit 2, Safety Evaluation Report, dated December 12, 1984.

3. S. E. Quinn to Document Control Desk, USNRC, "Proposed Technical Specification for H2 Recombiners", August 21, 1996.

4. Letter from Jeffrey F. Harold, NRC to A. Alan Blind, Con Edison “Issuance of Amendment and Bases Change for Indian Point Nuclear Generation Unit No. 2 (TAC No. M96475)”, dated April 27,1999.

5. EPRI ALWR Report GC-108771 “Effects of Inhibitors and Poisons on the Performance of Passive Autocatalytic Recombiners (PARs) for Combustible Gas Control in ALWRs”, June 28, 1999.

6. Regulatory Guide 1.7, Control of Combustible Gas Concentrations in Containment Following a Loss-of-Coolant Accident (Rev. 2)

7. Letter from P. Milano, NRC to M. Kansler, Entergy; “Issuance of Amendment (243) Eliminating Requirements for Hydrogen Recombiners and Hydrogen Monitors”, dated April 14, 2005.

6.8 FIGURES

APPENDIX 6A
EFFECTIVENESS OF THE CONTAINMENT SPRAY SYSTEM
TO REMOVE AIRBORNE ACTIVITY FOLLOWING A LOCA

In the event of a postulated Loss-of-Coolant Accident (LOCA) with degraded core, there would be a substantial release of core fission product activity to the containment atmosphere. The core degradation source term is assumed to release the following fractions of core activity to the containment atmosphere consistent with the source term described in Regulatory Guide 1.183 (Reference 1):

Noble gases	1.0
Iodines	0.4
Alkali metals	0.3
Tellurium group	0.05
Barium & Strontium	0.02
Noble metals	0.0025
Cerium group	0.0005
Lanthanides	0.0002

With the exception of the noble gases, the activity released to the containment atmosphere is subject to removal by the containment spray system. As defined in Regulatory Guide 1.183, the iodine activity entering the containment atmosphere is assumed to exist primarily as an aerosol (95% of the iodine is assumed to be in the form of cesium iodide) with the remainder existing as elemental iodine vapor (4.85%) or as organic compounds (0.15%). The activity for the remaining nuclide groups is assumed to all be in the aerosol form.

The containment spray system is one of the engineered safety features systems employed following a LOCA to reduce the pressure and temperature in the containment. The spray system also affords an excellent means of removing both elemental iodine vapor and aerosols from the containment atmosphere. Organic iodine compounds are assumed not to be subject to removal by sprays.

During the spray injection phase the spray consists of boric acid solution taken from the refueling water storage tank. This spray solution has a pH of ~4.5 – 5.0. While sprays are an effective means to remove airborne iodine, retention of iodine in the sump solution requires that the solution pH be raised to 7.0 or above. This pH adjustment is provided by the sodium tetraborate stored in baskets in the sump area (See Section 6.3).

The removal of airborne activity by the containment sprays is expressed by the following equation which calculates the amount of activity remaining at a given time:

$$C = C_O\, e^{-(\lambda_s t)} \qquad (6A\text{-}1)$$

where:

C = Current activity, Ci
C_o = Initial activity, Ci
λ_s = Spray removal coefficient, hr^{-1}
t = time, hr

6A.1 SPRAY REMOVAL COEFFICIENT FOR PARTICULATES

The spray removal coefficient for particulates is determined using the model described in SRP Section 6.5.2 (Reference 2):

$$\lambda_p = 3hFE / 2VD \qquad (6A\text{-}2)$$

where:

λ_p = Particulate removal rate constant due to spray removal, hr^{-1}
h = Drop fall height, ft
F = Spray flow rate, ft^3/hr
V = Volume Sprayed, ft^3
E = Single drop collection efficiency
D = Drop diameter, ft

The value for E/D is conservatively defined in SRP Section 6.5.2 to be 10 m^{-1} (3.048 ft^{-1}) initially and is reduced by a factor of 10 after the suspended aerosol mass has been depleted by a factor of 50 (i.e., after 98% of the aerosols have been removed).

Spray Fall Height and Sprayed Volume

The spray system nozzle and header arrangement is designed to cover a maximum area in the upper containment. Four headers, arranged in concentric circles, are located in the containment dome at elevations of 213.5, 218.6, 223.6, and 228.6 feet.

Credit is taken for spray coverage of the total volume above the operating deck. The spray fall height from the lowest of the spray headers is 118.5 feet. It is conservatively assumed that all spray falls through the same distance (ignoring the additional fall-time associated with the higher spray headers). The sprayed volume is 80 percent of the containment free volume (i.e., 2.088×10^6 cubic feet).

Containment Spray Flow Rate

During the spray injection phase it is assumed that only one of the two spray pumps is operating, drawing water from the refueling water storage tank. The minimum spray flow rate is 2135 gpm (17,646 ft^3/hr) for one spray pump under design basis accident conditions.

Once the inventory of the refueling water storage tank is depleted, there is a switch to the recirculation spray phase. The containment spray pump is not used during the spray recirculation phase; instead, the flow to the spray headers is obtained from the recirculation pump which recirculates water from the sump to the reactor vessel. The system resistance provided by the physical configuration of the recirculation piping and components is hydraulically balanced such that sufficient flow is established to the core and to the spray headers. The minimum recirculation spray flow rate of 1100 gpm (8823 ft^3/hr) is half the injection spray flow rate.

Spray Removal Coefficient

Using the above-defined values, the spray removal coefficient for aerosols is determined as follows:

λ_p = 3hFE / 2VD

Injection Phase: $\lambda_p = 4.5\ hr^{-1}$

Recirculation Phase: $\lambda_p = 2.28\ hr^{-1}$

These values are reduced by a factor of 10 once the aerosol inventory is reduced to two percent of the total aerosol inventory released to the containment atmosphere.

6A.2 SPRAY REMOVAL COEFFICIENT FOR ELEMENTAL IODINE

The original design of the containment spray system included spray additive (sodium hydroxide) to increase the pH of the boric acid solution to approximately 9.5. The purpose of the pH adjustment was to increase spray removal of elemental iodine. However, as discussed in SRP Section 6.5.2 (Reference 2), it has been determined that the boric acid spray without pH adjustment to alkaline conditions is an effective means to remove airborne elemental iodine. The spray removal coefficient for elemental iodine is determined using the model described in SRP Section 6.5.2:

$$\lambda_s = 6K_g tF / VD \qquad (6A\text{-}3)$$

where

- λ_s = Elemental iodine removal rate constant due to spray removal, hr^{-1}
- K_g = Gas phase mass transfer coefficient, ft/min
- t = Average spray droplet fall time, min
- F = Spray flow rate, ft^3/hr
- V = Volume sprayed, ft^3
- D = Average drop diameter, ft

The values for spray flow rate and sprayed volume are defined in Section 6A.1. The gas phase mass transfer coefficient (K_g) is conservatively defined as 3 m/min (9.084 ft/hr) in Reference 3 based on a number of experimental studies.

SRP Section 6.5.2 specifies an upper limit for λ_s of 20 hr^{-1} for fresh solution.

Drop Diameter

There is a spectrum of drop sizes in the containment atmosphere at any time. Measurements of the drop size distribution of the Sprayco 1713 nozzle have shown that the drop size varies between 80 and 3800 μm, with ~80 percent of the generated drops being 500 μm or smaller (Reference 4). The initial distribution of drops after release from the nozzles is affected during the fall through the containment atmosphere by coalescence and by condensation of steam onto the drops. The increase in drop size associated with condensation increases the total surface area of the droplets that would be available for absorbing iodine. While coalescence of

drops also increases individual droplet size, it decreases the number of droplets and thus decreases the total surface area of the airborne droplets. Increases in droplet size also decrease the fall time during which the drops are available to remove elemental iodine from the containment atmosphere.

Since the spray droplets enter the containment at temperatures far below that of the initial temperature of the containment atmosphere, condensation of steam from the containment air-steam mixture will increase the initial size of the drops until they are in thermal equilibrium with the ambient.

From an energy balance on the drop:

$$mh + m_c\, h_g = m'h_f \qquad (6A\text{-}4)$$

where:

m = mass of drop before condensation
h = enthalpy of drop at spray inlet temperature
m_c = mass of condensed steam
m' = mass of drop after condensation
h_g = saturation vapor enthalpy at containment condition
h_f = saturation liquid enthalpy at containment condition

From a mass balance: $m_c = m' - m$

Substituting into equation 6A-4: $mh + (m' - m)h_g = m'h_f$

or: $m'/m = (h_g - h)/(h_g - h_f)$

But, $m = 4\pi r^3/3v$ and $m' = 4\pi r'^3/3v_f$

where v is the specific volume at inlet conditions and v_f is the specific volume at containment conditions.

The increase in the drop radius due to condensation then is:

$$\frac{r'}{r} = \left(\frac{v_f\left(h_g - h\right)}{v\left(h_g - h_f\right)} \right)^{\frac{1}{3}} \qquad (6A\text{-}5)$$

Just as the spray droplets can remove aerosols from the containment atmosphere, the droplets are capable of removing other droplets. This collision and resulting coalescence result in an increase in the average drop size. Coalescence efficiency is the probability that a collision between two drops will result in the formation of a single larger drop. It is conservatively assumed that all droplet collisions result in coalescence. This is a very conservative assumption which results in the prediction of an average droplet size significantly larger than would be expected to occur.

Taking the effects of coalescence and condensation into consideration, the value for the mean droplet diameter is 1200 μm (3.94E-3 ft).

Spray Droplet Fall Time

One of the simplifying assumptions of the model is that the residence time of the drop in the containment atmosphere may be approximated by fall height divided by the terminal velocity of the drop (h/U_t).

However, the actual residence time of the drop is considerable longer, since the drops do not leave the nozzle with only a vertical velocity component, but with an additional horizontal component, which causes the droplets to fall along a trajectory, which increases the residence time of the droplet. (This fact is further amplified by the 45-degree nozzles.) Thus, the use of h/U_t, combined with defining the spray fall height as the fall height for the lowest elevation spray ring header, adds further conservatism to the model. This model gives a minimum residence time of 10.7 seconds for a 1200 μm drop.

Spray Removal Coefficient

Using the above-defined values, the spray removal coefficient for aerosols is determined as follows:

$\lambda_s = 6K_g tF / VD$

Injection Phase:	$\lambda_s = 20\ hr^{-1}$	The calculated value of 22.5 hr^{-1} is reduced to 20 hr^{-1} consistent with the upper limit defined in SRP Section 6.5.2.
Recirculation Phase:	$\lambda_s = 5.6\ hr^{-1}$	The calculated value of 11.2 hr^{-1} is reduced by 50% to 5.6 hr^{-1} to address the loading of the recirculating solution with elemental iodine.

6A.3 LIMITS OF REMOVAL

There is no defined limit in SRP Section 6.5.2 on the removal of aerosols from the atmosphere. A conservative approach, which is used for Indian Point Unit 2, is to limit credit for aerosol removal to a DF of 1000 (i.e., 0.1% of the original inventory release to the containment atmosphere remaining airborne). If recirculation spray is terminated prior to reaching this limit, sedimentation will continue to remove aerosols until the DF limit is reached. The removal rate associated with sedimentation is addressed in Section 14.3.6.1.

SRP Section 6.5.2 specifies a limit on elemental iodine removal of a DF of 200 (i.e., 0.5% of the original inventory release to the containment atmosphere remaining airborne). This is dependent on the sump solution pH being adjusted to ≥7.0. As discussed in Section 6.3.2, the mass of Sodium Tetraborate stored in the containment is sufficient to assure that following a LOCA a sump solution pH of ≥7.0 is achieved and maintained.

REFERENCES FOR APPENDIX 6A

1. USNRC Regulatory Guide 1.183, “Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors,” July 2000.

2. NUREG-0800, USNRC Standard Review Plan, Section 6.5.2, “Containment Spray as a Fission Product Cleanup System,” Revision 2, December 1988.

3. R. E. Davis, H. P. Nourbakhsh, and M. Khatib-Rahbar, “Fission Product Removal Effectiveness of Chemical Additives in PWR Containment Sprays,” Technical Report A-3788, Brookhaven National Laboratory, 8/12/86.

4. WCAP-8659, “CIRCUS Computer Code – Calculation of Vapor Phase Elemental Iodine Removal in the Reactor Containment by Chemical Additive Spray,” June 1975, (Westinghouse Proprietary)

APPENDIX 6B
PRIMARY SYSTEM LEAK DETECTION INTO CONTAINMENT VESSEL, INDIAN POINT UNIT 1

Table of Contents

Section and Title

6B.0 OPERATIONAL EXPERIENCE

During the lifetime of Indian Point 1, primary system leakage was minimal. A combination of all the following instrumentation was used to detect several leaks ranging in size from 0.1 to 3 gal/hr. However, because of the magnitude of these leaks, positive identification resulted only from visual inspection during containment entries made after scheduled plant shutdown.

Small leaks that developed in the primary system pressure boundary could be detected by several continuously recording instruments available to the plant operators. The most sensitive of these detectors was the radioactive air particulate monitor, which continuously sampled the air in the containment cooling system. The purpose of the containment cooling system was to maintain proper ambient temperatures for equipment in the containment vessel. This system took air from the upper elevations of the vessel and recirculated it through cooling coils on the suction side of the supply fan. This air was then discharged at a rate of 40,000 cfm through steam coils. The turnover rate of the containment vessel as a result of this system was approximately once every hour. By sampling air from the discharge of the containment cooling system supply fan, leak rates as small as 0.3 gal/hr (20 cm^3/min) could be detected.

Another detector, the radiogas monitor, sampling air from the same position as the air particulate monitor, continuously analyzed air from the containment cooling system for gaseous radioactivity. This monitor was capable of detecting a leak rate of about 100 gal/hr (6500 cm^3/min).

In addition to measuring changes in the radioactivity of the containment vessel, dewpoint sensors continuously sampled the air from the suction side of the containment cooling system supply fans. These instruments could detect a primary coolant leak rate of approximately 4 gal/hr (250 cm^3/min) by measuring changes in the moisture content of the containment vessel.

By the use of the above instruments, plant operators could continuously monitor the containment vessel for primary system leakage and take any steps necessary to operate the facility safely. Measurements made by the New York University Medical Center, Institute of Environmental Medicine, have shown that the samples analyzed by these instruments are representative of the containment vessel and that samples take manually to backup these detectors were accurate to within a factor of 2.

Other methods for detecting and locating primary system leakage included visual inspection for escaping steam or water, boric acid crystal formation, component and primary relief tank levels, hydrogen concentration and radioactivity, containment sump level, and manual samples for tritium radioactivity in condensed moisture from the containment vessel.

6B.1 ASSUMPTIONS

1. Uniform mixing in containment occurs within 1 hr after a leak, based upon one containment cooling fan in service at 40,000 cfm.

2. The smallest significant changes instrumentation are as follows:
   a. Radiogas monitor on the containment cooling system
      1 count per sec is equivalent to 3×10^{-7} µCi/cm^3
   b. Particulate monitor 8 counts per sec is equivalent to 8×10^{-9} µCi/cm^3

   c. Dewpoint 4°F

An 8-hr period was used to evaluate these changes; that provides time for checking instrumentation and determining the cause of the changes. The 8-hr evaluation period was predicted on determination of the magnitude of small leaks. Large leaks would of course be evaluated much sooner.

6B.2 BASIC DATA USED FOR CALCULATIONS

1. Sphere volume.

   1.8×10^6-ft^3 (5.05×10^{10} cm^3)

2. Sphere environment.

   a. Average temperature - 120°F
   b. Dewpoint - 70°F

3. Normal containment cooling radioactivity.

   a. Radiogas 2.5 counts per sec (7.5×10^{-7} µCi/cm^3)
   b. Particulate 16 counts per sec (1.6×10^{-3} µCi/cm^3)

4. Normal primary coolant radioactivity after 1 hr.

   a. Radiogas activity 5×10^{-3} µCi/ml H_2O
   b. Particulate 5×10^{-2} µCi/ml H_2O

6B.3 SAMPLE CALCULATIONS

1. Dewpoint in containment cooling system.

   a. At 120°F and 70°F dewpoint - the water content of the sphere would be 0.016 lb of water per pound of dry air

b. At a dewpoint of 74°F the water content of the sphere would be 0.018 lb of water per pound of dry air let X = the leak rate into the sphere in gallons per hour

$$X = \frac{(0.018 - 0.016)\,\text{lb } H_2O\,/\,\text{lb dry air} \times 1.8 \times 10^{6}\,\text{ft}^3 \times 0.081 \times \frac{109}{121}\,\text{lb}\,/\,\text{ft}^3}{8\text{hrs} \times 8.3\text{lb}\,/\,\text{gal}}$$

= 3.95 gal/hr or 100 gal/day

2. Radioactivity in containment cooling system.

a. Radiogas activity

(1) Increase in activity 1 cps on installed monitor
The radiogas activity increase = 3×10^{-7} μCi/cm^3

(2) Let Y = leak rate into a sphere in gallons per hour

$$Y = \frac{3.0 \times 10^{-7}\,\mu\text{Ci}\,/\,\text{cm}^3\,\text{air} \times 5.05 \times 10^{10}\,\text{cm}^3\,\text{air}}{8\text{hr} \times 5 \times 10^{-3}\,\mu\text{Ci}\,/\,\text{ml}\ \ H_2O \times 3.8 \times 10^{3}\,\text{ml}\,/\,\text{gal}}$$

= 99.8 gal/hr or 2400 gal/day

b. Particulate activity in containment cooling system 8 counts per sec on the installed monitor
Radioactivity increase = 8×10^{-9} μCi/cm^3 air
Let Z = leak rate into the sphere in gallons per hour

$$Z = \frac{8 \times 10^{-9}\,\mu\text{Ci}\,/\,\text{cm}^3\,\text{air} \times 5.05 \times 10^{10}\,\text{cm}^3\,\text{air}}{8\text{hr} \times 5 \times 10^{-2}\,\mu\text{Ci}\,/\,\text{ml}\ H_2O \times 3.8 \times 10^{3}\,\text{ml}\,/\,\text{gal}}$$

= 0.265 gal/hr or 6 gal/day

APPENDIX 6C
POST-ACCIDENT CONTAINMENT ENVIRONMENT, REVISION 1

TABLE OF CONTENTS

Section and Title

Appendix 6C
POST-ACCIDENT CONTAINMENT ENVIRONMENT

LIST OF TABLES

LIST OF FIGURES

Appendix 6C
POST ACCIDENT CONTAINMENT ENVIRONMENT

6C.1 DEFINITION OF POSTACCIDENT CONTAINMENT ENVIRONMENTAL CONDITIONS

As part of the initial license application, an evaluation of the suitability of materials of construction for use in the reactor containment system was performed considering the following:

1. The integrity of the materials of construction of engineered safeguards equipment when exposed to post-design-basis accident conditions.

2. The effects of corrosion and deterioration products from both engineered safeguards (vital equipment) and other (non vital) equipment on the integrity and operability of the engineered safeguards equipment.

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

The original chemistry for the Containment Spray System utilized an alkaline adjusted sodium borate containment spray with the pH adjusted by sodium hydroxide. Use of solid Trisodium Phosphate for pH control of the solution has been used at a number of plants and was implemented at Indian Point 2 (IP2) in 1997. Reference 24 (WCAP-14542) discusses the benefits and justification for this change. In response to NRC Generic Letter 2004-02, Trisodium Phosphate (TSP) has been replaced by Sodium Tetraborate (STB) for pH control in order to reduce the risk of sump screen plugging due to the formation of chemical products. Replacement of TSP with STB was evaluated in WCAP-16596-NP (Reference 41) and concluded STB was the most comparable alternative to TSP and NaOH. This section was updated to incorporate this change and much of the updated information was drawn from WCAP-16596-NP. Updated references were also added as appropriate.

6C.1.1 Design-Basis Accident Temperature-Pressure Cycle

Figures 6C-1 and 6C-12 present the temperature-pressure-time relationship following the design-basis accident. These figures represent the containment condition for the following safety feature operation. One of the two spray pumps is considered to inject into the containment. When the refueling water storage tank is empty, the recirculation pumps can supply flow to the spray headers. Recirculation flow through one recirculation pump is cooled in the residual heat exchanger.

Figures 6C-2 and 6C-3 present materials evaluation test conditions for the containment and core environment, respectively.

Materials evaluations were performed, in general, for conditions either simulating the time-temperature conditions of Figure 6C-2 or conservatively considering higher temperatures for longer periods. The basis for each material evaluation is described with the discussion of its particular suitability.

6C.1.2 Design-Basis Accident Radiation Environment

The evaluation of materials for use in containment includes a consideration of the radiation stability requirements for the particular materials application. Figures 6C-4 and 6C-5 present the post-design-basis accident containment atmosphere direct gamma dose rate and the integrated direct gamma dose, respectively. These data were calculated on the basis of a core meltdown and assume the following fission product fractional releases, consistent with the TID-14844 model:

1. Noble gases, 1.0
2. Halogens, 0.5
3. Other isotopes, 0.01

6C.1.3 Design Chemical Composition Of The Emergency Core-Cooling Solution

Nuclear Regulatory Commission Branch Technical Position MTEB 6-1 (ref 31) specifies a minimum pH level of 7.0 of the post-accident emergency coolant water, operation higher in the pH 7.0 to 9.5 range for greater assurance that no stress corrosion cracking will occur, and if pH greater than 7.5 is used consideration should be given to hydrogen generation from aluminum.

The system designs provide for the use of alkaline-adjusted boric acid solution as the spray and core-cooling fluid. Initially the injection solution is not alkaline-adjusted since the RWST contains only boric acid and not STB. It is not until re-circulation from the sump, where the injected water has dissolved STB, that the spray and core-cooling fluid is alkaline adjusted.

Plant designs that use the spray solution for retention of fission product iodine in solution, as well as containment cooling, include provisions for addition of chemical additive to the emergency core cooling system. Originally, that additive was a concentrated sodium hydroxide solution but a number of plants have converted to dry Trisodium Phosphate granules and IP2 has since converted to Sodium Tetraborate granules. Boric acid solution, containing approximately 2600 ppm boron, is pumped from the refueling water storage tank to the containment system by means of the safety injection system pumps, residual heat removal pumps, and the spray pumps.

Granular Sodium Tetraborate is stored in baskets strategically located in the post-accident flooded region of the containment. Initially the containment spray will be boric acid solution from the refueling water storage tank which has a pH of approximately 4.5. As the initial spray solution and subsequently the recirculation solution comes in contact with the Sodium Tetraborate, the STB dissolves raising the pH of the sump solution to an equilibrium value between 7.0 and 7.6 (Reference 42, pg 18).

Based on Reference 42, 8096 pounds of Sodium Tetraborate Decahydrate is sufficient to assure a post-LOCA sump pH of 7.0 at 30 days with a margin of approximately 2 (in terms of strong acid addition) to reach a pH of 7.0 (Reference 42, Figure 2). Titration curves for TSP in Boric acid developed in CN-CDME-00-10 (Ref. 32) are shown in figure 6C-6. Based on these

curves, 8000 pounds of Trisodium Phosphate Dodecahydrate ($Na_3PO_4 \cdot 12H_2O$) is sufficient to assure a post-LOCA sump pH of 7.0 (Ref. CN-CRA-96-005, Rev. 2) with a margin of approximately 41% to account for formation of acids over time in the solution. In addition the maximum pH due to Trisodium Phosphate is 7.61.

Reference 24A addresses the time required to submerge and dissolve the TSP and concludes that conservatively the dissolution times would be 1.7 hours for 6000 pounds of TSP and 2 hours for 8000 pounds. Realistically, the dissolution time would be considerably less. Dissolution testing performed in support of the buffer change to STB concluded that the rate of dissolution for STB was much faster than TSP (References 41 & 42).

The solutions are considered aerated through the entire exposure period.

6C.1.4 <u>Trace Composition Of Emergency Core Cooling Solution</u>

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

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

To identify the trace elements in the containment that may have a deleterious effect on engineered safeguards equipment, one must first establish, which elements are potentially harmful to the materials of construction of the safeguards equipment, and second, ascertain the presence of these elements in forms, which can be released to the emergency core cooling solution following a design-basis accident. Table 6C-1 presents a listing of the major periodic groups of elements. Elements that are known to be harmful to various metals are noted and potential sources of these elements are identified. A discussion of the effects of these elements is presented in later sections.

6C.2 MATERIALS OF CONSTRUCTION IN CONTAINMENT

All materials in the containment are reviewed from the standpoint of ensuring the integrity of the equipment of which they are constructed and to ensure that deterioration products of some materials do not aggravate the accident condition. In essence, therefore, all materials of construction in the containment must exhibit resistance to the post-accident environment or, at worst, contribute only insignificant quantities of the trace contaminants that have been identified as potentially harmful to vital safeguards equipment. This must be true for these materials in both the new condition and for the aged condition at which the post-accident environment might be more likely to be encountered. In addition to the integrity of major components (e.g. piping, supports, vessels, containment structures, etc.), environmental qualification of Class 1E equipment must not be affected. Section 6C.9 addresses requalification of Class 1E equipment.

Table 6C-2 lists typical materials of construction used in the reactor containment system. Examples of equipment containing these materials are included in the table.

Corrosion testing, described in Section 6C.3, showed that of all the metals tested only aluminum alloys were found incompatible with the alkaline sodium borate solutions. Aluminum was observed to corrode at a significant rate with the generation of hydrogen gas. Since hydrogen generation can be hazardous to containment integrity, a detailed survey was conducted to identify all aluminum components in the containment.

Table 6C-3 lists the nuclear steam supply system aluminum inventory that is present in the reactor containment. Included in the table is the mass of metal and exposed surface area of each component. The 1100 and the 6000 series aluminum alloys are generally the major types found in the containment. This inventory reflects the determination to exclude as much as practicable the actual use of aluminum in the containment.

6C.3 CORROSION OF METALS OF CONSTRUCTION IN DESIGN-BASIS EMERGENCY CORE COOLING SOLUTION

Emergency core cooling components are primarily austenitic stainless steel and hence are quite corrosion resistant to the alkaline sodium borate solution as demonstrated by corrosion tests performed at Westinghouse and Oak Ridge National Laboratory.[1] The general corrosion rate, for type 304 and type 316 stainless steels was found to be 0.01 mils per month in pH 10 solution at 200°F. Data on corrosion rates of these materials in the alkaline sodium borate solution have also been reported by Oak Ridge National Laboratory [2,3] to confirm the low values.

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

Metals tested included Zircaloy, Inconel, aluminum alloys, cupro-nickel alloys, carbon steel, galvanized carbon steel, and copper. The results of the corrosion testing of these materials are reported in detail in Reference 1. Of the materials tested, only aluminum was found to be incompatible with the alkaline sodium borate solution. Aluminum corrosion is discussed in Section 6C.5. The following is a summary of the corrosion data obtained on various materials of construction exposed for several weeks in aerated, alkaline (pH 9.3 to 10.0) sodium borate solution at 200°F. The exposure condition is considered conservative since the test temperature (200°F) is considerably higher than the long-term design-basis accident temperature.

Material	Maximum Observed Corrosion Rate (mil/month)
Carbon Steel	0.003
Zircolay-4	0.004
Inconel 718	0.003
Copper	0.03
90-10 Cu-Ni	0.002
70-30 Cu-Ni	0.051
Galvanized carbon steel	0.031
Brass	-0.01

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

Corrosion tests at both Westinghouse and Oak Ridge National Laboratory have shown copper suffers only slight attack when exposed to the alkaline sodium borate solution at design-basis accident conditions. The corrosion rate of copper, for example, in alkaline sodium borate solution at 200°F is approximately 0.03 mils per month.[1] The corrosion of copper in an alkaline sodium borate environment under spray conditions at 284°F and 212°F have been reported by Oak Ridge National Laboratory. Corrosion penetration of less than 0.02 mil was observed after 24-hr exposure at 284°F (see Reference 3, Table 3.13) and a corrosion rate of less than 0.3 mils per month was observed at 212°C (see Reference 2, Table 3.6).

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

The corrosion rate of galvanized carbon steel in alkaline sodium borate (3000 ppm boron, pH 9.3) is also low. Tests conducted in aerated solutions showed the corrosion rate to be 0.003 mils per month (0.046 mg/dm^2-hr) and 0.002 mils per month (0.036 mg/dm^2-hr) for temperatures of 200 and 150°F, respectively. Therefore, it can be seen that the corrosion of zinc (galvanized) in alkaline borate solution is minimal and will not contribute significantly to the post-accident hydrogen buildup.

Consideration was given to possible caustic corrosion of austenitic steels by the alkaline solution. Data presented by Swandby (Figure 6C-7) show that these steels are not subject to caustic stress cracking at the temperature (285°F and below) and caustic concentrations (less than one weight percent) of interest. It can be seen from Figure 6C-7 that the stress cracking boundary minimum temperature as defined by Swandby coincides with a high free caustic concentration (approximately 40-percent) and is considerably above (approximately 80°F) the long-term post-accident design temperature. Further, from Figure 6C-7 a temperature in excess of 500°F is required to produce stress corrosion cracking at a sodium hydroxide concentration greater than 85-percent.

6C.4 CORROSION OF METALS OF CONSTRUCTION BY TRACE CONTAMINANTS IN EMERGENCY CORE COOLING SOLUTION

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

6C.4.1 MERCURY

The use of mercury or mercury-bearing items, however, is restricted in the containment. This includes mercury vapor lamps, fluorescent lighting, and instruments that employ mercury for pressure and temperature measurements and for electrical equipment. The use of mercury is limited to the refueling cavity lights. Potential sources of mercury therefore, are generally excluded from the containment and hence, no hazard from this element is recognized.

The refueling cavity lights contain a small amount of mercury in the arc tube, enclosed in a quartz enclosure to preclude breakage due to thermal shock. An evaluation has demonstrated that neither the arc tube nor the outer quartz protective tube will break during a seismic event and they can withstand both the pressure and temperature expected during a loss-of-coolant accident.

6C.4.2 Chlorine

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

6C.4.2.1 Low Temperature of Emergency Core Cooling Solution

The temperature of the emergency core cooling solution is reduced after a relatively short period of time (i.e., a few hours) to about 150°F. While the influence of temperature on stress corrosion cracking of stainless steel has not been unequivocally defined, significant laboratory work and field experience indicate that lowering the temperature of the solution decreases the probability of failure. Hoar and Hines[5] observed this trend with austenitic stainless steel in 42 weight percent solutions of $MgCl_2$ with temperature decrease from 310°F to 272°F. Staehle and Latanision[6] present data that also show the decreasing probability of failure with decreasing solution temperature from about 392°F to 302°F. Staehle and Latanision[6] also report the data of Warren[7] that showed the significant change with decrease in temperature from 212°F to 104°F. The work of Warren, while pertinent to the present consideration in that it shows the general relationship of temperature to time to failure, is not directly applicable in that the chloride concentration (1800 ppm Cl) believed to have effected the failure was far in excess of reasonable chloride contamination, which may occur in the emergency core cooling solution. More recent articles by Sedricks (ref 28), Moller (ref 29), and Macdonald & Cragnolino (ref 30) all state the importance of temperature as a variable in determining whether chloride stress corrosion cracking will occur but yet do not provide any definitive guidance. Jiang and Staehle (ref 34) correlate data from 17 references and conclude that for constant stress an Arrhenius form equation is reasonable for presenting the dependencies on t_f.

6C.4.2.2 Low Chloride Concentration of Emergency Core Cooling Solution

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

1. Restriction in chloride content of water used in concrete.
2. Prohibition of use of chloride in cleaning agents for stainless steel components and surfaces.
3. Prohibition of use of chloride in concrete etching for surface preparation.
4. Use of non-chloride bearing protective coatings in containment.
5. Restriction in chloride concentration in safety injection solution, 0.15-ppm chloride maximum.

The effect of decreasing chloride concentration on decreasing the probability of failure of stressed austenitic stainless steel has been shown by many experimenters. Staehle and Latanision[6] present data of Staehle that show the decrease in probability of failure with decrease in chloride concentration at 500°F. Edeleanu[8] shows the same trend at chloride concentrations from 40 to 20-percent as $MgCl_2$ and reported no failures in this experiment at less than about 5-percent $MgCl_2$. Westinghouse corrosion tests (ref. 22) intended to simulate design basis accident conditions showed that crack initiation time increases with decreasing chloride concentration in tests.

Instances of chloride cracking at representative emergency core cooling solution temperatures and at low solution chloride concentration have generally been on surfaces on which concentration of the chloride occurred. In the emergency core cooling system, concentration of chlorides is not anticipated since the solution will operate sub-cooled with respect to the containment pressure and further, the containment atmosphere will be 100-percent relative humidity.

6C.4.2.3 Alkaline Nature of the Emergency Core Cooling Solution

The emergency core cooling solution will have a solution pH of greater than 7.0 after dissolution of the Sodium Tetraborate additive stored in the sumps. Numerous investigators have shown that increasing the solution pH decreases the probability of failure. Thomas et al.,[9] showed that the failure probability decreases with increasing pH of boiling solutions of $MgCl_2$. More directly applicable, Scharfstein and Brindley[10] showed that increasing the solution pH to 8.8 by the addition of NaOH prevented the occurrence of chloride stress corrosion cracking in a 10-ppm Chloride (as NaCl) solution at 185°F. Thirty stressed stainless-steel specimens including 304 as received, 347 as received, and 304 sensitized were tested. No failures were observed.

Other test runs by Schafstein and Brindley showed the influence of solution pH on higher chloride concentrations, up to 550 ppm Chloride; however, in these tests the pH-adjusting agents were either sodium phosphate or potassium chromate. The authors express the opinion, however, that in the case of the chromate solution, chloride cracking inhibition was simply because the hydrolysis yielded a pH of 8.8 and not because of an influence of the chromate anion. A similar hydrolysis will occur in the borate solution.

Studies conducted at Oak Ridge National Laboratory by Griess and Bocarella[11] on type 304 and type 316 stainless-steel U-bend stress specimens exposed to an alkaline borate solution (0.15M NaOH - 0.28M H_3BO_3) containing 100-ppm chloride (as NaCl) showed no evidence of cracking after 1 day at 140°C, 7 days at 100°C, 29 days at 55°C. These extreme test conditions, combined with the fact that some parts of the test specimens were subjected to severe plastic deformation and intergranular attack before exposure, show that the probability of chloride induced stress corrosion cracking in a post-accident environment is very low indeed.

Westinghouse corrosion tests (ref. 22) showed that at pH 7, 100 ppm Cl, sensitized and non-sensitized samples of 304 stainless steel cracked in approximately 7.5 and 10 months, respectively. Based primarily on those results, Branch Technical Position MTEB 6-1 (ref. 31) recommends that the minimum pH of the sump solution should be 7.0 and that the higher the pH, in the range 7 to 9.5, the greater the assurance that stress corrosion cracking will not occur.

As discussed in section 6C.1.3, the initial pH of the solution will be approximately 4 and will increase after the Sodium Tetraborate dissolves. Therefore, for some period of time the spray solution will be below 7.0. This period of time was conservatively estimated at 2.0 hours for Trisodium Phosphate (Reference 24) and would be less than that for Sodium Tetraborate (References 41 & 42). Westinghouse corrosion tests (Ref. 22) indicate that the minimum time to crack (100% crack of a 304 SS welded single U-bend) in a pH 4.5, 100 ppm Cl solution is 3 days and no cracking of any test materials was observed before 8 hours. Thus crack initiation occurred between 8 hours and three days. pH adjustment must occur prior to initiation of cracking. Hence, based on these results, it is necessary that the pH of the sump solution be raised above 7.0 within 8 hours.

Chlorides are not expected to instantaneously appear in the sump solution in concentrations sufficient to initiate cracking. The initial spray and safety injection solution is drawn from the refueling water storage tank where the chloride concentration is limited to 0.15 ppm. The Westinghouse tests indicate that crack initiation in boric acid with 0.4 ppm chloride and pH of approximately 4 requires extended exposure times (12 months in one example). Hence, cracking will not occur during the relatively brief spray and safety injection.

During recirculation, as the solution washes over the containment structures and components, chlorides and other contaminants will be removed from the surfaces and dissolved in the solution. Concrete is potentially a significant chloride source but is painted with a nuclear qualified coating which is expected to greatly impede chloride leaching. An extended time period will be required for chloride concentration to build up to critical concentrations (if they ever do). Since the time required to adjust the sump solution pH to greater than 7.0 by dissolution of Sodium Tetraborate is much less than 8 hours, pH adjustment will occur well before chloride concentrations have built up to a critical level.

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

6C.5 CORROSION OF ALUMINUM ALLOYS

Corrosion testing has shown that aluminum alloys are not compatible with alkaline borate solution. The alloys generally corrode fairly rapidly at the post-accident condition temperatures with the liberation of hydrogen gas. A number of corrosion tests were conducted in the Westinghouse laboratories (ref. 1, 23) and at Oak Ridge National Laboratory facilities. A review of applicable corrosion data is given in Table 6C-4 and on Figure 6C-8.

For purposes of the resolution of Generic Letter 2004-02 in regards to chemical effects on sump strainers, the methodology provided in WCAP-16530-NP-A was used for the prediction of the postulated chemical compounds produced in precipitate form from the corrosion of Aluminum and other materials in containment subject to sump and spray fluid. The results from the WCAP were employed, in conjunction with scaled strainer tests, to quantify head losses to be considered for the strainers when calculation NPSH for the Recirculation and RHR Pumps during the recirculation phase of a LOCA.

6C.5.1 Aluminum Corrosion Products In Alkaline Solution

The corrosion of aluminum in alkaline solution expected following a design-basis accident, has been shown to proceed with the formation of aluminum hydroxide[12-14] and the aluminate ion, as well as with the production of hydrogen gas.

The design-basis accident conditions expected for the Indian Point Unit 2 plant include the establishment of an alkaline emergency core cooling solution having a total volume of liquid of 4.47×10^5 gal after actuation of the engineered safety features.

As mentioned above, aluminum is known to corrode in alkaline solution to give a precipitate of $Al(OH)_3$, which in turn can redissolve in an excess of alkali to form a complex aluminate. Van Horn[12] noted that the precipitation of $Al(OH)_3$ begins about pH 4 and is essentially complete at pH 7. A further increase in pH to about 9 causes dissolution of the hydroxide with the formation of the aluminate.

Therefore, it can be seen that the solubility of aluminum corrosion products is a function of the pH of the environment. Consistent with this, the corrosion of aluminum is also strongly dependent on the solution pH, because when the corrosion products are dissolved from the metal surface, corrosion of the base metal can proceed more freely.

Figure 6C-9 presents a plot of aluminum corrosion rate as a function of solution pH.[1] The corrosion rate of aluminum is seen to decrease by a factor of 21 (1/0.048) as the pH decreases from 9.3 to 8.3 and by a factor of 83 (1/0.012) as the pH decreases from 9.3 to 7.0.

Therefore, one must consider both corrosion and the dissolution of the corrosion products at specific reference conditions, since the two are directly related.

The corrosion reactions that are of interest in the design-basis accident condition here would include the reaction of aluminum in alkaline solution to form aluminum hydroxide, i.e.,

$$2\,Al + 6\,H_2O \Leftrightarrow 2\,Al\,(OH)_3 + 3\,H_2 \qquad (6C\text{-}1)$$

and dissolution of the hydroxide to form the aluminate, i.e.,

$$Al\,(OH)_3 \Leftrightarrow Al0_2^- + H^+ + H_2O \qquad (6C\text{-}2)$$

Knowledge of the solubility product of the aluminum hydroxide in an alkaline solution allows the determination the solubility expected for the hydroxide in the design-basis accident environment.

Deltombe and Pourbaix[15] have determined the solubility product of aluminum hydroxide. Using the value of 2.28×10^{-11} for K_{sp}, as reported by Deltombe and Pourbaix, the following calculation can be made.

The solubility of Al $(OH)_3$ is determined from Equation 6A-2

$$Al\,(OH)_3 \Leftrightarrow Al0_2^- + H^+ + H_2O \qquad (6C\text{-}3)$$

$$K_{SP} = (AlO_2^-)\ (H^+)$$

$$2.28 \times 10^{-11} = (AlO_2^-)\ (H^+)$$

at pH = 9.3

$$(AlO_2^-) = \frac{2.28 \times 10^{-11}}{5 \times 10^{-10}} = 4.6 \times 10^{-2} \text{ moles/liter} \qquad (6C\text{-}4)$$

Therefore, the solubility of $Al(OH)_3$ in a pH 9.3 solution at 25°C (77°F) is 4.6×10^{-2} moles per liter or 3.0×10^{-2} lb/gal. Expressed as aluminum, the solubility at these conditions is 1.05×10^{-2} lb/gal.

The solubility of the aluminum corrosion products in the post-accident environment is a function of both solution pH and temperature. Figure 6C-10 presents plots of the corrosion product solubility, expressed in terms of aluminum versus solution pH for temperatures of 77 and 150°F. The change in solubility with temperature is found using the relationship of the free energy of formation, temperature, and the solubility product.

With the data available from Figures 6C-9 and 6C-10 and a knowledge of the reference aluminum corrosion behavior for any specific plant, one can calculate the expected solubility limits for the corrosion reaction.

For Indian Point Unit 2, there are 4.47×10^5 gal of emergency core cooling solution after actuation of the safety features. The total amount of aluminum present in the Indian Point Unit 2 containment is given in Table 6C-3. Table 6C-5 shows the corrosion of aluminum with time for the original (NaOH additive) design basis pH 9.3 post-accident environment. CN-CRA-96-005 calculates a maximum pH of 7.61 with Trisodium Phosphate used for solution pH control and IP-CALC-07-00129 calculates a maximum pH of 7.6 with Sodium Tetraborate used for pH control.

Table 6C-6 presents a summary of the applicable solubility and corrosion parameters for various conditions. The table lists the applicable solubility products (K_{sp}) and solubilities at the various temperatures and solution pH together with the soluble aluminum limit for the Indian Point Unit 2 system at the specific conditions. The last values in the table given the aluminum solubility margin after 100 days of corrosion; that is, the soluble aluminum limit divided by the aluminum corroded. It can be seen that in all cases, including the very conservative low-temperature and low-pH conditions; the emergency core cooling solution is not expected to be saturated with aluminum corrosion products. Furthermore, within the expected design conditions for temperature and pH, the aluminum solubility margin ranges from approximately 20 to 106.

The preceding analysis is based on the original design basis with NaOH addition and a pH of 8.5 to 10.0 in the solution. Use of Trisodium Phosphate or Sodium Tetraborate reduces the long term pH to a minimum of 7.0 and a maximum of approximately 7.6 (Ref. 27 & 42). Figures 6C-9 and 6C-10 show significant (orders of magnitude) decreases in the corrosion rate and solubility of aluminum when the pH is reduced from the 9.3 range to the low 7 range. This is also shown in reference 23 (WCAP-8776) for Trisodium Phosphate. Thus the corrosion rate of aluminum and the production of hydrogen due to that corrosion can be expected to significantly decrease in the post accident environment due to replacement of the NaOH pH control additive by Trisodium Phosphate. Although corrosion of aluminum is greater for Sodium Tetraborate it is not excessive (Ref. 41).

It is concluded therefore, that the corrosion products of aluminum will be in the soluble form during the post-accident period considered and hence, there is no potential for deposition on flow orifices, spray nozzles, or other equipment.

6C.5.2 Behavior Of Circulating Aluminum Corrosion Products

The solubility of aluminum corrosion products has shown that for this plant, the entire inventory produced after 100 days of exposure to the post-design-basis accident condition would remain in solution. The review also indicates that the emergency core cooling solution is only approximately 17-percent saturated at 77°F and less than 1-percent saturated at 150°F.

However, it is of interest to review the experience of facilities that have operated with insoluble aluminum corrosion products and to relate their conditions with those expected in the post-accident environment.

The most significant experience available is that of Griess[16] who operated a recirculating test facility to measure the corrosion resistance of a variety of materials in alkaline sodium borate spray solution.

Tests were conducted on 1100, 3003, 5052, and 6061 aluminum alloys exposed at 100°C in pH 9.3 sodium borate solution (0.15M NaOH - 0.28M H_3BO_3). It was reported that even though the solution contained copious amounts of flocculent aluminum hydroxide, it had no effect on flow through the spray nozzle (0.093-in. orifice). The pH of the solution did not change because of the increase in the corrosion products.

Griess[16] in describing his observations regarding aluminum corrosion product deposition potential stated that:

1. No significant deposition was observed on the cooling coil installed in the solution.
2. No significant deposition was observed on the heated surfaces of the facility.
3. No significant deposition was observed on isothermal facility surfaces.

The amounts of aluminum corroded to the solution in the tests conducted by Griess at 55 and 100°C were approximately 4.0 and 18.6 g, respectively. The concentration of aluminum present in the recirculation stream, therefore, was approximately 0.2 and 1 g/liter, respectively. This value is a factor of about 5 above the aluminum concentration expected in the post-accident emergency core cooling solution at Indian Point Unit 2 in a pH 9.3 (NaOH additive) solution after 100 days, and many times that when the lower pH Trisodium Phosphate additive is considered. Although corrosion of aluminum is greater for Sodium Tetraborate it is not excessive (Ref. 41).

Hatcher and Rae[17] describe the appearance of turbidity in the NRU reactor and "purpose" that deposition of aluminum corrosion products may have occurred on heat exchanger surfaces, although they do not report any specific examination results. Moreover, Hatcher and Rae report no operations problems associated with the presence of aluminum corrosion product turbidity in the NRU reactor. The overall heat transfer coefficient for each NRU reactor heat exchanger was measured after 2 yr. of full-power operation on several occasions and within the limit of accuracy of the measurements, reported at approximately 5-percent, no change in the thermal resistance had been observed.

It is concluded, therefore, both from the work of Griess and Hatcher and Rae, that the deposition of aluminum corrosion products on heat exchangers surfaces will not be significant in the post-accident environments even for the circumstances of insoluble product formation.

6C.6 COMPATIBILITY OF PROTECTIVE COATINGS WITH POST-ACCIDENT ENVIRONMENT

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 presented in WCAP-7198[18], showed that several inorganic zincs, modified phenolics, and epoxy coatings are resistant to an environment of high temperature (320°F maximum test temperature) and alkaline sodium borate. Long-term tests included exposure to spray solution at 150°F to 175°F for 60 days, after initially being subjected to the conservative design-basis accident cycle shown in Figure 6C-3. The protective coatings, which were found to be resistant to the conditions, that is, exhibited no significant loss of adhesion to the substrate nor formation of deterioration products, comprise virtually all of the protective coatings recommended for use in the containment. Hence, the protective coatings will not add deleterious products to the core-cooling solution.

It should be pointed out that several test panels of the recommended types of protective coatings were exposed for two design-basis accident cycles and showed no deterioration or loss of adhesion with substrate.

In July 1973, Regulatory Guide 1.54, "Quality Assurance Requirements for Protective Coatings Applied to Water-Cooled Nuclear Power Plants", (35) was issued to describe an acceptable method for complying with the NRC's quality assurance requirements with regard to protective coatings applied to ferritic steels, stainless steel, zinc-coated (galvanized) steel, concrete, or masonry surfaces of water-cooled nuclear power plants. Revision 1 of Regulatory Guide 1.54 was issued in 2000. Plants whose design basis that includes a commitment to Regulatory Guide 1.54 require that protective coatings be qualified and capable of surviving a design basis accident without adversely affecting safety related structures, systems, or components needed to mitigate the accident. NRC Generic Letter 98-04 (Ref 36) identified the potential for degradation and failure of "qualified" protective coatings applied to exposed surfaces within nuclear power plant containments. The NRC sponsored work at the Savannah River Technology Center to investigate the performance and potential for failure of Service Level 1 coating systems used in nuclear power plant containments (37). The Nuclear Energy Institute has issued "Condition Assessment Guideline: Debris Sources Inside PWR Containments" (38) that provides guidance to PWR operators in assessment of condition of coatings (among other items).

6C.7 EVALUATION OF THE COMPATIBILITY OF CONCRETE AND EMERGENCY CORE COOLING SOLUTION IN THE POST ACCIDENT ENVIRONMENT

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

The purpose of this study was to establish:

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

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

For these tests the total surface area of concrete in the design containment, which may be exposed to the emergency core cooling solution following a design-basis accident was estimated at 6.3 x 10^4-ft^2. This value includes both coated and uncoated surfaces. The emergency core cooling solution volume for a reference plant was considered at approximately 313,000 gal and the surface to volume ratio from these values is approximately 29 in.^{2}/gal. The surface to volume ratios for the concrete-boron tests used were between 28 and 78 in.^{2}/gal of solution. Table 6C-7 presents a summary of the data obtained from the concrete-boron test series.

Testing of uncoated concrete specimens in the post-accident environment showed that attack by both boric acid and the alkaline boric solution is negligible and the amount of deterioration product formation is insignificant. Other specimens covered with modified phenolic and epoxy protective coatings showed no deterioration product formation. These observations are in agreement with Orchard[19] who lists the following resistances of Portland cement concrete to attack by various compounds:

1. Boric acid, little or no attack.
2. Alkali hydroxide solution under 10-percent, little or no attack.
3. Sodium borate, mild attack.
4. Sodium hydroxide over 10-percent, very little attack.

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

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

Griess and Bacarella (ref. 21) report on tests of concrete cylinders poured from the actual construction mix at Browns Ferry Nuclear Plant in a simulated post accident environment. None of the cylinders underwent any visible changes and in all cases the strength of the concrete after exposure exceeded the design strength. Solutions in these tests picked up about 10 ppm chloride (leached from the concrete) and the boron concentration was reduced by about 10%. However, the ratio of surface area to solution volume was very high (240 in^2/gal), about 10 times higher than the ratio calculated above, and the results are therefore substantially greater than would be expected in the actual containment.

6C.8 MISCELLANEOUS MATERIALS OF CONSTRUCTION

6C.8.1 Sealants

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

Table 6C-8 presents a summary of the sealant materials tested, together with a description of the panel's appearance after testing. Three generic type of sealants were tested: butyl rubber, silicone, and polyurethane. Each of the materials was the "one package" type, that is, no mixing of components was necessary prior to application. The materials were applied on stainless steel allowed to cure well prior to testing.

The test results showed that the silicone sealants tested were chemically resistant to the design-basis accident environment and are acceptable for use in containment. Sealant 780 by Dow Corning Corporation would be acceptable for use in the containment. Major applications of this sealant could be as concrete expansion joint sealant on the liner insulation panels. Sealant 780 will contribute no deterioration products to the emergency core cooling solution

during the post design-basis accident period and will maintain its structural integrity and elastic properties.

6C.8.2 Polyvinyl Chloride Protective Coating

Tests were conducted to determine the stability of the polyvinyl chloride (PVC) protective coating, of the type, which might be used on conduit in the design-basis accident environment. Samples of the polyvinylchloride exposed to alkaline sodium borate solutions at design-basis accident conditions showed no loss in structural rigidity and no change in weight or appearance.

A sample of polyvinylchloride coated aluminum conduit (1-in. OD x 8-in. long) was irradiated by means of a Co-60 source, at an average dose rate of 3.2 x 10^6 rads/hr, to a total accumulated dose of 9.1 x 10^7 rads. The specimen was immersed in alkaline sodium borate solution (pH 10, 3000 ppm boron) at 70°F. Visual examination of the coating after the test showed no evidence of cracking, blistering, or peeling; and the specimen appeared completely unaffected by the gamma exposure. Chemical analysis of the test solution indicated that some bond breakage had occurred in the polyvinyl-chloride coating as evidenced by increase in the chloride concentration. The gamma exposure of approximately 10^8 rad resulted in a release to the solution of 26 mg of chloride per square foot of exposed polyvinylchloride surface. Considering a total surface area of polyvinylchloride coating present in containment (approximately 500-ft^2) and the emergency core cooling solution volume of 313,000 gal, the chloride concentration increase in the emergency core cooling solution due to irradiation of the coating, would be approximately 0.01 ppm.

Therefore, it is concluded that polyvinylchloride protective coating will be stable in the design basis accident environment.

6C.8.3 Fan Cooler Materials

Samples of the following air-handling system materials were exposed in an autoclave facility to the design-basis accident temperature-pressure cycle:

1. Moisture separator pad.
2. High efficiency particulate media.
3. Asbestos separator pads.
4. Adhesive for joining separator pads and high-efficiency particulate air filter media corners.
5. Neoprene gasketing material.

The materials were exposed in both the steam phase and liquid phase of a solution of Sodium Tetraborate (15ppm boron) to simulate the concentrations expected down stream of the fan-cooler coils. Examination of the specimens after exposure showed the following:

1. Moisture separator pads were somewhat bleached in color but maintained their structural form and showed good resiliency as removed in both liquid and steam phase exposure.

2. High-efficiency particulate filter media maintained its structural integrity in both the liquid and steam phase. No apparent change.

3. Asbestos separator pads showed some slight color bleaching. How ever, both steam and liquid phase samples maintained their structural integrity with no significant loss in rigidity.

4. Adhesive material for the HEPA/separator pad edges showed no deterioration or embrittlement and maintained its adhesive property.

5. Neoprene gasketing material is also satisfactory in both the steam and liquid phase. The material showed only weight gain and a shrinkage of 15 to 30-percent based on a superficial, one flat side area. The gasket thickness decreased about 10-percent. The gasket material was unrestrained during the exposure, and hence the dimensional changes experienced are greater than those which would result as installed in the air-handling side of the fan coolers.

6C.8.4 Polyvinyl Chloride Insulation

The containment liner is insulated with a polyvinyl chloride (PVC) insulation enclosed in a stainless steel jacket. Table 14.3-40 lists 6940 ft^2 of 1.25 inch thick PVC insulation enclosed with 0.019 inch jacket and 7434 ft^2 of 1.5 inch thick PVC insulation enclosed with a 0.025 inch jacket. This totals about 1652 ft^3 of PVC installation for this application. Approximately 25 ft^2 of the 1.25 inch thick insulation was replaced with fiberglass insulation. Materials with high chloride content (like PVC) are subject to thermal and/or radiological degradation and are not normally left in containment when a plant returns to power. This insulation material is enclosed in stainless steel jackets which if intact and containing no penetrations should keep the spray solution from contacting the insulation or its degradation products. However, if access paths in the jackets allow sump solution or spray to come in contact with the insulation material, it might be a large source of chlorides.

PVC degradation releases HCL which dissolves in water to form hydrochloric acid. Properties of PVC have been reported to change at ~1.9 x 10^7 rads (ref 39) and classified in reference 40 as having excellent resistance to radiation (noticeable changes in properties occur above 10^7 rads). Reference 40 also indicates that polymers are sensitive only to total radiation dose and not to dose rate so that the dose accumulated to the accident must also be considered. That reference also lists a continuous high temperature limit of 75 °C (167°F) for PVC. Degradation of PVC has caused failure of a number of components at nuclear power plants as reported in various NRC reports.

6C.9 ENVIRONMENTAL REQUALIFICATIONS

Qualification of electrical equipment for harsh environment is discussed in Section 7.1. The impact of changing the post accident spray solution pH control chemical from sodium hydroxide injected from the spray additive tank to Trisodium Phosphate allowed to dissolve in the solution collected in the sump on Westinghouse supplied Class 1E equipment was evaluated and documented in WCAP-14495 (ref. 25). Some equipment installed at IP2 was of an older vintage, not qualified in accordance with WCAP-8587 (ref 33), or was provided and/or qualified by a vendor other than Westinghouse and was therefore outside of the scope of the WCAP-14495. A number of additional non-metallic materials, not considered here, contained in the Class 1E electrical equipment are also considered in that evaluation. That evaluation concluded that “spray additive tank elimination will not affect the qualification of Westinghouse

supplied Class 1E equipment located inside containment at Indian Point Nuclear Generating Station Unit No. 2 (IP2) However, the Notes at the bottom of Table 2.0-1 states that "Evaluation of this equipment is representative of equipment installed at IP2. Additional evaluation will be necessary to affirm applicability/completeness for equipment installed at IP2".

An evaluation (Reference 43) was performed to determine the impact of the change in post-LOCA buffered sump chemistry on the existing EQ equipment qualified using Trisodium Phosphate. The evaluation concluded that due to the similarities between post-LOCA Trisodium phosphate and Sodium Tetraborate buffered sump solutions; the equipment qualified for Trisodium Phosphate remains qualified using the new Sodium Tetraborate buffered solution. Therefore, there would be no impact on existing IP2 EQ equipment as a result of the subject post-LOCA buffered sump chemistry change.

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TABLE 6C-1 (Sheet 1 of 2)
Review of Sources of Various Elements in Containment
and Their Effects on Materials of Construction

Group	Representative Elements	Corrosivity of Elements	Sources of Elements
0	He, Ne, Kr, Xe	No effect on any materials construction	Fission product release
I a	Li, Na, K	Generally corrosion inhibitive properties for steels and copper alloys - harmful to aluminum	Li - coolant pH-adjusting agent Na - spray additive solution, concrete leach product K - concrete leach product
II a	Mg, Ca, Sr, Ba	Generally not harmful to steel or copper base alloys	Concrete leach products - deteriorated insulation
III a	Y, La, Ac	Not considered harmful in low concentrations	Fission product release
IV a	Ti, Zr, Hf	Not considered harmful to any materials	Fuel rod cladding, control rod material, alloying constituent
V a	V, Nb, Ta	Not considered harmful to any materials	Alloying constituents in low concentration
VI a	Cr, Mo, W	Not considered harmful to any materials	Alloying constituents in equipment
VII a	Mn, Tc, Re	Not considered harmful	Mn - alloy constituent

TABLE 6C-1 (Sheet 2 of 2)
Review of Sources of Various Elements in Containment
and Their Effects on Materials of Construction

Group	Representative Elements	Corrosivity of Elements	Sources of Elements
VIII	Fe, Ni, Cr, Os	Fe, Ni, Cr - not harmful to any materials	Fe, Ni, Cr - alloying constituents. Others have no identifiable sources
I b	Cu, Ag, Au	Not harmful to any materials	Cu present as material of construction and alloying constituent
II b	Zn, Cd, Hg	Hg - harmful to stainless steel, Cu alloys, aluminum Zn - unknown Cd - unknown	Hg has been entirely excluded from use in the containment; Cd finish plating on components. Zn galvanizing and alloying constituent
III b	B, Al, Ga, In	Not harmful to material	B - neutron poison additive Al - materials of construction
IV b	C, Si, Sn, Pb	C, Si, Sn not harmful to materials Pb considered harmful to nickel alloys	Si - concrete leach product Pb - alloy constituent in some brazes
V b	N, P, As, Sb, Bi	No effect from N unless ammonia is formed; others unknown	N - containment air;Others not identified in significant materials
VI b	O, S, Se, Te	S possibly harmful to nickel alloys	Te - fission product S - oils, greases, insulating materials
VII b	F, Cl, Br, I	F considered potentially harmful to Zircaloy Cl potentially harmful to stainless steel Br and I not generally harmful	Cl - concrete leach product, general contamination F - organic materials I and Br - fission products, low concentration

TABLE 6C-2
Materials of Construction in Reactor Containment

Material	Equipment Application
300 series stainless steel	Reactor coolant system, residual heat removal loop, spray system
400 series stainless steel	Valve materials
Inconel (600, 718)	Steam generator tubing, reactor vessel nozzles, core supports, and fuel rod grids
Galvanized steel	Ventilation duct work, CRDM shroud material, I and C conduit
Aluminum	Nuclear detectors, I and C equipment, CRDM connectors, paints
Copper	Service water piping, fan cooler material
70-30 Cu Ni	Fan cooler material
90-10 Cu Ni	Fan cooler material
Carbon steel	Component cooling loop, structural steel, main steam piping, etc.
Monel	Possibly instrument housings
Brass	Possibly instrument housings
Polyvinylchloride	Conduit sheathing, electrical insulation, containment liner insulation
Protective coatings	General use on carbon steel structures and equipment, concrete
Inorganic zincs Epoxy Modified phenolics	
Silicones - neoprene	Ventilation duct work gasketing, sealants

TABLE 6C-3
Inventory of Aluminum in Containment

	Item	Mass (lb)	Surface Area (ft^2)
1.	CRDM connectors	122	42
2.	Reactor vessel insulation foil	269	Very high
3.	Area monitors	6	4
4.	Source, intermediate, and power range detectors	140	40
5.	Process instrumentation and control	420	84
6.	Lighting fixtures and equipment	1061	380
7.	Paint on steam generator, pressurizer and reactor vessel	140	Very high
8.	Contingency	250	85

TABLE 6C-4
Corrosion of Aluminum Alloys in Alkaline Sodium Borate Solution

Data Point	Temperature (°F)	Alloy Type	Test Duration	Corrosion Rate mg/dm2/hr	pH	Exposure Condition	Reference
1	275	5053	3 hr	96.2	9	Solution	WCAP-7153, Table 9
2	275	5005	3 hr	840	9	Solution	WCAP-7153, Table 9
3	200	6061	320 hr	15.4	9.3	Solution	WCAP-7153, Table 8 WCAP-7153, Figure 9
4	210	5052	7 days	53.0	9	Solution	WCAP-7153, Table 7 WCAP-7153, Figure 8
5	210	5052	2 days	14.0	9	Solution	WCAP-7153, Table 5
6	210	5005	2 days	27.1	9	Solution	WCAP-7153, Table 5
7	284	5052	1 day	54	9.3	Spray	ORNL-TM-2425 Table 3.13
8	284	5052	1 day	31.5	9.3	Solution	ORNL-TM-2425 Table 3.13
9	212	6061	3 days	126	9.3	Spray	ORNL-TM-2368 Table 3.6
10	212	6061	3 days	110	9.3	Solution	ORNL-TM-2368, Table 3.6
11	150	6061	7 days	2.9	9.3	Solution	Westinghouse data
12	150	5052	7 days	4.2	9.3	Solution	Westinghouse data

TABLE 6C-5
Corrosion Products of Aluminum Following Design Basis Accident, Indian Point Unit 2

Time After Reactor Trip (days)	Mass of Aluminum Corroded (lb x 10^{-2})	Hydrogen Produced (SCF x 10^{-3})	Mass of Al $(OH)_3$ Formed (lb x 10^{-2})
1	1.71	3.41	4.94
5	4.31	8.60	12.4
10	4.50	8.98	13.0
20	4.88	9.75	14.1
30	5.26	10.5	15.2
40	5.66	11.3	16.4
50	6.06	12.1	17.5
60	6.41	12.8	18.5
70	6.81	13.6	19.7
80	7.21	14.4	20.9
90	7.61	15.2	22.0
100	7.97	15.9	23.0

TABLE 6C-6
Summary of Unit 2 Aluminum Corrosion Product Solubility Data

	Solution Temperature			
	77°F		150°F	
Parameter	pH 9.3	pH 8.3	pH 9.3	pH 8.3
Solubility product K_{sp}	2.28×10^{-11}	2.28×10^{-11}	4.16×10^{-10}	4.16×10^{-10}
Aluminum solubility, lb Al/gal	1.05×10^{-2}	1.05×10^{-3}	1.9×10^{-1}	1.9×10^{-2}
Soluble aluminum Limit[1] for emergency core cooling system, lb	4.69×10^{3}	4.69×10^{2}	8.49×10^{4}	8.49×10^{3}
Aluminum corrosion rate (normalized)	(Not used)	(Not used)	1	0.048
Aluminum corroded after 100 days, lb	(Not used)	(Not used)	795	438[2]
Aluminum solubility margin at 100 days	5.9[3]	1.1[3]	106	19

1. Indian Point Unit 2 solution volume 4.48×10^{5} gal.

2. Value assumes rapid corrosion of all aluminum paint and reactor vessel foil insulation.

3. Note corrosion rate at 150°F was used for "aluminum corroded" value; hence, value is very conservative.

TABLE 6C-7
Concrete Specimen Test Data

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

1. These specimens coated with Phenoline 305. All others were uncoated.

2. These tests were at high-temperature design-basis accident transient conditions. All others at 195°F - 205°F.

TABLE 6C-8
Evaluation of Sealant Materials For Use In Containment

Sealant Type	Manufacturer	Posttest Appearance
Butyl rubber	A	Unchanged, flexible.
Silicone	B	Unchanged, flexible.
Silicone	B	Unchanged, flexible
Polyurethane	C	Sealant bubbled and became very soft; solution permeated into bubbles.
Polyurethane	C	Sealant swelled and became soft; solution permeated into material.
Polyurethane	C	Sealant swelled and became very soft and tacky; solution permeated into material.

6C FIGURES

APPENDIX 6D
SPRAY SYSTEM MATERIALS COMPATIBILITY FOR LONG-TERM STORAGE OF SODIUM HYDROXIDE

(RETAINED FOR HISTROICAL PURPOSES)

SPRAY SYSTEM MATERIALS-COMPATIBILITY FOR LONG-TERM STORAGE OF SODIUM HYDROXIDE

TABLE OF CONTENTS

LIST OF TABLES

LIST OF FIGURES

Appendix 6D

SPRAY MATERIALS COMPATIBILITY FOR LONG-TERM STORAGE OF SODIUM HYDROXIDE

[Historical Information] A materials compatibility review for the spray additive tank and associated equipment during long-term storage of sodium hydroxide is presented below. The exposure conditions are shown in Table 6D-1, and the materials for the various components are shown in Table 6D-2. The corrosion rates for the various materials at or near the long-term exposure conditions with air contamination are shown on Table 6D-3. The immunity of most of the materials in Table 6-2 to caustic cracking at the exposure conditions listed in Table 6D-1 has been reported by Logan[1] (see Figure 6D-1). No caustic cracking of 17-4 pH or Stellite has been reported.[2]

The effect of carbon dioxide from air exposure on the corrosion of iron is shown in Figure 6D-2.[3] At pH 14, no additional corrosion is observed over that observed in carbon-dioxide-free solution. In the Indian Point Unit 2 system, a nitrogen blanket is continuously maintained over the sodium hydroxide solution in the spray additive tank, thus essentially eliminating any carbon dioxide contamination of the solution.

The Nordel[*] rubber diaphragm material used in the tank valves was exposed in 33 wt percent sodium hydroxide solution at 110°F for 6 months and found to be unaffected by the simulated spray additive tank solution. The integrity of the structural materials in the spray additive tank system would not be adversely affected even using the corrosion rate presented in Table 6D-3 where air contamination is present. In the Indian Point Unit 2 system, where nitrogen blanketing of the spray additive tank would prevent air contamination, the corrosion rates would be even lower with even less effect on the material integrity.

Diamond Shamrock Company[4] reported that no galling of steel valves occurred after exposure to 50-percent sodium hydroxide at 120° to 140°F for greater than 3 years. Stainless steel valves, exhibiting lower corrosion rates, would have an even lower propensity toward galling than steel. Therefore, no galling should occur on the valves exposed to the long-term storage conditions. The total corrosion product released to the spray additive tank as oxide would be less than 1000 g/yr with aerated solution and would be much less with the air-free solution (i.e., the Indian Point Unit 2 solution). This small quantity of corrosion product should not present any problems with clogging of delivery lines.

No sodium hydroxide precipitation will occur for a 30 wt percent solution if the temperature of the tank and liners is maintained above 35°F. Since this system is located in an area of the auxiliary building, which is continuously heated to maintain a 50°F minimum temperature, no solid sodium hydroxide would be present and, therefore, no clogging of the lines could occur.

[*] Nordel is a product of E.I. Du Pont de Nemours and Company.

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TABLE 6D-1
Exposure Conditions

Temperature, ^{o}F	110
Nitrogen overpressure	Slight positive pressure
Sodium hydroxide concentration, w/o	30
Oxygen concentration, normal	Nitrogen blanketed
Carbon dioxide concentration, normal	Nitrogen blanketed

TABLE 6D-2
Component Materials

Spray additive tank	304 stainless steel cladding on steel A-516 GR-70
Piping	304 stainless steel
Valve bodies	304 and 316 stainless steel
Valve seats	Austenitic stainless steel or Stellite
Valve stems	17-4 pH and 410 stainless steel
Valve diaphragm	Ethylene-propylene dipolymer (Nordel rubber by DuPont)

TABLE 6D-3
Corrosion Rates

Material	Temperature (°F)	NaOH Concentration (ppm)	Aeration	Corrosion Rates (mils/yr)	Reference No.
304 S/S	136	22 to 50	Yes	<0.1	1
316 S/S	125	30	Yes	<2	2
Steel	179	30 to 50	Yes	<20	2
410 S/S	125	30	Yes	<2	2
17-4 pH	176	30	Yes	3 to 6	7
Stellite	150	50	Yes	<0.6	4
Nordel rubber	110	33	Yes	<0.004	5

6D FIGURES