SECTION 6

ENGINEERED SAFETY FEATURES

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SECTION 6

ENGINEERED SAFETY FEATURES

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

1. Design of the reactor core in conjunction with the Reactor Control and Protection Systems to preclude release of fission products from the fuel (Sections 4 and 7).

2. Retention of fission products in the reactor coolant for whatever leakage occurs (Sections 5 and 6).

3. Retention of fission products by the containment for operational and accidental releases beyond the reactor coolant boundary (Section 3.8 and Section 6).

4. Limiting fission product dispersal to minimize population exposure for an accidental release beyond the containment (Sections 2, 12, and 15).

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

The engineered safety features in this station are:

1. The steel-lined, reinforced concrete containment, concrete cylindrical wall, and reinforced concrete base and dome. These form a virtually leak-tight barrier to the escape of fission products should a loss-of-coolant accident (LOCA) occur - detailed in Section 3.8.

2. The Emergency Core Cooling System (ECCS), which provides borated water to cool the core in the event of an accidental depressurization of the Reactor Coolant System (RCS). The combination of the control rods and the boron in the injected water provides the necessary control of reactivity required - detailed in Section 6.3.

3. The Containment Spray System which is used to reduce containment pressure and remove iodine from the containment atmosphere - detailed in Section 6.2.

4. The Containment Fan Cooling System is used to recirculate and cool the containment atmosphere in the event of a LOCA - detailed in Section 6.2.

Evaluations of techniques and equipment used to accomplish the central objectives including accident cases are detailed in Sections 3, 5, 6, and 15.

The design philosophy with respect to active components in the Engineered Safety Systems is to provide duplicate equipment so that maintenance is possible during operation without impairment of the safety function of the systems. Routine servicing and maintenance of equipment of this type would generally be scheduled for periods of refueling and maintenance outages.

Conditions on continued reactor operation during such outages that are provided in the Technical Specifications will conform to reasonable experienced judgment and industry practice and will be shown to ensure safe operation.

6.1 CRITERIA

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

Those criteria which are specific to one of the engineered safety features are discussed in the description of that system.

6.1.1 Engineered Safety Features Criteria

The criteria applying to all engineered safety features are given below.

6.1.1.1 Engineered Safety Features Basis for Design

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 pipe break up to and including the circumferential rupture of reactor coolant pipe assuming unobstructed discharge from both ends, and to cope with any steam or feedwater line break.

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 concrete reactor containment with liner weld channels and high integrity piping penetrations utilizing partial penetration seal welds between the containment penetration sleeve/seal plate and the process piping to form a virtually leak-tight barrier preventing the escape of fission products should a LOCA occur.

b. Isolation of process lines by the Containment Isolation System which imposes double barriers for each line which penetrates the containment.

2. Reducing the fission product concentration in the containment atmosphere. This is accomplished by chemically treated spray which removes elemental iodine vapor from the containment atmosphere by washing action, and by recirculation of containment atmosphere through high-efficiency particulate air filter units.

3. Reducing the containment pressure and thereby limiting the driving potential for fission product leakage. This is accomplished by the Containment Spray System which cools the containment atmosphere or by recirculation of containment atmosphere through fan cooler units.

6.1.1.2 Reliability and Testability of Engineered Safety Features

A comprehensive program of plant testing is formulated for all equipment systems and system control vital to the functioning of engineered safety features. The program consists of performance tests of individual pieces of equipment in the manufacturer's shop, integrated tests of the system as a whole, and periodic tests of the activation circuitry and mechanical components to assure reliable performance, upon demand, throughout the station lifetime.

The initial tests of individual components and the integrated test of the system as a whole complement each other to assure 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 intended. In the event that one of the redundant components should require maintenance as a result of failure to perform during the test according to prescribed limits, the necessary corrections or minor maintenance are made and the unit retested immediately. Satisfactory performance of the remaining redundant component(s) is proof of the availability of that safety feature, and it is not necessary to adjust station load during the brief period that a safety feature component may be out of service.

6.1.1.3 Protection Against Dynamic Effects and Missiles

A LOCA or other plant equipment failure might result in dynamic effects or missiles. For such engineered safety features as are required to assure safety in the event of such an accident or equipment failure, protection from these dynamic effects or missiles is considered in the layout of plant equipment and missile barriers. Fluid and mechanical driving forces are calculated, and consideration is given to the possibility of damage due to fluid jets and missiles which might be produced by the action of such jets. Consideration is given during the design of the station to the following sources of missiles: instrument thimbles including installed sensors, bolts, and complete control rod drive shafts and/or mechanisms (refer to Sections 3 and 5).

Layout and structural design specifically protect safety injection lines to unbroken reactor coolant loops against damage as a result of the maximum reactor coolant pipe rupture. Injection lines penetrate the main missile barrier and the injection headers are located in the missile-protected area between the main missile barrier and the containment wall. Individual injection lines, connected to the injection header, pass through the barrier and then connect to the loops. Separation of the individual injection lines is provided to the maximum extent practicable. 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.

In addition, missile protection is provided for engineered safety features located outside the containment. The containment structure is capable of withstanding the effects of missiles originating outside the containment and might be directed toward it so that no LOCA can result. The control room enclosure is also capable of withstanding such credible missiles as may be directed toward it, assuring capability to maintain control of the station. Consideration is also given to the layout of other equipment outside the containment which is required to place the station in a safe shutdown condition and maintain it in that condition until repairs can be effected.

Missile protection will be afforded by:

1. Judicious location of piping and equipment otherwise subject to possible damage, behind existing wall or other barriers with appropriate credit for spatial separation of redundant components

2. Local shielding to stop potential missiles at their source

3. Addition of missile barriers to protect vulnerable piping and equipment

All hangers, stops, and anchors are designed in accordance with ANSI B31.1, Code for Pressure Piping, and ACI 318, Building Code Requirements for Reinforced Concrete, which provide minimum requirements for material, design, and fabrication with ample safety margins for both dead and dynamic loads over the life of the equipment.

6.1.1.4 Engineered Safety Features Performance Capability

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

During the recirculation phase the ECCS is tolerant of one active or one passive failure, but not in addition to a single failure in the injection phase. One active or passive failure in the systems required for long-term ECCS operation will not prevent the accomplishment of the ECCS objectives nor cause the total offsite dose to exceed 10CFR50.67 limits, with credit for detection and operator action.

In the particular case of an ECCS pump being out for maintenance, an additional active or passive failure is not considered. The maximum period that operation would be continued with one pump out for maintenance is specified in the Technical Specifications.

The extreme upper limit of public exposure is taken as the levels and time periods presently outlined in 10CFR50.67, i.e., 25 rem TEDE maximum in a 2 hour period at the exclusion radius, and 25 rem TEDE over the duration of the accident at the low population zone distance. The accident condition considered is the hypothetical case of a release of fission products as in Regulatory Guide 1.183. Also, the loss of outside power is assumed concurrently with this accident.

Under the above accident conditions, the Containment Spray and Fan Cooling Systems are designed and sized so that, operating with partial effectiveness, it can supply the necessary post-accident cooling capacity to assure the maintenance of containment integrity; that is, keep 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 operation of a system with at least one active component failure.

The fan cooling system's performance capability for defense against thermally induced overpressure, the development of two-phase flow regions, and column separation or voiding leading to the possibility of waterhammer events are analyzed for as part of the NRC's Generic Letter 96-06 modifications.

The ECCS and related pumps which must operate following the design basis accident include the residual heat removal, safety injection, containment spray, centrifugal charging, component cooling water, and service water pumps.

Minimum available Net Positive Suction Head (NPSH) to the safety injection, centrifugal charging, and containment spray pumps occurs when all are taking suction from the refueling water storage tank (RWST) during the injection operation immediately following the design basis accident.

Since maximum required NPSH and minimum available NPSH occur at the runout flow for the pumps, this flow was assumed for calculation purposes. The temperature of the RWST water varies between 40°F and 100°F.

Available NPSH at runout flow to these pumps at both the high and low temperatures was calculated. Suction line friction losses are higher at 40°F, but the higher vapor pressure of 100°F water leaves less available NPSH to the pumps. Friction losses were calculated using the conservative pipe and fitting resistances given in the Crane Co. Technical Paper Number 410.

The residual heat removal pumps take suction during the post-accident recirculation phase from the containment sump. The water is at a higher temperature than during injection, but the elevated containment pressure following a design basis accident somewhat offsets the higher vapor pressure of the water; however, no credit is taken for this. In addition, the piping to the pump suctions is quite direct; hence, friction losses are small.

Service water pumps are vertical turbine pumps taking suction directly at the plant intake. Suction location is 44 inches below low-low water elevation (Elevation 76 feet), temperature 90°F. The component cooling water pumps have suction head tanks which maintain pressure in the closed system equal to the maximum elevation of the system piping.

6.1.1.5 Engineered Safety Features Components' Capability

Active components of the ECCS and the Containment Spray System are located outside the containment and not subject to containment accident conditions.

6.1.1.6 Accident Aggravation Prevention

The reactor is maintained subcritical following a pipe rupture accident. Introduction of borated cooling water into the core does not result in a net positive reactivity addition. The control rods insert and remain inserted.

The supply of water by the ECCS to cool the core cladding does not produce significant water-metal reactions. The delivery of cold emergency core cooling water to the reactor vessel following accidental expulsion of reactor coolant does not cause further loss of integrity of the RCS boundary. Accumulator actuation, including possible nitrogen addition, is evaluated in Section 15 and is shown not to aggravate any LOCA.

Instrumentation, motors, cables, and penetrations located inside the containment which are required to function 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.

The ECCS pipes serving each loop are restrained at the missile barrier in each loop area to restrict potential accident damage to the portion of piping beyond this point. The anchorage is designed to withstand, without failure, the thrust force of any branch line severed from the reactor coolant pipe and discharging fluid to the atmosphere, and to withstand a bending moment equivalent to that which produces failure of the piping under the action of free end discharge to atmosphere or motion of the broken reactor coolant pipe to which the emergency core cooling pipes are connected. This prevents possible failure at any point upstream from the support point including the branch line connection into the piping header.

6.1.1.7 Sharing of Systems

For all shared systems and/or components, analyses confirm that there is no interference with basic function and operability of these systems due to sharing, and hence no undue risk to the health and safety of the public results.

The residual heat removal pumps and heat exchangers serve dual functions. Although the normal duty of the residual heat removal exchangers and residual heat removal pumps is performed during periods of reactor shutdown, during all station operating periods this equipment is aligned to perform the low head injection function of emergency core cooling. During the recirculation phase of the accident, the residual heat removal pumps take suction from the containment sump. Each pump has a separate suction line. Operational testing of the system, performed during each refueling period before station startup, provides assurance of correct system alignment for the safety function of the components.

During the injection phase, the safety injection and centrifugal charging pumps do not depend on any portion of other ECCSs. During the recirculation phase, if RCS pressure stays high due to a small break accident, suction to the high head and centrifugal charging safety injection pumps is provided by the residual heat removal pumps.

The ability of the above systems to perform their dual function is discussed in Sections 6.2 and 6.3 and in Sections 5 and 15.

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. Therefore, they are discussed in other sections, as listed.

Name	Discussion
Quality Standards	Section 17
Performance Standards	Section 3.1
Records Requirements	Section 17
Instrumentation and Control Systems	Section 7
Engineered Safety Features Protection System	Section 7
Emergency Power	Section 8
Seismic Design Criteria	Section 3.7

6.2 CONTAINMENT SYSTEMS

6.2.1 Containment Functional Design

6.2.1.1 Design Basis

The reactor containment completely encloses the entire Reactor Coolant System (RCS) and ensures that post-accident leakage is limited to a safe rate of 0.1 percent of the containment free volume per day at the design pressure of 47 psig. A steel liner and leak-tight penetrations are provided to ensure that the leakage limits are not exceeded. The structure provides biological shielding for both normal and accident situations.

The reactor containment is designed to safely withstand the loading combinations described in Section 3.8.

Containment and associated systems are designed, fabricated, and erected to quality and performance standards with appropriate testing and inspection requirements. Records of design, fabrication, construction, and testing of the containment are maintained throughout the life of the plant.

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

The design pressure and temperature of the containment is equal to or greater than the peak pressure and temperature occurring as the result of the complete blowdown of the reactor coolant through any rupture of the RCS up to and including the complete severance of a reactor coolant pipe. Energy contribution from the steam generators is included in the calculation of the containment pressure transient due to the reverse heat transfer through the steam generator tubes. The RCS supports are designed to withstand the blowdown forces associated with the sudden severance of the reactor coolant piping but not steam piping since the coincidental rupture of the steam system is not credible. In addition, containment design pressure is not exceeded during any subsequent long-term pressure transient determined by the combined effects of heat sources such as residual heat and limited metal-water reactions, structural heat sinks and the operation of the engineered safeguards utilizing only the emergency onsite electric power supply.

In a design basis accident (DBA), reactor coolant is released through a double-ended break of the largest reactor coolant pipe, causing a rapid pressure rise in the containment. The reactor coolant pipe used in the accident is the 29-inch inside diameter section because rupture of the 31-inch inside diameter section requires that the blowdown go through both the 29-inch and the 27 1/2 inch inside diameter pipes and would, therefore, result in a less severe transient.

Additional energy release was considered from the following sources:

1. Stored heat in the reactor core.

2. Stored heat in the reactor vessel piping and other RCS components.

3. Residual heat production.

4. Limited metal-water reaction energy and resulting hydrogen-oxygen reaction energy.

Details of mass and energy releases are provided in Section 15.4.8.

The containment is also designed to withstand credible external pressures. In the event of inadvertent spray actuation, the containment would depressurize until the temperature of the atmosphere was approximately the temperature of the spray. A bounding calculation was performed to determine the maximum outside to inside pressure differential. The following initial conditions were assumed:

1. The containment is initially at 120°F which maximizes the temperature differential between the containment atmosphere and the spray, which is at a temperature of 40°F.

2. The containment pressure is 14.7 psia.

3. The relative humidity is at a maximum value of 100 percent.

As the air temperature is reduced from 120 to 40°F, the partial pressure of the air decreases from 12.91 to 11.13 psi. The steam partial pressure decreases from 1.6927 to 0.12163 psi. Thus, a containment equilibrium pressure of 11.25 psia is produced. This causes a differential pressure (d/p) of 3.45 psi across the containment shell, with no credit taken for the operation of the containment Pressure-Vacuum Relief System. In the long-term, the Pressure-Vacuum Relief System will be operated to return the containment pressure to normal.

The d/p between the design and maximum calculated negative pressure is 0.05 psi. This margin is adequate due to the conservatism used in the external pressure analysis.

The containment design provides limited access through personnel hatches with the reactor at power. This type of access is intended primarily for inspection and maintenance of the air recirculation equipment, incore ion chamber drives, seal table, operating deck, and reactor coolant drain tank. Opening of the containment equipment hatch or both doors in the personnel locks is limited by the Technical Specifications.

After shutdown, the containment is purged to reduce the concentration of radioactive gases and airborne particulates. A purge system is provided to reduce the radioactivity level to an exposure of less than 40 Derived Air Concentration-Hours (DAC-Hours), as defined by 10 CFR 20, in a 40-hour occupational work week, within 2 hours after plant shutdown, based on 1-percent fuel defects. To assure removal of particulate matter, the purge air is passed through a high-efficiency filter before being released to the atmosphere through the plant vent.

The primary reactor shield is designed so that access to the primary equipment is limited by the activity of the primary system equipment and not the reactor.

6.2.1.2 Containment Structural Acceptance Test

6.2.1.2.1 General Description

The completed containment structure was tested for structural integrity by subjecting the structure to an air pressure test of 54 psig, which is equivalent to 115 percent of the design pressure. The basic requirements of Regulatory Guide 1.18, "Structural Acceptance Test for Concrete Primary Reactor Containments," were satisfied in the performance of the test.

Containment pressurization was accomplished in incremental steps to 12 psig, 24 psig, 36 psig, 47 psig, and a final test pressure of 54 psig. Except for the final pressure level, the containment pressure was increased to 1 psig above the level at which measurement readings were to be taken. The pressure was then reduced to the specified value and, after a minimum time delay of 10 minutes to permit equalization of strains in the structure, the observations and measurements were made.

The final test pressure of 54 psig on the building was maintained for a period of 1 hour. During this time, measurements and observations were made to verify the adequacy of the structural design.

After the structural integrity test at 54 psig (held for 1 hour minimum) the pressure was reduced in the same incremental steps to 0 psig prior to performance of the containment liner leakage test.

Temperature, barometric pressure, and weather conditions were recorded hourly during the test period.

Prior to the strength test, predicted stress and strain at various locations were developed for an internal pressure of 54 psig. Although strain gages were installed on designated areas of the liner and concrete reinforcement, the analytically derived strains were not used as acceptance figures for the actual value. Values obtained, however, were analyzed and evaluated to determine the magnitude and direction of principal strains. Test data in excess of the predicted extremes required resolution through review of the design, evaluation of measurement errors and material variability and, if necessary, exploration of the structure.

Excessive crack widths, if any, observed during the test were required to be satisfactorily resolved in a manner similar to that discussed above for displacements.

6.2.1.2.2 Test Measurements and Instrumentation

An instrumentation program to determine the degree of agreement between predicted and observed deflection values at various points on the pressurized structure was employed to verify the design.

Radial and vertical growth of the cylinder was measured using linear motion transducers wired to electrical indicators along four approximately equally spaced meridians. Due to the equipment layout, it was not possible to run transducer wires across at six points at each circumference as recommended in Regulatory Guide 1.18. However, numerous additional strain gages were used on the liner plate and rebar to supplement the measurements. The radial deflections of the containment were measured at the spring line, mid-height of the cylinder and at 13.5 feet above the structural mat. Vertical deflections were measured at the apex and spring line of the dome.

Longitudinal and circumferential growth of the liner was measured by means of electrical strain gages attached to the exposed face of the liner in an area which is subjected solely to membrane forces (see Figure 6.2-1).

Strain gages were attached to selected hoop and meridional bars in the cylindrical wall and dome, as well as selected radial and circumferential top and bottom bars in the base slab. Also, strain gages were attached to representative circumferential bars around the equipment access opening and around both of the personnel access openings. Approximately 200 sets of strain gages had been attached to reinforcing bars at various locations in the containment structure.

Strain gages were attached to the steel liner to record strains at the junction with the mat liner, at mid-height, at the spring line, and in the dome. Additional strain gages were attached to the liner around the equipment access and personnel hatches.

Redundancy of instrumentation was attained through multiplicity of points and gages at which measurements were made, such that loss or damage to any one position would not be critical.

Two basic types of gages were used: (1) BLH, or equivalent, foil gages bonded to the members with epoxy cement, and (2) Microdot, Inc., weldable gages spot-welded to the members.

Where possible, gages were installed on reinforcing bars in the laboratory and the bars cadwelded in place.

Measurements around the personnel and equipment hatches were made using linear motion transducers between the hatches and the polar crane wall or other fixed supports as shown on Figure 6.2-2. Twelve linear motion transducers at each equipment and personnel hatch were used to measure the deflections, in accordance with Regulatory Guide 1.18.

During the structural acceptance test, all gages were read and recorded with a multichannel data acquisition system. Readings were obtained just prior to pressurization, at the various selected incremental pressures during pressurization and depressurization, and after depressurization.

The Unit 2 containment is a nonprototype structure, not requiring strain measurement. However, a small number of rebar and liner strain gauges were read for comparison and study at locations that had exhibited high strain when the test was performed on Unit 1.

Limited variable differential transmitter measurements were not taken on the Unit 2 personnel hatches, since the test performed on Unit 1 demonstrated that the personnel hatches were structurally loaded in a manner similar to the equipment hatch.

Crack patterns in the concrete were measured and recorded at the quarter points of circumference at the maximum test pressure. A strain sensitive coating was used to make the crack pattern more discernible (see Figure 6.2-3). Crack patterns in the areas of the large penetrations were visually checked to ascertain agreement with predicted stress patterns.

The range of strains and deformations expected at the 54 psig test pressure were as follows:

1. Increase in containment diameter: not more than 1.75 inches.

2. Maximum vertical elongation of the structure: not more than 2 inches.

3. Maximum width of new cracks or increase in existing cracks: not more than 0.03 inch.

4. Residual width of new cracks or increased width of existing cracks (after containment pressure is reduced to atmospheric): not more than 0.02 inch.

Since the containment structure was expected to remain in the elastic range during the pressure test, there was not expected to be any permanent distortion in the liner or in the concrete once the pressure was reduced to atmospheric or below. However, it was fully expected that small residual cracks in the concrete would appear as a result of concrete creep during pressurization.

6.2.1.2.3 Acceptance Criteria

The structural acceptance test determined whether the containment structure is capable of withstanding the magnitudes of loading used in the design. The acceptance criteria is that under the test load. The behavior of the structure under the test load must be such as to indicate its ability to withstand the loadings used for design.

Were the test acceptance criterion to equal or exceed the stresses computed under the factored loadings, then destruction of some elements would result.

It was not necessary to test up to design stresses to verify the structural integrity of the containment. Prediction and verification of deformation patterns, using the same design and analysis procedures for both design and test conditions, serves to verify the design.

Tensile stresses in the liner plate during the structural acceptance test were expected to be greater than those which would occur under the accident condition. The reason for this was that there was no temperature rise associated with the test condition. Compressive stresses would be created by the high temperatures associated with an accident condition, which overcome the tension in the liner. Stresses in the reinforcing bars were expected to be lower during the test condition than the values calculated for the accident condition.

With regard to the liner, the largest number and length of seams occurs in the cylinder and dome and, therefore, the greatest potential for leakage. The test condition was expected to yield tensile stresses in the dome and most of the cylinder that are higher than the design condition. The exception was the lower cylinder wall, where design tensile stresses are expected to be higher. With the exception of this area, the test placed a greater stress condition on the potential leakage paths than any of the design conditions.

The acceptance criterion requires demonstration that the overall structure exhibited elastic behavior throughout the test range. Inelastic behavior at localized stress concentrations was considered acceptable. Greatest agreement between the computed strains and those actually observed was anticipated to have been in the shell of the containment. Greater disparity between observed and calculated strains was contemplated around openings and at other discontinuities, where theoretical analysis becomes more complex. The acceptance criterion for cracking was based on the width and spacing of cracks, as determined through review of predicted crack size and crack spacing. Data obtained during the test were evaluated and a comparison with the values predicted by design was made to assess the structural behavior of the containment with regard to local and overall response.

6.2.1.3 Containment Overall Integrated Leakage Rate Tests

6.2.1.3.1 Preoperational Test

The preoperational containment overall integrated leakage rate test was performed following successful completion of the structural acceptance test. The test was performed to satisfy the requirements of 10CFR50, Appendix J, "Primary Reactor Containment

Leakage Testing for Water Cooled Power Reactors," for Type A tests.

The test was performed according to the peak pressure test program, using the "absolute" method, to ascertain that the leakage rate did not exceed 0.1 percent of the containment free volume per day at the design pressure of 47 psig. The test was performed at 47 psig.

6.2.1.3.2 Periodic Tests

The overall integrated leakage rate tests shall be in accordance with 10CFR50.54(o) in conformance with Appendix J of 10CFR50, Option B, using the methods and provisions of Regulatory Guide 1.163, September, 1995 as modified by approved exemptions. If the Type A test frequency is performed at 10 year intervals, two additional containment surface inspections shall be performed at approximately equal intervals during shutdowns between Type A tests.

The performance of these tests will be limited to periods when the plant is nonoperational and secured in the shutdown condition.

The periodic tests will be performed at a peak pressure of 47 psig.

Detailed test requirements are contained in the Technical Specifications. Should deviations become necessary, they will be the subject of License Change Requests (LCR) accompanied by appropriate justification. LCR 83-04, Public Service Electric & Gas (PSE&G) memo Liden to Varga, dated July 22, 1983, documents such a request for Unit 1.

6.2.1.4 Penetration Leakage Rate Tests

6.2.1.4.1 Preoperational Tests

Penetration leakage rate tests (Type B tests) were performed in accordance with 10CFR50, Appendix J, "Primary Reactor Containment Leakage Testing for Water Cooled Power Reactors." Only the free volume of the double penetrations was included in the test.

Because this volume is very small when compared to the containment free volume, the sensitivity and accuracy attainable in this leakage rate test was increased correspondingly over that attainable through integrated leakage rate testing.

All containment piping penetrations fitted with bellows are tested at Pa. Each bellow in penetrations utilizing more than one bellow is subjected to Type B testing.

The penetration leakage rate tests were performed with the penetrations pressurized to 47 psig, and the Containment Building at atmospheric pressure.

The combined leakage rate for the double penetrations and isolation valves was limited to less than 0.06 percent of the containment free volume per day.

6.2.1.4.2 Periodic Tests

Periodic leakage rate testing for penetrations will be conducted in a manner similar to the preoperational tests. The periodic tests will be performed according to the required frequencies set forth in 10CFR50, Appendix J, "Primary Reactor Containment Leakage Testing for Water Cooled Power Reactors," Option A, for Type B tests.

6.2.2 Containment Heat Removal Systems

Adequate post-accident heat removal capability for the containment is provided by two separate, Engineered Safety Features (ESF) Systems. These are the Containment Spray System described in Section 6.2.2.1, and the Containment Fan Cooling System, described in Section 6.2.2.2. These systems are of different engineering principles and serve as independent sources of containment cooling to assure that post-accident containment atmospheric temperature and pressure do not rise beyond their design basis values.

In addition to its ability to remove elemental iodine from the containment atmosphere, the heat removal function of the containment spray system is similar to that of the containment fan coil units. As described in section 15.4.8, "Containment Pressure Analysis", a minimum of three containment fan coil units in operation with a single containment spray train is capable of maintaining post-accident containment temperature and pressure below their design basis values, assuming a worst-case single active failure. Thus, design margin exists for the containment heat removal system.

6.2.2.1 Containment Spray System

6.2.2.1.1 Design Bases

The primary purpose of the Containment Spray System is to spray cool water into the containment atmosphere in the event of a loss-of-coolant accident (LOCA) and thereby ensure that containment pressure does not exceed the design value of 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 pipe. Pressure and temperature transients for LOCA are presented in Section 15. Although the water in the core after a LOCA is quickly subcooled by the Safety Injection System (SIS), the Containment Spray System design is based on the conservative assumption that the core residual heat is released to the containment as steam.

The Containment Spray System is designed to spray at least 2600 gpm of borated water into the Containment Building whenever two out of four (hi-hi) containment pressure signals occur or a manual signal is given.

Either of two subsystems containing a pump and associated valving and spray headers are independently capable of delivering 2600 gpm.

The design basis is to provide sufficient heat removal capability to maintain the post-accident containment pressure below the design pressure assuming that the core residual heat is released to the containment as steam.

A second purpose served by the Containment Spray System, including the recirculation phase, is to remove fission products (primarily iodine) from the containment atmosphere should it be released in the event of a LOCA. The analysis of offsite dose after a hypothetical LOCA is presented in Section 15. Iodine removal effectiveness is described in Section 6.2.3.

The Containment Spray System is designed to operate over an extended time period, following a primary coolant system failure, as required to restore and maintain containment conditions at near atmospheric pressure. It has the capability of reducing the containment post-accident pressure taking into account any reduction in capacity due to a single failure as defined in Section 6.2.2.

Portions of other systems which share functions and become part of the Containment Spray System, when required, are designed to meet the criteria of this section. Any single failure of an active component in either spray subsystem does not degrade the minimum containment cooling as defined in Section 6.2.2 or fission product removal capability of the Containment Spray System, as the containment pressure-temperature analysis in Section 15 assumes the most restrictive single failure.

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

1. Adequate shielding to maintain radiation levels within the limits of 10CFR50.67 (Section 11.2).

2. Collection of discharges from pressure relieving devices into closed systems.

3. Means to limit radioactivity leakage to the environs, consistent with limits set forth in 10CFR50.67.

System active components are redundant. System piping located within the containment is redundant and separable in arrangement.

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

Table 6.2-1 tabulates the codes and standards to which the Containment Spray System components are designed.

6.2.2.1.2 System Design

System Description

Adequate containment cooling and iodine removal are provided by the Containment Spray System shown on Plant Drawings 205235 and 205335 whose components operate in sequential modes. These modes are:

1. Spray a portion of the contents of the refueling water storage tank (RWST) into the containment atmosphere using the containment spray pumps. During this mode, the contents of the spray additive tank (sodium hydroxide) are mixed into the spray stream to enhance the iodine removal capability of the Containment Spray System.

2. Recirculation of water from the containment sump is provided by the diversion of a portion of the recirculation flow from the discharge of the residual heat removal (RHR) heat exchangers to the containment spray header after injection from the RWST has been terminated.

The bases for the selection of the various conditions requiring system actuation are presented in Section 15.

The principal components of the Containment Spray System are: two pumps, one spray additive tank, two eductors, spray ring headers and nozzles, and the necessary piping and valves. The containment spray pumps and the spray additive tank are located in the Auxiliary Building and the spray pump suctions are normally lined up to the RWST. Following an accident, the containment spray pumps are utilized until the water in the RWST is depleted.

During the recirculation phase, the system utilizes the two RHR pumps, two residual heat exchangers and associated valves and piping of the SIS.

The spray system is actuated by two out of four hi-hi containment pressure signals. The starting signal energizes the pumps and opens the discharge valves to the spray headers. The valves associated with the spray additive tank are opened on the same signal. If necessary, the operator can manually actuate the entire system from the control room.

During the period of time that the spray pumps draw from the RWST, a small portion of the spray flow is diverted from the spray pump discharge line through the eductor and back to the pump suction. Valve CS14 in the spray additive tank discharge line is provided with redundant position indication to assure effective chemical addition to the spray system. The liquid from the spray additive tank then mixes with the liquid entering the suction of the pumps. The result is a solution suitable for the removal of iodine from the containment atmosphere. The analysis of the iodine removal capability of the Containment Spray System, presented in Section 6.2.3, shows that most of the removable iodine in the containment atmosphere is washed out in the injection phase.

After the injection operation, spray pump flow is discontinued when the water in the RWST is depleted. Containment pressure control can then be maintained with the RHR System functioning through the containment spray headers.

If, for any reason, the containment pressure should be observed to increase, the operator can direct part of the discharge flow from the residual heat exchangers to the spray headers, thereby initiating recirculation spray flow.

The procedure for the change-over from injection to recirculation and cooling water for the residual heat exchangers is described in Section 6.3.

Components

All associated components, piping, structures, and power supplies of the Containment Spray System are designed to Class I (seismic) criteria.

The Containment Spray System shares the RWST liquid capacity with the SIS. Refer to Section 6.3 for a detailed description of this tank.

Pumps

The two containment spray pumps are of the horizontal centrifugal type, driven by electric motors which can be supplied with power from the standby ac power supply.

The design head of the pumps is sufficient to continue at rated capacity with a minimum level in the RWST against a head equivalent to the sum of the design pressure of the containment, the head to the uppermost nozzles, and the line and the nozzle pressure losses. Pump motors are direct-coupled and large enough for the maximum power requirements of the pumps. The materials of construction are stainless steel or equivalent corrosion-resistant material. Design parameters are presented in Table 6.2-2 and the pump head characteristic curve is presented on VTD 121398.

The containment spray pumps are designed in accordance with the specifications discussed for the pumps in the SIS, Section 6.3.

The pump motors are non-overloading to the end of the pump curve.

Each containment spray pump motor is provided with a shroud to prevent water spray damage from MEL piping as described in Section 3.6.5.12.5.

Details of the component cooling pumps and service water pumps, which serve the SIS, are presented in Section 9.

Spray Headers and Nozzles

The containment spray header piping arrangement is shown on Plant Drawings 207466 and 207467. These drawings illustrate the spray nozzle orientation, which has been designed to provide maximum spray coverage of the containment. The arrangement consists of four 360 degree ring headers at different elevations, with alternate headers connected. The header diameters are 101 feet at Elevation 244 feet-6 inches, 96 feet at Elevation 247 feet-0 inch, 53 feet at Elevation 266 feet-6 inches, and 48 feet at Elevation 269 feet-0 inch.

The spray headers are stainless steel of a hollow-cone pressure nozzle design, with a 3/8-inch diameter orifice. The nozzles have no internal parts which would be subject to clogging. The nozzles produce a drop size spectrum with a Sauter mean drop size of less than 1000 microns with the spray pump operating at design conditions and the containment at full design pressure and temperature.

The spray header supports are shown on Plant Drawings 223112, 223114 and 223123. These figures illustrate the relationship of the support steel to the headers and the containment building wall. The supports are designed such that interference with the spray pattern is kept to a minimum and their structural integrity under accident and seismic conditions is maintained.

The design is such that the alternate connected ring headers and corresponding sections of riser (from the last anchor point on the containment wall) will act as a unit under design thermal and seismic conditions. The pipe hangers and restraints are designed to support and restrain the pipe under design thermal and seismic conditions.

Spray Nozzles

The spray nozzles are of a hollow-cone pressure nozzle design without any internal parts subject to clogging. The nozzles produce a drop size spectrum with a Sauter mean drop size less than 1000 microns with the spray pump operating at design conditions and the containment at design pressure and temperature.

During spray recirculation operation, the water is screened through 1/12-inch (2.1 mm) diameter holes before leaving the containment sump. The spray nozzles are stainless steel and have a 3/8-inch diameter orifice. The nozzles are connected to four 360-degree ring headers of ring headers (alternating headers connected) of diameter 101 feet (Elevation 244 feet-6 inches), 53 feet (Elevation 266 feet-6 inches), 96 feet (Elevation 247 feet), 48 feet (Elevation 269 inches).

The nozzles and headers are so oriented as to maximize coverage of the containment volume.

All stresses are within those allowed by ANSI B31.1 Piping Code. Heavier walled pipe is used at anchor points and points of restraint to eliminate high stress regions.

Containment Dome Access System - Unit 2

Unit 2 utilizes a different design, the Containment Dome Access System. This system serves the dual purpose of supporting the Containment Spray System ring header piping and providing access for maintenance and inspection to the ring headers and the containment dome liner (see Vendor Technical Document 142864). This system consists of the following components:

1. An orbital inclined service bridge and trolley capable of carrying personnel and material, including an auxiliary hoist. It is designed to provide maximum coverage of both the containment dome liner and the spray header piping (see Vendor Technical Document 142864).

2. A structural steel girder, beam, and the support structure for the access bridge and spray piping (see Vendor Technical Document 142850).

Both components are seismic Class I and have been statically dynamically designed to withstand the effects of the design basis earthquake. They have combined total weight of 394,000 lb.

The support beams for this system penetrate the containment liner plate and are anchored into the concrete wall of the Reactor Containment Building. In order to maintain containment integrity, the penetrations through the liner plate are seal welded into place and vacuum box tested. A leak chase box is installed around each embedded beam to enable leak rate testing of the welds at any time (see Plant Drawing 224351).

The orbital service bridge, the spray header support or basket, and the spray piping were mathematically modeled as a system of node points interconnected by various weightless springs. The springs were assigned and stiffness characteristics of the structural beam and functional pipe elements of the system. All weights and inertias were distributed among the nodes. The degrees of freedom of the nodes were chosen to closely simulate the response of the system to external loading; the materials were assumed to be linearly elastic.

Static analysis was performed to obtain the maximum stresses under dead load and thermal variations.

Using the above mathematical model, a dynamic modal analysis was also performed to determine the modal frequencies and mode shapes. Safe Shutdown Earthquake (SSE) response spectra with 0.5-percent damping factor at the proper structural elevations were used as the input for the response spectrum analysis. The element stresses of those modes with meaningful participation for a given excitation direction were summed as a square-root-of-the-sum-of-the-squares (SRSS). When mode frequencies occurred within 10 percent of each other, an absolute summation of stresses was made prior to root mean square (RMS) summation.

The design stresses for the system are the summations of the maximum static and dynamic stresses for the respective members.

The analysis assumed the orbital bridge was locked to the rail in its storage location, the personnel cage was locked in the down (stored) position on the bridge with no load on the hoist, and the containment spray piping empty of liquid. This analysis simulates actual conditions during reactor operation.

The calculations performed on the Dome Access System indicate that none of the elements are subjected to loads beyond the allowable value of 32.40 ksi, which is 90 percent of the minimum yield strength of A36 steel. The loads obtained from the calculations for the Dome Access System were then used to design the Dome Access System containment interface tie-supports. These tie-supports which are made of A442 Grade 60 steel with an allowable stress of 19 ksi, will be subjected to a stress of only 10.92 ksi.

The allowable load on the access system will not be exceeded due to required administrative control.

The Dome Access System, consisting of the orbital service bridge and supporting basket and the spray header piping, was analyzed for an SSE using response spectrum curves at 0.5 percent damping with the bridge in the storage location. The bridge, basket, and piping were mathematically modeled as a multi-degree-of-freedom system with node points interconnected by various springs. ANSYS, a large scale, general purpose computer program, was used to perform the modal analysis.

Spray Additive Tank

The Spray Additive Tank holds a solution containing sodium hydroxide. The concentration of this solution assures that the injection spray pH will be at least 8.5.

The capacity of the tank is sufficient to contain enough sodium hydroxide solution which, upon mixing with the refueling water from the RWST, the boric acid from the boron injection tank (BIT), the borated water contained within the accumulators, and primary coolant, will bring the containment sump to a pH greater than 7.0. This assures adequate retention of the absorbed iodine in the sump liquid, and minimizes chloride induced stress corrosion cracking of stainless steel. Although iodine removal capability is maintained under these conditions, no credit is taken for any iodine removed after decontamination factor limitations specified by the Standard Review Plan, Section 6.5.2 (Ref. 26) are reached during the injection and recirculation phases. A level indicating alarm is provided in the Control Room if, at any time, the solution tank contains less than the required amount of sodium hydroxide solution. Periodic sampling confirms that proper sodium hydroxide concentration exists in the tank. Also, a flow indication is provided in the Control Room to alert the operator if there is low flow from the tank when required.

The tank design parameters are given in Table 6.2-3.

Heat Exchangers

The two residual heat exchangers that are used during the recirculation phase are described in Section 6.3.

Valves

The valves for the Containment Spray System are designed in accordance with the specifications for the valves in the SIS.

Valving descriptions and valve details are described in Section 6.3.

Piping

The piping for the Containment Spray System is designed in accordance with the specifications for piping in the SIS (Section 6.3).

The system piping is designed for 250 psig at 150°F.

Motors for Pumps and Valves

The motors for the Containment Spray System are designed in accordance with the specifications discussed for motors in the SIS (Section 6.3).

6.2.2.1.3 Design Evaluation

Range of Containment Protection

During the injection phase following the maximum LOCA (i.e., during the time that the containment spray pumps take their suction from the RWST) the Containment Spray System provides the design heat removal capacity for the containment. After the injection phase, each train of the Recirculation System provides sufficient cooled recirculated water to keep the core flooded as well as providing, if required, sufficient flow to the containment spray headers to maintain the containment pressure below the design value. This applies for all reactor coolant pipe break sizes up to and including the hypothetical instantaneous circumferential rupture of a reactor coolant pipe. Only one spray header is required to operate for this capability at the earliest time recirculation is initiated.

The Containment Spray and Fan Cooler Systems are capable of removing sufficient energy to maintain the pressure below the containment design pressure even in the event of a single failure. Each of these systems consists of independent equipment and components supplied from separate power sources. One containment spray train and three of five fan coolers, along with one train of the ECCS, is sufficient to ensure containment integrity.

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 116 feet from the lowest spray ring headers to the operating deck. The actual fall path is longer due to the trajectory of the droplets sprayed out from the ring header. Heat transfer calculations show that thermal equilibrium is reached by all droplets in the first few feet of their fall. Thus, the spray water reaches essentially the containment saturation temperature. The model for spray heat removal is discussed in Section 15.

In addition to heat removal, the Spray System is effective in scrubbing fission products from the containment atmosphere. Credit is taken for the removal of fission products (primarily iodine) in the analysis of the hypothetical LOCA (Section 15). A discussion of the effectiveness of containment spray as fission product removal process is contained in Section 6.2.3.

One containment spray pump and recirculation spray provide sufficient iodine scrubbing capability to ensure that post-accident fission product leakage (based on Reg. Guide 1.183 release fractions) would not result in doses exceeding the limits of 10CFR50.67.

System Response

The starting sequence of the containment spray pumps and their related emergency power equipment is designed so that delivery of the required spray into the containment is reached in 85 seconds following the appropriate initiating trip signal. This time delay for initiation of containment spray has included consideration of signal delay, assumed loss of offsite power, diesel start time, breaker closure, SEC sequencing and the time required for the spray pumps to reach full speed and to fill the spray headers and piping. The above delay time is consistent with the safety analysis described in Chapter 15 and the Tech Spec limit required for containment spray pump response.

Single Failure Analysis

A failure analysis has been made on all active components of the system to show that the failure of any single component will not prevent fulfilling the design function. This analysis is summarized in Table 6.2-4.

The LOCA analysis presented in Section 15 reflects the single failure analysis.

Reliance on Interconnected Systems

The Containment Spray System initially operates independently of other ESF following a LOCA. It provides containment cooling in combination with the Containment Fan Cooling System. For extended operation in the recirculation mode, water is supplied through the RHR pumps and heat exchangers.

During the recirculation phase, some of the flow leaving the residual heat exchangers may be bled off and sent to either the discharge of the containment spray pumps or to the suction of the safety injection pumps and centrifugal charging pumps. Minimum flow requirements will be set for the flow being sent to the core and for the flow being sent to the containment spray pump discharge. Sufficient flow instrumentation is provided so that the operator can perform appropriate flow adjustments with the remote throttle valves in the flow path.

Shared Function Evaluation

Table 6.2-5 presents 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.

Net Positive Suction Head (NPSH) and Spray Water Entrapment

Spray recirculation has been evaluated considering loss of water through entrapment outside the containment sump. There are three areas within the containment where reactor coolant blowdown liquid and spray water may become trapped: the reactor cavity, the refueling canal, and the reactor instrumentation tunnel. The reactor cavity has ventilation openings around the reactor that would allow spray water to drain to the lower elevations of the containment. The refueling canal is normally isolated from the Fuel Handling Building and would trap no more than 9,500 gallons of liquid from Containment Spray System. The instrumentation tunnel has a water capacity of approximately 70,000 gallons, none of which would drain to the sump.

The total quantity of water released to the containment at the beginning of the recirculation phase of the Containment Spray System operation, assuming a DBA with reactor coolant loop piping half full of water, is approximately 275,000 gallons. Discounting the water volume trapped in the refueling canal and the reactor instrumentation tunnel, the volume available at the suction of the RHR pump used for containment spray is approximately 190,000 gallons. The required NPSH for the RHR pump is a water level relative to the bottom (Elevation 70 feet) of the 8-foot deep containment sump. The indicated available water volume is a water level several feet above the containment sump top. There is therefore no significant effect on the required static head for the RHR pump.

Available and required NPSH for the containment spray pumps and the RHR pumps are provided in Table 6.2-6. Compliance with Regulatory Guide 1.1 is discussed in Appendix 3A.

Environmental Protection

During operation, a moveable shield provides missile protection for the area immediately above the reactor vessel. The spray headers are therefore protected from missiles originating within the shield.

Active components of the Containment Spray System are located outside the containment, and hence are not required to operate in the steam-air environment produced by the accident.

Material Compatibility

Parts of the system in contact with borated water, sodium hydroxide spray additive, or mixtures of the two are stainless steel or an equivalent corrosion-resistant material.

6.2.2.1.4 Tests and Inspections

Inspection Capability

Where practicable, all active components and passive components of the Containment Spray System are inspected periodically to assure system readiness. The pressure-containing systems are inspected for leaks from pump seals, valve packing, flanged joints, and safety valves. During operational testing of the containment spray pumps, the portions of the systems subjected to pump pressure are inspected for leaks. Design provisions for inspection of the SIS, which also functions as part of the Containment Spray System, are described in Section 6.3.

System and Component Testing

Active components of the Containment Spray System were adequately tested both in pre-operational performance tests in the manufacturer's shop and in place after installation. Thereafter, periodic tests are also performed after component maintenance.

Means are provided to test initially, under conditions as close to design as is practical, the full operational sequence that would bring the Containment Spray System into action.

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

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

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

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

Flow bypass through the eductors was checked during the initial preoperational tests of the Spray System. Subsequent system tests will be made with the spray additive tank bypass valves closed.

The air test lines for checking spray nozzles connect downstream of the isolation valves. Air flow through the nozzles is monitored as required by the Technical Specifications.

The functional test of the ECCS described in Section 6.3 includes the operation of the Containment Spray System. A test signal simulating the containment spray initiating signal is used to demonstrate operation of the Spray System up to the isolation valve on the pump discharge.

Spray Nozzles

The spray nozzles are of a hollow-cone pressure nozzle design without any internal parts subject to clogging. The nozzles produce a drop size spectrum with a Sauter mean drop size less than 1000 microns with the spray pump operating at design conditions and the containment at design pressure and temperature.

During spray recirculation operation, the water is screened through 1/12-inch (2.1 mm) diameter holes before leaving the containment sump. The spray nozzles are stainless steel and have a 3/8-inch diameter orifice. The nozzles are connected to four 360-degree ring headers of ring headers (alternating headers connected) of diameter 101 feet (Elevation 244 feet-6 inches), 53 feet (Elevation 266 feet-6 inches), 96 feet (Elevation 247 feet), 48 feet (Elevation 269 feet).

The nozzles and headers are so oriented as to maximize coverage of the containment volume.

6.2.2.2 Containment Fan Cooling System

6.2.2.2.1 Design Basis

The Containment Fan Cooling System is designed to recirculate and cool the containment atmosphere in the event of a LOCA 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 LOCA is quickly subcooled by the SIS, the Containment Fan Cooling System is designed on the conservative assumption that the core residual heat is released to the containment as steam.

The Containment Ventilation System (Section 9.4) which includes the Containment Fan Cooling System, 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 LOCA containment pressure transient, to keep the containment pressure from exceeding the design pressure. The fan cooler units continue to remove heat after the LOCA and reduce the containment pressure close to atmospheric within the first 24 hours.

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

1. Each of the five fan-cooler units is normally capable of transferring heat at the rate of at least 44 x 10^6 Btu/hr from the containment atmosphere at post-accident peak conditions, i.e., a saturated air-stream mixture of 43.5 psig and 265.9°F. The accident analyses of Section 15 determined a minimum number of three fan-cooler units, along with other containment heat sinks, are needed to maintain containment integrity. This correlates to a cumulative heat transfer rate of at least 132 x 10^6 Btu/hr. This heat transfer rate exceeds the analyzed value assumed in the accident analyses of Section 15.

   The establishment of basic heat transfer design parameters for the cooling coils of the fan-cooler units, and the calculation by computer of the overall heat transfer capacity are discussed in Section 15.4.

2. In removing heat at the design basis rate, the cooler coils are capable of discharging the resulting condensate without impairing the air flow capacity of the fan coolers 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.

In addition to the above design bases, the equipment is designed to operate at the post-accident conditions of 47 psig and 271°F for 3 hours, followed by operation in an air-steam atmosphere at 20 psig, 219°F for an additional 21 hours. The equipment design will permit subsequent operation of an air-steam atmosphere at 5 psig, 152°F for an indefinite period.

All components are capable of withstanding or are protected from d/ps which may occur during the rapid pressure rise to 47 psig in 10 seconds.

Portions of other systems which share functions and become part of this Containment Cooling System, when required, are designed to meet the criteria of this section. 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.

Where portions of these other 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, to the limits consistent with limits set forth in 10CFR50.67.

3. The RCFC (or CFCU) units are able to deliver their designed cooling capacity under all normal and abnormal conditions. Two-phase flow regions within the RCFC cooling coils following a LOCA/MSLB concurrent with a Loss of Offsite Power event are prevented by maintaining the RCFC cooling coils water solid during all normal and abnormal conditions.

   The waterhammer issues and modifications to the SW system (see Section 9.2, Service Water System) addressed in Generic Letter 96-06, preclude the possibility of the detrimental heat transfer effects resulting from the development of two-phase flow regions within the RCFC cooling coils.

6.2.2.2.2 System Description

The Containment Fan Cooling System is illustrated on Plant Drawings 205238 and 205338.

Individual system components and their supports meet the requirement for Class I (Seismic) structures and are isolated from fan vibration.

The Containment Fan Cooling System consists of five air handling units, each including motor, fan, motor heat exchanger, cooling coils, roughing filters, dampers, duct distribution system, instrumentation, and controls. The units are located on the operating floor, between the containment wall and the polar crane wall.

Each fan is designed to supply a nominal 110,000 cfm during normal operation and 40,000 cfm during accident operation. The fans are direct driven, centrifugal type, and the coils are plate fintube type. Each fan-cooler unit is normally capable of removing at least 44 x 10^6 Btu/hr or a cumulative of 132 x 10^6 Btu/hr for three fan-cooler units from the containment atmosphere under accident conditions. A minimum of 1300 gpm of service (cooling) water is supplied to each unit during accident conditions. The design maximum river water inlet temperature is 90°F, which results in an outlet temperature of 160°F under design basis conditions or 205°F for zero fouling case. Assuming a single active failure of the CFCU high speed breaker to open following an SEC MODE I or III accident signal, the outlet temperature could reach 209°F.

The Section 15 accident analysis also assumes an additional degradation in the heat transfer rate of 10% for the first two minutes of diesel powered fan cooler operation. This assumption accounts for nitrogen gas that could be released from solution from the service water system accumulators that provide a part of the resolution of Generic Letter 96-06.

Duct work distributes the cooled air to the various containment compartments and areas. During normal operation, the flow sequence through each air handling unit is as follows: inlet dampers, roughing filters, cooling coils, fan, discharge header. During post-accident operation, air is drawn through a moisture separator, a post-accident high-efficiency particulate air (HEPA) filter section and cooling coils and is discharged to the duct header.

Tight closing dampers isolate the post-accident filter section from the normally operating components. These dampers are tripped to the accident position upon either manual or automatic actuation of the respective fan. Electrically operated four-way solenoid valves control instrument air to the damper control cylinders. On a loss of either control air or control power the dampers fail to the accident (open) position.

The Fan Cooling System is actuated (in the post-accident mode) by a safety injection signal. The accident analysis assumes the CFCU initiating safety injection signal was containment high pressure because this is the limiting time delay case. Either all five fans or a minimum of three fans are started by the safeguards equipment controller, depending on the availability of emergency power.

A flow switch at each fan indicates whether air is circulating in the intended normal or post-accident flowpath. Indication and alarms are provided in the Control Room.

Flow Distribution and Flow Characteristics

The location of the distribution ductwork outlets, together with the location of the fan cooler unit inlets, ensures that the air will be directed to all areas requiring ventilation before returning to the units.

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

The air discharged inside the periphery of the polar crane rail circulates and rises above the operating floor through openings around the steam generators where it mixes with air displaced from the dome area. This mixture is returned to the fan coolers located on the operating floor. The temperature of this air will be essentially the design ambient for the containment vessel (120°F average maximum).

The steam-air mixture from the containment entering the cooling coils initially during the accident will be at approximately 271°F and have a density of 0.172 pounds per cubic foot. Most of the water vapor will condense on the cooling coil, and the air leaving the fan cooler will be saturated at a temperature slightly below 271°F.

With a flow rate of 39,000 cfm from each of 5 fans under accident conditions and a containment net free volume of 2,620,000 ft^3, the recirculation rate with five fans operating is approximately 4.5 containment volumes per hour.

Cooling Water for the Fan Cooler Units

The cooling water requirements for all five fan cooler units during a LOCA and recovery are supplied by the Service Water System. The Service Water System is described in Section 9. The design basis river water temperature for service water to the containment fan coolers is 90°F, although river water temperatures throughout the year are normally less. The service water temperature rise through the containment fan coolers is approximately 8°F for normal operation. In the unlikely event of an accident, this temperature rise will be a maximum of approximately 70°F* for a period of less than 1 hour, after which it will decrease. It is not expected that any significant amount of calcium carbonate precipitation on the heat exchanger surfaces will occur at these temperatures, and, therefore, there will be no subsequent plugging of the fan coolers.

As part of the issues addressed in Generic Letter 96-06, certain design constraints have been applied to the SW system for the Containment Cooling System CFCU (RCFC)

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*101°F for the zero fouling case and 131°F for the accident and single active failure of the flow controller to reset to accident flow.

units. These constraints and modifications are discussed in further detail in Section 9.2 (Service Water System). The following constraints apply to the cooling water for the fan cooler units:

1. Voids or column separation induced by siphoning effects, or changes in elevation cannot be tolerated in the SW system. These locations must remain water solid during all operating conditions.

2. The pressure in the flowing portions of the SW system must remain above the fluid saturation pressure for all operating conditions. Flashing or boiling resulting from increased temperatures or decreased pressures cannot occur. This constraint precludes the possibility of waterhammer or other hydraulic events due to steam bubble collapse, two-phase flow, or steam propelled water slugs.

3. The SW system containment penetrations and the containment closed systems must have pressure relief capability, or be shown not to be susceptible to large increases in internal pressure due to increased fluid temperatures. This constraint prevents failures of the containment boundary due to thermally induced overpressures.

Service water discharge from the cooling coils is subsequently mixed with the circulating water where radiation monitors R13A and R13B sample the effluent prior to discharge to the river. An alarm is annunciated in the control room upon detection of high radioactivity in an effluent line.

Flow and temperature indication is provided outside containment for service water flow to and from each fan cooler unit. Abnormal flow alarms for inservice fan cooler units are provided in the control room.

With the CFCU fixed resistance control scheme, the restricting orifices along with the flow control valve have been sized/set to establish a target flow of approximately 1900 gpm to 2100 gpm for each CFCU, dependent on the service water header pressure at the CFCUs. This flow rate is significantly higher than the nominal flow rate required to maintain the containment ambient temperature less than or equal to the Tech Spec value of 120°F. This should help in maintaining lower containment ambient temperature during times of high river water temperature. During normal operation, the flow control valve will open to its open limit stop position (approx. 50% open) to provide the target flow rates described above.

During a safety injection, the flow control valve will also open to its open limit stop position to provide minimum flow of 1300 gpm to each CFCU. The control valve closes when the fan cooler unit is not in use.

The solenoid valve associated with the flow control valve is de-energized to apply control air header pressure to the valve operator (closing the valve) whenever the RCFC fans are not running, or DC power is lost. The solenoid valve would then be energized to apply the pneumatic control signal to the flow control valve operator when the RCFC fan is operating in high or low speed.

Components

Roughing and HEPA Filters

The roughing filters in each fan cooler unit are designed to remove the larger particles of suspended dust and dirt from the containment atmosphere during normal power operation, normal reactor shutdown and loss of offsite power conditions. Removal of the particles also prevents buildup on the cooling coils, thus avoiding a reduction in heat transfer.

The roughing filters are arranged in two banks, each consisting of structural steel frame and removable filter cells. Each filter cell contains a fiberglass media which is capable of removing 90 percent of visible dust particles. The media efficiency is 70 percent on National Bureau of Standards type test ratings.

The HEPA filters in each fan cooler are provided to remove any particulate matter from the containment atmosphere. The HEPA filters are arranged in a structural steel frame and are individually removable. The filter media is fiberglass with asbestos separators and is capable of collecting 99 percent of particles 0.3 micron and larger in size from a saturated (100-percent relative humidity) 271°F atmosphere processed through the filter at 250-300 fpm. The HEPA filter media meets MIL-F-51079 and MIL-STD-282.

<u>Fan-Motor Units</u>

The five containment cooling fans are of the centrifugal, non-overloading direct drive type. Each fan provides a minimum flow rate of 39,000 cfm when operating against the system resistance existing during accident conditions (0.172 lb/ft^3 density, a containment pressure of 47 psig, and temperature of 271°F).

The two-speed containment fan cooler motors are totally enclosed, fan cooled (TEFC), 300 hp (high speed), induction type, 3 phase, 60 cycle, 1200 RPM, 460 volt with ample insulation margin. At low speed the motor delivers 100 hp. Insulation is Class F (NEMA rated total temperature 155□C) Westinghouse Thermalastic. It is impregnated and coated to give a homogeneous insulation system which is highly impervious to moisture. Internal leads and the terminal box-motor interconnection are given special design consideration to assure that the level of insulation matches or exceeds that of the basic motor system. At incident ambient and/or accident load conditions (271□F and 100 hp), the motor insulation hot spot temperature is not expected to exceed 113□C.

Fan cooler motors are cooled by an air-to-water heat exchanger which 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. Two vent valves permit containment ambient air to enter the cooling compartment (on increasing containment pressure) so the motor bearings will not be subjected to an excessive d/p. An open condensate drain line will enable the cooling compartment to equalize with the containment pressure as containment pressure is reduced by the motor heat exchanger. Cooling water is supplied by the Service Water System (SWS).

The motors are equipped with high temperature grease lubricated ball bearings to withstand the design basis incident ambient temperatures. Continuous bearing temperature monitoring is provided which will alarm in the control room. Fan motor leads are brought out of the frame through a seal and into a motor junction box. The motor leads are spliced to the field cables using environmentally qualified splice kits. 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.

In addition to the usual quality control tests which are performed to give assurance that the motors meet design specifications, special tests are performed to demonstrate that insulation margins are built in as expected. The completely wound stators are given a special high potential test to ground. The stators are immersed in water, meggered, and given a high potential test while immersed. After passing the water tests, the motor is baked and given a final coating dip. The stator and rotor are then baked again.

Cooling Coils

Coils are fabricated of AL-6X tubing. The heat removal capability of the cooling coils is at least 44×10^6 Btu/hr per fan cooler unit at peak saturation conditions (265.9°F, 43.5 psig). The design internal pressure of each coil is 200 psig and the coils can withstand postulated design basis accident pressures and temperatures without damage.

Each recirculating unit consists of 12 coil units mounted in two banks of 6 coils high. These banks are located one behind the other for horizontal series air flow, and the tubes of the coil are horizontal with vertical fans.

A moisture separator in each fan cooler removes the larger droplets of suspended moisture from the containment atmosphere in the event of a LOCA. Removal of the droplets prevents any significant water deluge over the face of the HEPA filters and thus avoids a serious reduction in filter effectiveness. The separator consists of a structural steel frame with removal separator elements. Each element is capable of removing 95 percent of water droplets 10 microns and larger in size.

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

Ducting

The ducts are designed to withstand the sudden release of RCS energy and energy from associated chemical reactions without failure due to shock or pressure waves by incorporation of damper along the ducts which open at slight overpressure (3.0 psi). The ducts are designed and supported to withstand thermal expansion during an accident. The seismic design and analysis methodologies used to qualify all ductwork and the contained equipment are described in Section 3.8.4.4.1.

Ducts are of welded and flanged construction. All longitudinal seams are welded. Where flanged joints are used, joints are provided with gaskets that are suitable for postulated design basis accident conditions. Ducts are constructed of galvanized sheet metal.

Dampers

All air control dampers that are an integral part of the fan coolers are designed to Class I seismic criteria. The damper construction is designed to withstand the design basis earthquake (DBE) concurrent with the pressure transients, thermal energy, and chemical activity resulting from a LOCA. Each damper is constructed of specially painted steel, with multiple blades that operate in unison and edge seals to minimize air leakage.

The backdraft damper at the discharge of each fan cooler is a normally closed counter-weighted device that opens automatically when the fan operates. It is designed to remain intact and operable during any LOCA by withstanding an approximate 7-psi air pressure surge over a 10-second period. This damper prevents the pressure surge from damaging the fan-motor assembly.

Two-position shut-off dampers are provided at each fan cooler to divert air flow through the HEPA filters and moisture separators during any LOCA or through the roughing filters during normal operation. The roughing filter dampers are normally open and fail closed. The HEPA filter dampers are normally closed and fail open. Both sets of dampers revert to their fail positions after a safety injection signal.

Each two-position shut-off damper is provided with redundant pneumatic operators that can provide 150 percent of the design operating torque. Each damper assembly is designed to remain intact and operable during any LOCA by withstanding a 3-psi air pressure surge over a 10-second period.

The fan coolers are equipped with pressure relief dampers in the filter enclosures. These dampers are normally closed counter-weighted devices that open progressively as the d/p across them exceeds 0.25 psi. In the event of a LOCA, the pressure-relief dampers limit the d/p to 3 psi and thus maintain the structural integrity of the fan coolers during the pressure transient.

6.2.2.2.3 Design Evaluation

Range of Containment Protection

The Containment Fan Cooling System provides the design heat removal capacity for the containment following a LOCA assuming that the core residual heat is released to the containment as steam. The system accomplishes this by continuously recirculating the air-steam mixture through cooling coils to transfer heat from containment to service water. The heat removal function of the containment fan coil units is similar to that of the containment spray system. As described in section 15.4.8, "Containment Pressure Analysis", a minimum of three containment fan coil units in operation with a single containment spray train is capable of maintaining post-accident containment temperature and pressure below their design basis values, assuming a worst-case single active failure. Thus, design margin exists for the containment heat removal system.

The performance of the Containment Fan Cooler System in pressure reduction is discussed in Section 15.

System Response

Automatic starting of the standby fan cooler units (under design conditions, up to four of the fans, and two service water pumps operate during normal power operations for containment ventilation) and the related emergency power equipment is designed so that the required air flow and cooling water flow for an accident condition is reached within the time delay for starting fan cooler units assumed in the containment pressure analyses.

(This text has been deleted)

The water valves and air dampers are actuated to the accident position by closure of the fan cooler low speed breaker.

Single Failure Analysis

A failure analysis for all active components of the system shows that the failure of any single active component will not prevent fulfilling the design function. This analysis is summarized in Table 6.2-7.

The analysis of the LOCA presented in Section 15 is consistent with the single failure analysis.

Reliance on Interconnected Systems

The Containment Fan Cooling System is dependent on the operation of the SWS. Cooling water to the coils is supplied from the SWS. Six service water pumps are provided, only two of which are required to operate during the post-accident period.

Shared Function Evaluation

Table 6.2-8 is an evaluation of the main components which have been discussed previously and a brief description on how each component functions during normal operation and during the accident.

Reliability Evaluation of the Fan Cooler Motor

The design of the motor and motor heat exchanger is such that the accident environment is prevented, in a significant sense, from entering the motor winding. When entering in the very limited amount required to equalize motor interior pressure, the incoming atmosphere is directed to the heat exchanger coils where moisture is condensed. If some quantity of moisture should pass through the coil, the motor interior environment would "clean up" since interior air continually recirculates through the heat exchanger.

The motor insulation hot spot temperature is not expected to exceed 113°C even under accident conditions; normal life would be expected with a continuous hot spot of 155°C. The insulation has resistance to moisture, and tests indicate that the insulation system would survive the accident ambient moisture condition without failure. The heat exchanger system of preventing moisture from reaching the winding keeps the winding in much more favorable conditions. In addition, the motors are furnished with an insulation voltage margin beyond the operating voltage of 480 V.

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 same type was subjected to prolonged exposure to 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 further assure that the unit performed as designed. The post-testing inspection showed no degradation of the motor components (1). The fan motor bearings are designed to perform in the accident ambient temperature conditions. However, the interior bearing housings are cooled by the heat exchanger. It is expected that bearing temperatures would be 125°C to 140°C, under accident conditions. The heat exchanger is designed using a conservative 0.002 fouling factor.

Throughout the lifetime of the plant, these motors perform the normal heat removal service and are loaded to approximately 275 hp.

Environmental Protection

All of the fan cooler units are located on the operating floor adjacent to the containment wall. The distribution header is located below the operating floor, between the polar crane wall and the containment wall. This arrangement provides missile protection for all components.

System control and instrumentation devices required for post-accident operation are also installed in locations such as to minimize the danger of control loss due to missile damage.

The fan motor enclosures, electrical insulation, and bearings are designed for operation during accident conditions. Surfaces in contact with the containment atmosphere are protected against corrosion.

6.2.2.2.4 Tests and Inspection

Component and System Testing

Each fan cooling unit was tested after installation for proper flow through the Duct Distribution System.

The Containment Fan Cooling System is designed such that the components can be tested periodically, and after any component maintenance, for operability and functional performance.

Four of the fan cooling units are in use during normal operation. The fan not in use can be started from the control room to verify readiness. The dampers directing flow through the post-accident filter section can be tested when the fan is running on low speed.

The functional test of the ECCS described in Section 6.3 will demonstrate proper transfer of the fan units in the event of a loss-of-power. A test signal is used to initiate damper motion and fan starting. This test will verify proper functioning of the air flow switch provided for each fan.

Inspection

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

6.2.3 Containment Atmosphere Iodine Removal

6.2.3.1 Introduction

The Containment Spray System is an Engineered Safety System employed to reduce pressure and temperature in the containment following a postulated LOCA. For this purpose, subcooled water is sprayed into the containment atmosphere through a large number of nozzles from spray headers located in the containment dome.

Because of the large surface area between the spray solution and the containment atmosphere, the Containment Spray System also serves as a removal mechanism for fission products postulated to be dispersed in the containment atmosphere. Radioiodine in its various forms is the fission product of primary concern in the evaluation of a LOCA. The major benefit of the containment spray is its capacity to absorb molecular iodine from the containment atmosphere. To enhance this iodine absorption capacity of the spray, the spray solution is adjusted to an alkaline pH which promotes iodine hydrolysis to nonvolatile forms.

According to the known behavior of elemental iodine in highly dilute solutions, the hydrolysis reaction:

$$I_2 + OH \rightarrow HIO + I$$

proceeds nearly to completion (2) at pH > 8. The iodide form is highly soluble, and HIO readily undergoes additional reactions to form iodate.

The overall reaction is:

$$3I_2 + 3H_2O \rightarrow 5HI + HIO_3$$

Values for the spray removal half-life of the molecular iodine in a typical containment are on the order of minutes, or less. This makes the Containment Spray System a very efficient fission product removal system, in comparison to such alternatives as charcoal filtration systems.

For the small break loss-of-coolant accidents for which containment spray is not automatically initiated, offsite dose analysis was performed by Westinghouse using the methodology suggested in Reference 21. The results of this analysis are presented in Reference 22. It has been demonstrated that the consequences of a small break LOCA without containment spray actuation are bounded by those of the large break LOCA.

6.2.3.2 Iodine Removal Model

Containment spray performance has been determined using the spray model developed by Westinghouse. This model includes the effects of spray drop size distribution, droplet coalescence, and liquid phase mass transfer resistance. Its use results in conservative values of spray iodine removal constants when compared with test results.

Method of Calculation

In order to eliminate the need of scale-up factors from experimental results to full-sized reactor containments, the size-dependent calculations in this model were programmed for discrete size parameters, i.e., the calculations are repeated for incremental height steps, and for 40 different drop-size groups to represent the effects of the drop-size distribution. No significant effect on results was observed by increasing the number of groups. The resulting model with discrete size-dependent parameters has been programmed for a digital computer.

In the computer code, the sprayed volume of the containment is divided into layers of incremental height and area equal to the total sprayed area at any height z. The height-dependent calculations, such as drop trajectories and the change in the drop size distribution due to coalescence, are performed for each height step, using the parameters calculated in the previous step as input for the next step.

Drop-Size Distribution

The drop-size distribution used in the model is based on data obtained from measurements of the actual size distribution from the Spraco 1713 nozzle for the range of pressure drops encountered during operation of the Spray System. The results obtained for 20, 30, 40, and 50 psi pressure drops across the nozzle have been used in this evaluation.

Analysis of these drop-size measurements shows that the drop-size distribution from this nozzle may be represented by a continuous distribution function, which is used as the input to the computer code.

Condensation

As the spray solution enters the high temperature containment atmosphere, steam will condense on the spray drops. The amount of condensation is easily calculated by a mass balance on the drop:

$$mh + m_c h_g = m' h_f$$

where:

m and m'	= the mass of the drop before and after condensation, lb
m_c	= the mass of condensate, lb
h	= the initial enthalpy of the drop, Btu/lb
h_g and h_f	= saturation enthalpy of water vapor and liquid, Btu/lb

The increase in each drop diameter in the distribution, therefore, is given by:

$$\left(\frac{d'}{d}\right)^{(3)} = \left(\frac{v}{v_f}\right)\left(\frac{h_g - h}{h_{fg}}\right)$$

where:

v_f	= the specific volume of liquid at saturation, ft^3/lb
v	= the specific volume of the drop before condensation, ft^3/lb
h_{fg}	= the latent heat of evaporation, Btu/lb
h_g	= the enthalpy of steam at saturation, Btu/lb
d	= the drop diameter, cm before condensation
d'	= the drop diameter, cm after condensation

This increase in drop size due to condensation is expected to be complete in a few feet of fall for the majority of drop sizes in the distribution. More detailed calculations by Parsley (3) show that even for the largest drops in the distribution thermal equilibrium is reached in less than half of the available drop fall height. The change in the drop-size distribution due to condensation was conservatively modeled by a step increase to the equilibrium size immediately after the drops emerge from the nozzle.

Drop Trajectories

A description of the actual drop trajectories is required to obtain accurate drop residence times, and to obtain the trajectory angle required for the coalescence calculations described below. These trajectories are obtained by integrating the equations of motion for each drop size.

The equations of motion were integrated numerically, with the drag coefficient being determined iteratively from Reynolds number and terminal velocity.

These calculations yield the following results:

1. Spread and Nozzle Interference

   Trajectory results for a range of drop sizes show that the horizontal velocities of the drops are quickly attenuated. For the smaller drop sizes ($<400\mu$), the trajectory essentially is a straight fall. Even for 1000μ drops, the horizontal velocity component diminished to less than 10 percent of the total velocity in less than 10 feet. The effect of temperature and pressure on drop trajectories has also been calculated. The resulting spray envelope is a smaller diameter at higher temperatures and pressure.

2. Drop Residence Time

   For downward-directed spray nozzles, the initial vertical velocity is higher than the terminal velocity, resulting in a slightly shorter residence time than that calculated with the assumption of terminal velocity. An accurate account of the residence time is obtained from consideration of the actual trajectories followed by the drop.

Correction factors are calculated for each drop size in the spectrum, so that the drop fall-times used for the iodine removal calculations are the actual drop residence times.

A measure of conservatism is added to the drop residence calculations by the use of the drop diameters after condensation. Actually, the drop velocities would have been attenuated to a fraction of the initial nozzle velocity by the time condensation is complete.

Drop Coalescence

This effect will tend to decrease the overall surface-to-volume ratio of the spray, thereby affecting the fission product removal capability of the system. Concern has been centered particularly on the effect of coalescence on scale-up factors applied to data obtained from small-scale experiments. The effects of this phenomenon are accounted for by a mathematical model which is dependent of the containment size. The mathematical model used to account for drop coalescence due to the effects of overlapping spray patterns and due to larger drops overtaking smaller ones shows the number of coalescences to be functions of the collision and coalescence efficiencies, as well as the trajectory angle, drop velocities, drop size, and drop density.

The coalescence efficiency is the probability that a collision will result in the formation of a single larger drop. The collision efficiency describes the probability that two drops on a geometric collision course, (i.e., their centers of motion are separated by a distance less than the sum of the radii of the two drops), will actually collide.

The results calculated with the drop coalescence model show that the smaller drops with diameters near the mode of the distribution are affected most. This is expected, since these sizes have the highest density of drop population. Due to the considerably larger volumes of the larger diameter drops, however, the increase in the larger drop population is not very pronounced.

Mass Transfer

The basic equation for the iodine concentration in the containment atmosphere is derived from a material balance of the elemental iodine in the containment. The iodine removal by the spray system may be expressed by:

$$V_c \frac{dCg}{dt} = -EF(HC_g - C_{L2})$$

where:

- V_c = containment free volume in cc
- C_g = the iodine concentration in the containment atmosphere, gm/cc
- H = the iodine partition coefficient, (gm/liter of liquid)/ (gm/liter of gas)
- F = the spray flow rate, cc/sec

The resulting change in the drop size distribution is taken into consideration in the mass transfer calculations described below.

The variable E is the absorption efficiency, which may also be described as the fractional approach to saturation:

$$E = \frac{C_{L2} - C_{L1}}{C*_L - C_{L1}}$$

where:

C_{L1} = the iodine concentration in the liquid entering the dispersed phase, gm/cc

C_{L2} = the iodine concentration in the liquid leaving the dispersed phase, gm/cc

C^*_L = the equilibrium concentration in the liquid, gm/cc

This absorption efficiency may be calculated from the time-dependent diffusion equation for a rigid sphere, with the gas film mass transfer resistance as a boundary condition. This mass transfer model was suggested by L. F. Parsley (4), who gave the solution to the diffusion equation with the above-mentioned boundary condition as:

$$E = 1 - \sum_{n=1}^{\infty} \frac{6Sh^2 \exp(-\alpha_n^2 \theta_f)}{\alpha_n^2[\alpha_n^2 + Sh(Sh-1)]}$$

where:

Sh = is the dimensionless group

$$\frac{k_g a}{HD_L}$$

a = the drop radius, cm

k_g = the gas film mass transfer coefficient, cm/sec

D_L = the liquid diffusivity, cm^2/sec

θ_f = the dimensionless drop residence time

α_n = the eigenvalues of the solution

It is noted that this solution, which applies to the rigid drop model is based on the assumption that molecular diffusion is the only mechanism for the transport of iodine from the surface to the interior of the drop. Since a high degree of mixing is expected in the drops, particularly in the presence of sizable temperature and concentration gradients, it is apparent that this stagnant drop model presents a conservative approach to the calculation of iodine absorption by the drops.

The absorption efficiency calculated with the model described above is a function of drop size. The removal constant, λs in reciprocal hours, for the entire spray, therefore, is obtained by an appropriate summation over all drop size groups:

$$\lambda_s = \sum_{i=1}^{n} \frac{E_i F_i H}{V_C}$$

6.2.3.3 Experimental Verification of the Iodine Removal Model

To demonstrate that the ability of the model described above conservatively estimates actual spray performance, the Westinghouse model was applied to the test runs made at Oak Ridge National Laboratories (ORNL) and Battelle Northwest Laboratories (BNWL). Comparison of the results of these tests with the above described spray removal model show the spray removal model to be conservative in all cases.

6.2.3.4 Iodine Removal Evaluation

6.2.3.4.1 Injection Phase Operation

The analysis of iodine removal by containment spray water is based on the assumption that:

1. One of two spray pumps is operating.

2. One train of ECCS is operating at its maximum capacity.

The duration of injection spray is 48 minutes followed by recirculation spray.

An eductor system, described in Section 6.2.2.1, is used to maintain the injection spray solution at a pH in the range of 8.5 to 10.0 to ensure efficient and rapid removal of the iodine from the containment atmosphere.

The performance of the Spray System was conservatively evaluated at the peak temperature and pressure resulting from a double-ended rupture of the RCS, with no credit taken for the subcooling of the ECCS. These pressure and temperature conditions, listed in Table 6.2-9, were assumed throughout the injection and recirculation phases of operation of the Containment Spray System.

The injection and recirculation phase spray flow rates per pump, used in the calculation of λ, corresponding to this back-pressure in the containment are given in Table 6.2-9.

Since this peak pressure condition is expected to exist at most for a few minutes, and since both mass transfer parameters and spray flow rate improve with decreasing pressure, an appreciable conservatism is added to this evaluation by this assumption.

The removal constants for the spray system, in the injection and recirculation phases, calculated with the model described and with the above mentioned assumptions, is shown in Table 6.2-9.

6.2.3.4.2 Recirculation Phase

Under the assumptions stated in Section 6.2.3.4.1, the spray recirculation phase is analyzed to be initiated 58 minutes after the start of safety injection. Safety injection is assumed to last 48 minutes followed by an assumed conservative gap of 10 minutes without spray before recirculation spray is started. At this time, sump water would have reached its minimum equilibrium pH of at least 7.0. The iodine removal capability remains high under these conditions and credit is taken for iodine removal by sprays during the recirculation phase as shown in Table 6.2-9. During the spray recirculation phase, the sump pH will remain at equilibrium pH since no additional water is added to the system.

For those small-size primary breaks for which containment spray is not automatically actuated, sump solution pH will be adjusted to a minimum value of 7.0 within 48 hours of switchover to cold leg recirculation mode. However, no credit it taken for retention of iodine in solution in the offsite dose analysis summarized in Reference 22.

6.2.3.4.3 Re-Evolution of Iodine

Any re-evolution of dissolved iodine from the sump to the containment atmosphere is dependent on the concentration gradient between the liquid and vapor phases. The equilibrium between these concentrations is given by the partition coefficient, H, and, therefore, is a function of iodine concentration, pH, and temperature. A plot of the sump alkalinity, as a function of the time after the start of injection, is shown on Figure 6.2-14. The resulting partition coefficient, based on a constant iodine concentration equal to the concentration corresponding to a DF of 100 in the containment atmosphere, is shown on Figure 6.2-15 for sump temperatures of 150°F and 212°F. The equations given by Eggleton (5) were used to calculate the partition coefficient.

Although the iodate reaction, i.e.:

$$3I_2 + 3H_2O \rightleftharpoons IO^-_3 + 5I^- + 6H^+$$

is expected to contribute significantly (5) to the iodine partition at the high sump pH values, this reaction is conservatively neglected in these calculations.

From Figure 6.2-15 it is apparent that the partition coefficient of 4.3×10^3, which is required to maintain a DF of 100 in the vapor phase, is exceeded at all times during the recirculation phase.

6.2.4 Containment Isolation System

The Containment Isolation System provides the means of isolating the containment atmosphere and RCS as required to prevent the release of radioactivity to the outside environment in the event of a LOCA.

6.2.4.1 Design Bases

The following conditions and definitions are used in the design of the Containment Isolation System to assure that subsequent to an accident, there will be two barriers between the atmosphere outside the containment and the containment atmosphere.

1. The design parameters of all piping and connected equipment within the isolated boundaries are equal to or greater than the DBA environment of the containment, 47 psig, 271°F.

2. All valves and equipment which are isolation barriers are protected against missiles and water jets, both inside and outside the containment.

3. Lines which, due to safety considerations, must remain in service subsequent to certain accidents have, as a

minimum, one manual isolation valve outside the containment.

4. All isolation valves and equipment are designed to Class I seismic criteria.

5. Per acceptance methods of General Design Criteria 55 and 56 and ANS N271-1976/ANS 56.2 the two barriers may consist of:

   (a) two closed piping systems or vessels, one inside and one outside the containment, (b) two automatic isolation valves, one inside and one outside the containment, (c) an automatic isolation valve inside the containment and a closed system outside the containment, (d) an automatic isolation valve outside the containment and a closed system inside the containment, or (e) an automatic isolation valve outside containment and a closed system outside the containment.

6. A check valve on an incoming line or a normally closed valve is considered an automatic valve.

6.2.4.2 System Description

The following four classes of piping arrangement are provided in the Containment Isolation System. These classes are illustrated on Figure 6.2-16.

Class A

Class A piping is connected to a normally closed system outside the containment, and is separated from the RCS and the containment atmosphere by a closed system inside the containment.

For Class A piping, no additional valves are required for isolation.

Class B

Class B piping is connected to open systems outside the containment, and is connected to the RCS or is open to the containment atmosphere.

For Class B piping, the following is provided, as a minimum, for isolation:

1. Incoming Lines: Two auto-trip valves (one inside, one outside), or a check valve inside and an auto-trip valve outside.

2. Outgoing Lines: Two auto-trip valves (one inside, one outside).

Class C

Class C piping is connected to open systems outside the containment, and is separated from the RCS and the containment atmosphere by a closed system.

For Class C piping, the following is provided, as a minimum, for isolation:

1. Incoming Lines: One check valve or auto-trip valve outside. No valve inside.

2. Outgoing Lines: One auto-trip valve outside. No valve inside.

Class D

Class D piping is connected to a closed system outside the containment, and is connected to the RCS or is open to the containment atmosphere.

For Class D piping, the following is provided, as a minimum, for isolation:

1. Incoming Lines: One auto-trip valve or check valve inside. No valve outside.

2. Outgoing Lines: One auto-trip valve inside and no valve outside. Alternately, one auto-trip valve outside and no valve inside.

In addition to Classes B and C, for lines 1-inch nominal pipe size and larger which penetrate the containment and which are connected to the RCS, at least two valves are provided inside the containment. The valves are normally closed or have automatic closure. For incoming lines, check valves are permitted and are considered as automatic. Piping which penetrates the containment, but which represents normally closed lines, also falls under this criterion. In this case, manual isolation valves are acceptable.

In order to be considered a "closed" system inside containment, a system must meet the following requirements:

1. Does not communicate with either the RCS or the containment atmosphere.

2. Safety classification same as for engineered safety systems.

3. Must withstand external pressure and temperature equal to containment design pressure and temperature.

4. Must withstand accident transient and environment.

5. Must be missile protected.

In order to be considered a "closed" system outside containment, a system must meet the following requirements:

1. Does not communicate with the atmosphere outside the containment.

2. Safety classification same as for engineered safety systems.

3. Internal design pressure and temperature must be at least equal to containment design pressure and temperature.

For incoming lines to the containment, check valves are used whenever an additional barrier is provided. Use of check valves in this service is confined to either liquid lines or lines that are closed outside the containment. These check valves shut under a d/p when the higher pressure is on the containment side of the check valve.

These isolation valving arrangements were designed in accordance with Atomic Energy Commission (AEC) proposed General Design Criteria published in 1967, which were in effect at the Construction Permit stage. The valving arrangements that deviate from AEC General Design Criteria 55, 56, and 57 dated July 7, 1971, are the following:

1. RHR connections between the RCS and the RHR pumps. Redundant isolation protection is provided by a normally closed motor operated valve inside the containment and the closed system (RHR) outside the containment.

2. Seal water supply line from the seal water injection filters to the reactor coolant pump seals. Redundant isolation protection is provided by a check valve inside the containment and the closed system (CVCS) outside the containment.

3. Safety injection recirculating suction line from the containment sump to the suction of the RHR pumps. Redundant isolation protection is provided by normally closed motor operated valves inside protective chambers outside of containment and the closed system (RHR) outside the containment.

4. Containment instrument lines (see below).

5. The main feedwater lines are provided with one stopcheck valve (BF22) outside containment. These valves include remote-manual motor operators.

6. RHR pump discharge to cold leg Safety Injection. Redundant isolation is provided by the remote manual (SJ49) valves located outside containment and the RHR closed system outside containment. This is considered an acceptable isolation barrier per the "other defined basis" in ANSI N271-1976. This standard is endorsed by Regulatory Guide 1.141.

7. ECCS relief line discharge to the containment sump. Redundant isolation is provided by a check valve inside containment (PR25) and the closed system outside the containment.

8. Service Water system to and from the Containment Fan Coil Units. Redundant isolation is provided by remote manual valves outside containment and the closed Nuclear Class III system inside containment. Original system design complied with AEC General Design Criteria #53 and the system meets the definition for a Safety Class 2 system.

9. Component Cooling to and from the Excess Letdown Heat Exchanger. Redundant isolation is provided by automatic isolation valves outside containment and the closed Nuclear Class III system inside containment. Original system design complied with AEC General Design Criteria #53 and the system meets the definition for a Safety Class 2 system.

10. Main Steam supply lines to the Auxiliary Feed Pump Turbine, Radiation Monitors and the Steam Safety Valves support struts. These essential system branch lines off the Main Steam penetrations only utilize a single isolation barrier being the closed system inside containment. The calculated release through these paths is already bounded by the accident analysis for a primary to secondary leak and a complete blowdown of the Steam Generator.

Instrument Lines

Instrument lines which penetrate the containment are the following:

1. The containment pressure instrument used to initiate safeguards consists of four instrument lines penetrating the containment. Each line consists of a sealed, filled measuring system whose isolation consists of a diaphragm-type sensor which separates the containment atmosphere from the seal fluid and another diaphragm in the transmitter which separates the seal from the atmosphere outside the containment.

2. The containment air sample radiation monitor normal inlet and outlet sample lines are each equipped with two automatic trip valves, one inside and one outside the containment, which close upon receipt of a containment isolation phase A signal. The backup inlet and outlet sample lines are normally closed and under administrative control with two remote operated isolation valves, one inside and one outside the containment for each line.

3. The containment pressure instrument used for wide range monitoring consists of two instrument lines penetrating the containment. Each line consists of a sealed, filled measuring system whose isolation consists of a diaphragm-type sensor, which separates the containment atmosphere from the seal fluid and another diaphragm in the transmitter, which separates the seal from the atmosphere outside containment.

3. The pressurizer dead-weight pressure calibrator has a single line penetrating the containment. Isolation is accomplished with two manual valves located just outside the containment. These manual-valves are normally closed and are opened only under administratively controlled conditions.

4. Three lines penetrate the containment for instrumentation required for leak rate testing. Each line is isolated with two manual valves, one inside and one outside containment. These valves are normally closed and under administrative control.

These provisions meet the requirements of Regulatory Guide 1.11.

Containment Isolation Valve Summary

Table 6.2-10 lists the major piping penetrations through the reactor containment for each fluid system and summarizes the specific isolation provisions for each penetration. Valve positions during normal operation, shutdown, and accident conditions are also listed. Isolation valving arrangements are shown graphically on Figures 6.2-17 through 6.2-46.

The main steam isolation valves (MSIVs) fulfill their containment isolation function as remote-manual containment isolation valves. The automatic closure of the MSIVs is not required for containment isolation due to having a closed system inside containment. The remote-manual containment isolation function of the MSIVs can be accomplished through either the use of the hydraulic operator or when the MSIV has been tested in accordance with Technical Specification 4.7.1.5, the steam assist closure function can be credited.

Valve closing time using the hydraulic actuator is approximately six minutes. The closure time for establishing containment isolation is that necessary to significantly limit the release of radioactivity to the environment. MSIV fast closure is not required for containment isolation in any operating Mode because the steam generator shell and main steam piping serve as the primary barrier for a LOCA. For the LOCA, the design basis does not assume a concurrent feedwater or steam line break. The main steam system does not directly connect to the reactor coolant system or the containment atmosphere. However, a steam generator tube break or rupture makes a connection between the RCS and the secondary side systems via the main steam system. The Chapter 15 SGTR accident analysis assumes a coincidental loss of offsite power that causes the steam dump valves to close, protecting the condensers. For the Mode 1 or 2 SGTR Chapter 15 accident analysis, isolation of the faulted steam generator is assumed to occur within 30 minutes as necessary to limit the release of radioactivity to the environment via the steam generator PORV or safety relief valves. Isolation of the faulted steam generator also limits the spread of radioactivity to the interconnected steam generators, at least one of which will be used to cooldown the RCS until RHR can be initiated at 32 hours post-accident. During low temperature (<375°F) Mode 3 and Mode 4 operations, there is insufficient energy transferred to the secondary side in a SGTR or steam generator tube leak to result in lifting the steam generator PORVs or safety relief valves and there will be no release of radioactivity to the environment. Use of the remote-manual, hydraulic actuator for containment isolation in low temperature (<375°F) Mode 3 and Mode 4 is satisfactory because even if isolation of the faulted steam generator fails, the failure will not increase the dose consequences beyond the existing Chapter 15 SGTR accident analysis that remains bounding.

The 20-inch inside diameter fuel transfer tube between the refueling canal inside the containment and the fuel transfer pool is sealed with a blind flange inside the containment, redundant isolation is provided by a double o-ring seal on the flange. The terminus of the tube outside the containment is closed by a gate valve which is not a containment isolation valve.

The equipment hatch (door) is under administrative control to assure that it is properly closed and sealed whenever containment integrity is required. No instrumentation is provided for the equipment hatch.

Actuation Provisions

Containment isolation is actuated under the following conditions:

1. A safety injection signal generates the containment isolation signal (Phase A), which actuates most containment isolation valves. The Phase A isolation signal closes all trip valves which are located in lines which are connected to the reactor coolant loops and penetrate the containment, thereby preventing loss of reactor coolant through the lines in which the automatic trip valves are located. Normally closed motor operated containment isolation valves in the SIS are opened by the safety injection signal to permit SIS operation.

2. A rise in containment pressure to the high containment pressure set point also generates the Phase A isolation signal.

3. A further rise in containment pressure, indicating a major LOCA, results in a containment high-high pressure signal which generates both the containment spray and containment isolation Phase B signal. All normally open lines which penetrate the containment which are not closed by the Phase A isolation signal are closed by the Phase B isolation signal. Normally closed motor-operated Containment Spray System valves are opened by the high-high containment pressure signal to permit Containment Spray System operation.

4. The CV68 and CV69 vlaves do not receive containment isolation signals (Phase A or Phase B). These valves get a close signal on a Safety Injection (SI) signal.

Lines which penetrate the containment and are normally closed by means of valves under administrative control are assumed to be already closed and do not receive an isolation signal.

Automatic containment isolation valves can be actuated from the control room if any of the valves fail to close in response to the Phase A or Phase B isolation signal.

6.2.4.3 Design Evaluation

The following provisions apply to all lines penetrating the containment to prevent inadvertent opening of these lines to the atmosphere outside the containment:

1. Automatic isolation valves can be opened only upon manual reset of the solid state logic without cessation of the actuating signal.

2. Automatic isolation valves are capable of manual actuation from the control room with the limitations for reopening of the valve noted in Item 1 (above).

3. Remote manual valves are operated only under administrative control.

4. Manual valves are operated under administrative control.

5. Check valves open only when the fluid pressure is higher on the side outside the containment.

6. The design pressure of all piping and connecting components within the isolation boundary is not less than the design pressure of the containment, 47 psig.

7. Automatic valves, once opened by a safety injection signal, can only be closed upon cessation and manual reset of the actuating signal.

For Items 1, 2, 3, and 4 (above), and for flanged closures, specific administrative procedures define the positioning of these closures in the Containment Isolation System during normal operation, shutdown, and accident conditions.

Instrumentation and adjunct control circuits associated with air operated automatic isolation valve closures are fail safe upon loss of voltage and/or control air. Such valves fail closed on loss of voltage, except for the outside containment isolation valves for the control air system (11, 12, 21 and 22CA330). The CA330's fail closed on loss of air, but fail as-is on loss of vital DC power. The control air system isolation valves inside containment (11, 12, 21 and 22CA360 check valves) prevent any single active failure from resulting in loss of the containment isolation function. The air operated isolation valves are air to open, spring return, diaphragm operated; thus providing a fail safe design. The automatic isolation valves inside the containment will function properly under all accident conditions. The isolation valve closing force is provided by a spring; control air is applied to the diaphragm of the isolation valve to open it. To close the isolation valve, an electrically operated solenoid valve located in the air supply line to the isolation valve operator vents the control air applied to the isolation valve diaphragm through the solenoid to the containment atmosphere, causing the spring to close the automatic isolation valve. Since the spring side of the isolation valve diaphragm is also vented to the containment atmosphere, the spring will force the valve to close when the solenoid vents the air line. Circuits which control redundant automatic valves are redundant in the sense that no single failure will preclude isolation. Means are provided to periodically test the functioning of the automatic isolation equipment such as the set point of sensors, speed of response, and operability of fail safe features. The containment isolation instrumentation is discussed in Section 7.

Valves used for containment isolation are capable of tight shutoff against gas leakage from containment design pressure down to zero psig.

Isolation valves and equipment are protected from missiles and water jets originating from the RCS. Missile protection for isolation valves, actuators, and controls is provided by locating isolation valves between the polar crane wall and the containment wall or locating isolation valves outside the containment structure. The pressure sensing devices which detect high containment pressure are located outside the containment. Location of the pressure sensing devices outside the containment protects them from missiles developed by a LOCA. Isolation valves and piping or vessels which provide one of the isolation barriers outside the containment are similarly protected.

Radiological Basis for Isolation Valve Closure Time

The closure times for the containment isolation valves are such that, in the event of a LOCA, no significant release of radioactivity to the environment through containment penetrations can occur.

In evaluating possible radioactive releases during a LOCA, the only release pathways considered were through those normally open penetrations associated with open systems outside the containment which are connected to the RCS or are open to the containment atmosphere (see Table 6.2-10).

A loss of offsite power was assumed coincident with a LOCA. The diesel-generators were assumed to be ready for loading in 13 seconds. The closure time for motor-operated valves is 10 seconds. Therefore, the total closing time for these valves is 23 seconds. It is conservatively assumed that these valves remain completely open for the time required to activate and completely close them.

The closure time for air operated valves is conservatively estimated to be 10 seconds. Operation of these valves is initiated when a containment pressure of 4.0 psig is reached. A conservative estimate of the time required to attain this pressure, assuming a double ended cold break, is 3 seconds. Therefore, the total closing time for these air operated valves is 13 seconds. One exception is the isolation valve for containment pressure - vacuum relief. This valve has a closure time of 2 seconds, resulting in a total closing time of 5 seconds for this analysis.

The activity released to the containment during the time required to close all isolation valves is limited to that contained in the RCS prior to the accident. This is based on the time required to close the isolation valves being sufficiently small that no clad perforation would occur before the valves were completely closed.

Total LOCA doses calculated in Section 15 which include the contribution of release through isolation valves and are within the limit values of 10CFR50.67.

Hence, it is concluded that the containment isolation valve closure times are sufficiently short and that there is no undue risk to the health and safety of the public.

6.2.4.4 Tests and Inspections

Preoperational Tests

Preoperational tests were performed on all valves in lines which penetrate the reactor containment and perform a containment isolation function to verify operability and leaktightness.

Valve operability testing was conducted prior to leakage testing. Each isolation valve was tested to demonstrate proper closure of normally open valves (or opening and closing of normally closed valves) upon receipt of an isolation signal. Closure of containment isolation valves was accomplished by normal operation and without any preliminary exercising or adjustment.

Valve leakage testing was performed by local pressurization in accordance with the applicable requirements of 10CFR50, Appendix J, "Primary Reactor Containment Leakage Testing for Water Cooled Power Reactors," for Type C tests. Valves were pressurized with air or nitrogen to a pressure of 47 psig. Where practical, pressure was applied in the same direction as the valve would experience during performance of its safety function.

Valve leakage was determined by measurement of the rate of pressure loss or by the flowrate of makeup air or nitrogen required to maintain test pressure.

The containment integrated leakage rate test procedure identified vent and drain valves which were opened in order to ensure exposure of the system piping penetrating containment to the full containment test pressure differential. Certain lines in the Service Water (SWS), Component Cooling Water, and RHR Systems are required for containment environmental control or decay heat removal and were not included in the integrated leakage rate test. The isolation valves in these lines were tested separately using a Type C test and any detected leakage was added to the Type A containment integrated leakage rate test results.

The combined leakage rate for the isolation valves and the double penetrations was limited to less than 0.06 percent of the containment free volume per day.

Periodic Tests

Periodic operability and leakage tests on isolation valves will be conducted throughout the lifetime of the plant according to the schedule specified in 10CFR50 Appendix J, "Primary Reactor Containment Leakage Testing for Water Cooled Power Reactors." The periodic isolation valve tests will be performed in accordance with the requirements for preoperational testing.

The piping arrangement provided to test the leaktightness of each isolation valve consists of a monitoring tap on the main line downstream of the valve. To test for valve tightness, the main piping section upstream of the valve is pressurized and evidence of leakage is checked at the downstream tap. When not in use, the monitoring lines are valved closed at the open end. Test pressures will be applied from the same direction as the pressure existing when the valve is required to perform its safety function.

The operability of the majority of containment isolation valves is fully testable at power except for those valves listed below. The valves are checked for circuit continuity up to and including the valve actuator during power operation by use of the Solid-State Protection System (SSPS) output test cabinet.

> Containment isolation valves: CA-330, SJ-12, SJ-13, CV-7 CV-68, CV-69, CV-116, CV-284, CC-118, CC-131, CC-136, CC-187, and CC-190
>
> Main steam isolation valves: MS-167

All other valves can be operationally tested at power from the SSPS output test cabinet to simulate accident operating conditions and verify the valve closing logic. All valves can be tested from the main control console as operating conditions permit.

6.2.5 Combustible Gas Control

6.2.5.1 Hydrogen Production

Hydrogen accumulation in the containment atmosphere following the DBA can be the result of production from several sources. The potential sources of hydrogen are the zirconium-water reaction, corrosion of construction materials, and radiolytic decomposition of the emergency core cooling solution. The latter source, solution radiolysis, includes both core solution radiolysis and sump solution radiolysis.

6.2.5.1.1 Methods of Analysis

The quantity of zirconium which reacts with the core cooling solution depends on the performance of the ECCS. The criteria for evaluation of the ECCS require that the zircaloy-water reaction be limited to 1 percent by weight of the total quantity of zirconium in the core. Emergency Core Cooling System calculations have shown the zircaloy-water reaction to be less than 0.1 percent, much less than required by the criteria.

The use of aluminum inside the containment is limited, and is not used in safety-related components which are in contact with the recirculating core cooling fluid. Aluminum is much more reactive with the containment spray alkaline borate solution than other plant materials such as galvanized steel, copper and copper nickel alloys. By limiting the use of aluminum the aggregate source of hydrogen over the long term is essentially restricted to that arising from radiolytic decomposition of core and sump water. The upper limit rate of such decomposition can be predicted with ample certainty to permit the design of effective countermeasures.

It should be noted that the zirconium-water reaction and aluminum corrosion with containment spray are chemical reactions and thus essentially independent of the radiation field inside the containment following a LOCA. Radiolytic decomposition of water is dependent on the radiation field intensity. The radiation field inside the containment is calculated for the maximum credible accident in which the fission product activities given in TID-14844 (6) are used.

Two hydrogen generation calculations are performed: one using the Westinghouse model (7), the other using the AEC model discussed in Safety Guide 7 (8).

6.2.5.1.2 Zirconium-Water Reaction

The zirconium-water reaction is described by the chemical equation:

$$Zr + 2H_2O \rightleftharpoons Zr\ O_2 + 2H_2 + Heat$$

The hydrogen generation due to this reaction will be completed during the first day following the LOCA. The Westinghouse model assumes a 2-percent zirconium-water reaction and the AEC model assumes a 5-percent zirconium-water reaction. The hydrogen generated is assumed to be released immediately to the containment atmosphere.

6.2.5.1.3 Corrosion of Plant Materials

Oxidation of metals in aqueous solution results in the generation of hydrogen gas as one of the corrosion products. Extensive corrosion testing has been conducted to determine the behavior of various metals used in the containment in the emergency core cooling solution at DBA conditions. Metals tested include Zircaloy, Inconel, aluminum alloys, coppernickel alloys, carbon steel, galvanized carbon steel, and copper. Tests conducted at ORNL (9, 10) have also verified the compatibility of the various metals (exclusive of aluminum) with alkaline borate solution. As applied to the quantitative definition of hydrogen production rates, the results of the corrosion tests have shown that only aluminum will corrode at a rate that will significantly add to the hydrogen accumulation in the containment atmosphere.

The corrosion of aluminum may be described by the overall reaction:

$$2\ Al + 3\ H_2O \rightleftharpoons Al_2O_3 + 3H_2$$

Therefore, three moles of hydrogen are produced for every two moles of aluminum that is oxidized. (Approximately 20 standard cubic feet of hydrogen for each pound of aluminum corroded.)

The time-temperature cycle (Table 6.2-14) considered in the calculation of aluminum corrosion is based on a conservative step-wise representation of the postulated post-accident containment transient. The corrosion rate design curve is shown on Figure 6.2-47. Aluminum corrosion data points include the effects of temperature, alloy, and spray solution conditions. Based on these corrosion rates and the aluminum inventory given in Table 6.2-15, the contribution of aluminum following the DBA has been calculated. For conservative estimation, no credit was taken for protective shielding effects of insulation or enclosures from the spray, and complete and continuous immersion was assumed.

Calculations based on Safety Guide 7 are performed by allowing an increased corrosion rate during the final step of the post-accident containment temperature transient (Table 6.2-14) corresponding to 200 mils/yr (15.7 $mg/dm^2/hr$). The corrosion rates earlier in the accident sequence are the higher rates determined from Figure 6.2-47.

Hydrogen is also produced through the corrosion of zinc inside containment. Sources of zinc within containment are the following:

1. Cable trays and hangers

2. Conduit

3. Junction Boxes

4. Ductwork

These components are galvanized with approximately 2 oz/ft (2 of zinc, and the surface area and weight of zinc associated with each is as follows:

Item	Sq. Ft.	Weight of Zn or Zinc
1. Cable trays and hangers	35,000	4,375 lb
2. Conduit	15,000	1,875 lb
3. Junction Boxes	1,500	188 lb
4. Ductwork	35,000	4,375 lb
		10,813 lb

The corrosion rate of zinc as a function of temperature is shown on Figure 6.2-48.

The experimental data used as the basis for hydrogen production due to zinc was obtained from Reference 11.

6.2.5.1.4 Radiolysis

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

$$H_2O \rightleftharpoons H_2 + 1/2\ O_2$$

Of interest here is the quantitative definition of the rates and extent of radiolytic hydrogen production following the DBA.

An extensive program has been conducted by Westinghouse to investigate the radiolytic decomposition of the core cooling solution following the DBA. In the course of this investigation, it became apparent that two separate radiolytic environments exist in the containment at DBA conditions. In one case, radiolysis of the core cooling solution occurs as a result of the decay energy of fission products in the fuel. In the other case, the decay of dissolved fission products, which have escaped from the core, results in the radiolysis of the sump solution. The results of these investigations are discussed in Reference 12.

Core Solution Radiolysis

As the emergency core cooling solution flows through the core, it is subjected to gamma radiation by decay of fission products in the fuel. This energy deposition results in solution radiolysis and the production of molecular hydrogen and oxygen. The initial production rate of these species will depend on the rate of energy absorption and the specific radiolytic yields.

The energy absorption rate in solution can be assessed from knowledge of the fission products contained in the core, and a detailed analysis of the dissipation of the decay energy between core materials and the solution. The results of Westinghouse studies show essentially all of the beta energy will be absorbed within the fuel and cladding and that this represents approximately 50 percent of the total beta-gamma decay energy. This study shows further that of the gamma energy, a maximum of 7.4 percent will be absorbed by the solution in core. Thus, an overall absorption factor of 3.7 percent of the total core decay energy ($\beta + \gamma$) is used to compute solution radiation dose rates and the time-integrated dose. Table 6.2-16 presents the total decay energy ($\beta + \gamma$) of a reactor core, which assumes a full power operation time of 830 days prior to the accident. For the maximum credible accident case, the contained decay energy in the core accounts for the assumed TID-14844 release of 50 percent halogens and 1 percent other fission products. To be conservative, the noble gases have been assumed to remain in the core, whereas in reality, the noble gases are assumed by the TID-14844 model to escape to the containment vapor space where little or no water radiolysis would result from decay of these nuclides.

The radiolysis yield of hydrogen in solution has been studied extensively by Westinghouse and ORNL. The results of static capsule tests conducted by Westinghouse indicate that hydrogen yields much lower than the maximum of 0.44 molecule per 100 eV would be the case in-core.

With little gas space to which the hydrogen formed in solution can escape, the rapid back reactions of molecular radiolytic products in solution to reform water is sufficient to result in very low net hydrogen yields.

However, it is recognized that there are differences between the static capsule tests and the dynamic condition in-core, where the core cooling fluid is continuously flowing. Such flow is reasoned to disturb the steady-state conditions which are observed in static capsule tests, and while the occurrence of back reactions would still be significant, the overall net yield of hydrogen would be somewhat higher in the flowing system.

The study of radiolysis in dynamic systems was initiated by Westinghouse, which formed the basis for experimental work performed at ORNL. Both studies clearly illustrate the reduced yields in hydrogen from core radiolysis, i.e., reduced from the maximum yield of 0.44 molecule per 100 eV. These results were recently published (12, 13).

For the purpose of this analysis, the calculations of hydrogen yield from core radiolysis are performed with the very conservative value of 0.44 molecule per 100 eV. That this value is conservative and a maximum for this type of aqueous solution and gamma radiation is confirmed by many published works. The Westinghouse results from the dynamic studies show 0.44 to be a maximum at very high solution flow rates through the gamma radiation field. The referenced ORNL (13) work also confirms this value as a maximum at high flow rates. A. O. Allen (14) presents a very comprehensive review of work performed to confirm the primary hydrogen yield to be a maximum of 0.44 to 0.45 molecule per 100 eV.

On the foregoing basis, the production rate and total hydrogen produced from core radiolysis, as a function of time, has been conservatively estimated for the maximum credible accident case.

Calculations based on Safety Guide 7 assume a hydrogen yield value of 0.5 molecule per 100 eV and that 10 percent of the gamma energy produced from fission products in the fuel rods is absorbed by the solution in the region of the core.

Sump Solution Radiolysis

Another potential source of hydrogen assumed for the post-accident period arises from water contained in the reactor containment sump being subjected to radiolytic decomposition by fission products. In this consideration, an assessment must be made as to the decay energy deposited in the solution and the radiolytic hydrogen yield, much in the same manner as given above for core radiolysis.

The energy deposited in solution is computed using the following basis:

1. For the maximum credible accident, a TID-14844 release model (which is more conservative than the guidance of Regulatory Guide 1.183 for fission product release assumptions) is assumed where 50 percent of the total core halogens and 1 percent of all other fission products, excluding noble gases, are released from the core to the sump solution.

2. The quantity of fission product release is equal to that from a reactor operating at full power for 830 days prior to the accident.

3. The total decay energy from the released fission products, both beta and gamma, is assumed to be fully absorbed in the solution.

Within the assessment of energy release by fission products in water, account is made of the decay of halogens, and a separate accounting for the slower decay of the 1 percent other fission products. To arrive at the energy deposition rate and time-integrated energy deposited, the contribution from each individual fission product class was computed. The overall contributions from each of the two classes of fission products is shown in Table 6.2-17.

The yield of hydrogen from sump solution radiolysis is more nearly represented by the static capsule tests performed by Westinghouse and ORNL with the alkaline sodium borate solution. The differences between these tests and the actual conditions for the sump solution, however are important and render the capsule tests conservative in their predictions of radiolytic hydrogen yields.

In this assessment, the sump solution will have considerable depth, which inhibits the ready diffusion of hydrogen from solutions, as compared to the case with shallow-depth capsule tests. This retention of hydrogen in solution will have a significant effect in reducing the hydrogen yields to the containment atmosphere. The build-up of hydrogen concentration in solution will enhance the back reaction to formation of water and lower the net hydrogen yield, in the same manner as a reduction in gas to liquid volume ratio will reduce the yield. This is illustrated by the data presented on Figure 6.2-49 for capsule tests with various gas to liquid volume ratios. The data show a significant reduction in the apparent or net hydrogen yield from the published primary maximum yield of 0.44 molecule per 100 eV. Even at the very highest ratios, where capsule solution depths are very low, the yield is less than 0.30, with the highest scatter data point at 0.39 molecule per 100 eV.

With these considerations taken into account, a reduced hydrogen yield is a reasonable assumption to make for the case of sump radiolysis. While it can be expected that the yield will be on the order of 0.1 or less, a conservative value of 0.30 molecule per 100 eV has been used in the maximum credible accident case.

Calculations based on Safety Guide 7 do not take credit for a reduced hydrogen yield in the case of sump radiolysis and a hydrogen yield value of 0.5 molecule per 100 eV has been used.

6.2.5.1.5 Coatings

Keeler & Long's group of Nuclear Level One Qualified Coatings, which consist exclusively of epoxy products, have been used on carbon steel components in the nuclear reactor containment. These coating products afford resistance of the steel substrate to corrosion caused by accidental spillage, environmental agents and DBA conditions of temperature, moisture and chemistry. Tests conducted by the Franklin Institute Research Laboratories (15) cover the examination of the coating systems used in the reactor containment. These examinations were for chalking, flaking, peeling, craking, checking and rusting. Tests conducted by Keller & Long and ORNL (16), in compliance with ANSI N101.2-1972 and ANSI N512-1974 also demonstrate that the coating systems are virtually unaffected by exposure to DBA test conditions.

The question of hydrogen gas generation from coating systems exposed to DBA conditions is of concern only in cases where zinc-based coatings materials have been used on carbon steel components inside the reactor containment. Epoxy coatings, such as those used on carbon steel components inside the Salem containments, are not considered to be prone to hydrogen gas generation.

Non-ferrous surfaces have been primed with a thin coat (0.5 mil dry thickness) of Con-Lux vinyl wash primer and catalyst 286.3 followed by a coat of Phenoline 305 finish. This primer contains 3.08% zinc by weight. When spread at a rate less than one mil wet thickness, the amount of zinc contained in this material is negligible inside the containment where the area of non-ferrous surfaces is approximately 86,500 sq. ft.

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6.2.5.1.6 Chemical and Volume Control System

The source of hydrogen from the Chemical and Volume Control System is automatically cut-off upon receipt of a safety injection signal.

6.2.5.1.7 Results

The results of the calculations for hydrogen production and accumulation from zirconium-water reactions, aluminum corrosion and radiolytic decomposition of core and sump solution are shown on Figures 6.2-50, 6.2-51, 6.2-52, and 6.2-53.

Figures 6.2-50 and 6.2-51 show the hydrogen production rate as a function of time following a LOCA up to 100 days for the maximum credible accident. Similar information for the first 10 days is shown on Figure 6.2-54.

Figures 6.2-52 and 6.2-53 show the total quantity of hydrogen accumulated in the containment as a function of time for the maximum credible accident case up to 100 days. The contribution of the individual source is also shown (note that zinc corrosion is not included).

Figure 6.2-55 shows the hydrogen production rate from aluminum and zinc corrosion for the first 10 days following a LOCA.

Total hydrogen accumulated from all sources inside containment was reanalyzed following the Three Mile Island (TMI) accident to show compliance with 10CFR50.44.

The requirements for a hydrogen control system to mitigate a hydrogen release were eliminated when 10CFR50.44 was revised and it no longer defined a design-basis LOCA hydrogen release.

6.2.5.2 Hydrogen Control

This section has been deleted based on Technical Specification Amendment numbers 281 and 264 to Facility Operating License numbers DPR-70 and DPR-75.

6.2.5.2.1 Hydrogen Recombiner Description

This section has been deleted based on Technical Specification Amendment numbers 281 and 264 to Facility Operating License numbers DPR-70 and DPR-75.

6.2.5.2.2 Recombiner Test Program

This section has been deleted based on Technical Specification Amendment numbers 281 and 264 to Facility Operating License numbers DPR-70 and DPR-75.

6.2.5.2.3 Recombiner Inservice Testing

This section has been deleted based on Technical Specification Amendment numbers 281 and 264 to Facility Operating License numbers DPR-70 and DPR-75.

6.2.5.2.4 Hydrogen Purge

There is no specific controlled hydrogen purge capability in the Salem design, other than the three different and independent purge modes described in Section 9.4. There is, however, an inherent "backup" in the multiple exhaust fans and filters that are available in the Purge System. Purge System valve actuation periodic surveillance requirements are included in the Technical Specifications.

6.2.5.3 Hydrogen Monitoring

A Hydrogen Monitoring System is provided for continuous measurement of hydrogen concentration at two sample locations within containment. Data from sample locations allows for diagnosing beyond design basis accidents. The system is designed in accordance with NUREG-0737 and Regulatory Guide 1.97.

The analyzing unit is mounted inside containment such that only electrical penetrations are required. Equipment located inside containment is operable under post-accident conditions of pressure, temperature, and radiation. All system components are seismically designed.

Hydrogen concentration is measured by a hydrogen partial pressure sensor in conjunction with a total pressure sensor. The partial pressure sensor is galvanic in nature, consisting of a platinum black electrode and a platinum oxide counter electrode within a polysulfone housing.

The range of measurement is 0 to 10 volume percent with an accuracy of 2 percent of full scale. Output is displayed in one Control Room. Alarms are provided for high hydrogen concentration, power failure, system error, and calibration mode.

Power is supplied from vital sources.

In addition to the Hydrogen Monitoring System, hydrogen concentration may be determined by taking a grab sample using the containment air particulate detector (APD) skid.

In amendments 281 and 264 to Salem Units 1 and 2 Operating Licenses, a commitment was made to maintain the capability for the hydrogen monitoring system for diagnosing beyond design basis accidents. The functionality requirements of the containment hydrogen analyzers are contained in the Salem Technical Requirements Manual.

6.2.6 References for Section 6.2

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16. Report 78-0810-1, "Radiation Tolerance, Decontamination, Design Basis Accident, Physical Properties & Chemical Properties Tests for Carbon Steel & Concrete Coating Systems (1977 ORNL Test Series)," Keller & Long, Inc., August 1978.

17. Deleted.

18. Deleted.

19. Cranston, G. V., "Testing Criteria for Integrated Leakage Rate Testing of Primary Containment Structures for Nuclear Power Plants," Bechtel Topical Report BN-TOP-1 Revision 1, November 1, 1972

20. United States Atomic Energy Commission, Letter from R. C. DeYoung to R. D. Allen, February 1, 1973.

21. Nuclear Safety Advisory Letter NSAL-93-016, Revision 1, "Containment Spray System Issues," Westinghouse, October 4, 1993.

22. Westinghouse Letter PSE-94-500, J. Huckabee to E. S. Rosenfeld, PSE&G, "Small Break LOCA Offsite Dose Analysis," January 5, 1994.

23. VTD 326657, Cooling Fan Motor Drawing, August 20, 1994.

24. USNRC Regulatory Guide 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors", July 2000.

25. Section 50.67 of title 10 of the Code of Federal Regulations (10CFR 50.67), "Accident Source Term".

26. USNRC Standard Review Plan Section 6.5.2, "Containment Spray as a Fission Product Cleanup System", Revision 2, 1988.

27. Technical Specification Amendment Numbers 281 and 264 to Facility Operating License numbers DPR-70 and DPR-75.

TABLE 6.2-1

CONTAINMENT SPRAY SYSTEM - CODE REQUIREMENTS

Component	Code
Spray Additive Tank	ASME Section VIII
Valves	ANSI B16.5
Piping (including headers and spray nozzles) pumps	ANSI B31.1[(1)]

(1) For piping not supplied by the NSSS supplier, material inspections, fabrication and quality control conform to ANSI B31.7. Where not possible to comply with ANSI B31.7, the requirements of ASME III-1971, which incorporated ANSI B31.7, were adhered to.

TABLE 6.2-2

CONTAINMENT SPRAY PUMP DESIGN PARAMETERS

Quantity	2
Design Pressure, discharge, psig	250
Design Temperature, °F	150
Design Flow Rate, gpm	2600
Design Head, ft	450
Shutoff Head, ft	~530
Motor, hp	400
Type	Horizontal-Centrifugal

TABLE 6.2-3

SPRAY ADDITIVE TANK DESIGN PARAMETERS

Number	1
Total Volume (empty), gal.	4000
NaOH concentration, w/o	30
Design temperature, °F	300
Design pressure, psig	14
Material	Austenitic Stainless Steel

TABLE 6.2-4

SINGLE FAILURE ANALYSIS - CONTAINMENT SPRAY SYSTEM

	Component	Malfunction	Comments and Consequences
A.	Spray Nozzles	Clogged	Large number of nozzles render clogging of a significant number of nozzles as incredible.
B.	Pumps		
	1) Containment Spray Pump	Fails to start	Two provided. Evaluation based on operation of one pump in addition to three out of five containment cooling fans operating during injection phase.
	2) Residual Heat Removal Pump	Fails to start	Two provided. Evaluation based on operation of one pump.
	3) Service Water Pump	Fails to start	Six provided. Operation of two pumps during recirculation required.
	4) Component Cooling	Fails to start	Three provided. Operation of one pump during recirculation required.
C.	Automatically Operated Valves: (Open on two out of four (HiHi) containment pressure signals)		
	1) Containment spray pump discharge isolation valve	Fails to open	Two complete systems provided.
D.	Valves Operated From Control Room		
	(a) Injection		
	1) Spray Additive Tank Outlet Isolation Valve	Fails to open	Two parallel valves provided. Operation of one required.

TABLE 6.2-4 (cont)

Component	Malfunction	Comments and Consequences
(b) Recirculation		
1) Containment Sump Isolation Valve	Fails to open	Two lines in parallel. One line required.
2) Containment Spray Header Isolation Valve from Residual Heat Exchangers	Fails to open	Two complete loops provided. Operation of one required.

TABLE 6.2-5

SHARED FUNCTIONS EVALUATION

Component	Normal Operating Function	Normal Operating Arrangement	Accident Function	Accident Arrangement
Spray Additive Tank	None	Lined up for spray water diversion	Source of sodium hydroxide for spray water	Lined up for spray water diversion
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, residual heat removal pumps, service water pumps, component cooling pumps, residual heat exchangers and component cooling heat exchangers which are also associated either directly or indirectly with the Containment Spray System.

TABLE 6.2-6

NET POSITIVE SUCTION HEADS FOR CONTAINMENT SPRAY

Pump	Elevation	Flow and Condition	Suction Source and Elevation	Minimum Available NPSH	Required NPSH	Maximum Water Temperature
Containment Spray	86'-3"	2600 gpm Rated flow	RWST 101'-8"	29.9'	10'	100°F
Residual Heat Removal (Unit 1 one Pump operation)	46'-10"	4850 gpm Recirculation Spray flow	Containment Sump 81'-8"	28.1'	22'	Saturation
Residual Heat Removal (Unit 2 one Pump operation)	46'-10"	4850 gpm Recirculation Spray flow	Containment Sump 81'-8"	25.7'	22'	Saturation

The available NPSH was calculated for the pumps indicated above using the following conservative assumptions:

1. All calculations assume an empty refueling water storage tank.
2. No credit is taken for RWST fluid temperature below 100°F.
3. No credit is taken for increased containment pressures following the LOCA.

TABLE 6.2-7

SINGLE FAILURE ANALYSIS - CONTAINMENT FAN COOLING SYSTEM

Component	Malfunction	Comments and Consequences
Containment Cooling Fan	Fails to start	Five provided. Evaluation based on three fans in operation and one containment spray pump operating during the injection phase.
Service Water Pumps	Fails to start	Six provided. Two required for operation.
Automatically Operated Valves	Fails to operate as required	Five RCFC units are provided. A failure of one valve to operate as required will result in no more than one RCFC becoming inoperable. Evaluations have demonstrated that three RCFC units in operation and one Containment Spray Pump operating, provide sufficient cooling during the injection phase of a LOCA event.

TABLE 6.2-8

SHARED FUNCTION EVALUATION

Component	Normal Operating Function	Normal Operating Arrangement	Accident Function	Accident Arrangement
Containment Cooling Fan Units (5)	Circulate and cool containment atmosphere	Up to four fan units in service	Circulate and cool containment atmosphere	Five fan units in service
Service Water Pumps (6)	Supply river cooling water to fan units	Four pumps in service	Supply river cooling water to fan units	Two pumps in service

TABLE 6.2-9

SPRAY EVALUATION PARAMETERS

Containment Pressure, psia	61.7
Containment Temperature °F	271
Injection Spray flow rate, gpm	2600
Recirculation Spray flow rate, gpm	1900
Injection Spray pH	8.5 to 10.0
Containment free volume, ft^3	2.6×10^6
Spray fall height, ft	116
Minimum spray coverage	0.75

Iodine spray removal coefficient λ_s credited in radiological evaluation during injection phase:

λ_s (hr^{-1}) (DF < 100) elemental	20
λ_s (hr^{-1}) (DF < 50) particulate	4.44

λ_s credited during the transition from injection phase to recirculation phase (i.e., removal is not credited):

λ_s (hr^{-1}) elemental & particulate	0.0

λ_s credited in radiological evaluation during recirculation phase:

λ_s (hr^{-1}) (DF < 100) elemental	14.6
λ_s (hr^{-1}) (DF < 50) particulate	3.24
λ_s (hr^{-1}) (DF > 100) elemental	0.0
λ_s (hr^{-1}) (DF > 50) particulate	0.32
λ_s (hr^{-1}) (>4 hours) particulate	0.0

TABLE 6.2-10
CONTAINMENT ISOLATION - MAJOR PIPING PENETRATIONS

THE INFORMATION CONTAINED IN THIS TABLE WAS RELOCATED TO THE SALEM TECHNICAL REQUIREMENTS MANUAL

TABLE 6.2-11

This page intentionally deleted

TABLE 6.2-12

This Table has been deleted

TABLE 6.2-13

This Table has been deleted

TABLE 6.2-14

POST-ACCIDENT CONTAINMENT TEMPERATURE TRANSIENT
USED IN THE CALCULATION OF ALUMINUM CORROSION

Time Interval (sec)	Temperature (°F)
0 - 300	271
300 - 1000	230
1000 - 2000	188
2000 - 4000	175
> 4000	147

TABLE 6.2-15

INPUT PARAMETERS AND ALUMINUM INVENTORY
PARAMETERS USED TO DETERMINE HYDROGEN GENERATION

Plant Thermal Power Rating	3575 MWt
Containment Temperature at Accident	120°F
Containment Free Volume	2,500,000 ft^3
Weight of Zirconium	47,946 lb
Hydrogen Generated by Zirc-Water Reaction	
Based on 2 percent value	7,575 SCF
Based on 5 percent value	18,940 SCF
Corrodible Metal	Aluminum

INVENTORY OF ALUMINUM IN CONTAINMENT (NUCLEAR STEAM SUPPLY SYSTEM)

Item	Weight (lbs)	Surface Area (ft^2)
Source, Intermediate and Power	244	83
Control Rod Drive Mechanism Connectors	193	42
Paint	140	18,000
Contingency (Nuclear Steam Supply System)	250	85
Flux Mapping Drive System	122	84
Miscellaneous Valves	230	86
CRDM Ventilation System Fan Motor Rotor [23] (Unit 1 & Unit 2)	71	3
Tri-band Antennas	4	
Permanent Shielding (Carabiners)	175	

TABLE 6.2-16

CORE FISSION PRODUCT ENERGY
AFTER 830 FULL POWER DAYS

Core Fission Product Energy/[1]

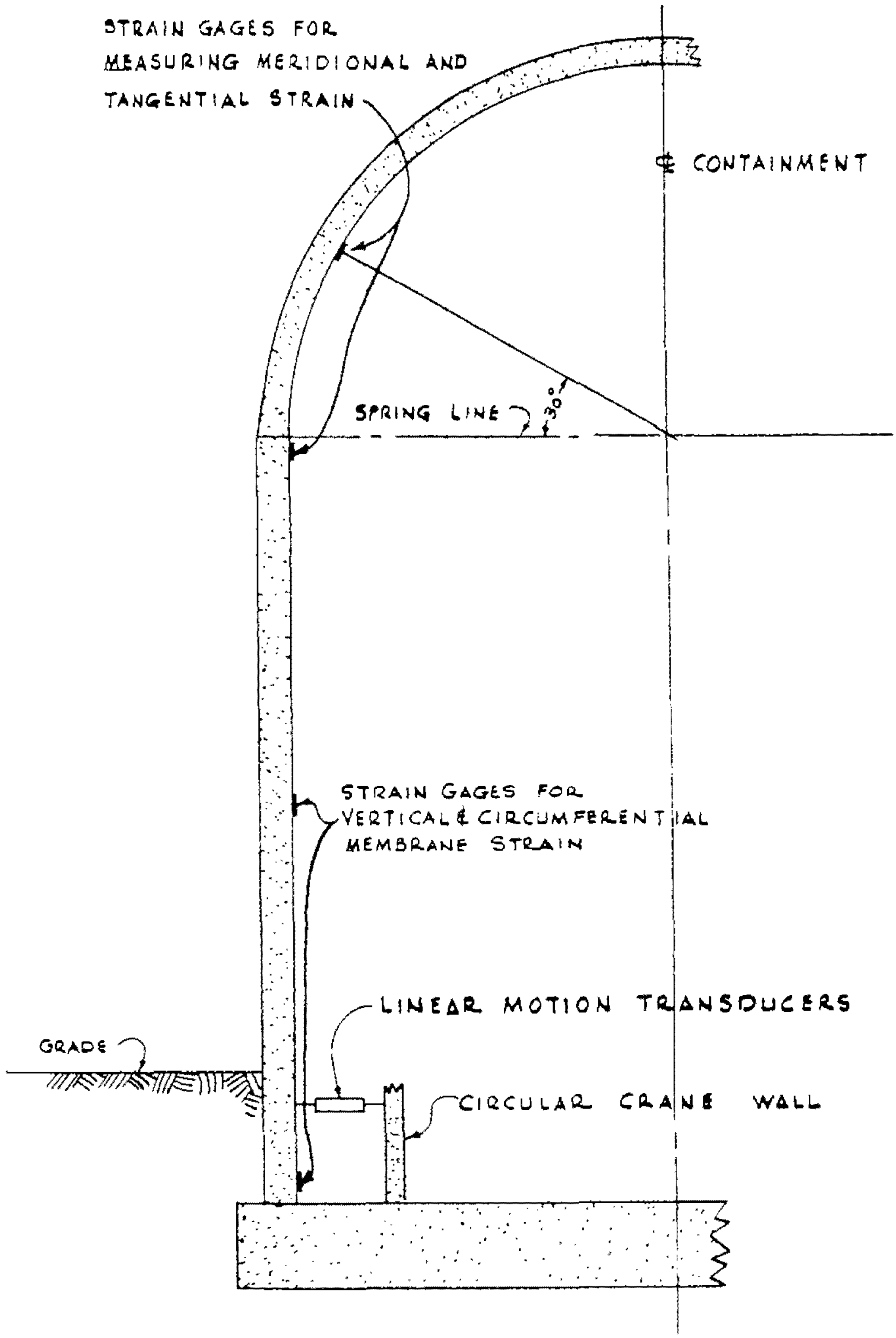
Time After Reactor Trip Days	Energy Release Rate Watts/MWtx10^{-3}	Integrated Energy Release Watt-Days/MWtx10^{-4}
1	3.887	0.574
5	2.595	1.777
10	2.211	2.967
20	1.760	4.934
30	1.475	6.541
40	1.291	7.919
50	1.163	9.143
60	1.068	10.259
70	0.992	11.289
80	0.926	12.249
90	0.867	13.139
100	0.814	13.979

[1] Assumes release of 50 percent core halogens +1 percent other fission products, includes 100 percent noble gases. Values are for total (β and γ) energy.

TABLE 6.2-17

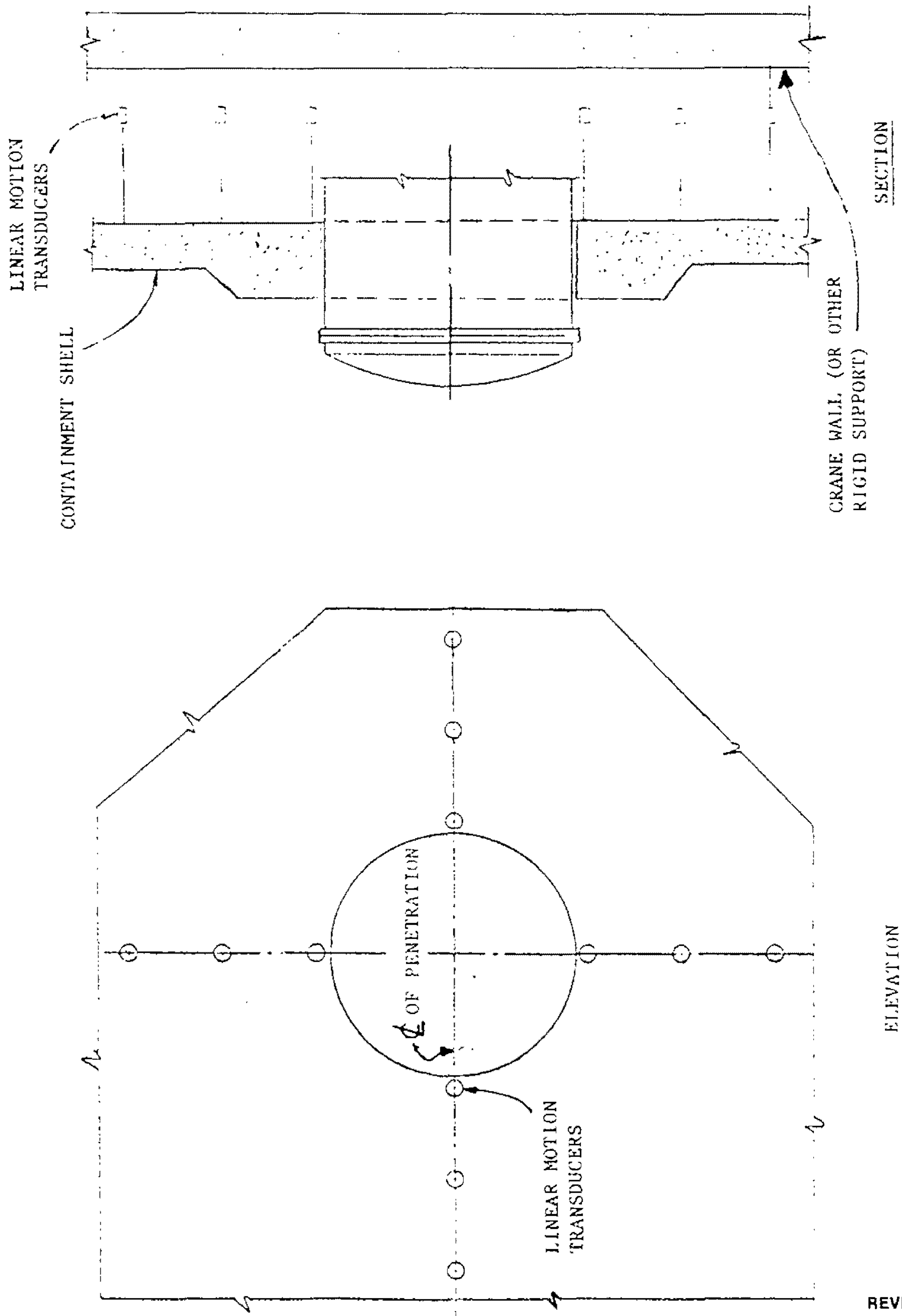
FISSION PRODUCT DECAY DEPOSITION IN SUMP SOLUTION

Time After Reactor Trip Days	50 Percent Halogens: Energy Release Rate Watts/MWt	50 Percent Halogens: Integrated Energy Release Watt-Day/MWtx10^{-2}	1 Percent Other Fission Products: Energy Release Rate Watts/MWtx10^{-1}	1 Percent Other Fission Products: Integrated Energy Release Watt-Day/MWtx10^{-2}	Total: Energy Release Rate Watts/MWtx10^{-1}	Total: Integrated Energy Release Watt-Day/MWtx10^{-3}
1	145	4.27	3.78	0.536	18.28	0.481
2	49.4	5.88	2.90	1.18	7.85	0.707
5	31.0	6.65	2.59	1.73	5.69	0.838
10	18.2	7.82	2.22	2.92	4.03	1.07
20	7.63	9.03	1.77	4.89	2.53	1.39
30	3.22	9.54	1.49	6.51	1.81	1.61
40	1.36	9.76	1.30	7.90	1.44	1.77
60	0.241	9.89	1.08	10.3	1.10	2.02
80	0.043	9.91	0.935	12.3	0.940	2.22
100	0.008	9.92	0.822	14.0	0.823	2.39



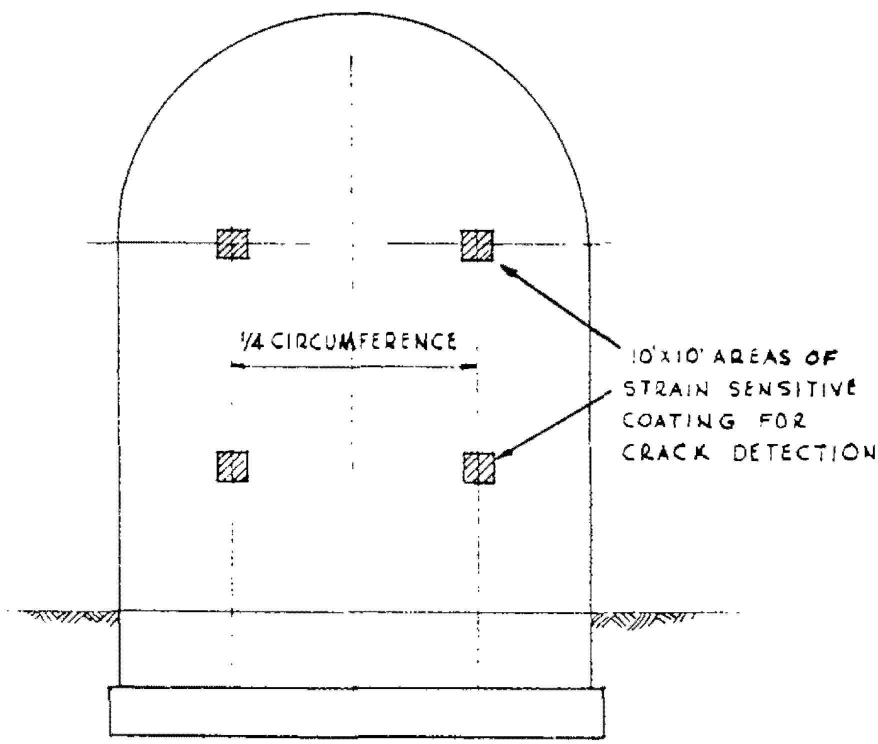
REVISION 6
FEBRUARY 15, 1987

PUBLIC SERVICE ELECTRIC AND GAS COMPANY SALEM NUCLEAR GENERATING STATION	Containment Instrumentation	
	Updated FSAR	Figure 6.2-1



REVISION 6
FEBRUARY 15, 1987

PUBLIC SERVICE ELECTRIC AND GAS COMPANY SALEM NUCLEAR GENERATING STATION	Large Penetration Instrumentation (Equipment and Personnel Hatches)
	Updated FSAR — Figure 6.2-2



REVISION 6
FEBRUARY 15, 1987

PUBLIC SERVICE ELECTRIC AND GAS COMPANY SALEM NUCLEAR GENERATING STATION	Location of Strain Sensitive Coatings on the Containment
	Updated FSAR — Figure 6.2-3

Figure F6.2-4A intentionally deleted.

Refer to plant drawing 205235 in DCRMS

Figure F6.2-4B intentionally deleted.

Refer to plant drawing 205335 in DCRMS

Figure F6.2-5 intentionally deleted.

Refer to VTD 121398 in DCRMS

Figure F6.2-6 intentionally deleted.

Refer to plant drawing 207446 in DCRMS

Figure F6.2-7 intentionally deleted.

Refer to plant drawing 207467 in DCRMS

Figure F6.2-8 intentionally deleted.

Refer to plant drawing 223112 in DCRMS

Figure F6.2-9 intentionally deleted.

Refer to plant drawing 223114 in DCRMS

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Refer to plant drawing 223123 in DCRMS

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Refer to VTD 142864 in DCRMS

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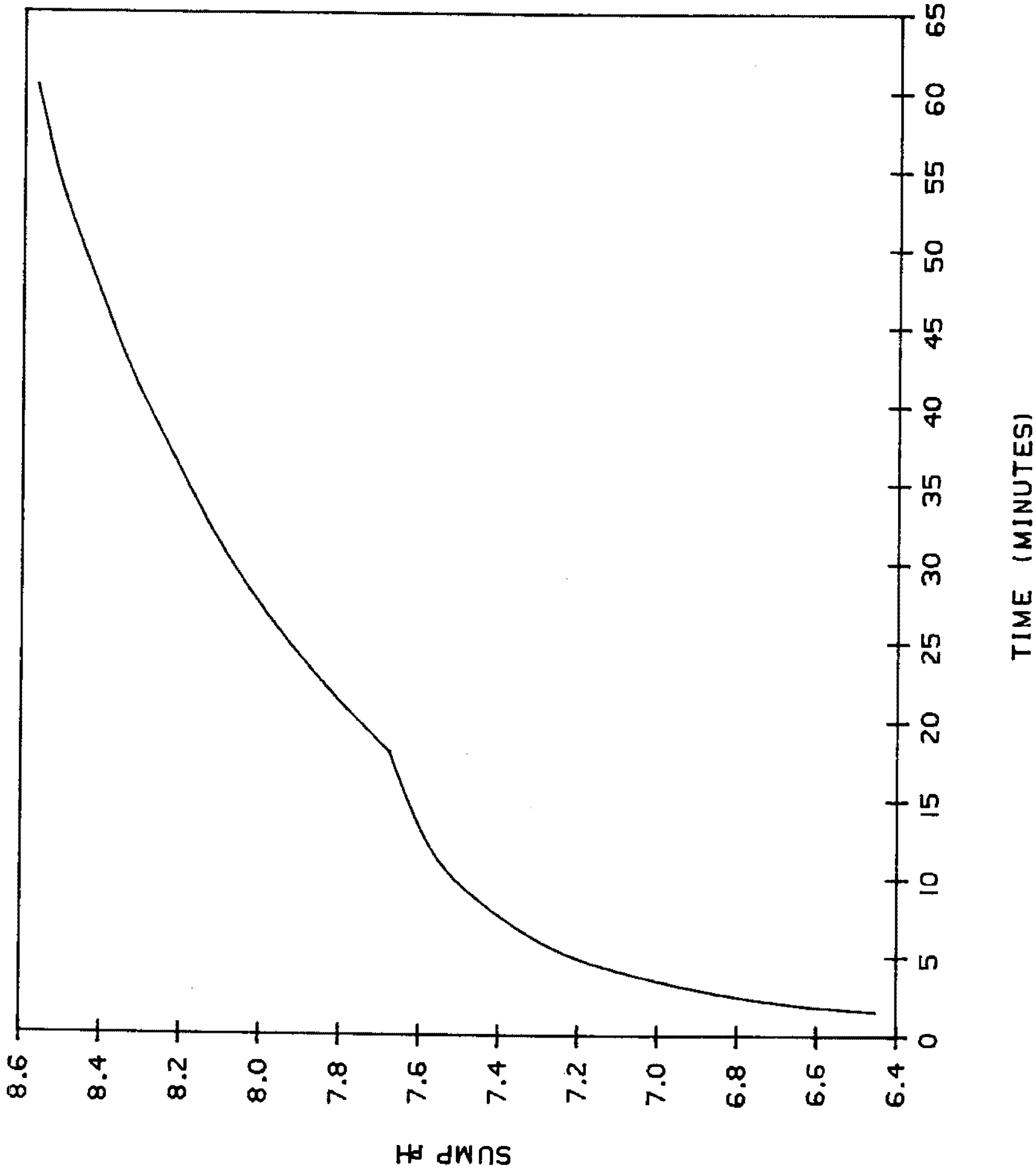
Refer to VTD 142850 in DCRMS

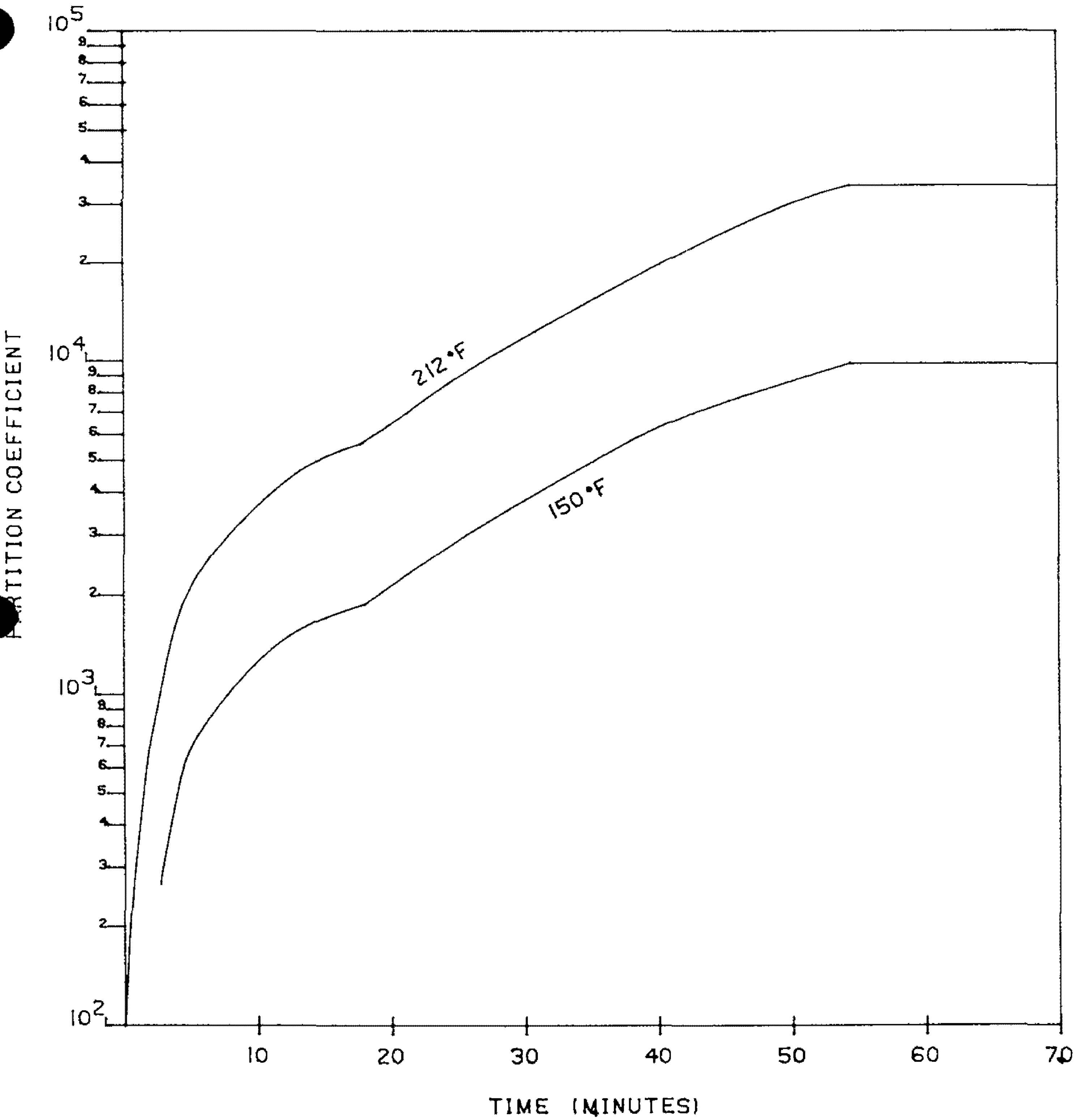
Figure F6.2-13 intentionally deleted.

Refer to plant drawing 224351 in DCRMS



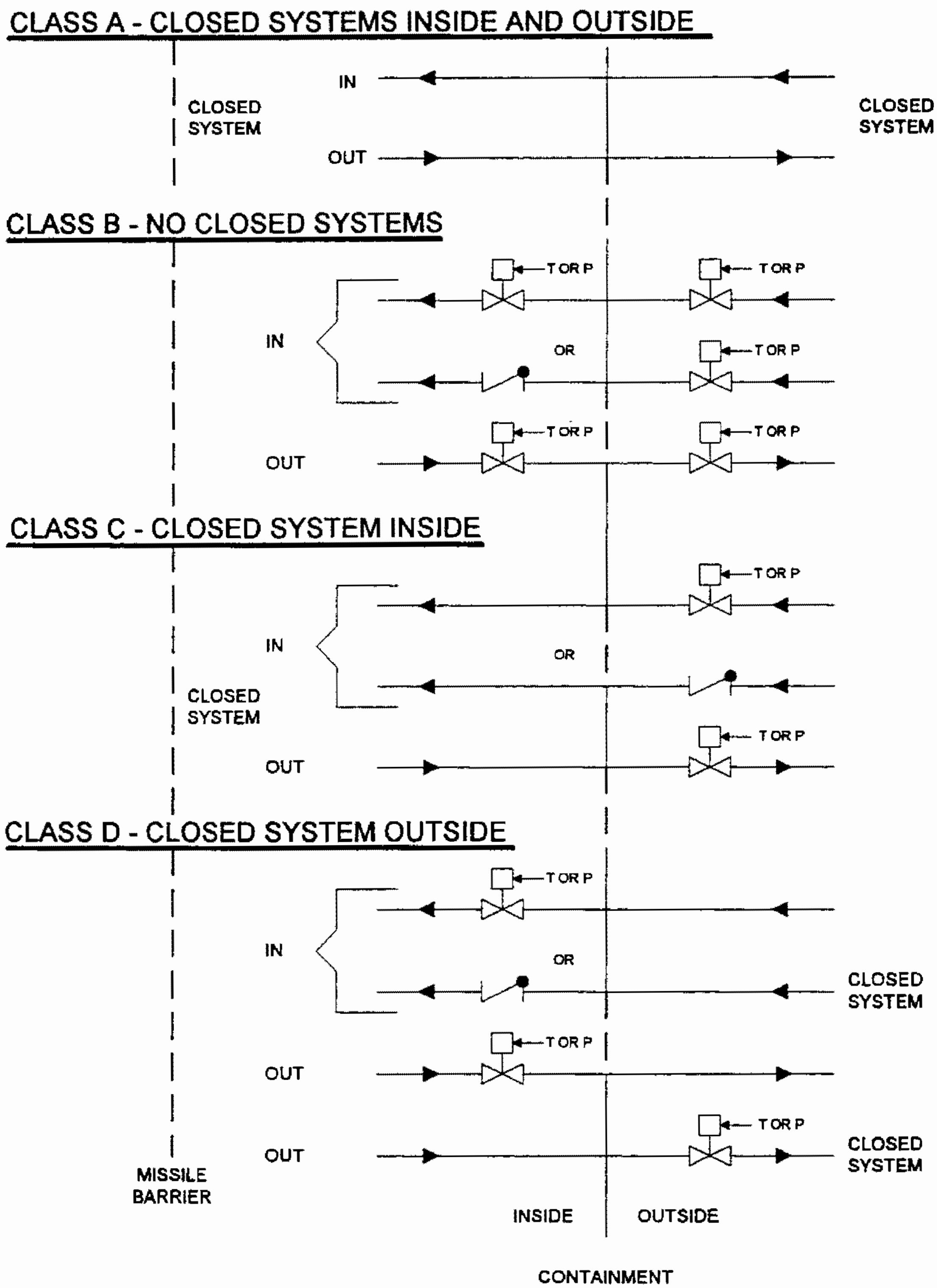
REVISION 6
FEBRUARY 15, 1987

PUBLIC SERVICE ELECTRIC AND GAS COMPANY SALEM NUCLEAR GENERATING STATION	Minimum Sump pH vs Time	
	Updated FSAR	FIG. 6.2-14

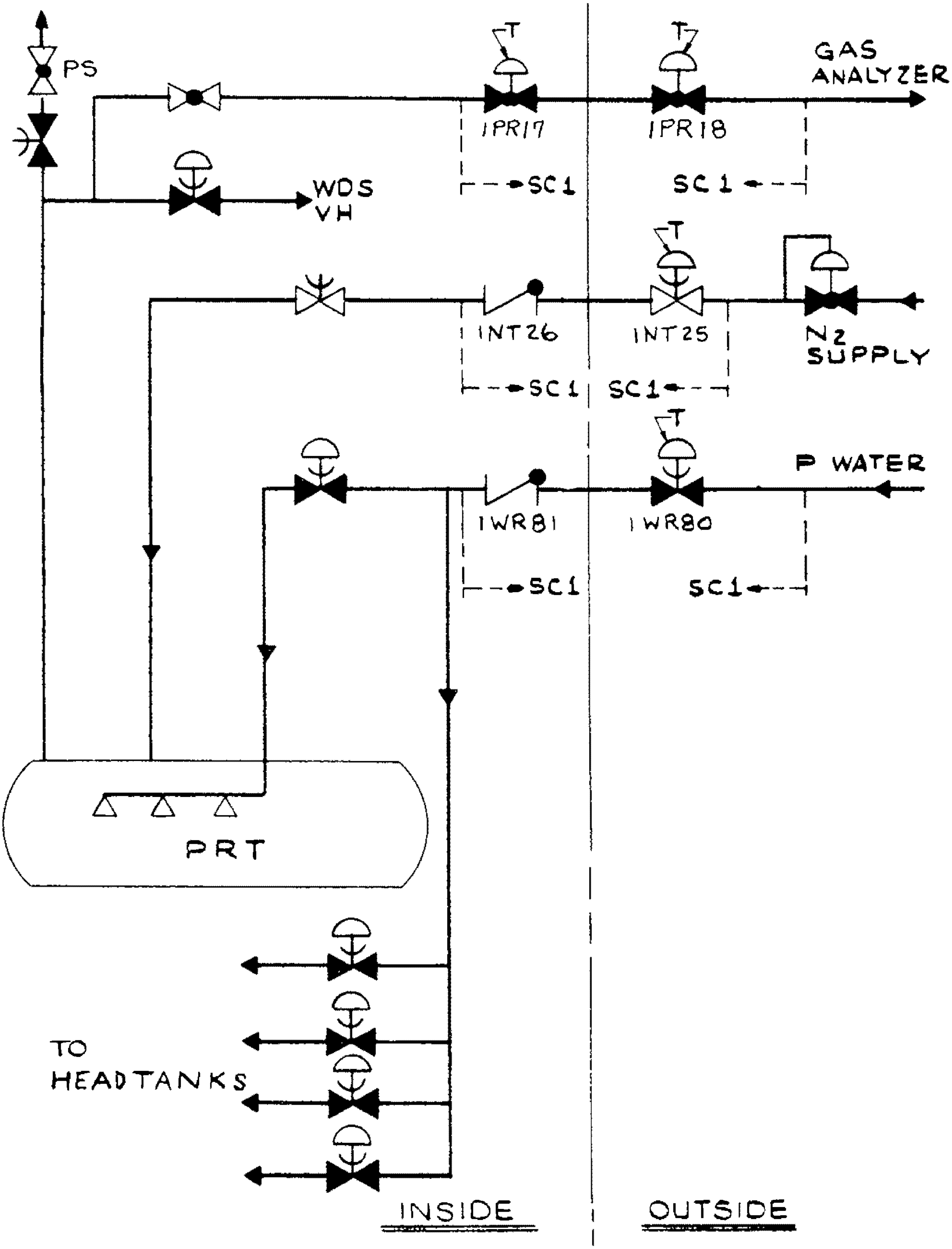


REVISION 6
FEBRUARY 15, 1987

PUBLIC SERVICE ELECTRIC AND GAS COMPANY SALEM NUCLEAR GENERATING STATION	Minimum Sump Partition Coefficient vs Time (Iodine Reaction not Included)	
	Updated FSAR	FIG. 6.2-15

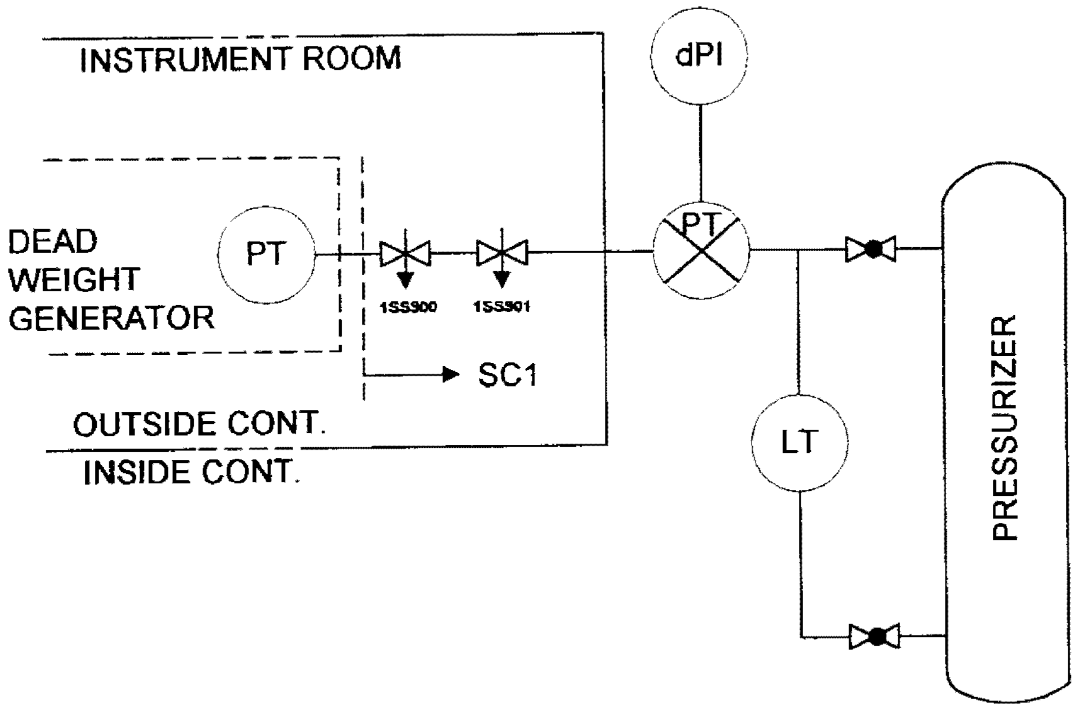


PUBLIC SERVICE ELECTRIC AND GAS COMPANY SALEM NUCLEAR GENERATING STATION	Containment Isolation Piping Classes Updated FSAR Revision 16 — Figure 6.2-16, January 31, 1998

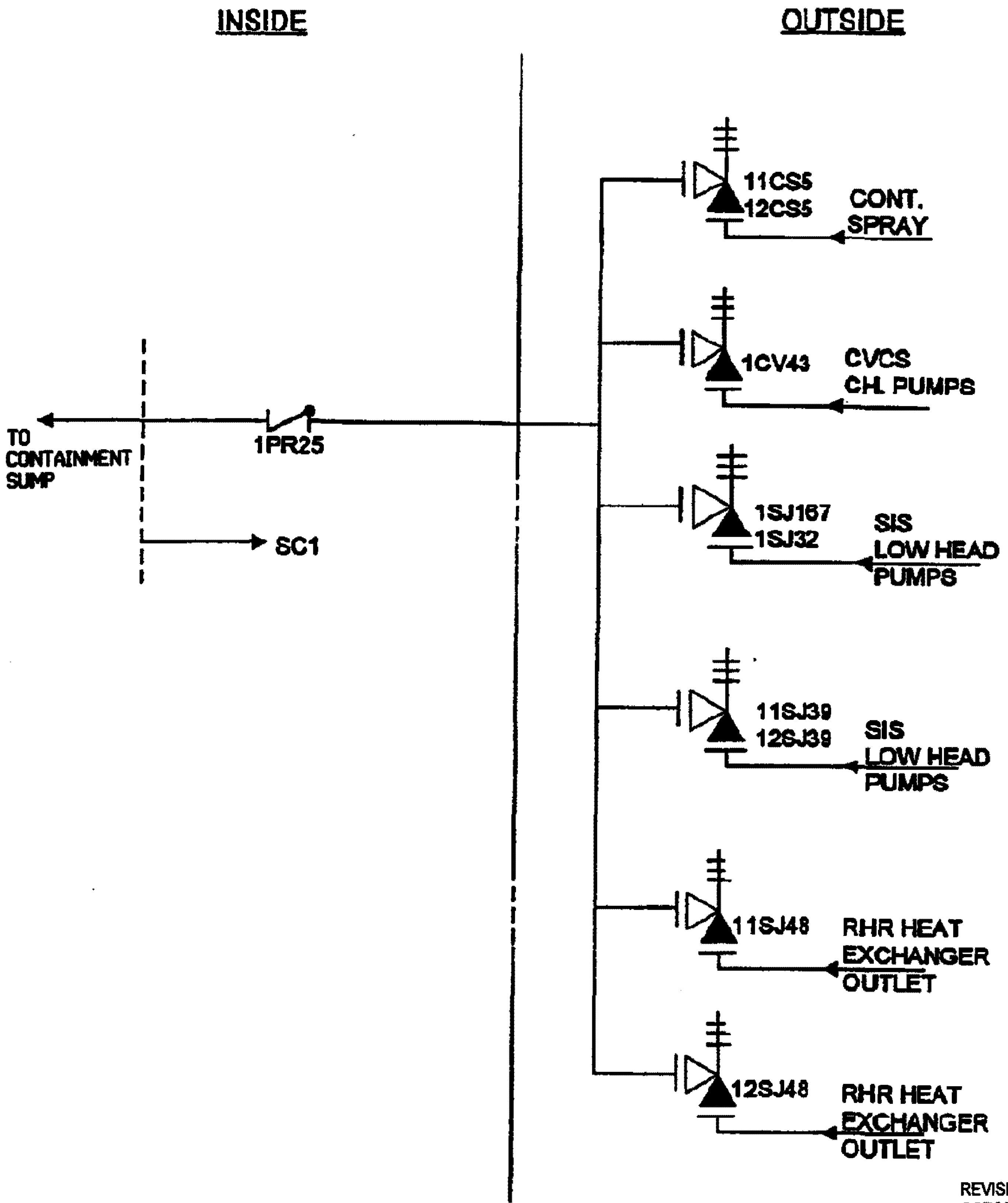


REVISION 6
FEBRUARY 15, 1987

PUBLIC SERVICE ELECTRIC AND GAS COMPANY SALEM NUCLEAR GENERATING STATION	Containment Isolation Pressurizer Relief Tank Connections
	Updated FSAR FIG. 6.2-17



PUBLIC SERVICE ELECTRIC AND GAS COMPANY SALEM NUCLEAR GENERATING STATION	Containment Isolation Dead Weight Calibrator
	Updated FSAR Revision 16 — Figure 6.2-18 January 31, 1998



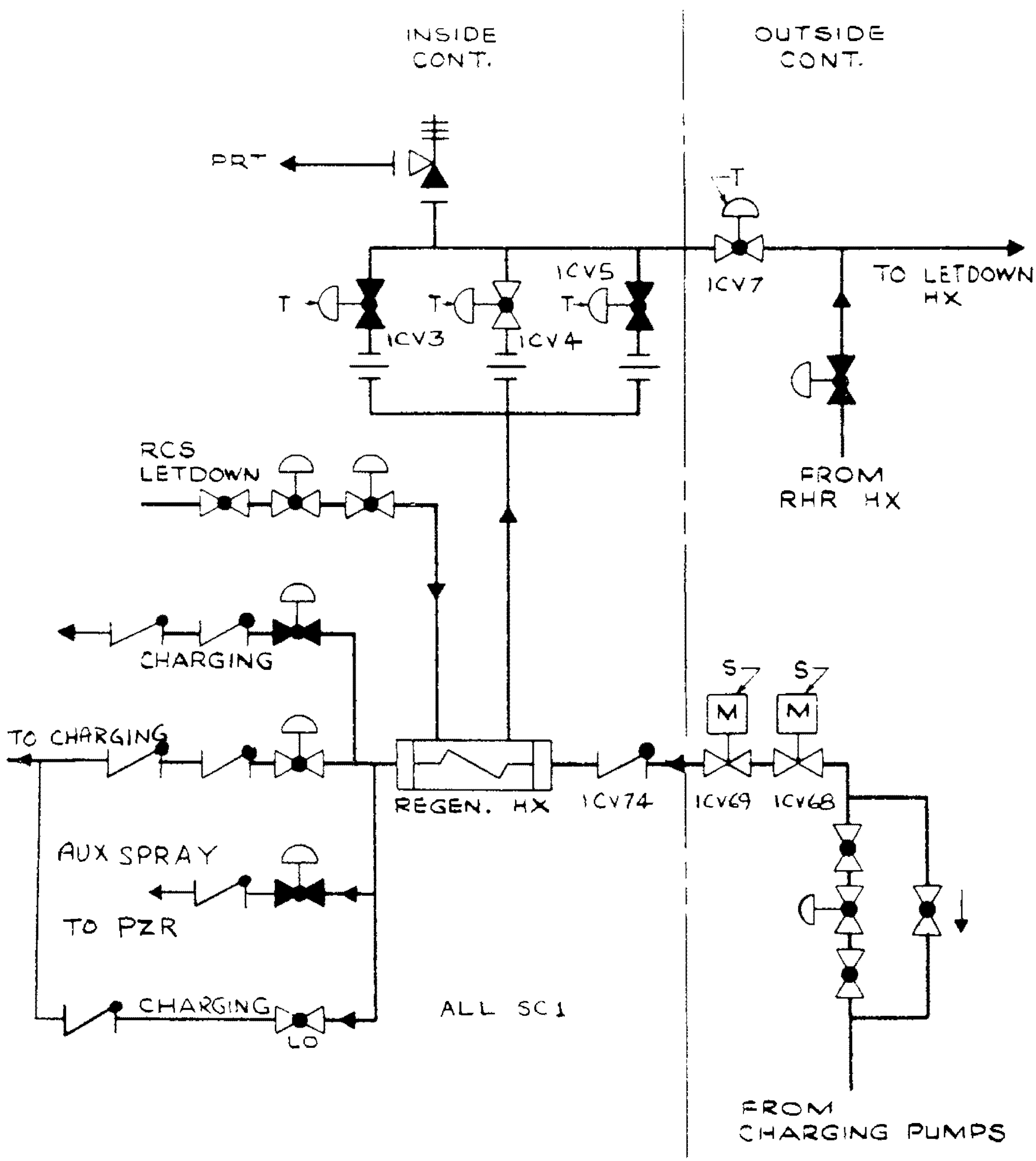
REVISION 17
OCTOBER 16 1998

PUBLIC SERVICE ELECTRIC AND GAS COMPANY
SALEM NUCLEAR GENERATING STATION

CONTAINMENT ISOLATION
RELIEF LINES TO CONTAINMENT SUMP

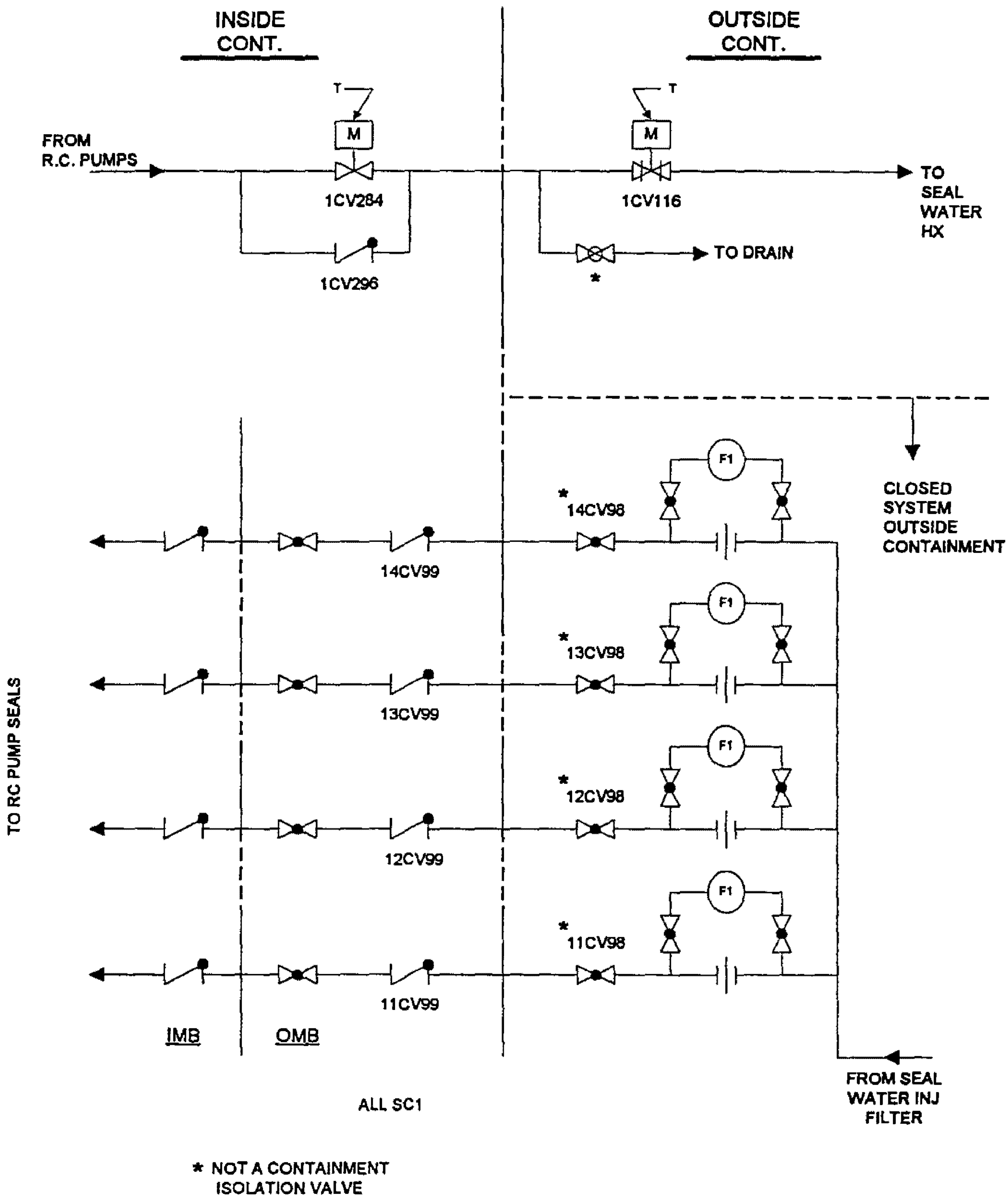
Updated FSAR

FIG. 6.2-19

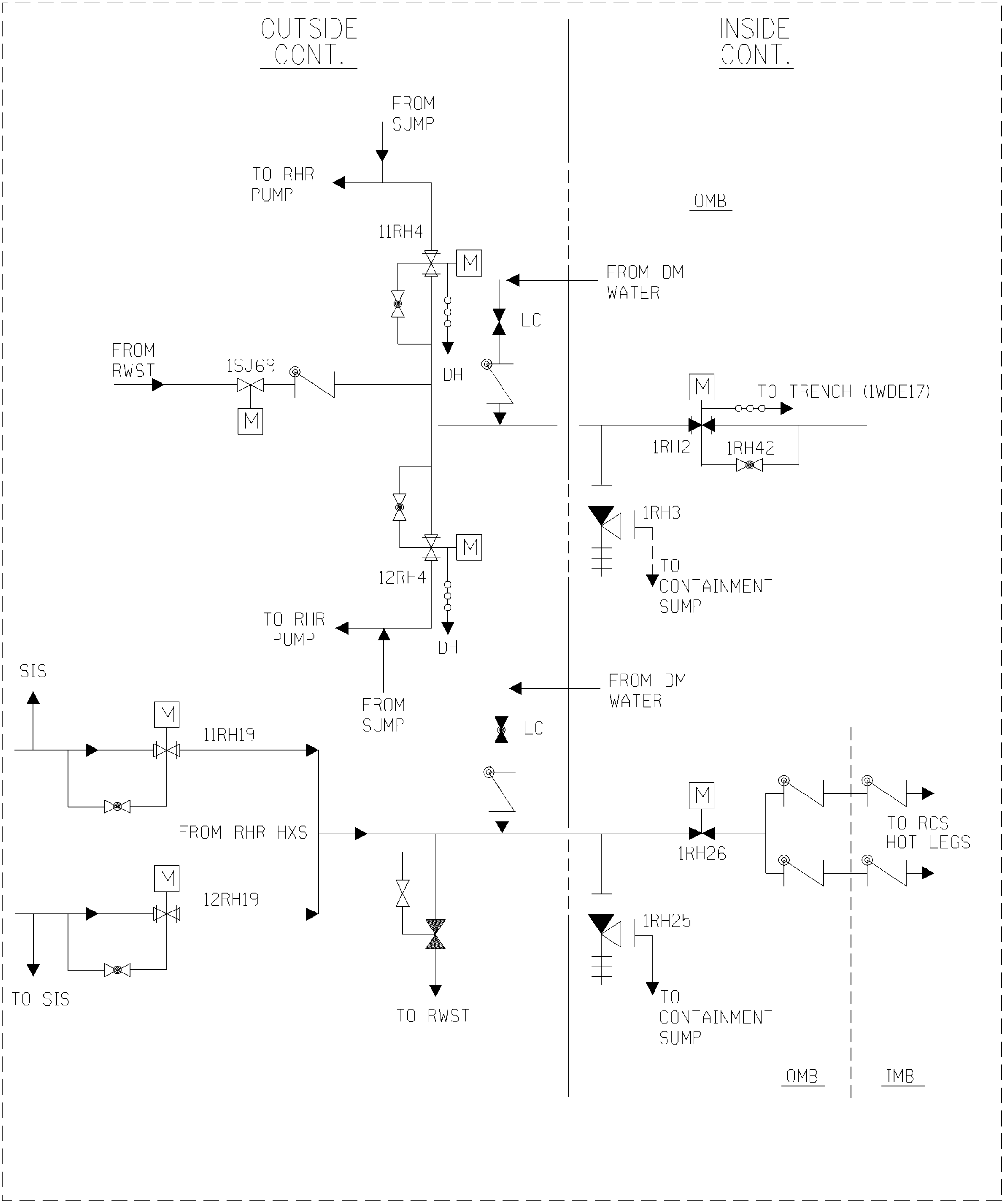


REVISION 6
FEBRUARY 15, 1987

PUBLIC SERVICE ELECTRIC AND GAS COMPANY SALEM NUCLEAR GENERATING STATION	Containment Isolation Letdown and Charging Lines
	Updated FSAR FIG. 6.2-20

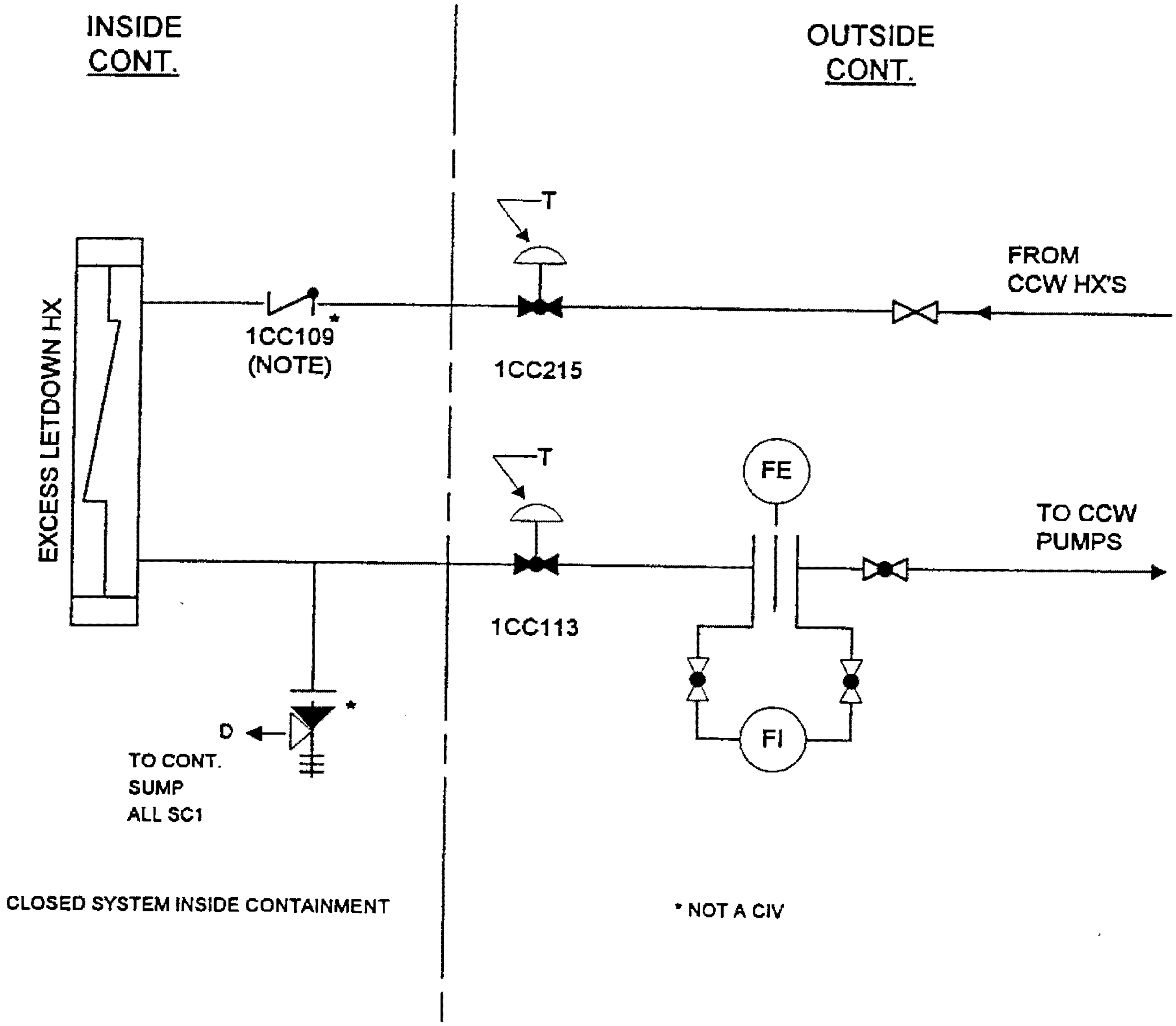


PUBLIC SERVICE ELECTRIC AND GAS COMPANY SALEM NUCLEAR GENERATING STATION	Containment Isolation Seal Water Supply and Return for R.C. Pumps
	Updated FSAR Revision 16 — Figure 6.2-21 January 31, 1998



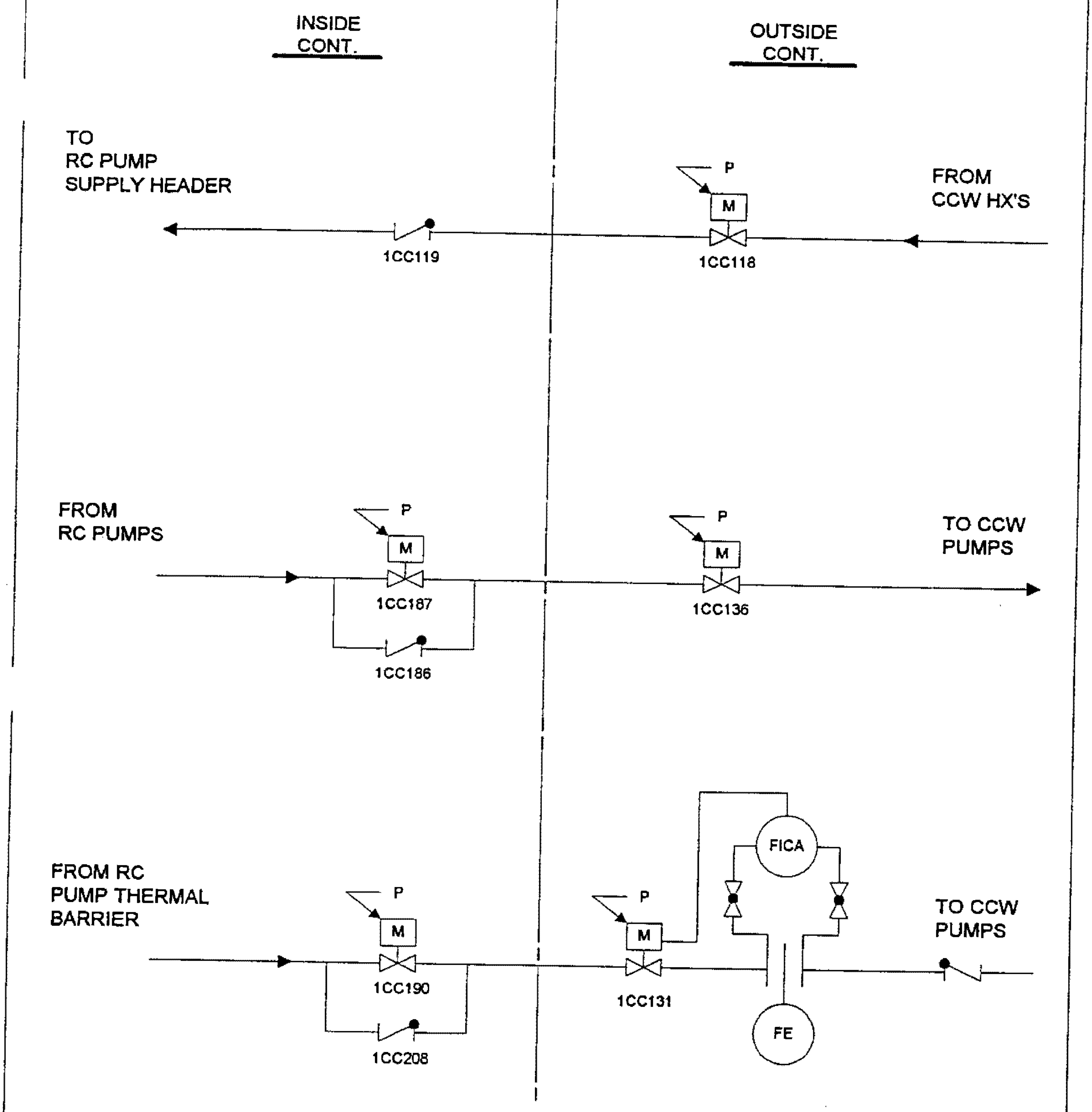
Revision 20, May 6, 2003

PSEG Nuclear, LLC SALEM NUCLEAR GENERATING STATION	Salem Nuclear Generating Station CONTAINMENT ISOLATION RESIDUAL HEAT REMOVAL CONNECTIONS
	Updated FSAR — Figure 6.2-22



NOTE: VALVE PROVIDED, BUT NOT REQUIRED FOR CONTAINMENT ISOLATION PURPOSES

PUBLIC SERVICE ELECTRIC AND GAS COMPANY SALEM NUCLEAR GENERATING STATION	Containment Isolation Component Cooling for Excess Letdown HX
	Updated FSAR Revision 16 — Figure 6.2-23 January 31, 1998



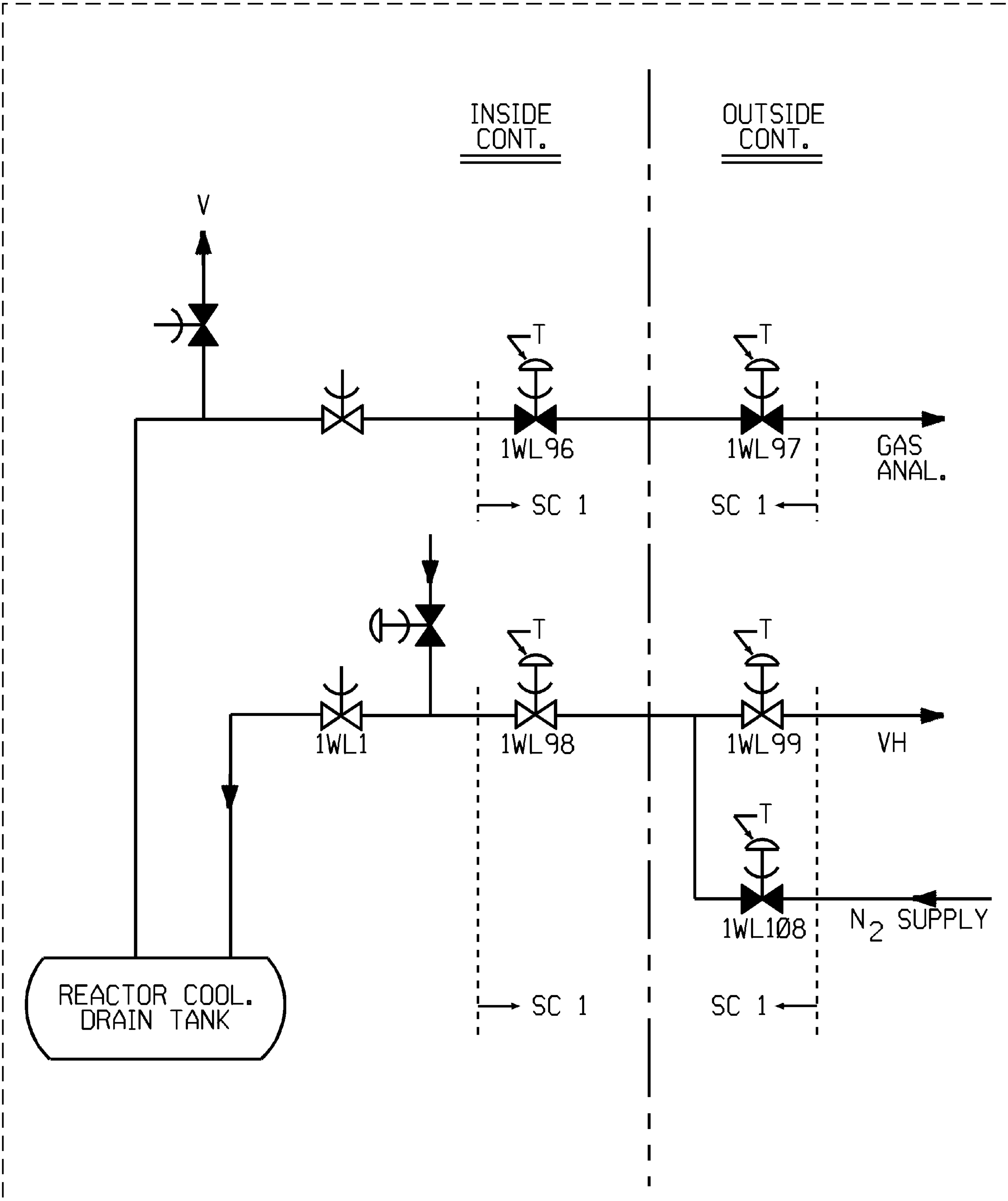
NOTE: VALVE PROVIDED, BUT NOT REQUIRED FOR CONTAINMENT ISOLATION PURPOSES

PUBLIC SERVICE ELECTRIC AND GAS COMPANY
SALEM NUCLEAR GENERATING STATION

Containment Isolation
Component Cooling for Reactor Coolant Pumps

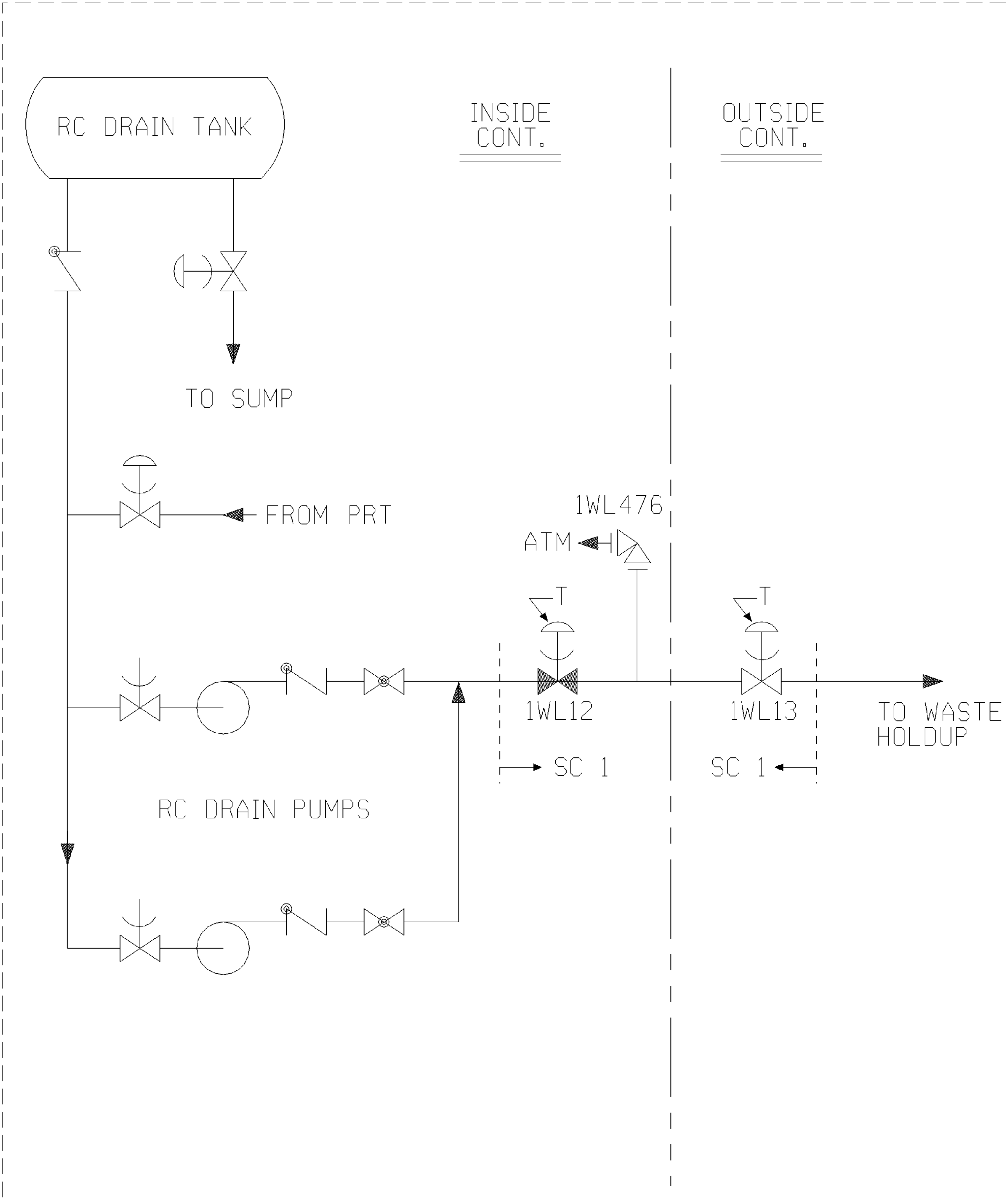
Updated FSAR
Revision 16

Figure 6.2-24
January 31, 1998



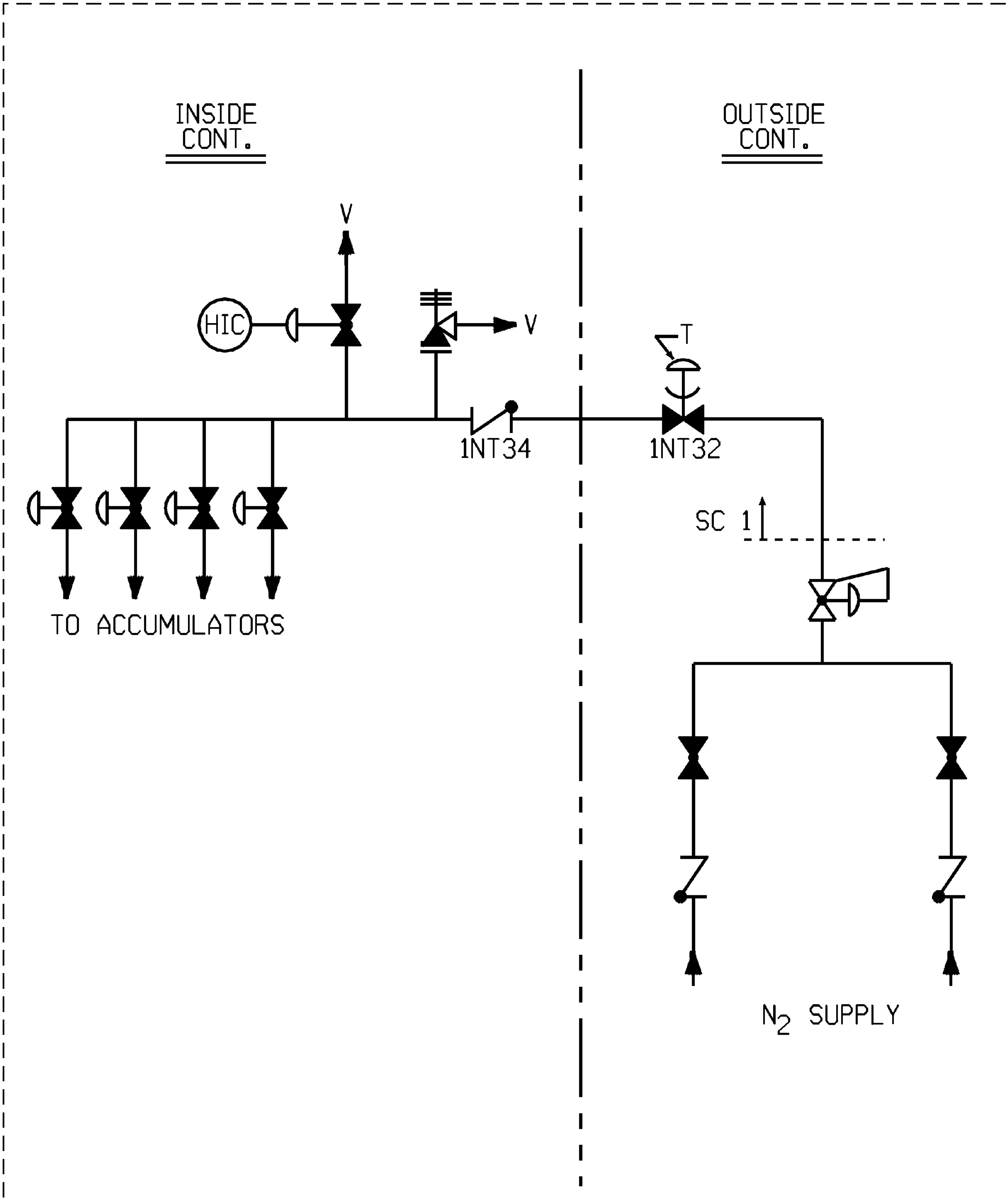
Revision 20, May 6, 2003

PSEG Nuclear, LLC SALEM NUCLEAR GENERATING STATION	Salem Nuclear Generating Station CONTAINMENT ISOLATION REACTOR COOLANT DRAIN TANK CONNECTIONS
	Updated FSAR Figure 6.2-25



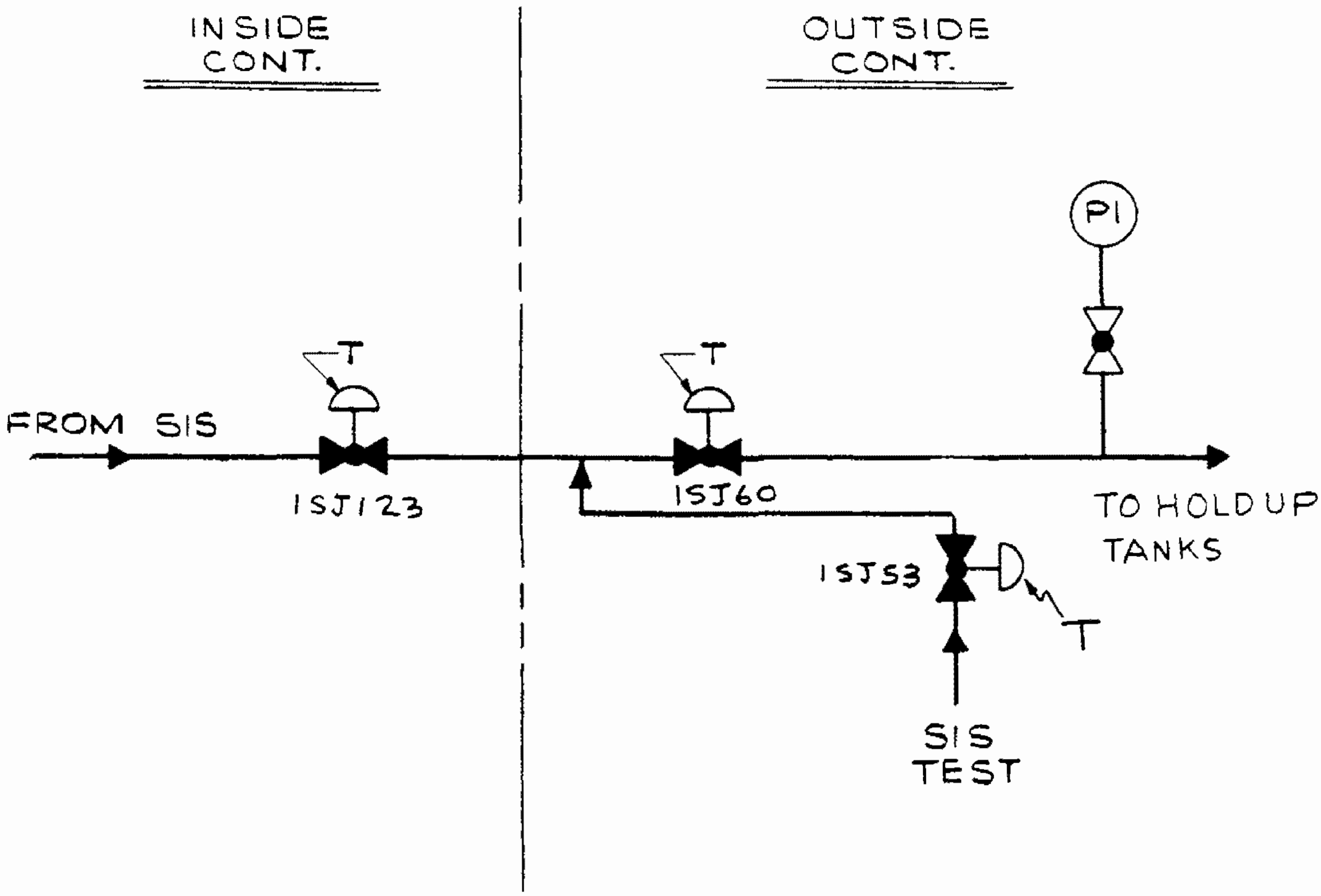
Revision 20, May 6, 2003

PSEG Nuclear, LLC SALEM NUCLEAR GENERATING STATION	Salem Nuclear Generating Station CONTAINMENT ISOLATION REACTOR COOLANT DRAIN TANK PUMPS
	Updated FSAR — Figure 6.2-26



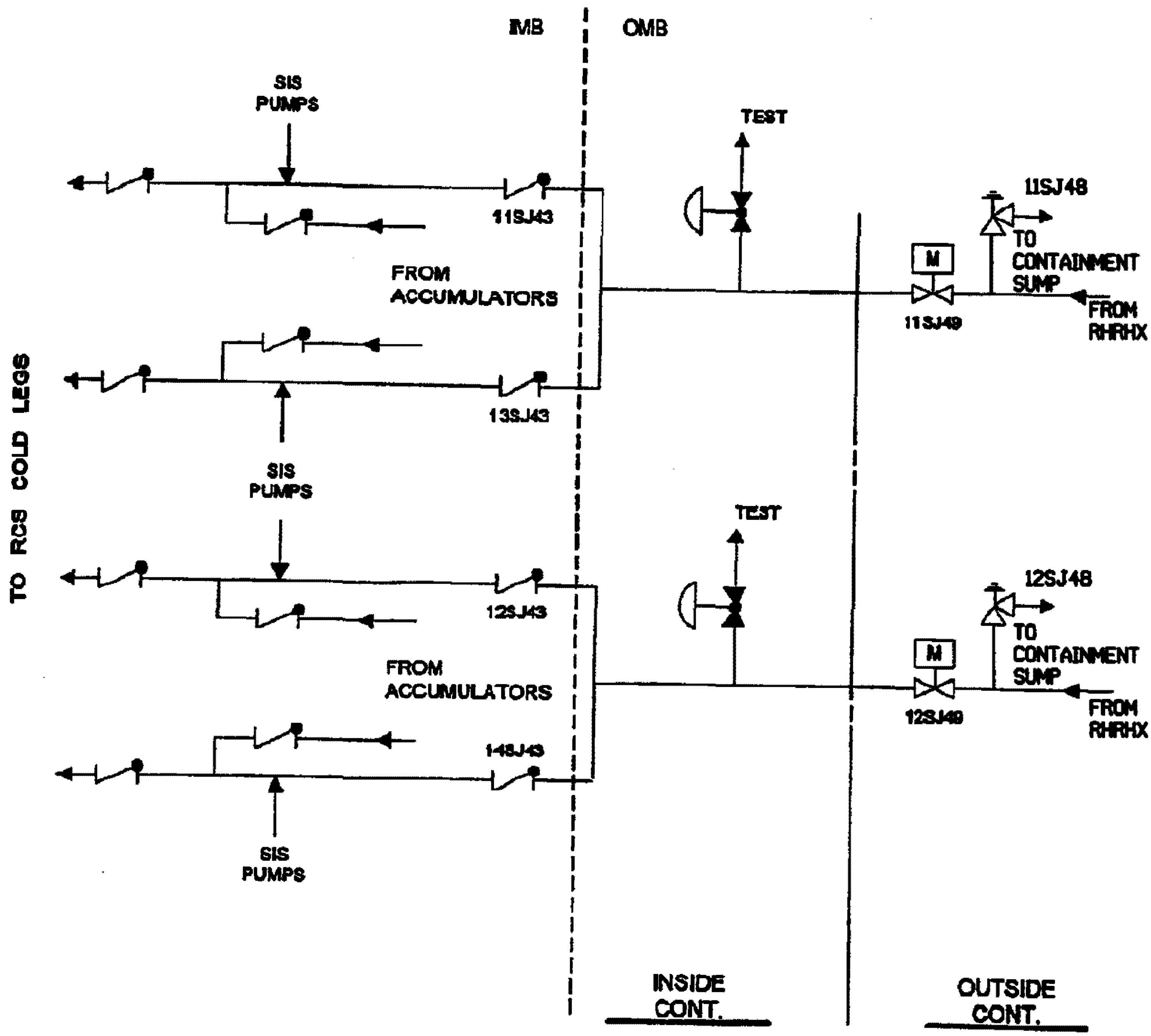
Revision 20, May 6, 2003

PSEG Nuclear, LLC SALEM NUCLEAR GENERATING STATION	Salem Nuclear Generating Station CONTAINMENT ISOLATION ACCUMULATOR NITROGEN SUPPLY
	Updated FSAR Figure 6.2-27



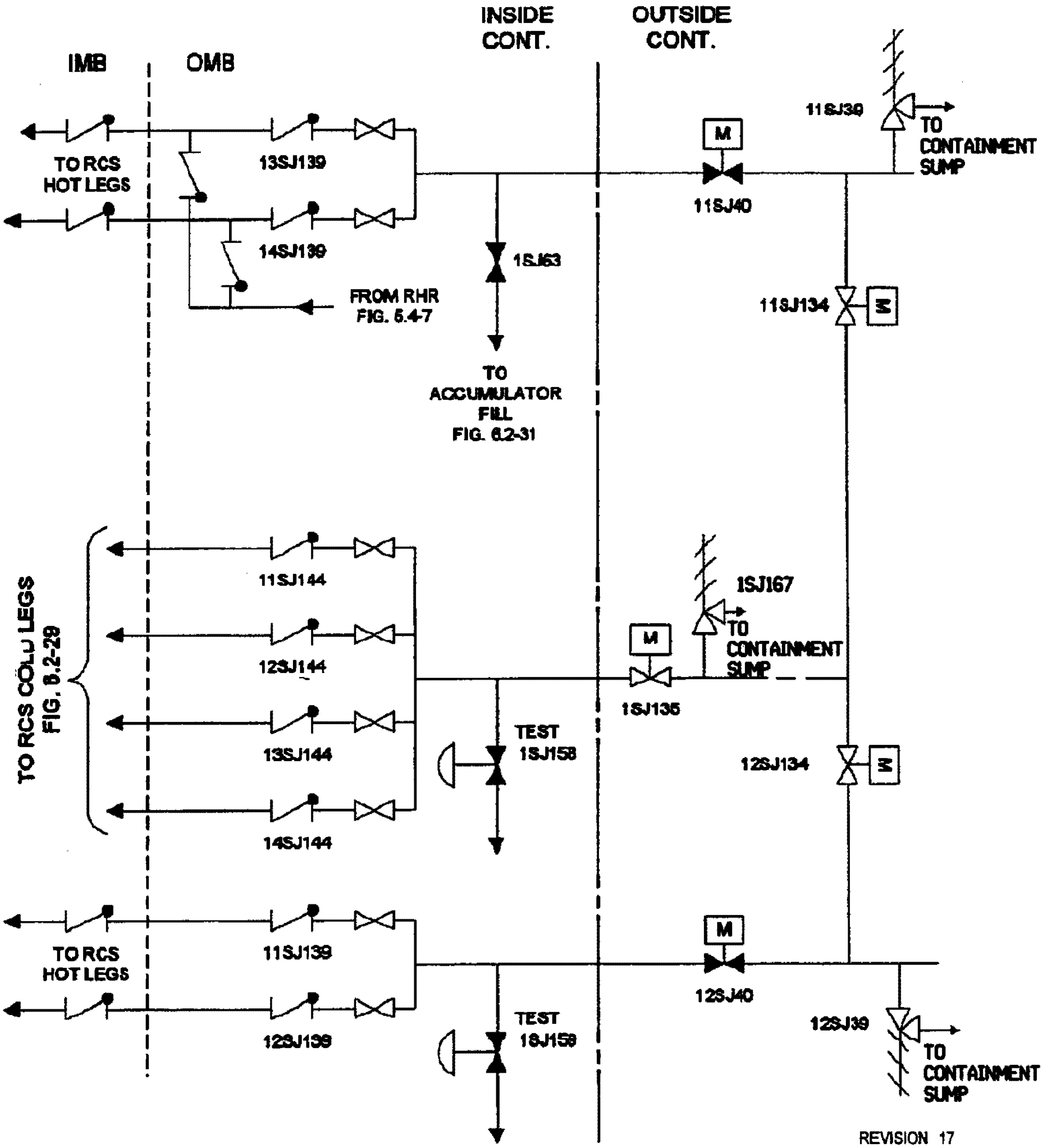
REVISION 6
FEBRUARY 15, 1987

PUBLIC SERVICE ELECTRIC AND GAS COMPANY SALEM NUCLEAR GENERATING STATION	Containment Isolation Safety Injection Test Line	
	Updated FSAR	FIG. 6.2-28



REVISION 17
OCTOBER 16 1998

PUBLIC SERVICE ELECTRIC AND GAS COMPANY SALEM NUCLEAR GENERATING STATION	CONTAINMENT ISOLATION RHR SAFETY INJECTION CONNECTIONS
	Updated FSAR FIG. 6.2-29



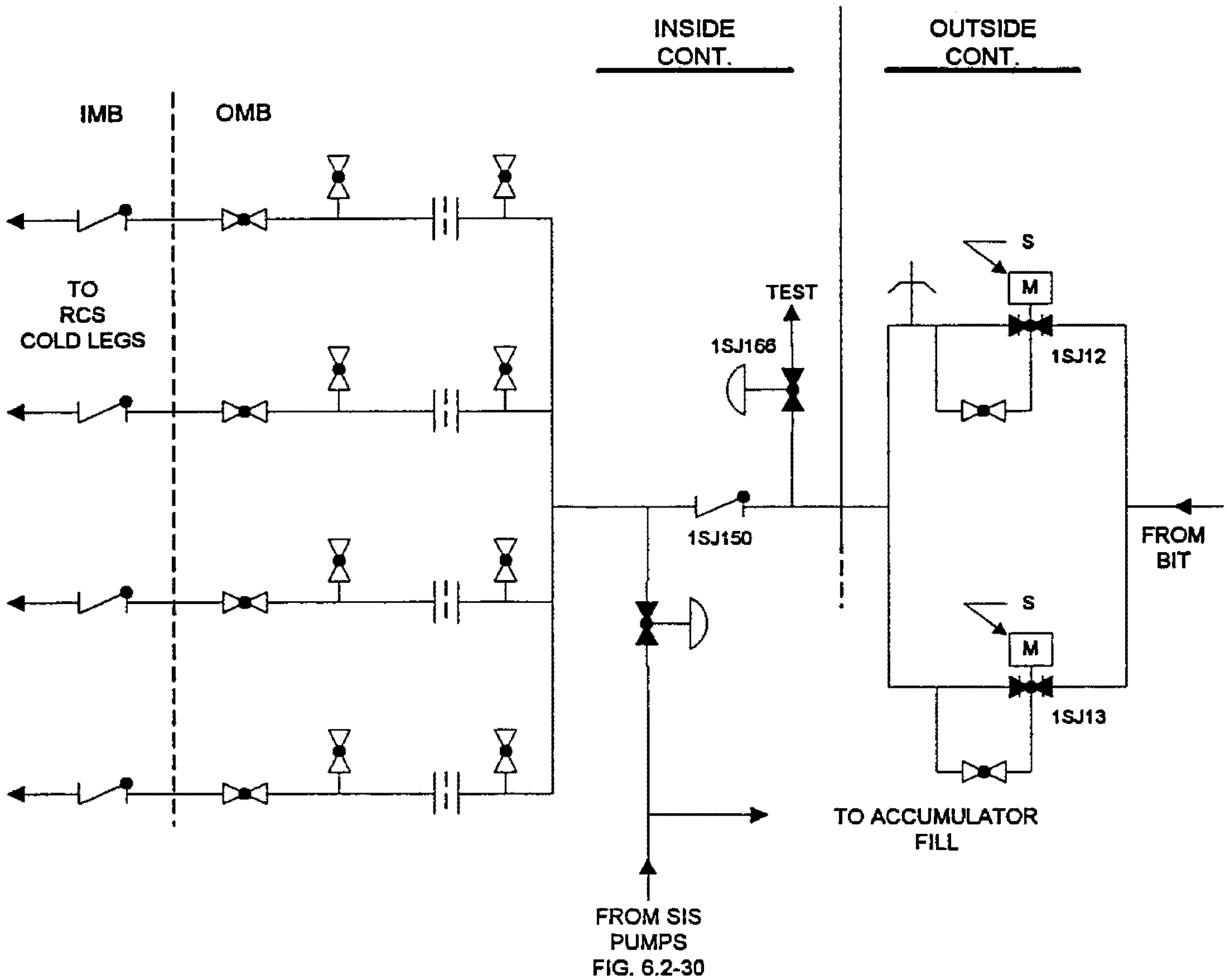
REVISION 17
OCTOBER 16 1998

PUBLIC SERVICE ELECTRIC AND GAS COMPANY
SALEM NUCLEAR GENERATING STATION

CONTAINMENT ISOLATION
SAFETY INJECTION PUMP CONNECTIONS

Updated FSAR

FIG. 6.2-30

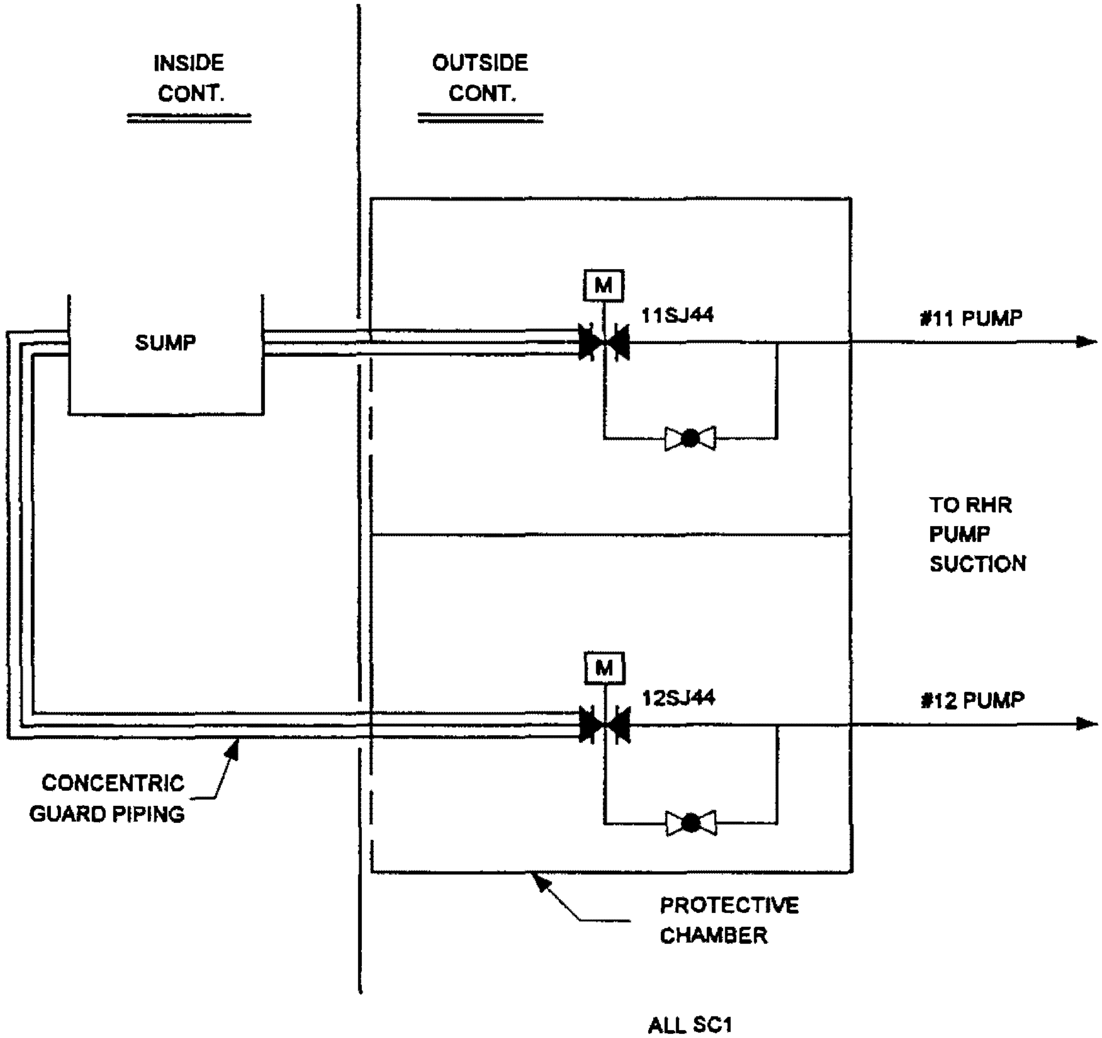


PUBLIC SERVICE ELECTRIC AND GAS COMPANY
SALEM NUCLEAR GENERATING STATION

Containment Isolation
Charging Pump Connections

Updated FSAR
Revision 16

Figure 6.2-31
January 31, 1998

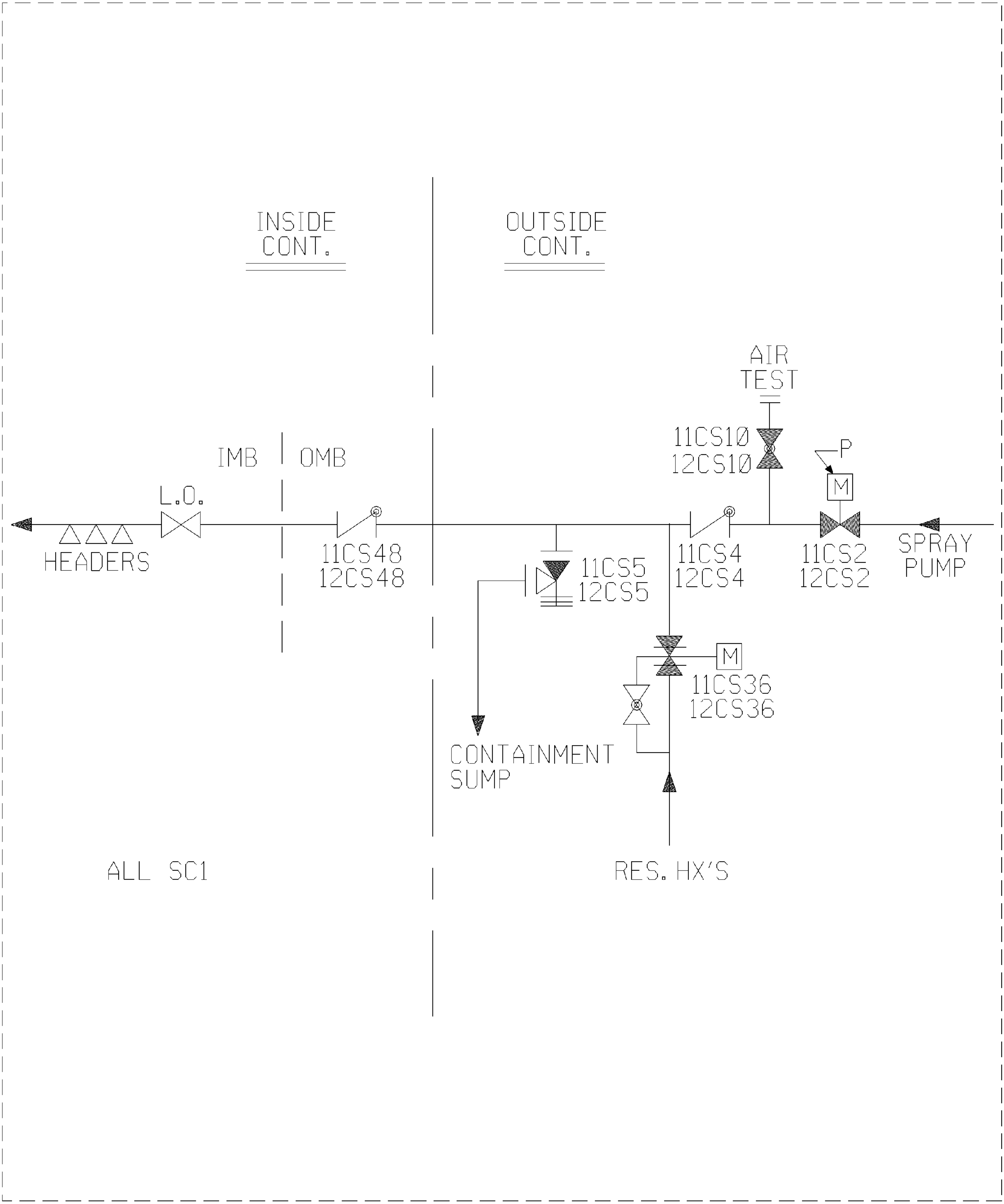


PUBLIC SERVICE ELECTRIC AND GAS COMPANY
SALEM NUCLEAR GENERATING STATION

Containment Isolation
Safety Injection Recirculation From Sump

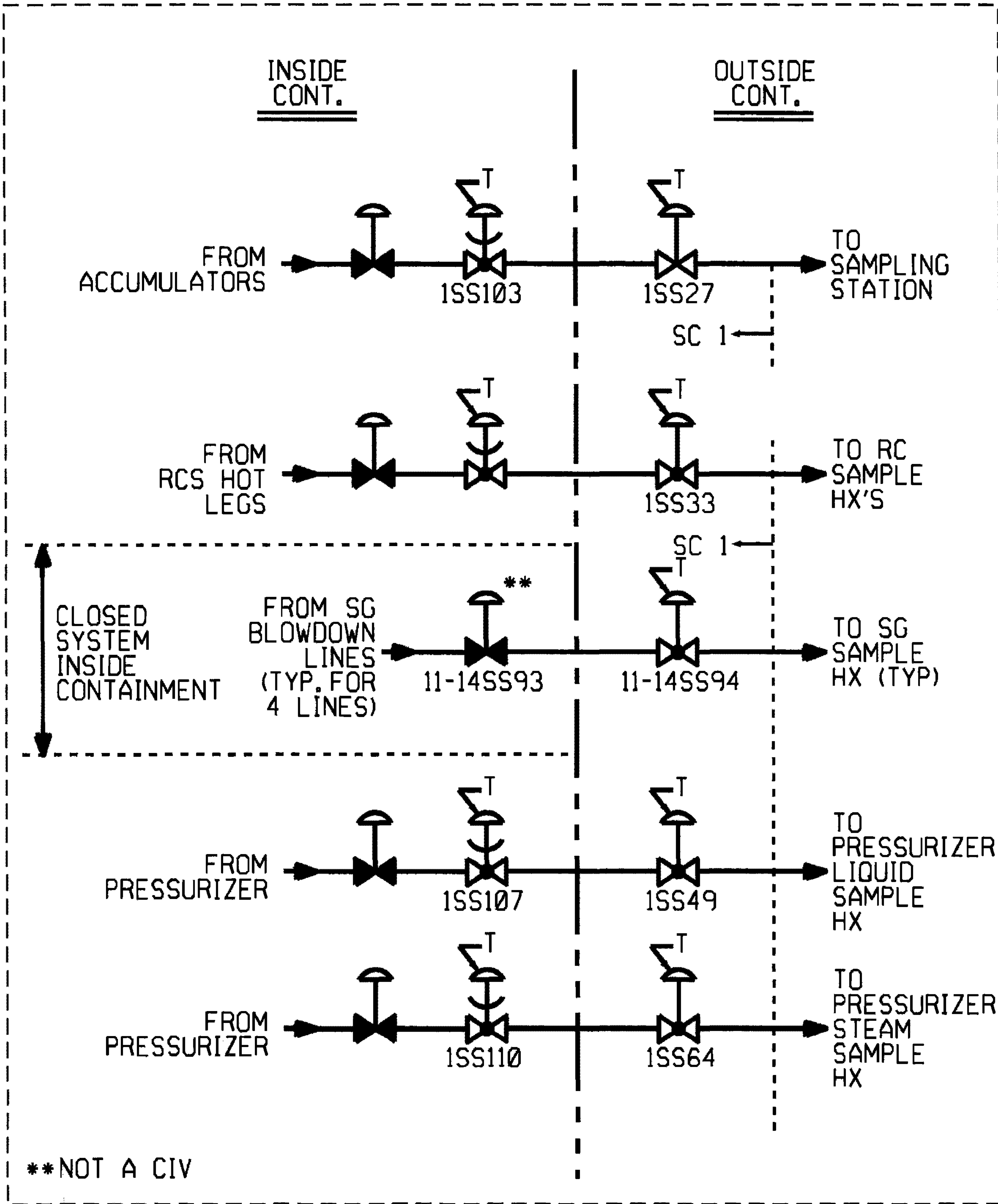
Updated FSAR
Revision 16

Figure 6.2-32
January 31, 1998



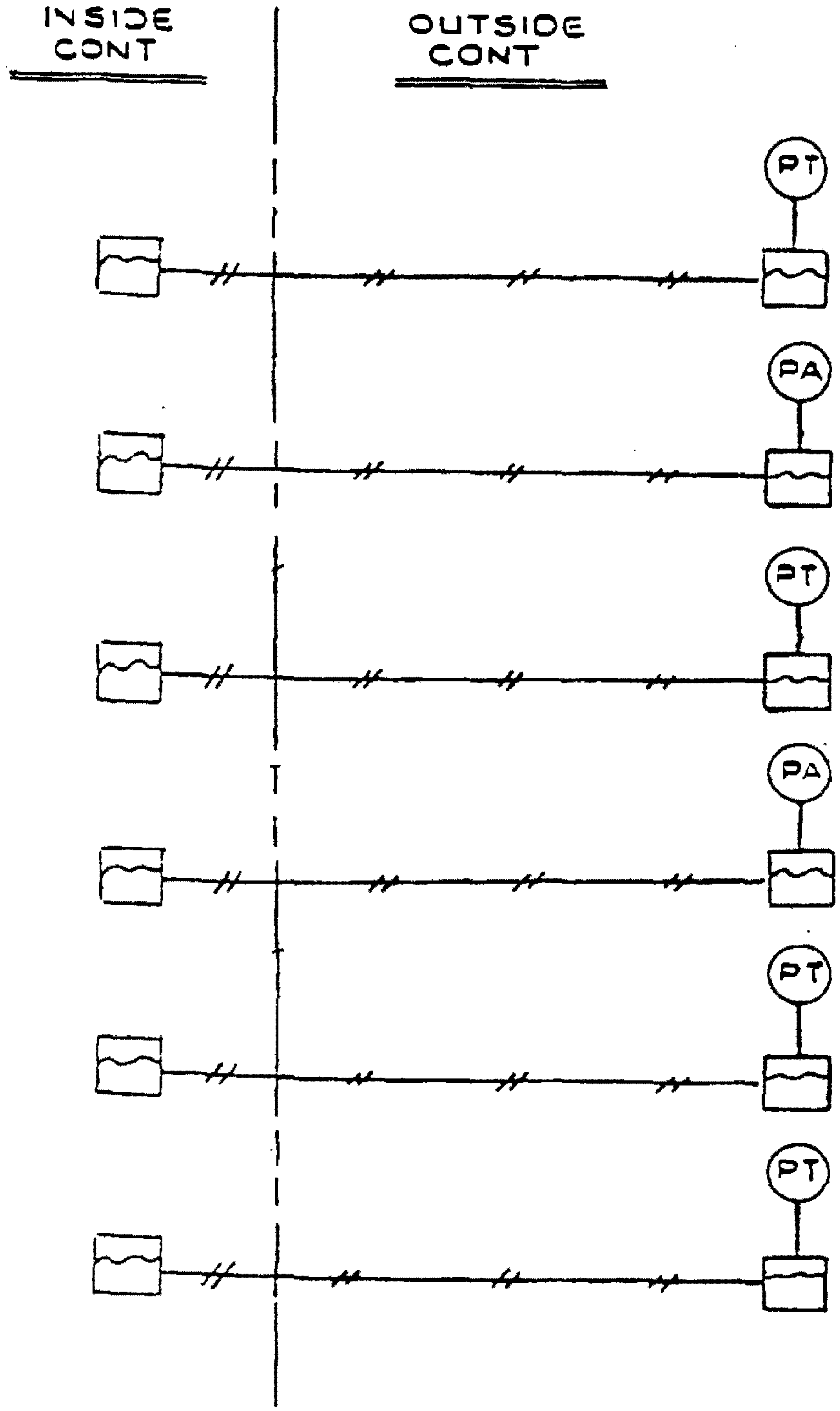
Revision 20, May 6, 2003

PSEG Nuclear, LLC SALEM NUCLEAR GENERATING STATION	Salem Nuclear Generating Station CONTAINMENT ISOLATION CONTAINMENT SPRAY SYSTEM
	Updated FSAR — Figure 6.2-33



Revision 25, October 26, 2010

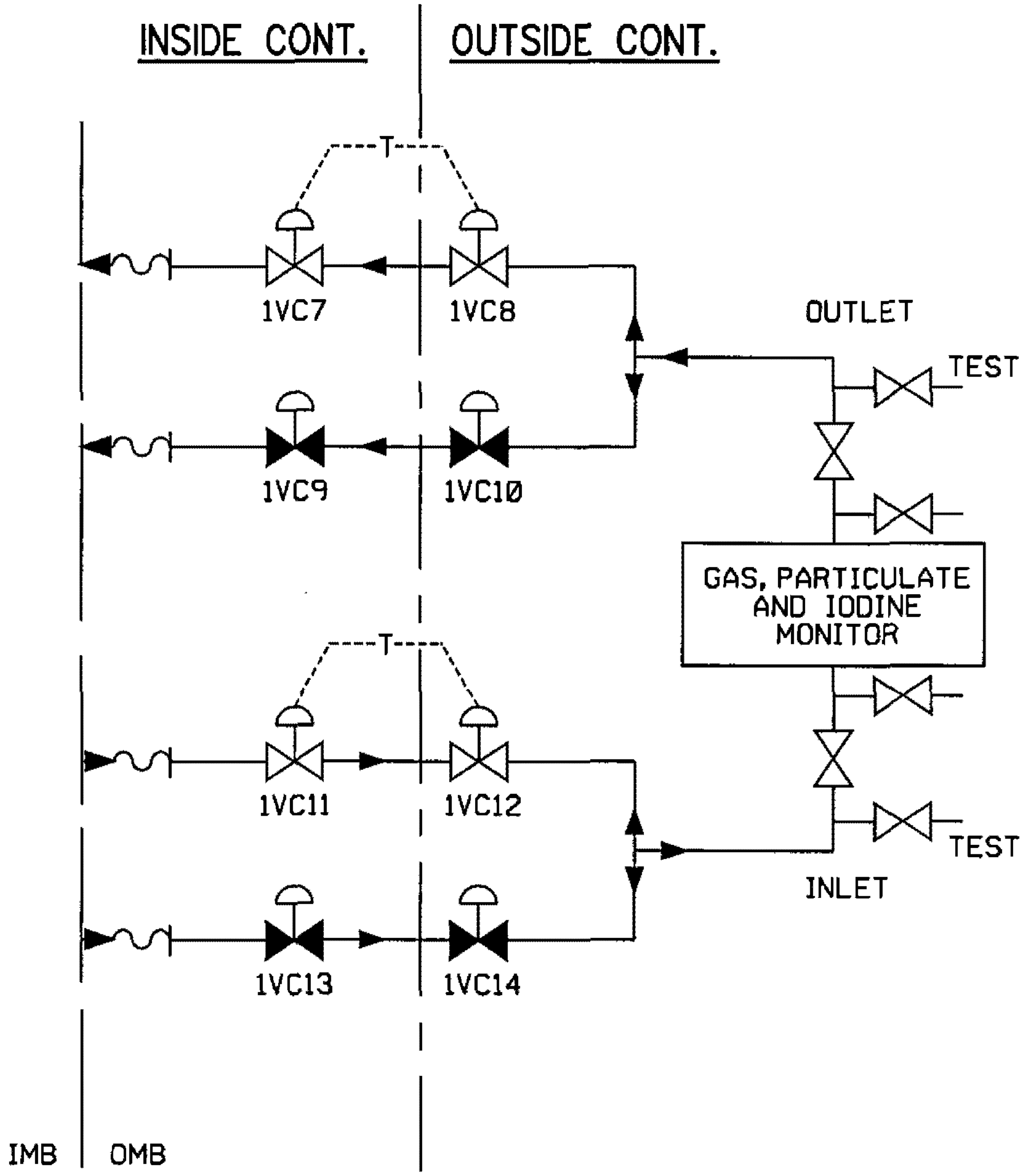
PSEG Nuclear, LLC SALEM NUCLEAR GENERATING STATION	Salem Nuclear Generating Station CONTAINMENT ISOLATION-REACTOR COOLANT, STEAM GENERATOR, PRESSURIZER, ACCUMULATOR SAMPLING
	Updated FSAR — Figure 6.2-34



SEALED SYSTEM

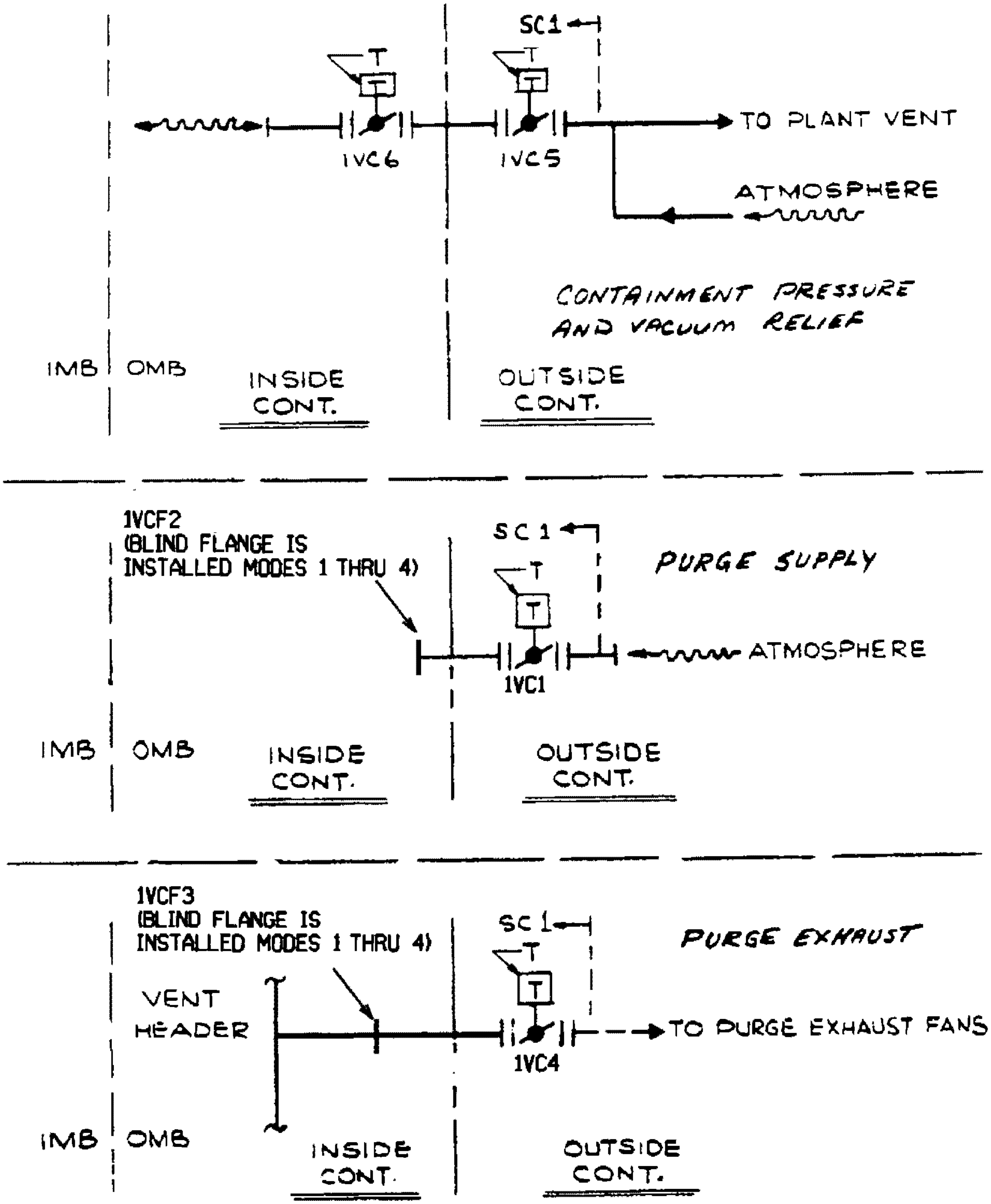
Revision 18, April 26, 2000

PSEG Nuclear, LLC SALEM NUCLEAR GENERATING STATION	Salem Nuclear Generating Station CONTAINMENT ISOLATION CONTAINMENT PRESSURE INSTRUMENTATION
	Updated FSAR — Figure 6.2-35



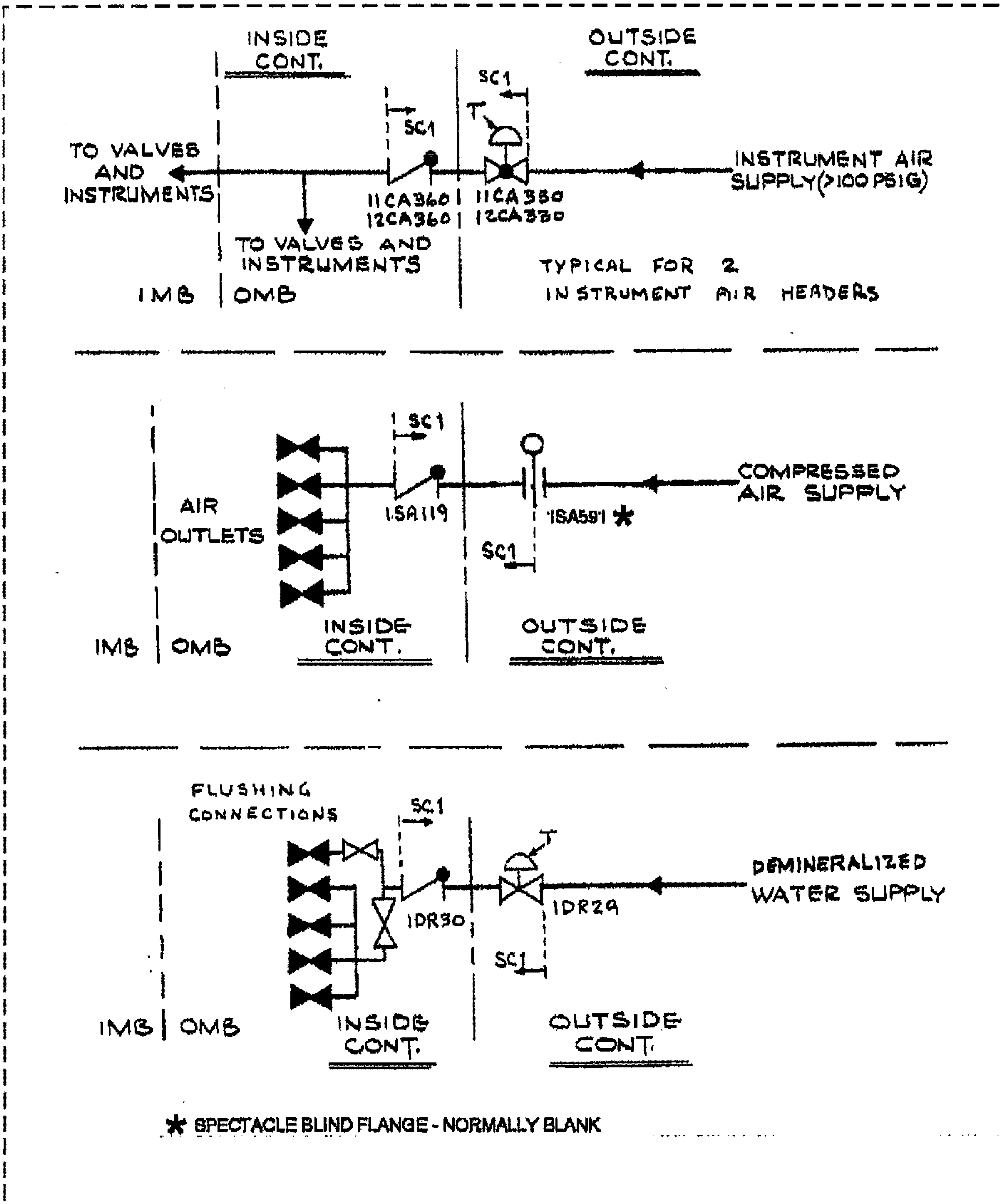
Revision 19, Nov. 19, 2001

PSEG Nuclear, LLC SALEM NUCLEAR GENERATING STATION	Salem Nuclear Generating Station CONTAINMENT ISOLATION CONTAINMENT AIR SAMPLER
	Updated FSAR Figure 6.2-36



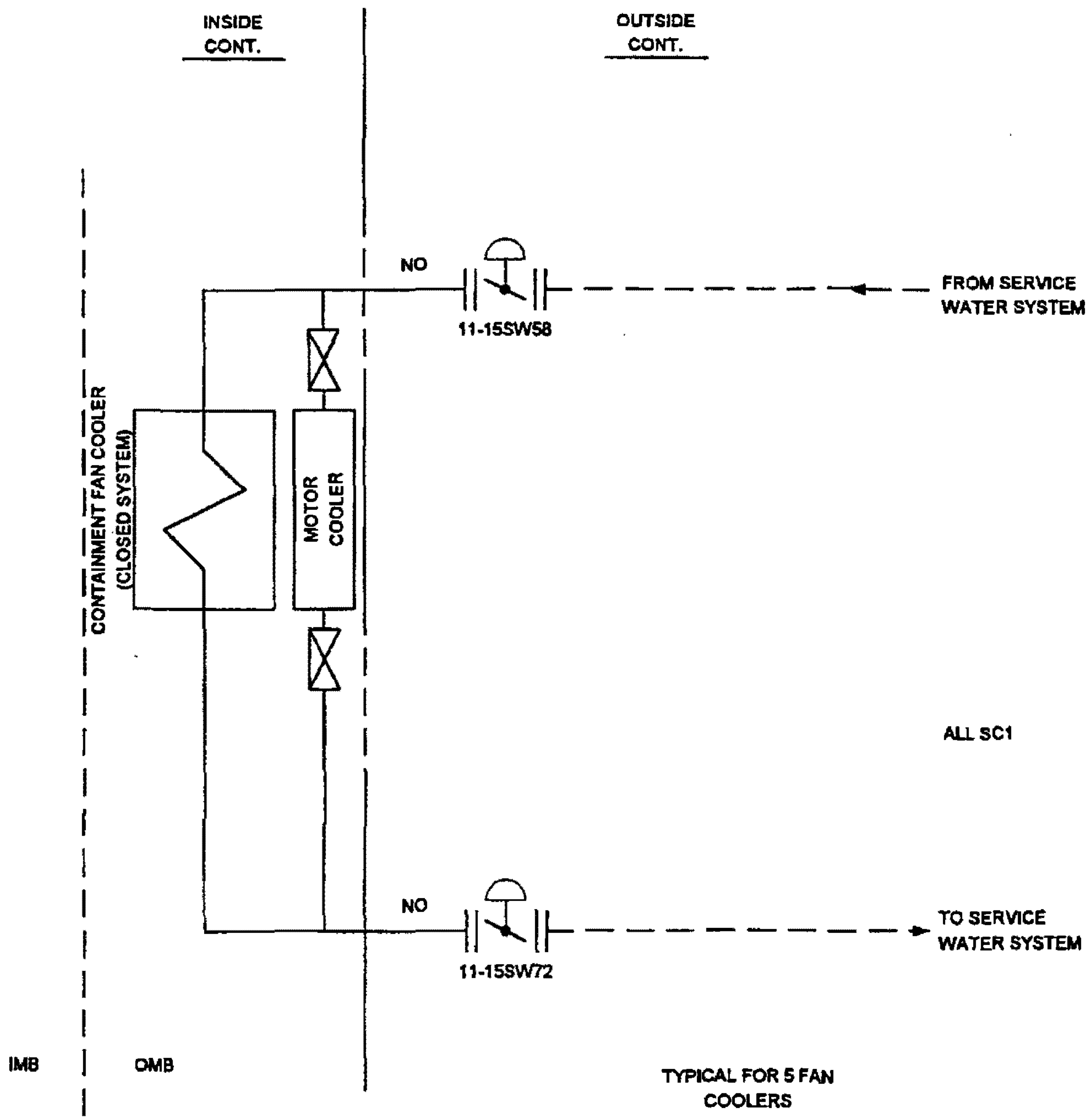
Revision 24
May 11, 2009

PUBLIC SERVICE ELECTRIC AND GAS COMPANY SALEM NUCLEAR GENERATING STATION	Containment Isolation Containment Pressure Relief and Purge
	Updated FSAR — FIG. 6.2-37



Revision 23, October 17, 2007

PSEG Nuclear, LLC SALEM NUCLEAR GENERATING STATION	Salem Nuclear Generating Station Containment Isolation Service Air, Instrument Air and Domestic Water
	Updated FSAR — Figure 6.2-38



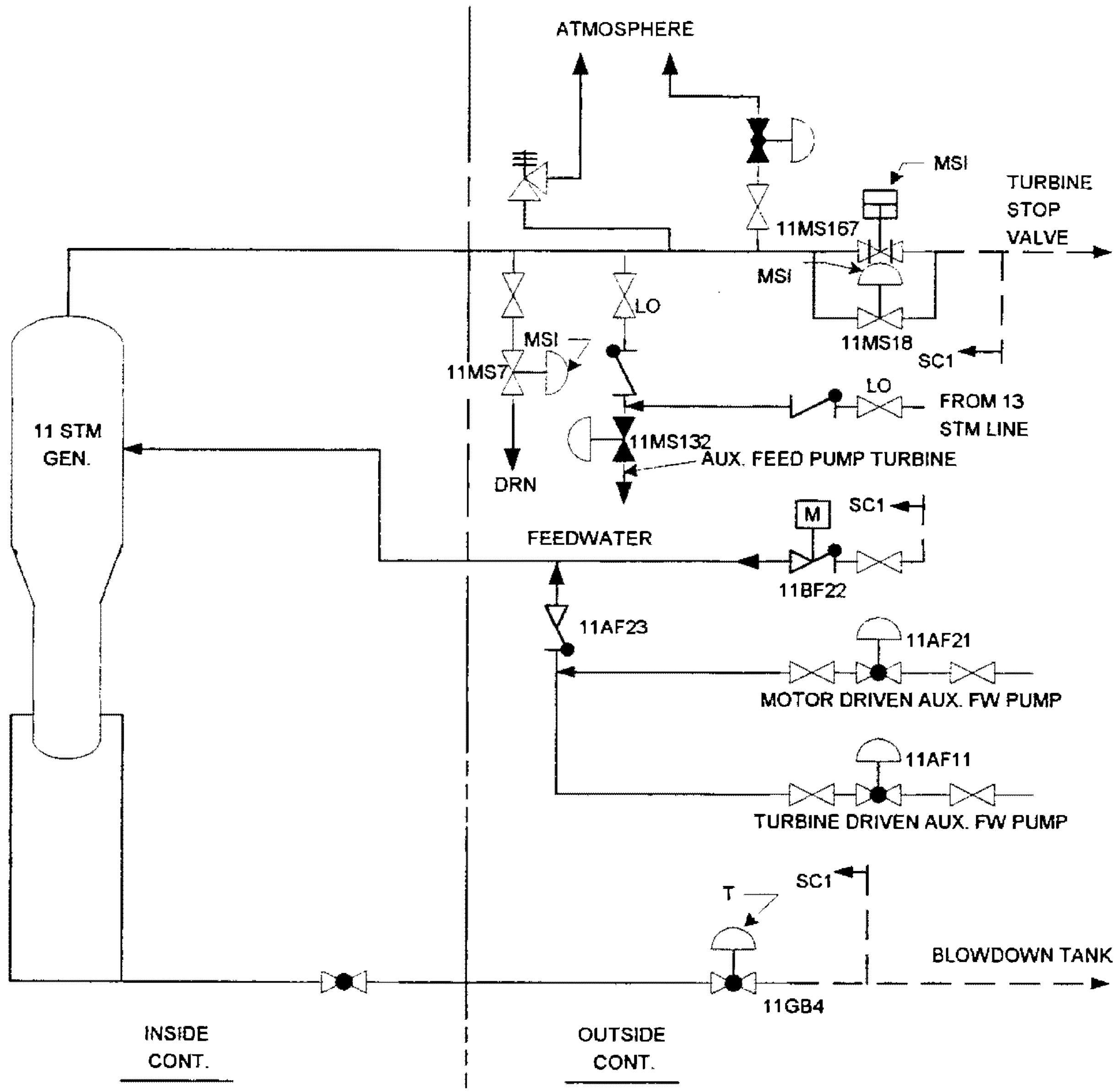
Revision 24
May 11, 2009

PUBLIC SERVICE ELECTRIC AND GAS COMPANY
SALEM NUCLEAR GENERATING STATION

Containment Isolation
Containment Fan Cooling Water

Updated FSAR

Figure 6.2-39

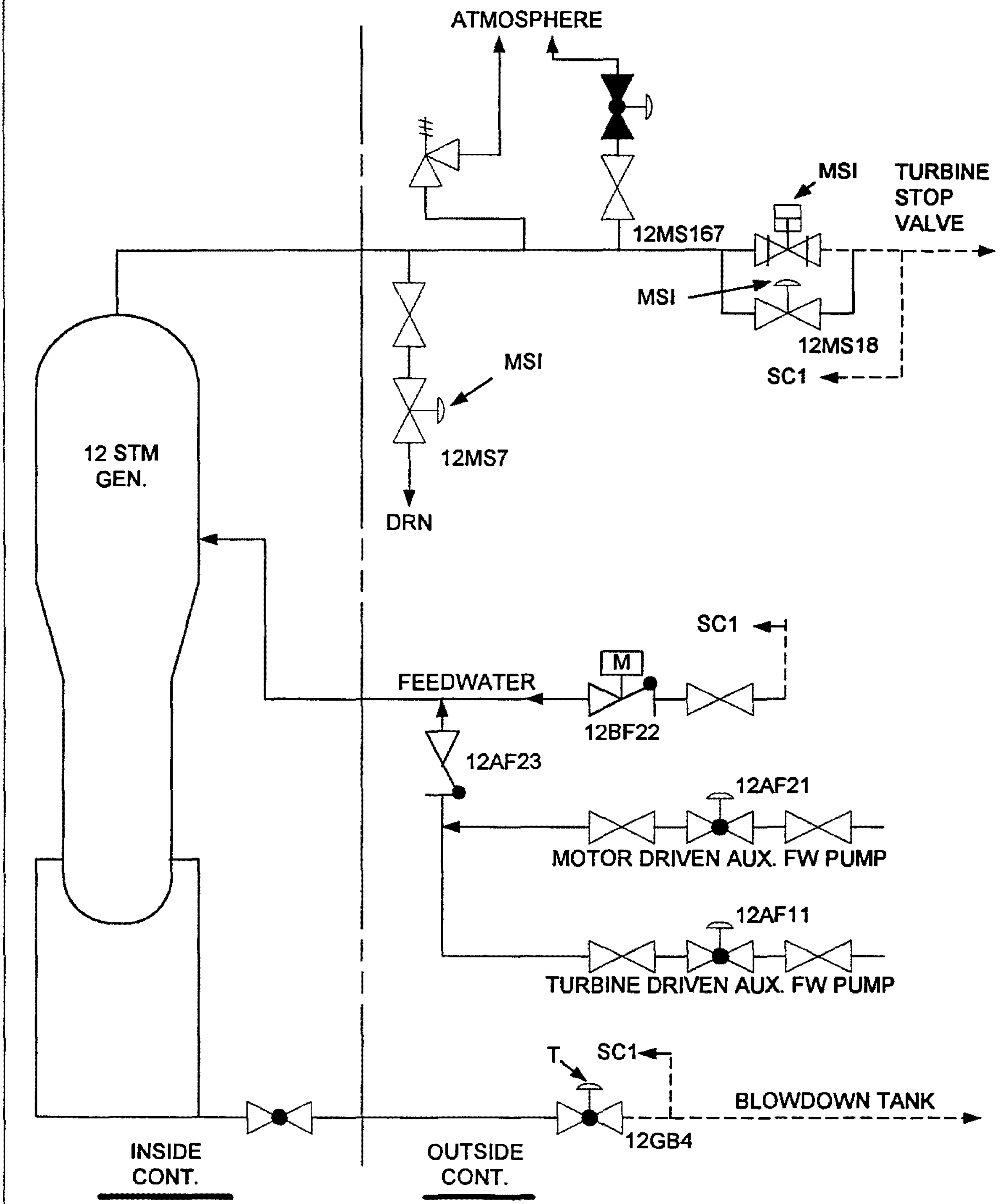


PUBLIC SERVICE ELECTRIC AND GAS COMPANY
SALEM NUCLEAR GENERATING STATION

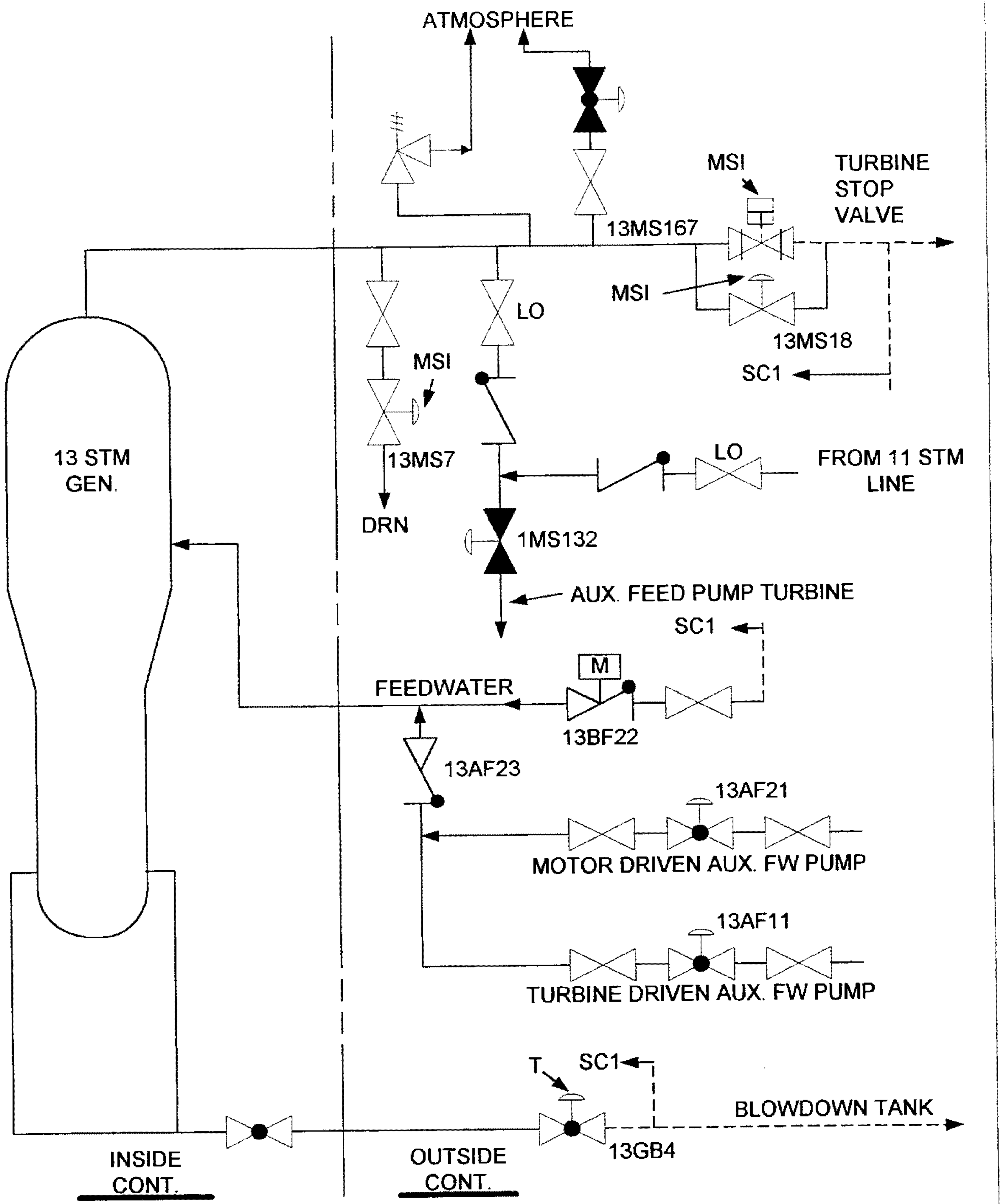
Containment Isolation - Main Steam, Feedwater and Blowdown (11 Stm Gen)

Updated FSAR
Revision 16

Figure 6.2-40
January 31, 1998



PUBLIC SERVICE ELECTRIC AND GAS COMPANY SALEM NUCLEAR GENERATING STATION	Containment Isolation - Main Steam, Feedwater and Blowdown (12 Stm Gen)
	Updated FSAR Revision 16 — Figure 6.2-41, January 31, 1998

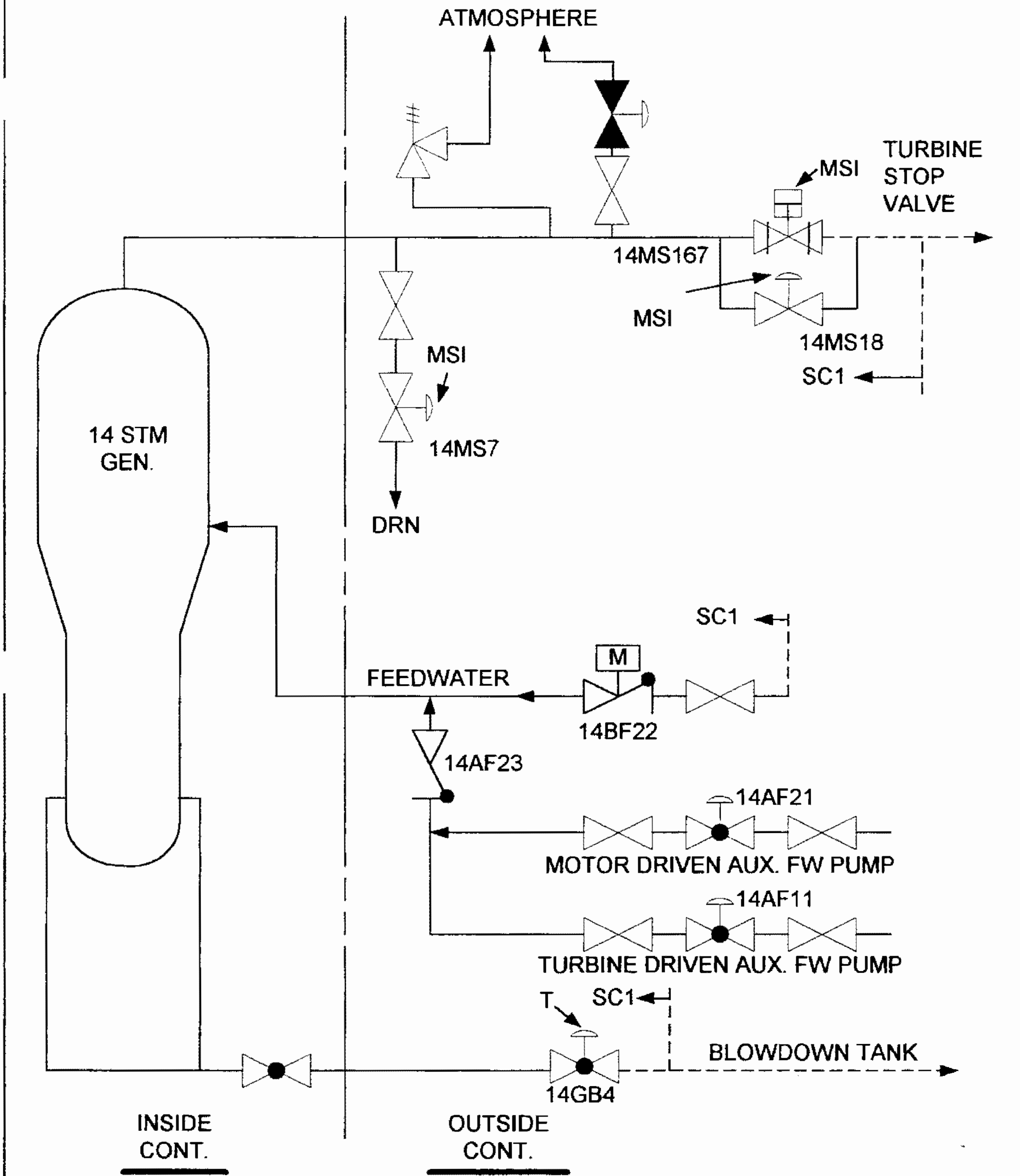


PUBLIC SERVICE ELECTRIC AND GAS COMPANY
SALEM NUCLEAR GENERATING STATION

Containment Isolation - Main Steam, Feedwater and Blowdown (13 Stm Gen)

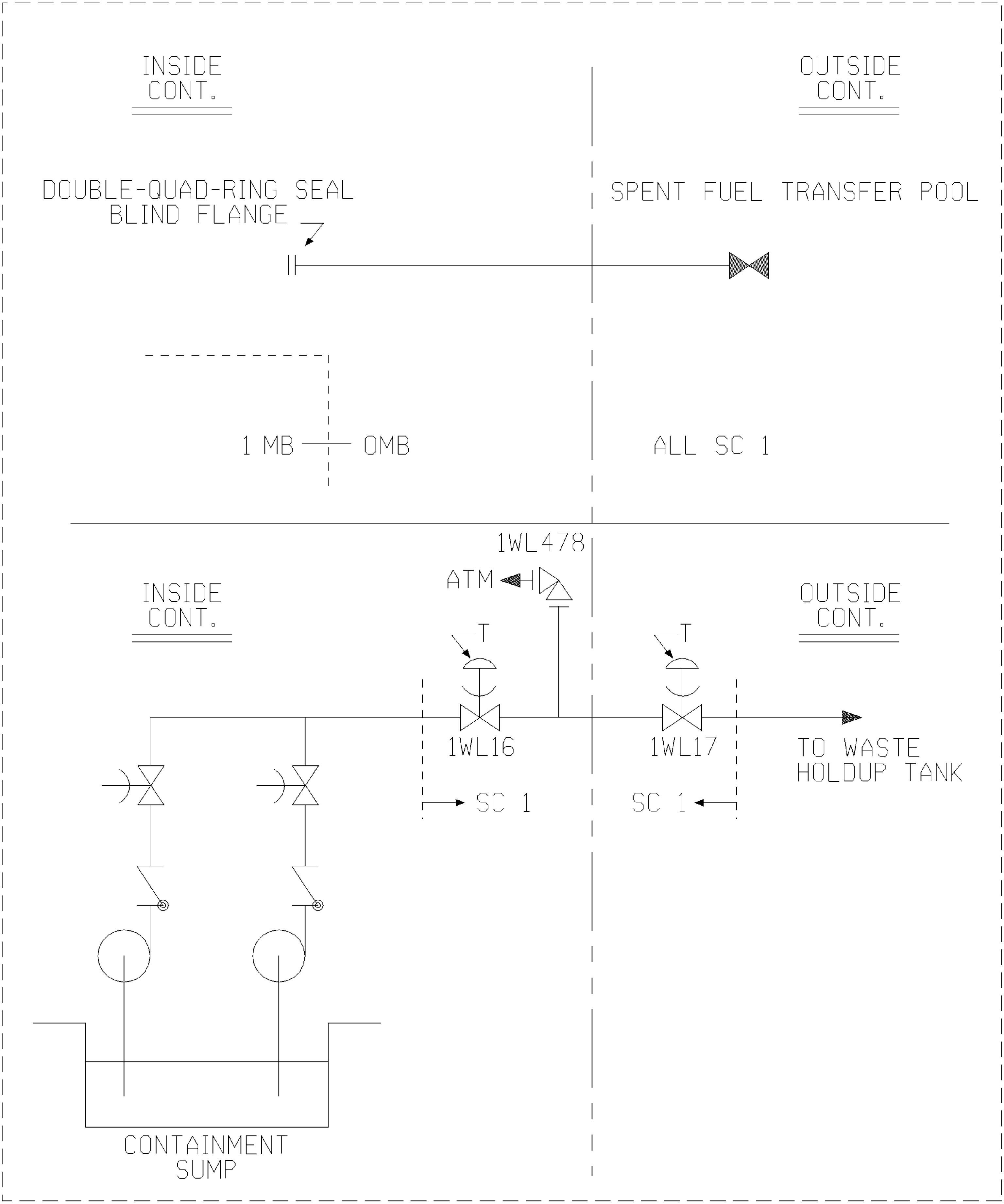
Updated FSAR
Revision 16

Figure 6.2-42
January 31, 1998



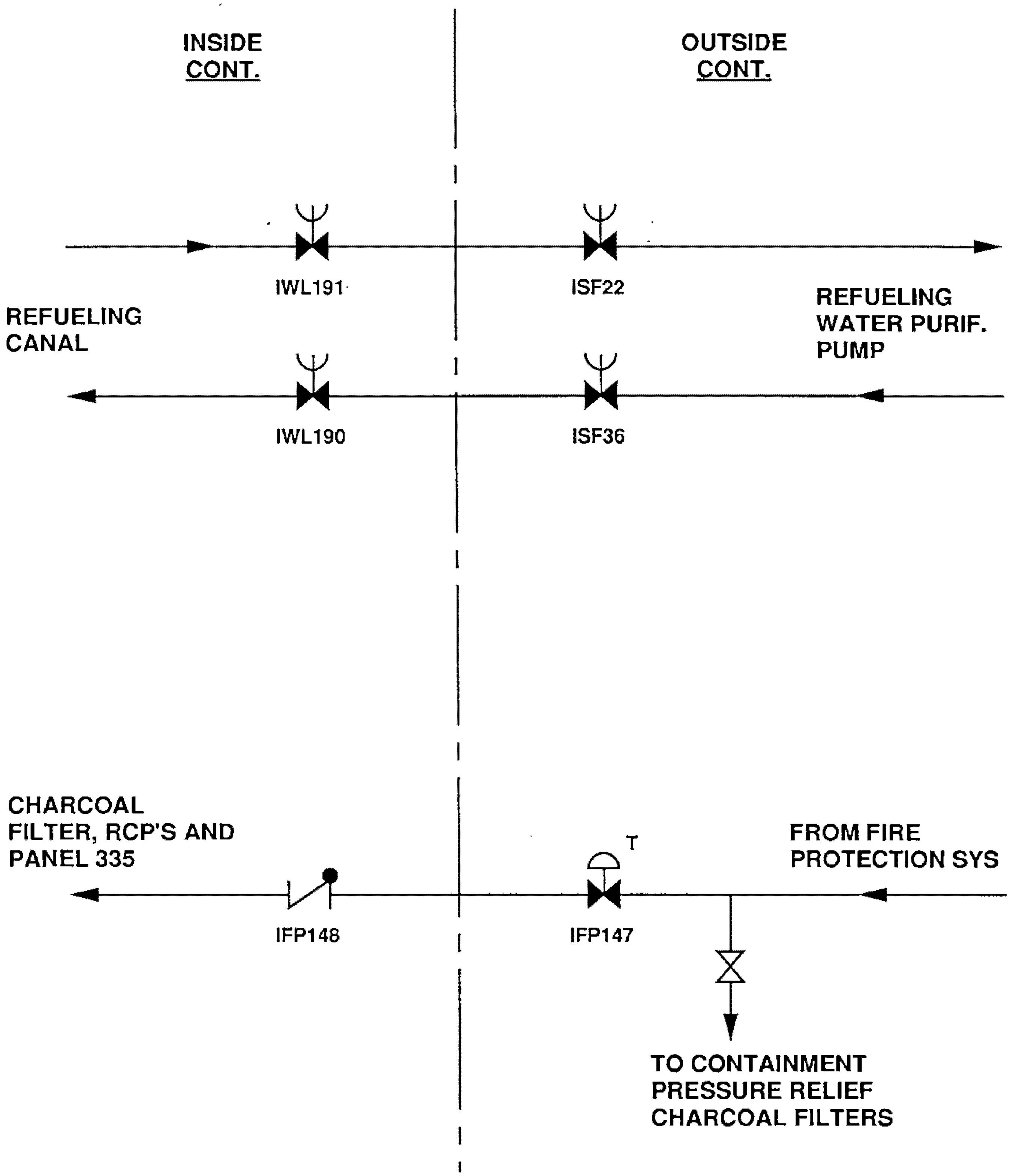
PUBLIC SERVICE ELECTRIC AND GAS COMPANY
SALEM NUCLEAR GENERATING STATION

Containment Isolation - Main Steam, Feedwater and Blowdown (14 Stm Gen)



Revision 20, May 6, 2003

PSEG Nuclear, LLC SALEM NUCLEAR GENERATING STATION	Salem Nuclear Generating Station CONTAINMENT ISOLATION CONTAINMENT SUMP DISCHARGE, FUEL TRANSFER TUBE Updated FSAR Figure 6.2-44



PUBLIC SERVICE ELECTRIC AND GAS COMPANY
SALEM NUCLEAR GENERATING STATION

CONTAINMENT ISOLATION -
REFUELING CANAL SUPPLY
AND DISCHARGE, FIRE WATER SUPPLY

Updated FSAR
Revision 12, July 22, 1992

Figure 6.2-45

THIS FIGURE HAS BEEN DELETED.

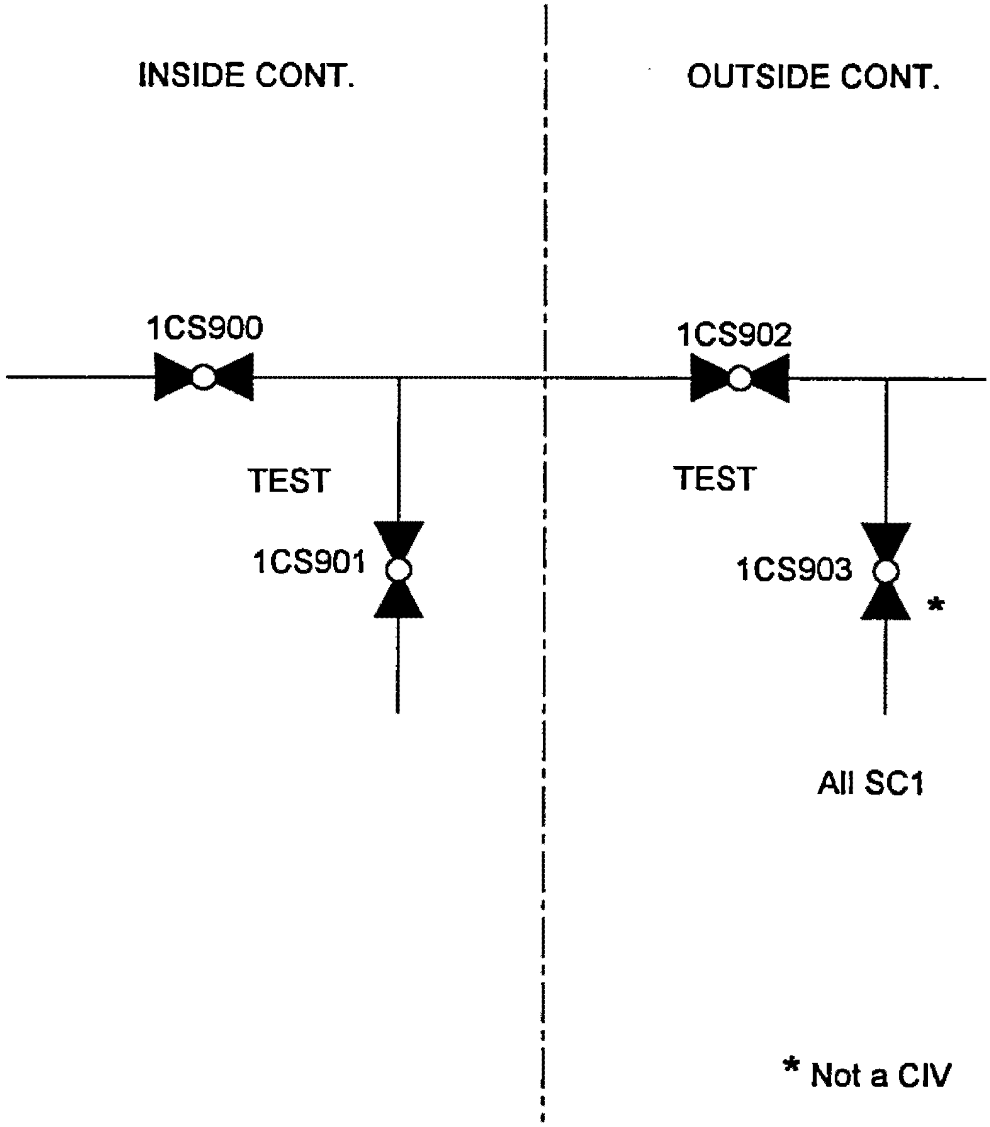
Revision 23, October 17, 2007

PSEG Nuclear, LLC SALEM NUCLEAR GENERATING STATION	Salem Nuclear Generating Station Containment Isolation - Post-Loca Atmospheric Sample	
	Updated FSAR	Figure 6.2-45A

THIS FIGURE HAS BEEN DELETED.

Revision 23, October 17, 2007

PSEG Nuclear, LLC SALEM NUCLEAR GENERATING STATION	Salem Nuclear Generating Station Containment Isolation - Post-Loca RCS Sample
	Updated FSAR Figure 6.2-45B

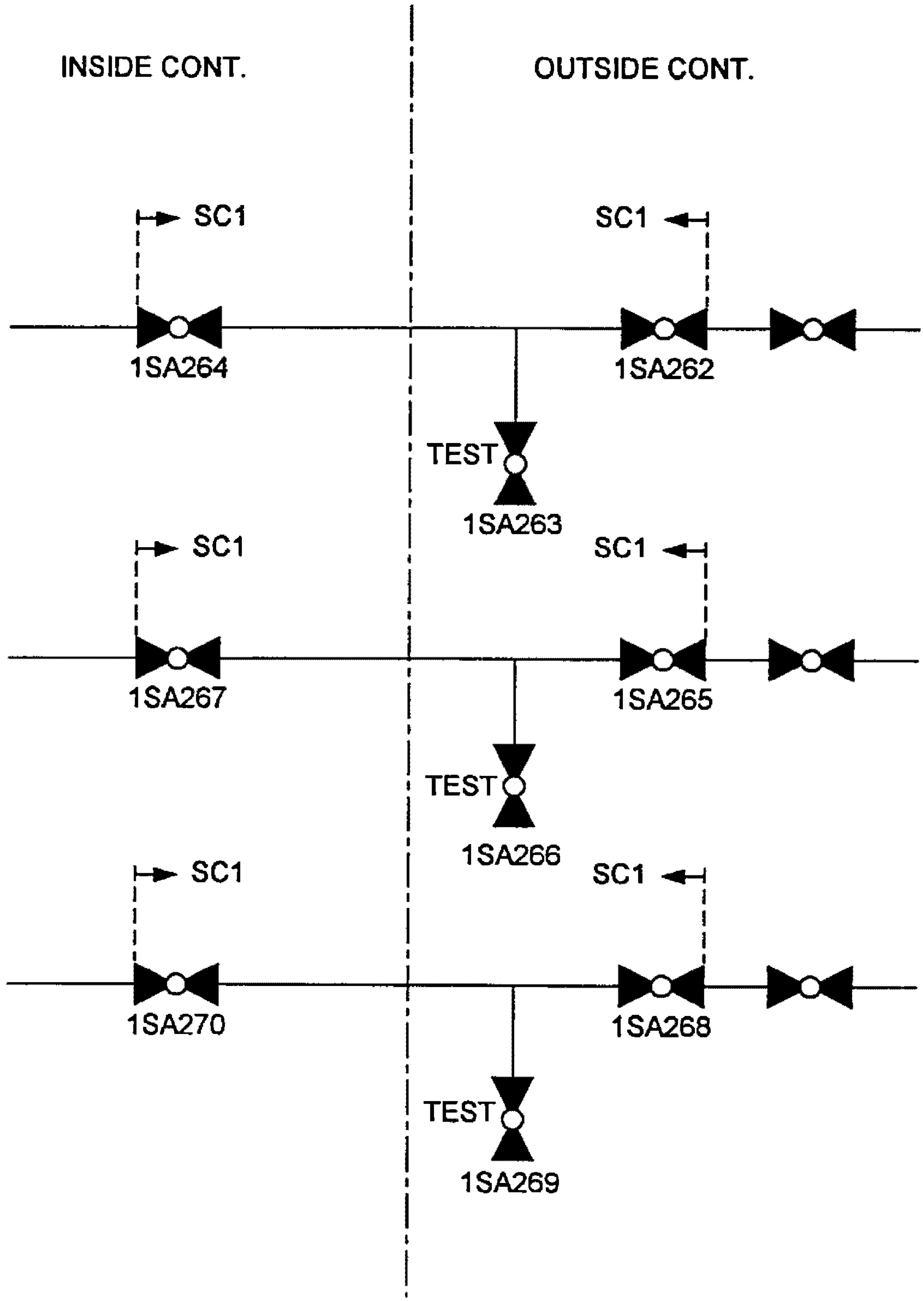


PUBLIC SERVICE ELECTRIC AND GAS COMPANY
SALEM NUCLEAR GENERATING STATION

Containment Isolation
Fill Line for Containment Pressure Instruments

Updated FSAR
Revision 16

Figure 6.2-45C
January 31, 1998

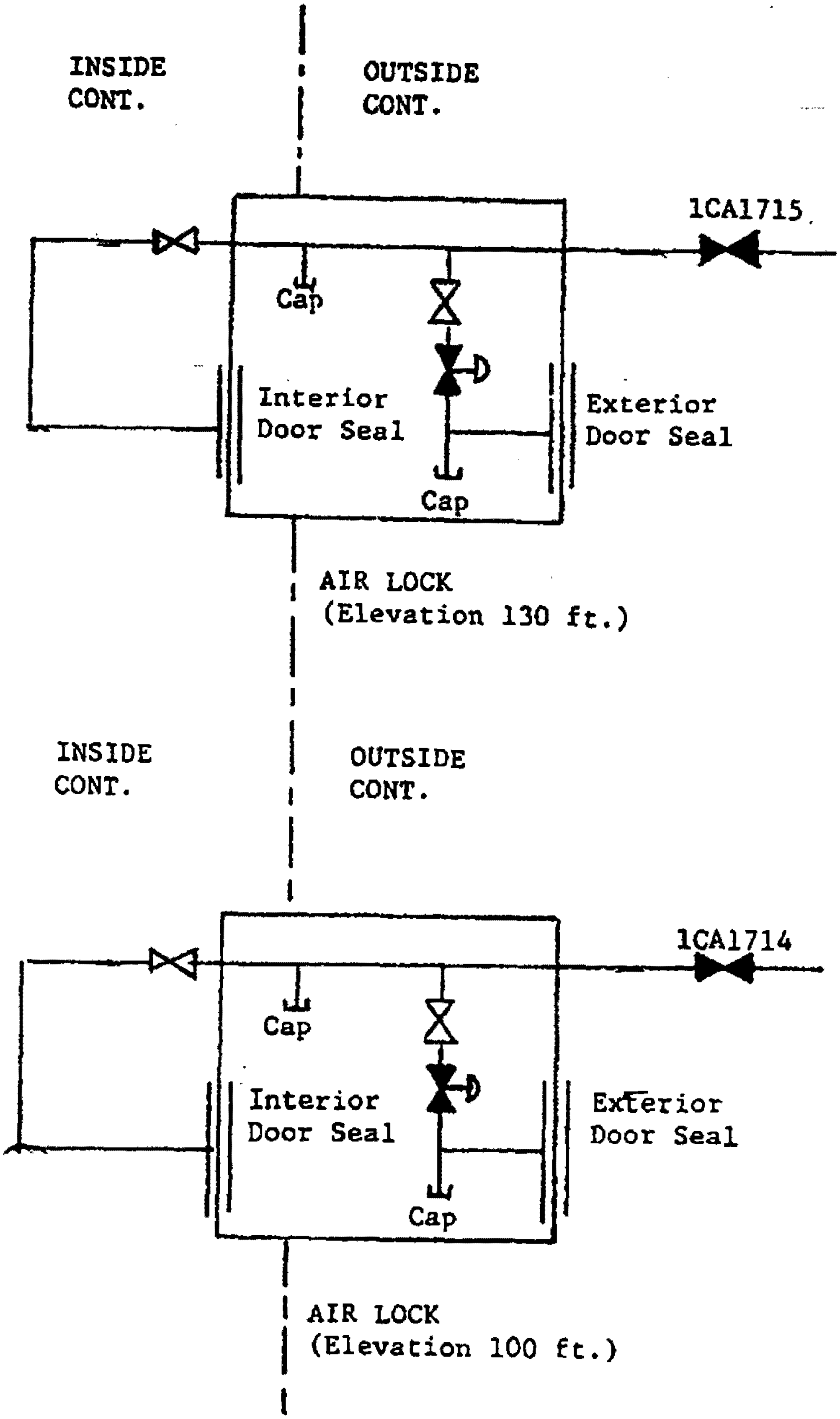


PUBLIC SERVICE ELECTRIC AND GAS COMPANY
SALEM NUCLEAR GENERATING STATION

Containment Isolation
Containment Pressure Test Instrumentation

Updated FSAR
Revision 16

Figure 6.2-45D
January 31, 1998



Revision 18, April 26, 2000

PSEG Nuclear, LLC SALEM NUCLEAR GENERATING STATION	Salem Nuclear Generating Station CONTAINMENT ISOLATION CONTAINMENT AIRLOCK TEST INSTRUMENTATION Updated FSAR — Figure 6.2-45E

VALVES

DOUBLE DISC GATE

GLOBE

DIAPHRAGM

GATE

CHECK

BUTTERFLY

SAFETY OR RELIEF

SELF CONTAINED
RPRESSURE REGULATOR

NEEDLE

STOP CHECK

OPERATORS

AIR OPERATOR

PISTON

MOTOR

DARKENED SYMBOL
INDICATES NORMALLY
CLOSED VALVE

VALVE STEM LEAKOFF

FILLED SYSTEM

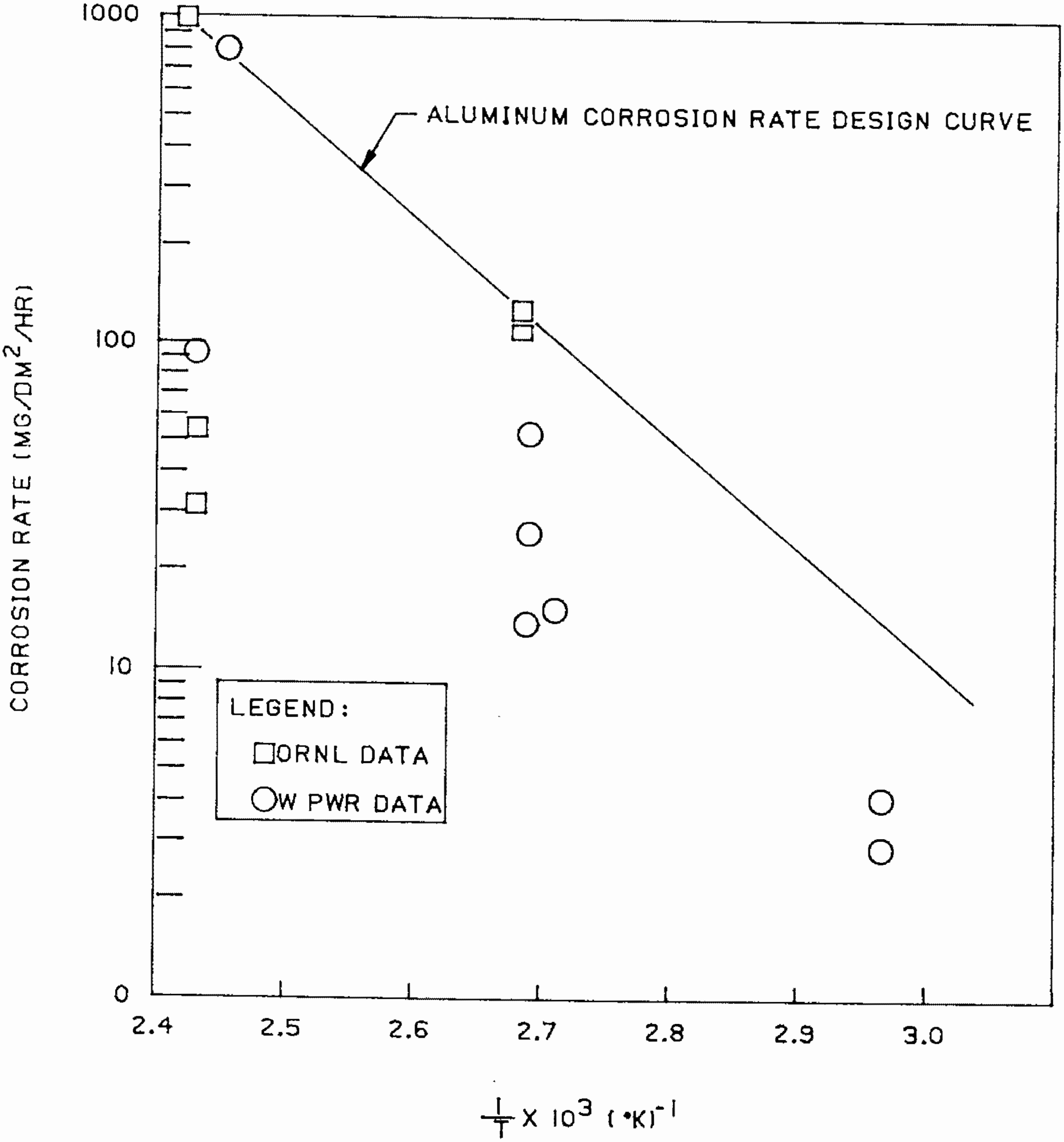
SAUNDERS VALVE

NOTATION

NO -NORMALLY OPEN
NC -NORMALLY CLOSED
FO -FAIL OPEN
FC -FAIL CLOSED
FAI -FAIL AS IS
LO -LOCKED OPEN
LC -LOCKED CLOSED
T -TRIPPED BY CONTAINMENT ISOLATION SIGNAL - PHASE A
P -TRIPPED BY CONTAINMENT ISOLATION SIGNAL - PHASE B
S -SAFETY INJECTION SIGNAL
SCL -SEISMIC CLASS I DESIGN
IMB -INSIDE MISSILE BARRIER
OMB -OUTSIDE MISSILE BARRIER
MSI -MAIN STEAM ISOLATION SIGNAL
FWI -FEEDWATER ISOLATION SIGNAL

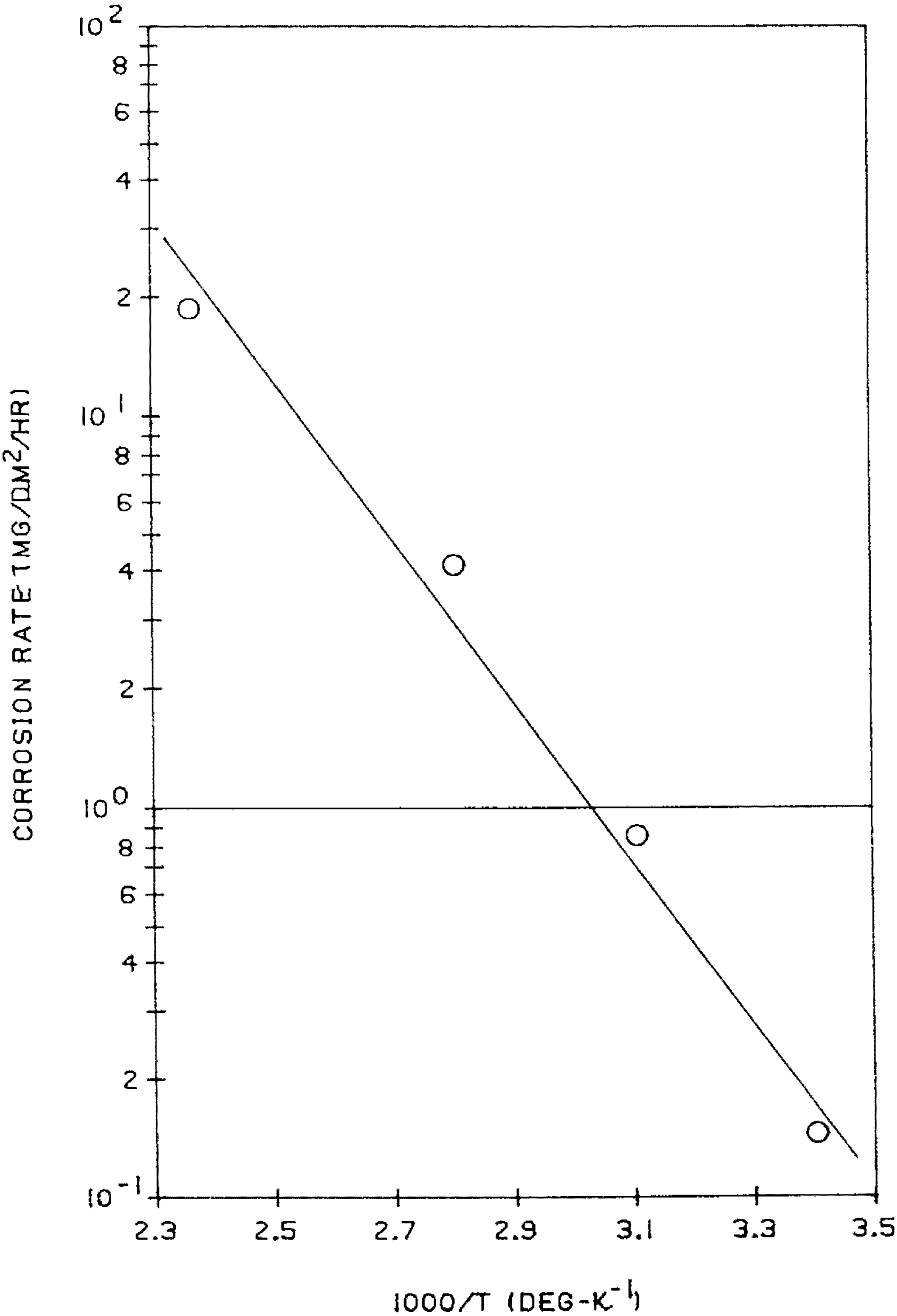
PUBLIC SERVICE ELECTRIC AND GAS COMPANY SALEM NUCLEAR GENERATING STATION	Containment Isolation Legend
	Updated FSAR FIG. 6.2-46

REVISION 6
FEBRUARY 15, 1987



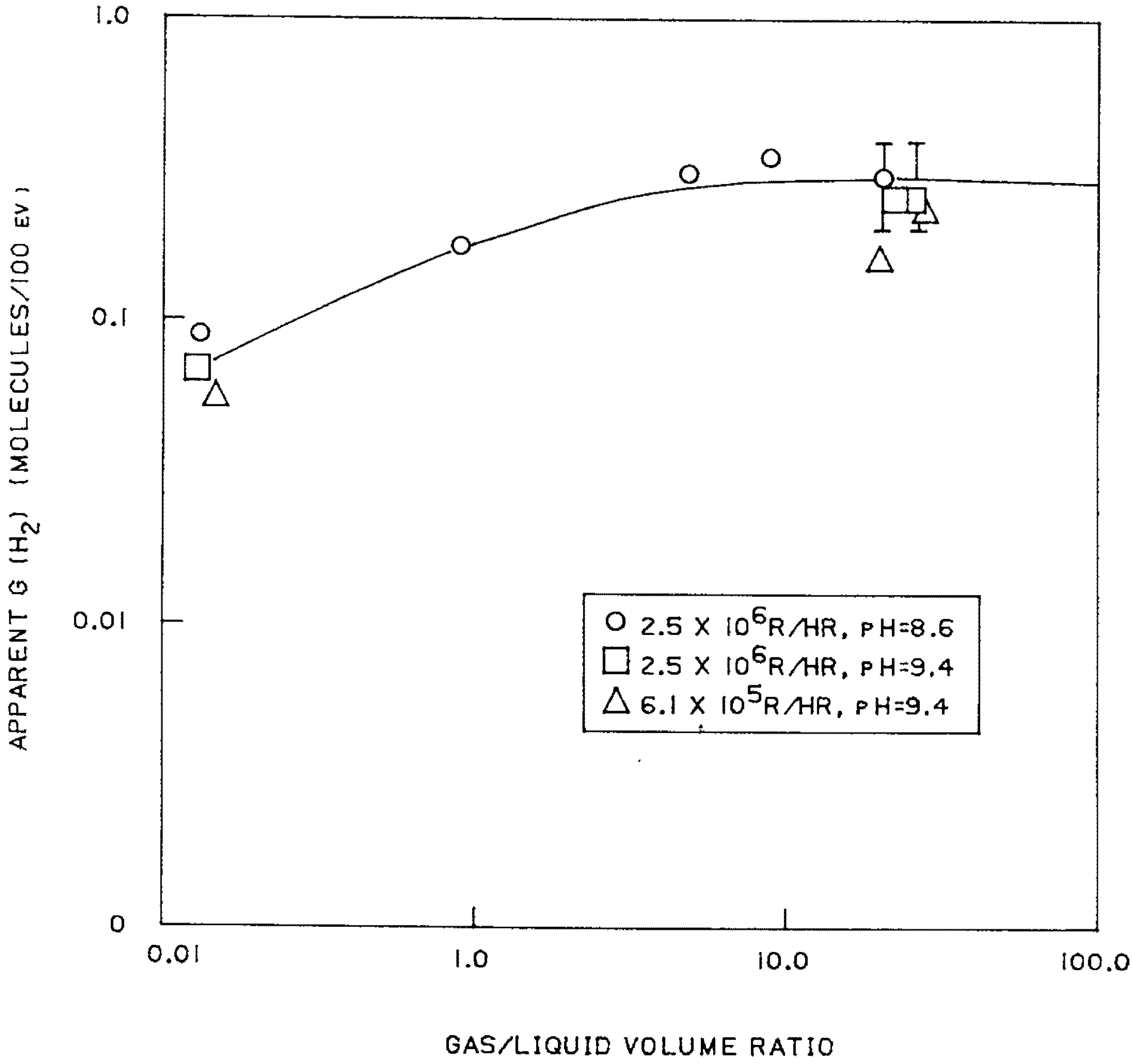
REVISION 6
FEBRUARY 15, 1987

PUBLIC SERVICE ELECTRIC AND GAS COMPANY SALEM NUCLEAR GENERATING STATION	Aluminum Corrosion in DBA Environment
	Updated FSAR FIG. 6.2-47



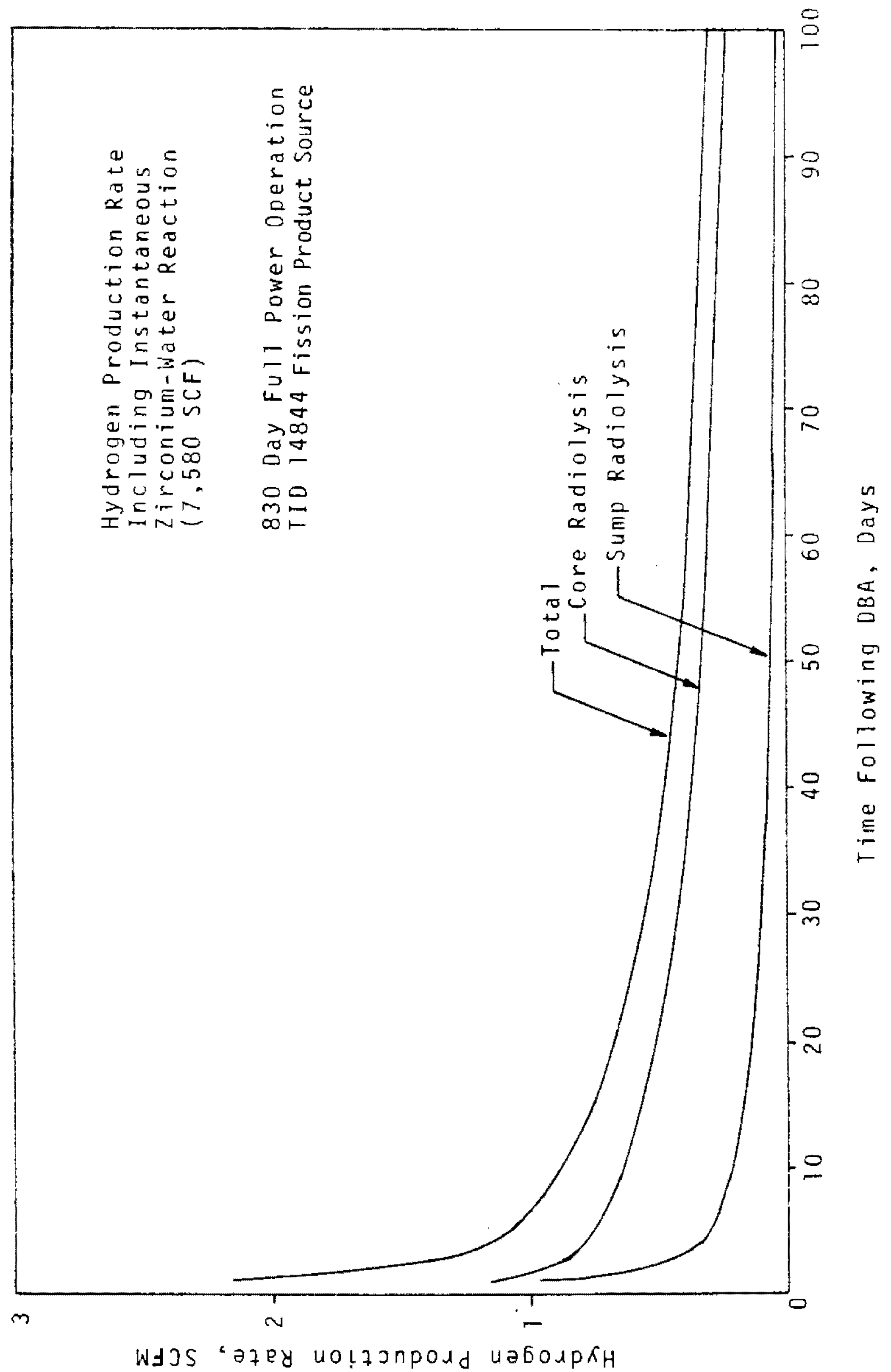
REVISION 6
FEBRUARY 15, 1987

PUBLIC SERVICE ELECTRIC AND GAS COMPANY SALEM NUCLEAR GENERATING STATION	Zinc Corrosion in DBA Environment	
	Updated FSAR	FIG. 6.2-48



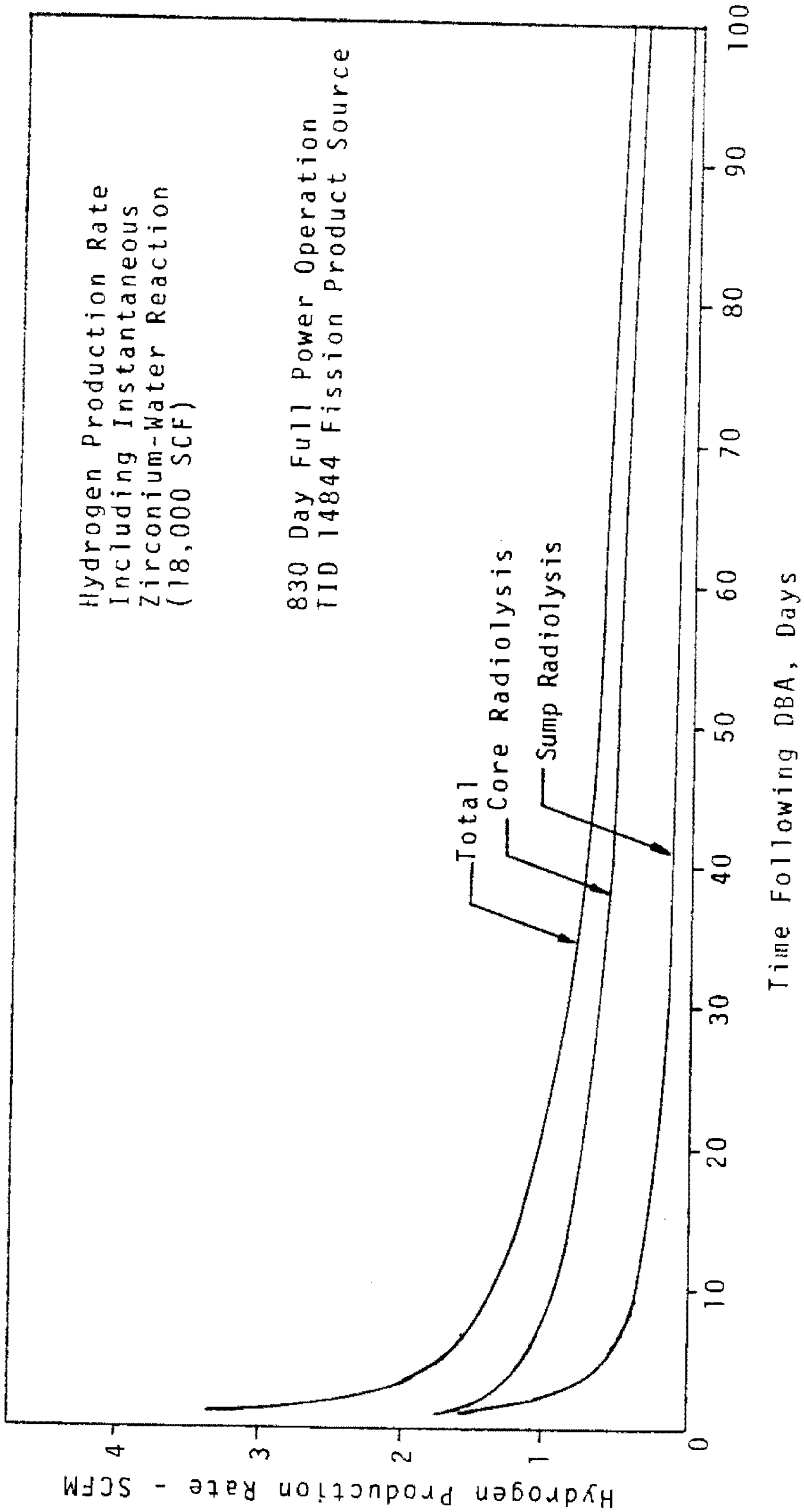
REVISION 6
FEBRUARY 15, 1987

PUBLIC SERVICE ELECTRIC AND GAS COMPANY SALEM NUCLEAR GENERATING STATION	Results of Westinghouse Capsule Irradiation Tests	
	Updated FSAR	FIG. 6.2-49



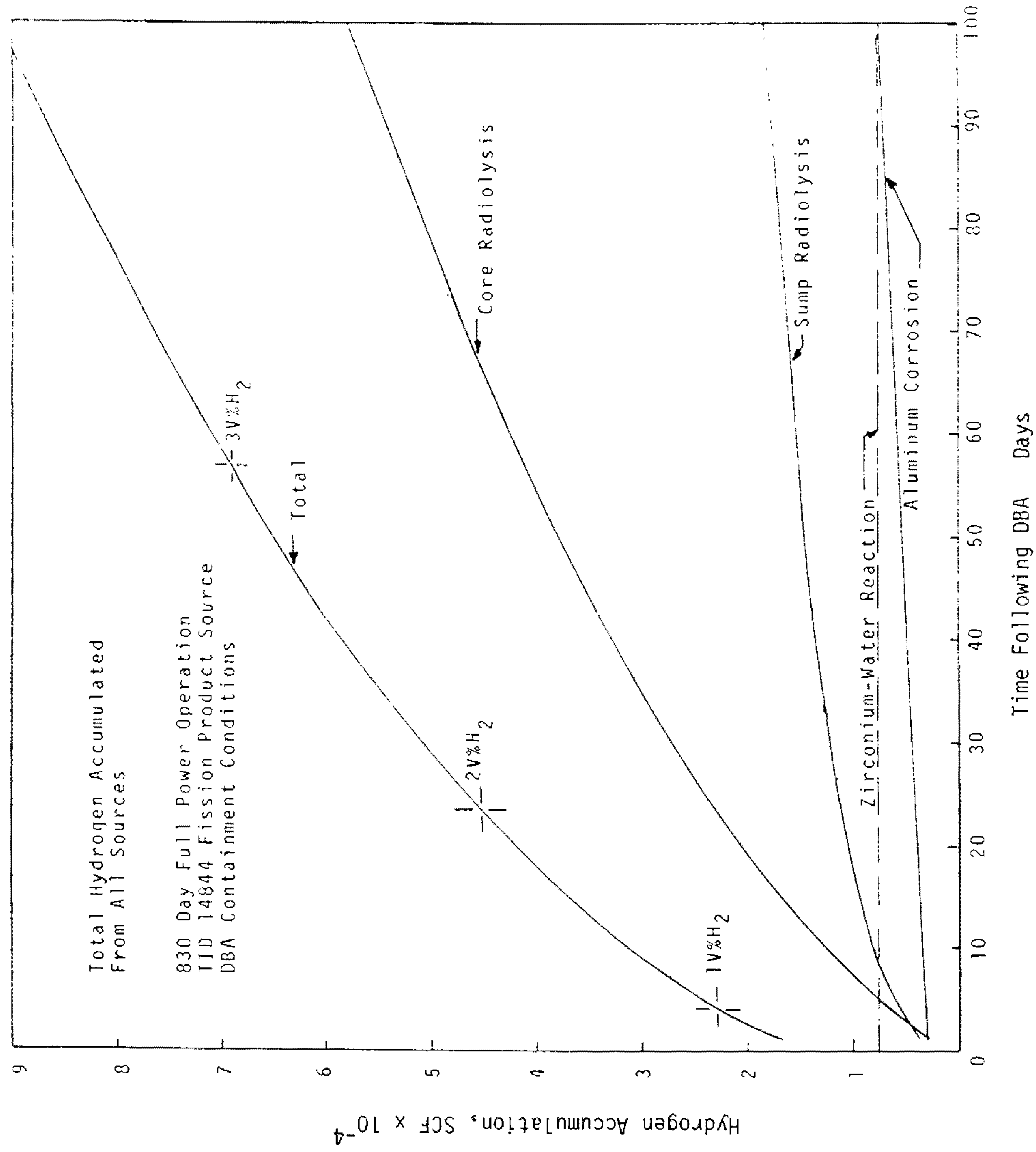
REVISION 6
FEBRUARY 15, 1987

PUBLIC SERVICE ELECTRIC AND GAS COMPANY SALEM NUCLEAR GENERATING STATION	Hydrogen Production Rate – Westinghouse Model	
	Updated FSAR	Figure 6.2-50



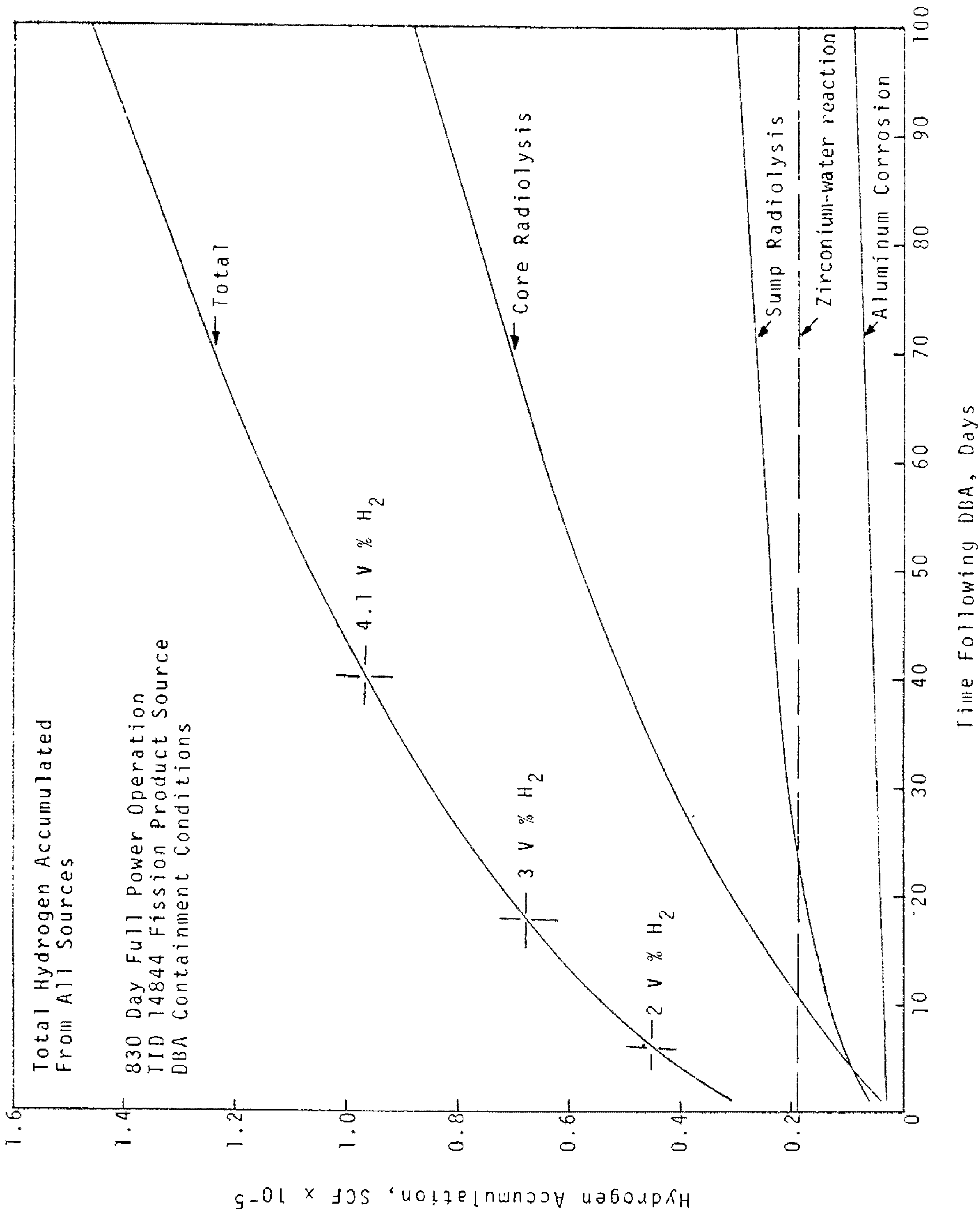
REVISION 6
FEBRUARY 15, 1987

PUBLIC SERVICE ELECTRIC AND GAS COMPANY SALEM NUCLEAR GENERATING STATION	Hydrogen Production Rate – AEC Model	
	Updated FSAR	Figure 6.2-51



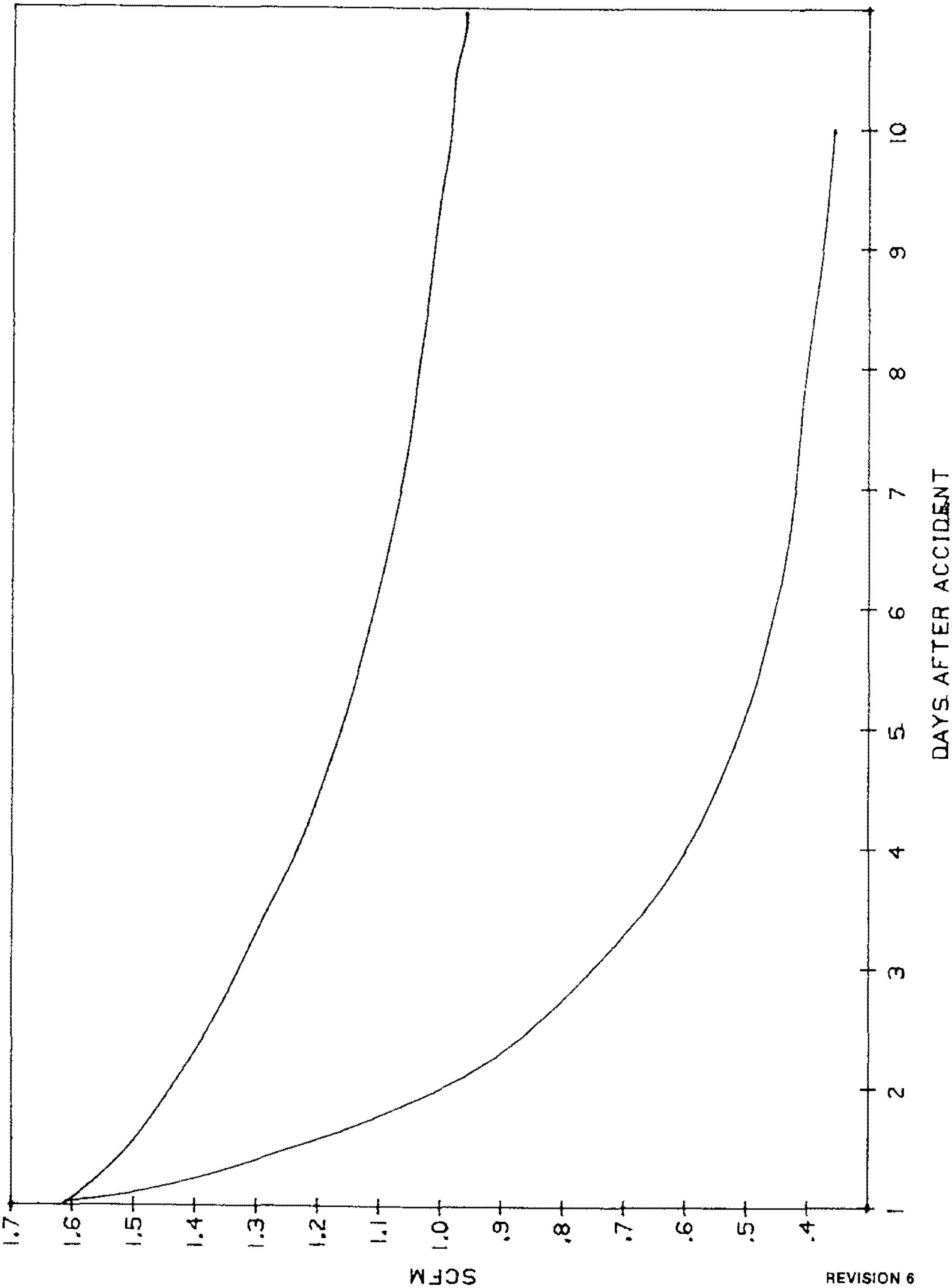
REVISION 6
FEBRUARY 15, 1987

PUBLIC SERVICE ELECTRIC AND GAS COMPANY SALEM NUCLEAR GENERATING STATION	Total Hydrogen Accumulated from All Sources Westinghouse Model
	Updated FSAR Figure 6.2-52



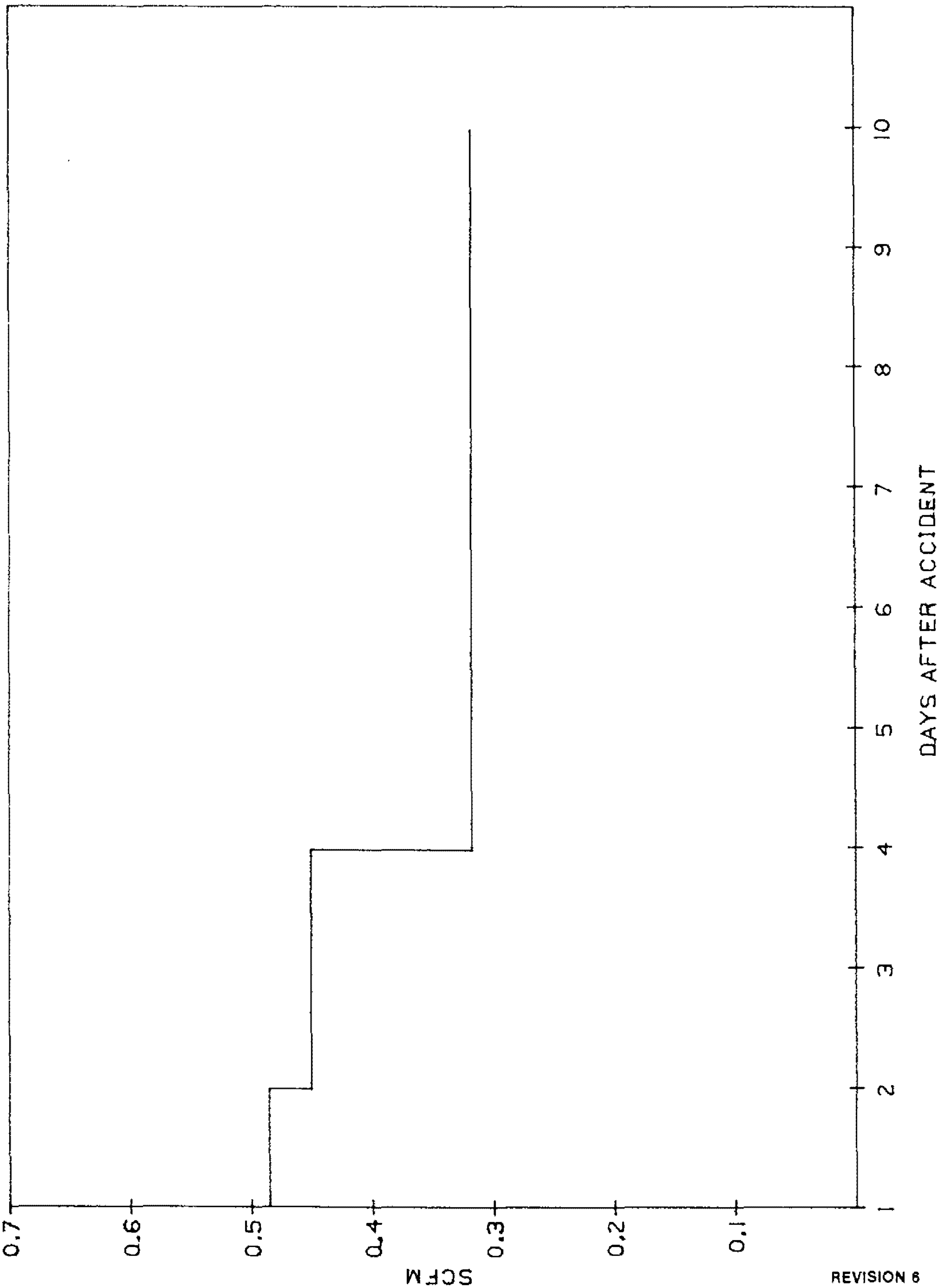
REVISION 6
FEBRUARY 15, 1987

PUBLIC SERVICE ELECTRIC AND GAS COMPANY SALEM NUCLEAR GENERATING STATION	Total Hydrogen Accumulated from All Sources AEC Model
	Updated FSAR — Figure 6.2-53



REVISION 6
FEBRUARY 15, 1987

PUBLIC SERVICE ELECTRIC AND GAS COMPANY SALEM NUCLEAR GENERATING STATION	Hydrogen Generated by Radiolysis
	Updated FSAR FIG. 6.2-54



REVISION 6
FEBRUARY 15, 1987

PUBLIC SERVICE ELECTRIC AND GAS COMPANY SALEM NUCLEAR GENERATING STATION	Hydrogen Generated from Metal Corrosion (Aluminum and Zinc)	
	Updated FSAR	FIG. 6.2-55

THIS FIGURE HAS BEEN DELETED.

Revision 23, October 17, 2007

PSEG Nuclear, LLC SALEM NUCLEAR GENERATING STATION	Salem Nuclear Generating Station TOTAL HYDROGEN GENERATED FROM ALL SOURCES
	Updated FSAR Figure 6.2-56

THIS FIGURE HAS BEEN DELETED.

Revision 23, October 17, 2007

PSEG Nuclear, LLC SALEM NUCLEAR GENERATING STATION	Salem Nuclear Generating Station HYDROGEN ACCUMULATION AFTER DBA
	Updated FSAR Figure 6.2-57

THIS FIGURE HAS BEEN DELETED.

Revision 23, October 17, 2007

PSEG Nuclear, LLC SALEM NUCLEAR GENERATING STATION	Salem Nuclear Generating Station ELECTRIC HYDROGEN RECOMBER Updated FSAR Figure 6.2-58

6.3 EMERGENCY CORE COOLING SYSTEM

6.3.1 Design Bases

6.3.1.1 Range of Coolant Ruptures and Leaks

The Emergency Core Cooling System (ECCS) automatically delivers cooling water to the reactor core in the event of a loss-of-coolant accident (LOCA). This limits the fuel clad temperature and thereby ensures that the core will remain substantially intact and in place, with its essential heat transfer geometry preserved. This protection is afforded for:

1. All pipe break sizes and locations in the Reactor Coolant System (RCS) 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. Pipe breaks in the steam system, up to and including the instantaneous circumferential rupture of the largest pipe in the steam system.

4. A steam generator tube rupture.

The criteria for LOCA evaluations are defined in Section 15.

Furthermore, for the rupture of any steam or feedwater line, the criteria are:

1. Assuming a stuck rod cluster control assembly (RCCA), with or without offsite power, and assuming a single failure in the engineered safety features, there is no

consequential damage to the primary system, and the core remains substantially in place and intact.

2. Energy released to the containment from the worst steam pipe break does not cause failure of the containment structure.

3. Assuming a stuck RCCA, there will be no return to criticality after reactor trip, for a break equivalent to the spurious opening, with failure to close, of the largest of any single relief or safety valve.

Redundancy and segregation of instrumentation and components are incorporated to assure that postulated malfunctions will not impair the ability of the system to meet the design objectives. The system is effective in the event of loss of normal 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 of a loss of coolant, the system is tolerant of a loss of any part of the flow path since backup alternative flow path capability is provided.

6.3.1.2 Fission Product Decay Heat

The ECCS removes the stored and fission product decay heat from the reactor core such that fuel rod damage, to the extent that it would impair effective cooling of the core, is prevented. The acceptance criteria for accidents, as well as accident analyses, are provided in Section 15.

6.3.1.3 Reactivity Required for Cold Shutdown

The ECCS provides shutdown capability for the accidents noted above by means of shutdown chemical (boron) injection. The most critical accident for shutdown capability is the steam line break. Following a steam line break, the RCS, in response to the apparent load increase, would increase reactor power. For larger breaks, an overpower reactor trip would occur. Continued secondary steam blowdown would cool the reactor coolant causing a positive reactivity insertion. Analyses described in Section 15 indicate that breaks large enough to produce a reactivity insertion sufficient to cause a return to criticality also produce sufficient depressurization and shrinkage of the primary coolant to initiate safety injection. The high pressure delivery of concentrated boric acid by the centrifugal charging pumps then re-establishes adequate shutdown margin even for the case where the highest worth control rod is stuck in the fully withdrawn position.

6.3.1.4 Capability to Meet Functional Requirements

In order to ensure that the ECCS will perform its desired function during the accidents listed above, it is designed to tolerate a single active failure during the short-term immediately following an accident, or to tolerate a single active or passive failure during the long-term following an accident.

The ECCS is designed to meet its minimum required level of functional performance with either onsite electrical power system operation (assuming offsite power is not available) or with offsite electrical power system operation for any of the above abnormal occurrences assuming a single failure as defined above.

Portions of the system located within the containment are designed to operate under the most adverse accident conditions without benefit of maintenance and without loss of functional performance for the duration of time the component is required following the accident.

The ECCS is designed to perform its function of ensuring core cooling and providing shutdown capability following an accident under simultaneous safe shutdown earthquake loading.

6.3.2 System Design

6.3.2.1 Schematic Piping and Instrumentation Diagrams

The flow diagram of the ECCS is shown on Plant Drawings 205234 and 205334. The codes and standards to which the individual components of the ECCS are designed are listed in Table 6.3-1. Pertinent design and operating parameters for the components of the ECCS are given in Tables 6.3-2 through 6.3-5.

The operation of the ECCS, following a LOCA, can be divided into two distinct phases: 1) the injection phase, in which any reactivity increase attending the accident is terminated, initial cooling of the core is accomplished, and coolant lost from the primary system is replenished, and 2) the recirculation phase, in which long-term core cooling is provided during the accident recovery period. A discussion of each phase is given below.

Injection Phase of Operation

The major equipment involved in the implementation of the injection phase functions are:

1. Two centrifugal charging pumps

2. Two safety injection pumps

3. Two residual heat removal (RHR) pumps

4. Four accumulators (one for each loop)

5. One boron injection tank (BIT)*

6. Refueling water storage tank (RWST)

* BIT only functions as part of pressure boundary within the safety injection path.

7. Associated valves and piping

The relative importance of the various pieces of injection equipment is dependent upon the size and location of the primary system break. For a large break, the accumulators represent the principal injection mechanism. They are the first piece of equipment to be effective. (For the double-ended cold leg break, they begin to inject approximately 10 seconds after the break, whereas the remainder of the system has a time delay associated with it on the order of 25 seconds). They deliver at a very high flow rate (approximately 47,000 gpm maximum for a double-ended break versus a maximum of 2,400 gpm for the remainder of the system).

The accumulators, utilizing the stored energy of the compressed nitrogen, inject borated water into the cold legs of the reactor coolant piping when the primary system pressure falls below 600 psig. One accumulator is provided for each cold leg of the RCS. They are located inside the containment but outside the missile barrier and are therefore protected against possible missiles. Accumulator water level can be adjusted remotely during normal power operation. Borated makeup water from the RWST is added using a safety injection pump. Water level is reduced by draining to the Chemical and Volume Control System (CVCS) holdup tanks. Samples of the solution in the accumulator tanks are taken at the sampling station for periodic checks of boron concentration. Provisions are also included for remote nitrogen makeup. The accumulators are passive components of the injection system because they require no external source of power or signal in order to function. The remainder of the major pieces of equipment comprising the Safety Injection System (SIS) are active components which are actuated by any of the following safety injection signals:

1. Low pressurizer pressure (2/3)

2. High containment pressure (2/3 Hi)

3. High steam line differential pressure between any two steam generators (2/3)

4. High steam line flow in two of four lines (1/2 measurements per line) in coincidence with either low-low T_{avg} (2/4) or low steam line pressure (2/4)

5. Manual actuation (1/2).

The safety injection signal initiates a reactor trip (this may have already occurred), starts the diesel generators, and initiates the safeguards sequence, which in turn initiates the required action. Finally, a safety injection signal will produce a Phase A containment isolation signal, which results in the closure of the majority of the automatic containment isolation valves.

The active components serve three functions during the injection phase:

1. Provide rapid injection of borated water as a shutdown chemical (boron dissolved in the form of boric acid).

2. Complete the reflooding process for large area ruptures where the initial refill is accomplished by the accumulators.

3. Provide injection for small area ruptures where the primary coolant pressure does not drop below the accumulator pressure for an extended period of time.

In accident analyses with coincident loss of outside power, full flow from the SIS occurs at no later than 25 seconds. The basis of this value is discussed in a later section. This delay time is independent of whether or not the accumulators have injected.

During safety injection, the centrifugal charging pumps deliver borated water at the prevailing RCS pressure to the four cold legs of the RCS. The injection points are separate from those used by the accumulators. The safety injection path is through the BIT. The BIT contains diluted boric acid at the same concentration range as RWST (0 to 2500 ppm). The BIT is normally isolated on the inlet and outlet lines from the cold legs by parallel motor operated gate valves. Both Unit 1 and Unit 2 BIT inlet and outlet isolation valves receive a safety injection signal. However, the diluted boric acid in the BIT is not credited for accident mitigation. The safety injection signal also operates motor-operated valves which transfer the suction of the centrifugal charging pumps from the volume control tank to the RWST.

The safety injection pumps take suction from the RWST and deliver borated water to four cold legs via the accumulator discharge lines. These pumps develop a maximum discharge pressure of about 1520 psig at shutoff, and as a result, deliver to the primary system only after its pressure is reduced below this value. Prior to this, they recirculate water back to the storage tank. This limitation on discharge pressure does not significantly reduce the effectiveness of the safety injection pumps since any break of sufficient size to require safety injection will reduce the coolant pressure below 1500 psig.

In the safety injection mode each of the RHR pumps takes suction from the RWST and delivers borated water to each of the four cold leg connections used by the safety injection pumps, i.e., via the accumulator discharge lines. To ensure that each RHR subsystem can meet this design requirement, the discharge cross tie valves, RH-19's, are required to be open. The RHR pumps deliver only when the RCS is depressurized to about 170 psig.

All active components of the SIS, which operate during the injection phase of a LOCA, are located outside the Containment. The centrifugal charging, safety injection, and RHR pumps discussed above are all located in the Auxiliary Building.

Changeover from Injection Phase to Recirculation Phase

The sequence, from the time of the safety injection signal, for the changeover from injection to recirculation is as follows:

1. First, containment sump level indication shows that sufficient water is delivered to the containment floor to provide adequate submergence of the sump strainer modules and to provide the net positive suction head (NPSH) required for the RHR pumps to change to recirculation.

2. Second, the low-level alarm on the RWST sounds. The operator, at this point, takes appropriate action to switch over to recirculation. One spray pump continues to run until the RWST is nearly empty. The spray additive tank is isolated when the sodium hydroxide solution is depleted.

3. Finally, the low-low level alarm on the RWST sounds. At this time, the operator stops the spray pump. Spraying is continued at this time for approximately 14 hours (Unit 1) and 6.5 hours (Unit 2) using the RHR pumps pumping to the spray header located at the RHR heat exchanger discharge.

The changeover from injection to recirculation is affected by the operator in the control room via a series of manual switching operations. The changeover sequence is given in Table 6.3-6.

Recirculation Phase of Operation

After the injection operation, water collected in the containment sump is cooled and returned to the RCS by the low head/high head recirculation flow path. The RCS can be supplied simultaneously from the RHR pumps, and from a portion of the discharge from the residual heat exchangers that is directed to the charging pumps and safety injection pumps, which return the water to the RCS. The latter mode of operation assures flow in the event of a small rupture where the depressurization proceeds more slowly, such that the RCS pressure is still in excess of the shutoff head of the RHR pumps at the onset of recirculation.

NRC issued Information Notice 87-63, which identified the possibility of unintended flow paths during the recirculation modes of operation, which could increase RHR pump maximum flow potential. With "loop around" flow considered, the highest RHR pump flow was calculated to occur during the cold leg recirculation mode of operation following a postulated failure of one of two operating RHR pumps.

At approximately 14.0 hours (Unit 1) and 6.5 hours (Unit 2) after the switchover to cold leg recirculation, hot leg recirculation will be initiated to assure termination of boiling. To ensure adequate flow performance, simultaneous flow delivery to the RCS cold legs and RCS hot legs are required. At a minimum, one safety injection pump is required to be aligned and operated in a hot leg recirculation flow mode. For a LOCA during Mode 4, with RCS cold leg temperature <312°F, a SI pump may not be available for the hot leg recirculation. In this instance, the RHR flowpath through RH26 would be utilized to provide hot leg recirculation.

Since the injection phase of the accident is terminated before the RWST is completely emptied, all pipes are kept filled with water before recirculation is initiated. Water level indication and alarms on the RWST inform the operator that sufficient water has been injected into the containment to allow initiation of recirculation with the RHR pumps and to provide ample warning to terminate the injection phase while the operating pumps still have adequate NPSH. In addition, two level switches are provided inside the containment that provide a signal to the control room console when the water level in containment is sufficient to provide adequate submergence of the strainer modules and to provide adequate NPSH to the RHR pumps.

Redundancy in the external recirculation loop is provided for by the inclusion of duplicate charging, safety injection, and RHR pumps and residual heat exchangers. Inside the containment, the High Pressure Injection System is divided into two separate flow trains. For cold leg recirculation, the charging pumps deliver to all four cold legs and the safety injection pumps also deliver to all four cold legs by separate flow paths. For hot leg recirculation, each safety injection pump delivers through separate paths to two reactor coolant loops.

The low head pumps take suction through separate lines from the containment sump and discharge through separate paths to the RCS. The sump design provides adequate NPSH for the RHR pumps to operate in the recirculation mode.

A debris interceptor is installed around the perimeter of each strainer to obstruct debris transport to the strainer. The debris interceptors consist of grating with 1/8" perforated plate attached to the downstream side of the grating. All generated debris is conservatively assumed to be transported to the debris interceptor or the strainer, even though there are some areas of the sump pool where debris would be stopped. The ECCS is not adversely impacted by the debris passing downstream of the sump screen.

The sump isolation valves are located in small steel-lined controlled leakage compartments. This arrangement contains any isolation valve leakage and assures that leakage during long-term recirculation will not impair the integrity of the containment or recirculation system.

The containment sump is described in Section 6.3.2.2. Special attention is paid to the design, materials, and fabrication of the sump, the suction piping, guardpipes, and isolation valves to provide assurance that the sump and piping will remain functional under the accident environment and continue to provide suction for the long-term recirculation.

A sample connection is provided in the RHR System to remotely sample recirculated liquid in the sample room during post-accident operations. Additives can be supplied to the sump through the existing plant design features within 48 hours from switchover to cold leg recirculation mode, if measurements indicate the sump liquid is outside the desired pH range of 7.0 to 10.0. A minimum sump liquid pH of 7.0 will minimize the potential for chloride induced stress corrosion cracking of stainless steel components (Reference 3). Note: Branch Technical Position MTEB 6.1 supports a lower limit of 7.0.

6.3.2.2 Equipment and Component Description

The major components of the ECCS are described below.

Accumulators

The accumulators are pressure vessels containing borated water and pressurized with nitrogen gas. During normal operation, each accumulator is isolated from the RCS by two check valves in series. Should the RCS pressure fall below the accumulator pressure, the check valves open and borated water is forced into the RCS. One accumulator is attached to each of the cold legs of the RCS. Mechanical operation of the swing disc check valves is the only action required to open the injection path from the accumulators to the core via the cold leg.

The accumulators are passive engineered safety features because the gas forces injection; no external source of power or signal transmission is needed to obtain fast-action, high-flow capability when the need arises. One accumulator is attached to each of the cold legs of the RCS.

The design capacity of the accumulators is based on the assumption that flow from one of the accumulators spills onto the containment floor through the ruptured loop, and the flow from the remaining accumulators provides sufficient water to fill the volume outside of the core barrel below the nozzles, the bottom plenum, and a portion of the core. This assumption is based on no water remaining in the vessel after blowdown but takes credit for the water delivered by three accumulators. All the effects that could cause loss of accumulator water are evaluated in Section 15.

The accumulators are carbon steel, 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. The accumulator design parameters are given in Table 6.3-2.

The margin between the minimum operating pressure and design pressure provides a band of acceptable operating conditions within which the Accumulator System meets its design core cooling objectives. The band is sufficiently wide to permit the operator to minimize the frequency of adjustments in the amount of contained gas or liquid to compensate for leakage.

The accumulator tank pressure and level are continuously monitored during plant operation and flow from the tanks can be checked at any time using test lines.

The accumulators and the safety injection piping up to the final isolation valve are maintained full of borated water at refueling water concentration while the plant is in operation. The accumulators and injection lines will be refilled with borated water as required by using the safety injection pump. Any excessive flow from the safety injection pumps can be recirculated back to RWST through a bypass line off the pump discharge header.

Level and pressure instrumentation are provided for each accumulator tank.

Boron Injection Tank

The BIT contains between 0 to 2500 ppm boric acid solution and is connected to the discharge of the centrifugal charging pumps. Upon actuation of the safety injection signal, the flow from the centrifugal charging pumps is routed through the BIT into the RCS. Although the BIT is part of the safety injection pressure boundary, the diluted form of boric acid in the BIT is not credited for accident mitigation.

The BIT is maintained in a 100-percent full condition. The BIT is kept 100% full administratively by filling and venting periodically using procedural controls. The parallel motor-operated gate valves at the inlet and outlet of the BIT are kept normally closed. The BIT pressure also can be monitored from the Control Room console.

Chapter 15, Accident Analysis, conservatively assumes that the BIT is filled with unborated water (0-ppm boric acid) when analyzing core response, containment integrity, and equipment environmental qualification.

The normal temperature of the BIT and its associated lines is at ambient temperature. Heaters in the BIT and associated line heat tracing are not required because of the low concentration of boric acid.

Additionally, a permanently installed facility is provided to enable periodic checking of boric acid concentration in the BIT to ensure that it is within acceptable levels.

The equipment employed with the BIT is designed to the same quality standards and codes as the rest of the engineered safety features equipment and is Seismic Class I design.

Refueling Water Storage Tank

In addition to its usual service of supplying borated water to the refueling canal for refueling operations, the RWST provides borated water to the centrifugal charging pumps, safety injection pumps, RHR pumps, and the containment spray pumps for the LOCA. During normal power operation, storage tank water is valved to the suction of the safety injection pumps, RHR pumps, and containment spray pumps. The suction of the centrifugal charging pumps is automatically valved to the storage tank by a safety injection signal. The positive displacement charging pump has a dedicated suction line to the RWST that may be used to support safe shutdown of the opposite unit if it loses all charging capability.

The minimum quantity of the RWST is 364,500 gallons and is based on the requirement for filling the refueling canal. This volume also provides a sufficient amount of borated water to meet the following conditions:

1. Adequate volume during the injection phase to meet ECCS design objective

2. Increase the concentration of recirculation water to a point that assures no return to criticality with the reactor at cold shutdown and all control rods, except the most reactive rod cluster control assembly, inserted into the core

3. Fill the containment sump to permit the initiation of recirculation

4. Fulfill spray requirements

The water in the tank is borated to a concentration that assures reactor shutdown by approximately 5 percent Δk/k when all rod cluster control assemblies are inserted and when the reactor is cooled down for refueling.

The design parameters are presented in Table 6.3-4.

The RWST is classified Class I for seismic design. This requires that there will be no loss of function or spillage of its contents for loads from two times the design earthquake when combined with the normal loads. The effect of water sloshing within the tank is considered in determining the seismic loads.

Compressive stresses in the shell of the tank are limited by allowable buckling stresses, determined in a manner similar to Paragraph I-1150 of the ASME Code, Section III. The 20" diameter suction line reinforcement plate on the Salem Unit 1 RWST is defined as the pressure boundary (Reference 4). The tank shell does not function as the pressure boundary at the area covered by the reinforcement plate.

The tank is provided with a high-level alarm, and the overflow line is piped to the diked area around the No. 13 Chemical and Volume Control holdup tank, from where any overflow can be pumped to the Liquid Waste Disposal System. The overflow line includes a collection pot, which is also provided with a high level alarm. Both alarms are indicated in the control room.

Anti-vortex plates are installed in the containment spray suction line from the RWST. Verification of vortex control in the containment sump is discussed in Section 6.3.4.4.

The temperature of the water in the RWST is prevented from dropping below 32°F by automatic initiation of a Circulating and Heating System, which draws water from the tank through the Safety Injection System suction pipe and the Containment Spray System suction pipe. The water is then pumped through a heat exchanger located in the Auxiliary Building and enters the tank via the return line from the refueling water purification pump. Thus, the water in the tank and the water in the connecting piping is heated and circulated.

The instrumentation that actuates the Heating System senses the temperature in the SIS suction pipe. This temperature is monitored and alarmed in the Control Room. The system has provisions for local manual actuation.

Electrical heat tracing is provided for the instrument connections to the tank and for that portion of the tank drain piping which could otherwise freeze. Thermal insulation is also provided for the exposed piping.

Valves 1SJ30 and 1SJ69 (in the RWST suction line to the ECCS pumps) are provided with the same type of redundant position indication as the accumulator discharge valves, described in Section 6.3.2.15.

Residual Heat Removal Pumps

The two RHR pumps are vertical electric motor driven single stage pumps. All parts of the pump in contact with the pumped fluid are stainless steel or of equivalent corrosion resistant material.

A minimum flow bypass line is provided for the pumps to recirculate through the residual heat exchangers and return the cooled fluid to the pump suction should these pumps be started with their normal flow paths blocked. Once flow is established to the RCS, the bypass line is automatically closed. This line prevents deadheading the pumps and permits pump testing during normal operation.

Centrifugal Charging Pump

The two centrifugal charging pumps are horizontal electric motor driven multistage pumps. All parts of the pump in contact with the pumped fluid are stainless steel or equivalent corrosion-resistant material. Min-flow protection is provided during ECCS operation. Because these pumps operate during normal operation as well as during safety injection, the min-flow valves, 1(2)CV139 and 140, are open at all times, unless specific direction is given by plant procedures.

Safety Injection Pump

The two safety injection pumps are horizontal electric motor driven multistage pumps. All parts of the pump in contact with the pumped fluid are stainless steel or equivalent corrosion-resistant material. A minimum flow bypass line is provided on each pump discharge to recirculate flow to the RWST in the event that the RCS pressure is above the shutoff head of the pumps. In Unit 1, a 100 gpm test line is provided in parallel to the min-flow line. This line is used for inservice testing and is locked out at other times. A similar 100 gpm test line is not provided in Unit 2.

Pump Design, Materials, and Fabrication

The pressure-containing parts of the pumps are stainless steel castings conforming to ASTM A-351 Grade CF8 or CF8M, stainless steel forgings procured per ASTM A-182 Grade F304 and F316, or carbon steel forgings to ASTM A-181, Grade 1, clad with austenitic steel. Parts fabricated of stainless plate are constructed to ASTM A-240, Type 304 or 316. All bolting material meets or exceeds ASTM A-193, A-194 and ASME SA-564 Grade 360 condition H1100 or other equally or greater rated fasteners.

Materials such as weld-deposited Stellite or Colmonoy are used at points of close running clearances in the pumps to prevent galling and to ehsure continued performance ability in high velocity areas subject to erosion. In other cases, wear points are of ASTM A-420 Grade stainless steel, heat treated to give the required antigalling properties.

All pressure-containing parts of the pumps are chemically and physically analyzed and the results are checked to assure conformance with the applicable American Society for Testing and Materials' specification. In addition, all pressure-containing parts of the pump are liquid penetrant inspected in accordance with Appendix VIII of Section VIII of the ASME Code. The acceptance standard for the liquid penetrant test is the ASME Pump and Valve Code.

Pump design is reviewed with special attention to the reliability and maintenance aspects of the working components. Specific areas include evaluation of the shaft seal and bearing design to determine that they are adequate for the specified service.

During pump fabrication and installation, where welding of pressure-containing parts was necessary, a welding procedure including joint detail was submitted for review and approval by Westinghouse.

This procedure included evidence of qualification necessary for compliance with Section IX of the ASME Code Welding Qualifications. This requirement also applied to any repair welding performed on pressure-containing parts. Subsequent to construction, Welding Procedure Specifications (WPS) are approved and qualified in accordance to the current requirements of Section IX of the ASME Boiler and Pressure Vessel Code.

The pressure-containing parts of the pump were assembled and hydrostatically tested to 1.5 times the design pressure for 30 minutes.

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

A qualitative analysis shows that any flooding resulting from a leak in one pumping train will not incapacitate the redundant pump.

Heat Exchangers

The two residual heat exchangers of the RHR system cool the recirculated sump water. These heat exchangers are sized for the cooldown of the RCS. The design parameters of the heat exchangers are presented in Section 5.5.

The residual heat exchangers are designed to the ASME Code and to conform to the requirements of the Tubular Exchanger Manufacturers' Association for Class R heat exchangers.

Additional design and inspection provisions include:

1. Confined-type gaskets

2. General construction and mounting brackets suitable for the plant seismic design requirements

3. Tubes and tube sheet capable of withstanding full shell side pressure and temperature with atmospheric pressure on the tube side

4. Radiographic inspection in accordance with Sections UW-11, UW-12-b, and UW52 of ASME Section VIII

5. Ultrasonic inspection in accordance with Paragraph N-324.3 of Section III of the ASME Code of all tubes before bending

6. Penetrant inspection in accordance with Paragraph N-627 of Section III of the ASME Code of all welds and all hot or cold formed parts

7. A hydrostatic test duration of not less than 30 minutes

8. The witnessing of hydro and penetrant tests by a qualified inspector

9. A thorough final inspection of the unit for workmanship and the absence of any gouge marks or other scars that could act as stress concentration points

10. A review of the radiographs and of the certified chemical and physical test reports for all material 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 a SA 515GR70 carbon steel shell, SA-213 TP-304 stainless steel tubes, SA-105 with Type 304 stainless steel overlay channel cover and a tube sheet of forged steel SA-105 GR.III with 1/4-inch minimum TP-304 weld overlay.

Valves (General)

Design features employed to minimize valve leakage include the following:

1. Other valves that are normally open, except those valves which perform a control function, are provided with backseats to limit stem leakage.

2. Normally closed globe valves are installed with recirculation fluid pressure under the seat to prevent stem leakage of recirculated (radioactive) water.

3. Relief valves are enclosed, i.e., they are provided with a closed bonnet and discharge to a closed system or the containment sump.

4. Control and motor operated valves (3 inches and above) exposed to recirculation flow may have double packed stuffing boxes and stem leakoff connections to the Waste Processing System.

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

Valving is specified for exceptional leak tightness. All valves, except those which perform a control function, are provided with backseats that are capable of limiting leakage to less than 1.0 cc per hour per inch of stem diameter, assuming no credit taken for valve packing. Normally closed globe valves are installed with recirculation flow under the seat to prevent leakage of recirculated water through the valve stem packing. Relief valves discharge to an enclosed system or the containment sump. Control and motor-operated valves, 3 inches and above that are exposed to recirculation, may be provided with double-packed stuffing boxes and stem leakoff connections which are piped to the Equipment Drain System.

The check valves that isolate the ECCS from the RCS are installed near the reactor coolant piping to reduce the probability of an injection line rupture causing a LOCA.

Portions of the ECCS piping are protected by relief valves. The relieving capacity of these valves is based on a flow several times greater than the expected leakage rate through the check valves. The valves relieve to the pressurizer relief tank.

The RHR System is protected by four relief valves: one on the header from the RCS to the pumps, two on the cold leg injection headers, and one on the hot leg return header.

These valves discharge to the containment sump.

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

Motor-Operated Valves

The pressure containing parts (body, bonnet, and discs) of the motor-operated valves employed in the SIS are designed per criteria established by the ANSI B16.5 or MSS SP66 specifications. The materials of construction for these parts are procured per ASTM A182, F316 or A351, GR CF8M, or CF8. All material in contact with the primary fluid, except the packing, is austenitic stainless steel or equivalent corrosion-resistant material. The pressure-containing cast components are radiographed in accordance with ASTM E-94 and the acceptance standard as outlined in ASTM E-71. The body, bonnet, and discs are liquid penetrant inspected in accordance with ASME Pump and Valve Code. The liquid penetrant acceptable standard is outlined in the ASME Pump and Valve Code.

When a gasket is employed, the body-to-bonnet joint is designed per ASME Code Section VIII and/or ANSI 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, or equivalent.

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 minutes. Any leakage is cause for rejection. The seating design is of the Darling parallel disc design, the Crane flexible wedge design, or the equivalent. These designs have the feature of releasing the mechanical holding force during the first increment of travel. Thus, the motor operator has to work only against the frictional component of the hydraulic unbalance on the disc and the packing box friction. The discs are guided throughout the full disc travel to prevent chattering and provide ease of gate movement. The seating surfaces are hard faced (Stellite No. 6 or equivalent) to prevent galling and reduce wear.

The stem material is ASTM A276, Type 316, Condition B or precipitation hardened 17-4 PH stainless procured and heat treated to Westinghouse Specifications. These materials are selected because of their corrosion resistance, high tensile properties, and their resistance to surface scoring by the packing. The valve stuffing box of motor-operated valves having leakoff is designed with a lantern ring leak-off connection with a minimum of a full set of packing below the lantern ring; a full set of packing is defined as a depth of packing equal to 1 1/2 times the stem diameter. The experience with this stuffing box design and the selection of packing and stem materials have been very favorable in both conventional and nuclear power plants.

The motor operator is extremely rugged and is noted throughout the power industry for its reliability. The unit incorporates a "hammer blow" feature that allows the motor to impact the discs away from the fore or backseat upon opening or closing. This "hammer blow" feature not only impacts the disc but allows the motor to attain its operational speed.

Each 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 fast operated valves up to and including 8 inches, 10-second maximum operators are provided. For all fast operated valves above 8 inches, the operating speed is 49 inches per minute. For slow operators, 12 inches per minute is specified for valves up to and including 8 inches. For all slow valves above 8 inches, 120-second maximum closing time is specified.

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.

The carbon steel valves are built to conform with ANSI B16.5. The materials of construction of the body, bonnet, and disc conform to the requirements of ASTM A105 Grade II, A181 Grade II, or A216, Grade WCB or WCC. The carbon steel valves pass only non-radioactive fluids and are subjected to hydrostatic test as outlined in MSS SP61 except that the test pressure is maintained for at least 30 minutes. Since the fluid controlled by the carbon steel valves is not radioactive, the double packing and seal weld provisions are not provided.

Accumulator Check Valves

The pressure-containing parts of this valve assembly are designed in accordance with ASME Boiler and Pressure Vessel Code, Section III, 1968. 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. The cast pressure-containing parts, machined surfaces, finished hard facings, and gasket bearing surfaces are liquid penetrant inspected per ASME Pump and Valve Code and the acceptance standard is as outlined in the ASME Pump and Valve Code. The final valve is hydrotested per MSS SP-66 except that the test pressure is maintained for at least 30 minutes. The seat leakage test is conducted in accordance with the manner prescribed in MSS SP-61 except that the acceptable leakage is 3 cc/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 resistance, tensile and bearing properties.

The disc and seat rings are manufactured from a forging. The mating surfaces are hard faced with Stellite No. 6 to improve the valve seating life. The disc is permitted to rotate, providing a new seating surface after each valve opening.

The valves are intended to be operated in the closed position with a normal differential pressure across the disc of approximately 1600 psi. The valves shall remain in this position except for testing and safety injection. Since the valves will not be required to normally operate in the open condition and hence be subjected to impact loads caused by sudden flow reversal, it is expected that these valves will perform their required functions without difficulty.

When the valve is required to operate, a differential pressure of less than 25 psig will shear any particles that may otherwise 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 the low boric acid concentrations used.

The experience derived from the check valves employed in the Emergency Injection System of the Carolina-Virginia Tube Reactor (CVTR) in a similar system indicates that the system is reliable and workable.

The CVTR Emergency Injection System, normally maintained at containment ambient conditions was separated from the main coolant piping by a single 6-inch check valve. A leak detection was provided at a proper elevation to accumulate any leakage coming back through the check valve and level alarm provided a signal on excessive leakage. The pressure differential was 1500 psi and the system was stagnant. The valve was located 2 to 3 feet from the main coolant piping, which resulted in some heatup and cooldown cycling. The CVTR went critical late in 1963 and operated until 1967, during which time the level sensor in the leak detector never alarmed due to check valve leakage.

Relief Valves

The accumulator relief valves are sized to pass nitrogen gas at a range in excess of the accumulator gas fill line delivery rate. The relief valves will also pass water in excess of the expected leak rate, but this is not necessary because the time required to fill the gas space gives the operator ample opportunity to correct the situation. For an inleakage rate 15 times the manufacturing test rate, there will be an excess of 1000 days before water will reach the relief valves. Prior to this, level and pressure alarms would have been actuated.

The ECCS relief valves are provided to relieve any pressure, above design, that might build up in the safety injection piping. The valve will pass a flow rate which is far in excess of the manufacturing design leak rate of 24 cc/hr.

Valve Leakage Specifications

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

1. Conventional globe - 3 cc/hr/in. of nominal pipe sizes

2. Gate valves - 3 cc/hr/in. of nominal pipe size; 10 cc/hr/in. for 300 and 150 pound ANSI Standard

3. Motor-operated gate valves - 3 cc/hr/in. of nominal pipe size; 10 cc/hr/in. for 300- and 150-pound ANSI Standard

4. Check valves - 3 cc/hr/in. of nominal pipe size; 10 cc/hr/in. for 300- and 150-pound ANSI Standard

5. Accumulator check valves - 3 cc/hr/in. of nominal pipe size

Piping

All ECCS piping in contact with borated water is austenitic stainless steel. All major piping joints are welded except for the flanged connections at pumps, heat exchangers, relief valves, filter housings, removable spools, and in-line flow instrumentation.

The piping beyond the accumulator stop valves is designed for RCS conditions.

The safety injection pump suction piping from the RWST is designed for low pressure losses to meet NPSH requirement of the pumps.

The safety injection high pressure branch lines are designed for high pressure losses to limit the flow rate out of the branch line in the event of rupture at the connection to the reactor coolant loop. The branch lines are sized so that a break will not result in a violation of the design criteria for the ECCS.

The piping is designed to meet the requirements set forth in (1) the ANSI B31.1 Code for Pressure Piping, (2) ANSI Standards B36.10 and B36.19, and (3) ASTM Standards.

Pipe fitting materials are procured in conformance with all requirements of the latest ASTM and ANSI specifications. All materials are verified for conformance to specifications 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 between the reactor coolant pipes and the isolation stop valves conform to ASTM A376 and meet the supplementary requirement S6 covering an ultrasonic test, on 100 percent of the pipe wall volume. The S6 supplementary requirement applies to pipes of nominal sizes 3 inches and larger.

3. Pipe fittings in the branch lines between the reactor coolant pipes and the isolation stop valves conform to the requirements of ASTM A403; all fittings have requirements for liquid penetrant examination.

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 1/2 inches and larger are of the full penetration type. Reducing tees are used where the branch size exceeds one-half of the header size. All welding is performed by welders and welding procedures qualified in accordance with the ASME Code Section IX, Welding Qualifications.

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 Appendix B of ANSI B31.7. In addition, butt welds are liquid penetrant examined in accordance with the procedures of Appendix B of ANSI B31.7. Finished branch welds are liquid penetrant examined on the outside and where size permits, on the inside root surfaces.

A post-bending 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 clean-up 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.

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 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.

Pump and Valve Motors

Emergency Core Cooling System pump motors are used on the following pumps:

1. Centrifugal charging

2. Safety injection

3. Residual heat removal

The motors are designed in accordance with the National Electric Manufacturers' Association (NEMA) Standards. These standards are used by the industry and provide requirements for construction, test, performance, and manufacture of ac and dc motors and generators, that by experience demonstrate a high quality level. (NEMA, Standard Publication for Motors and Generators, No. MG 1-1967.)

Emergency Core Cooling System motors are specified to an Equipment Specification and the following design classifications:

1. Drip proof enclosure

2. Class B insulation system or better

3. Service factor rating of 1.15

4. 80 percent starting voltage capability.

The integrity of the insulation system is considered of prime importance. To assure this integrity, motors are sized such that NEMA temperature limits for the service factor rating of the motor are not exceeded (NEMA MG 1).

Table 6.3-7 shows system parameters and brake horsepower for both normal and accident conditions. The brake horsepower requirements are well below NEMA horsepower ratings. These motors will operate below the temperature limits as specified by NEMA MG 1. Further, complete engineering tests are performed on all prototype motor frame sizes to confirm design calculations.

Motors Outside the Containment

Motor electrical insulation systems are supplied in accordance with USAS, IEEE, and NEMA standards and are tested as required by such standards. Temperature rise design selection is such that normal long life is achieved even under accident loading conditions.

Criteria for motors of the ECCS require that under normal plant operating conditions the motors operate below their nameplate rated horsepower, i.e., below a 1.0 service factor. For no other anticipated operating mode including safeguards operation do the motors exceed the maximum rating allowed by the nameplate, including their specified 1.15 service factor.

Motors Inside the Containment (Valve Motor Operators)

Tests which demonstrate the adequacy of valve motor operators to be functional after exposure to high temperatures, pressures, and radiation have been conducted. The results of the tests are confirmed in Reference 1.

Containment Sump

The physical location of the containment sump is shown on Plant Drawing 208915. All water entering the containment sump will have been strained by the train of strainer modules with 1/12-inch diameter holes that connect to the sump pit. Pump cavitation is minimized by the design of the sump enclosure, so an anti-vortex baffle is not needed.

The sump design differs from Regulatory Guide 1.82 in the following ways:

1. The small drainage sump for collecting and monitoring normal leakage within the containment is at the same location as the RHR sump (see Figure 6.3-3). The Liquid Radwaste and RHR Systems share a common sump pit. A plate is installed to isolate normal drainage from the RHR pump suction.

2. The containment sump screening consists of a train of strainer modules, interconnected with a channel leading back to the sump pit.

During a postulated Loss Of Coolant Accident (LOCA), debris is generated due to jet impingement from RCS pipe break within the bioshield area. All generated debris is conservatively assumed to be transported to the debris interceptor or the strainer, even though there are some areas of the sump pool where debris would be stopped. The debris is also transferred towards the sump by containment spray. The debris generated and subsequently transported to the containment sump is documented in References 6 and 7.

The following is a summary of insulation materials used inside containment:

Reflective: This is an all-metallic stainless steel insulation. This material is used on the RCS as well as portions of the SGBD and SGFW piping inside containment and on the Unit 2 steam generators and on the Unit 2 steam generator blowdown lines inside the steam generator cubicles.

Encapsulated: This is a ceramic fiber insulation "cera-blanket" totally enclosed in a rigid stainless steel structure. This material is used on the ECCS piping and equipment in the containment.

Semi-Encapsulated: (Kaowoll Ceramic Fiber) This application of "cera-blanket" insulation utilizes an outer heavy gage stainless steel surface with an interior surface of formed stainless steel foil or heavy gage stainless steel channels and straps (panel insulation). Foil-enclosed insulation is used for heat retention on Nuclear Class 3 piping and equipment.

Min "K":	This is a high-efficiency powder-like insulation totally enclosed in stainless steel. Small amounts of this insulation are used on the RCS where physical arrangement does not permit the use of thicker reflective insulation.
Fiberglass:	This is a fibrous insulation covered by stainless steel and a vapor barrier used to prevent sweating of cold water systems (Component Cooling and Service Water).
Fiberglass With Blanket (Nukon Insulation):	This fibrous glass insulation is enclosed in woven fibrous glass fabric between two layers of fiberglass scrim sewn to the insulation. This material is used on the Pressurizer and the Unit 1 Steam Generators. On most areas of the Pressurizer and the Unit 1 Steam Generators where this material is used, this insulation is covered with a stainless steel jacket or a stainless steel mesh. Additionally, this style of insulation is used on the integrated head assembly to insulate the ring beam in the area local to where the L-Panel and RVCH Dome insulation converge.
Calcium Silicate:	This rigid solid insulation is used on straight portions of the Feedwater and Main Steam Systems that are not exposed to a LOCA pipe break inside the bioshield. This material is also covered with stainless steel. Welds in these systems are covered with encapsulated insulation.
Mineral Wool:	This fibrous insulation is applied to the lower 34 feet of the containment liner and is also covered with stainless steel lagging and a vapor barrier.
Fiberglass Blanket:	This fiberglass blanket material is jacketed with fiberglass fabric impregnated with silicone. This insulation is used as anti-sweat insulation on Service Water piping of 3" & 2" dia. at CFCUs in the Containment Building Units 1 & 2.

6.3.2.3 Applicable Codes and Classifications

The codes and standards to which the individual ECCS components are designed are listed in Table 6.3-1

6.3.2.4 Materials' Specification and Compatibility

Materials are selected to meet the applicable material requirements of the codes in Table 6.3-1 and the following additional requirements:

1. All parts of components in contact with borated water are fabricated of or clad with austenitic stainless steel or equivalent corrosion-resistant material.

2. All parts of components in contact (internal) with sump solution during recirculation are fabricated of austenitic stainless steel or equivalent corrosion-resistant material.

3. Valve seating surfaces are hard faced with Stellite No. 6 or equivalent to prevent galling and to reduce wear.

4. Valve stem materials are selected for their corrosion resistance, high tensile properties, and resistance to surface scoring by the packing.

The elevated temperature of the sump solution during recirculation is well within the design temperature of all ECCS components. In addition, consideration has been given to the potential for corrosion of various types of metals exposed to the fluid conditions prevalent immediately after the accident or during long term recirculation operations.

6.3.2.5 Design Pressures and Temperatures

The component design pressure and temperatures are given in Tables 6.3-2 through 6.3-5. These pressure and temperature conditions are specified as the most severe conditions to which each component is exposed during either normal plant operation or during operation of the ECCS. For each component, these conditions are considered in relation to the code to which it is designed. By designing the components in accordance with applicable codes and with due consideration for the design and operating conditions, the fundamental assurance of the structural integrity of the ECCS components is maintained.

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6.3.2.6 Coolant Quantity

The minimum storage volume for the accumulator is given in Table 6.3-2. The total volume of the RWST is 400,000 gallons. At the minimum volume permitted by the Technical Specifications (364,500 gallons) approximately 313,000 gallons for Unit 1 and 313,000 gallons for Unit 2 are available to the ECCS pumps.

The RWST volume must be sufficient to support operator action time following the SI actuation and during the switchover alignment to cold leg recirculation. Following an SI actuation, all ECCS pumps (RHR, Charging/Safety Injection and Safety Injection) are automatically started. If the containment High-High setpoint is reached, the Containment Spray pumps are also automatically started with all pumps initially taking suction from the RWST. This time period is termed the injection phase of the RWST draindown. When the RWST reaches the low level setpoint, the operator begins to take action to switchover from the injection phase to the containment sump recirculation phase. During this switchover phase, the RHR pump suction is re-aligned to the containment sump and the charging/safety injection pump and safety injection pump suction are aligned to the RHR pump discharge. One of the two operating CS pumps is also stopped upon entering the switchover phase to reduce the outflow from the RWST. When the RWST reaches the low-low level alarm, the second CS pump is stopped. The RWST low-low level setpoint must also support NPSH requirements for all ECCS pumps and CS pumps taking suction on the RWST. Once the ECCS is aligned for containment recirculation, the RHR pump discharge may then be cross-tied to the containment spray header to provide containment recirculation spray flow after all CS pumps are stopped.

In addition, the amount of water during the injection phase of a LOCA must be sufficient to provide adequate RHR NPSH in the containment sump and adequate submergence of the strainer modules prior to switchover to recirculation. The RWST volume to meet this requirement is 193,000 gallons.

The available RWST water volume for the injection phase is the minimum volume between the Technical Specification requirement and the RWST low level setpoint. The available RWST water volume for the switchover phase is the minimum volume between the RWST low level setpoint and the RWST low-low level setpoint. These minimum volumes account for instrument accuracy in the RWST level channels which are used by the operators to monitor RWST inventory. The following water volumes are available:

	Salem 1	Salem 2
Injection Phase	207,800 gallons	204,500 gallons
Switchover Phase	105,200 gallons	108,500 gallons
Total	313,000 gallons	313,000 gallons

Injection Phase

The available RWST water volume for the injection phase provides sufficient time for the operators to proceed through the Emergency Operating Procedures (EOPs) to the point where switchover to cold leg recirculation may be required. Following an SI actuation with a containment high-high signal, the RHR pumps, charging pumps, SI pumps and CS pumps are all started automatically, taking suction on the RWST. The highest drain rate for the RWST occurs with all pumps operating for a design basis large break LOCA when the RCS is rapidly depressurized to containment pressure and the RHR pumps inject flow to the RCS cold legs. ECCS pump flow rates vary with the RWST level and the RCS/containment pressures. Significantly conservative assumptions are used in determining the RWST drain flow rates as follows:

(1) The RCS and Containment pressure average approximately 10 psig for the first 5 minutes following SI actuation. This is based on the minimum RCS/containment pressures calculated in the LOCA PCT analysis. The LOCA PCT analysis for minimum RCS/containment pressures predict conservatively low RCS/containment pressures based on maximum pump flow delivery similar to those used in the RWST draindown evaluation.

(2) After the first 5 minutes, the RCS/containment pressures are conservatively assumed to be 0 psig.

(3) The RHR pumps inject to the RCS in the same lines as the accumulators. Since the accumulators are at a higher pressure than the RHR pumps, the RHR pumps do not inject to the RCS until the accumulator pressures decrease. An average time period of 45 seconds for accumulator blowdown is assumed based on the LOCA analysis, during which time the RHR pumps do not drain down the RWST.

(4) The pump flowrates are based on the maximum expected flow with the maximum allowable pump curves and conservative modeling of the piping and component resistances. The charging pump and SI pump flows are based on the balancing criteria provided in the Technical Specification.

Based on these assumptions for pump flow rates and the available RWST water volume during the injection phase (207,800 gallons for Unit 1 and 204,500 gallons for Unit 2), the RWST low level alarm will be reached in 12.9 minutes (Unit 1) and 12.5 minutes (Unit 2). This time is sufficient for the operators to proceed through the EOPs and begin switchover to containment recirculation. The RWST water volume required for RHR pump NPSH is also met.

Switchover Phase

Additional water storage is required in the RWST to accommodate the operator actions necessary to align the ECCS pump suctions from the RWST to the containment sump. The required operator actions for Salem 1 and 2 are provided in Table 6.3-6. The switchover is similar for both units with one exception. Salem Unit 1 requires a manual transfer while Salem Unit 2 is semi-automatic. This means that for Salem Unit 1, the RHR pumps are manually stopped, the RHR pump suctions are re-aligned to the containment sump and the RHR pumps re-started. For Salem Unit 2, the semi-automatic switchover is armed and the RHR pump suction is automatically switched from the RWST to the containment sump without stopping the RHR pumps.

The time available for operators to complete the switchover is dependent on the flow rate out of the RWST and the available RWST volume. A conservative analysis was performed to show that sufficient water is provided in the RWST to complete the switchover for all RCS break sizes assuming a limiting single failure while maintaining long term cooling consistent with the 10CFR50.46 analysis of record. The available water volume is that contained between the RWST low level and the RWST low-low level, taking into account instrument inaccuracies. At the RWST low-low level, all pumps taking suction on the RWST would be stopped to protect the pumps. Three limiting break sizes have been specifically evaluated for the Salem 1 and 2 switchover.

Large Break LOCA - The design basis large break LOCA produces the lowest RCS pressure and the highest RWST draindown rate, which results in the limiting time for RWST drain down during switchover to containment recirculation. All pumps are assumed to inject to a 0 psig back pressure. To maintain long term core cooling, one RHR pump injecting to two RCS cold legs (with one leg spilling to the containment and one leg delivering flow to the core) is sufficient to maintain long term core cooling.

Small Break LOCA - The RWST drain down time for a small break LOCA is longer since the RCS pressure remains above the RHR pump discharge pressure. This provides the operator with additional time to complete the switchover and align the charging pumps and SI pumps to cold leg recirculation. To maintain long term core cooling, one charging pump and one SI pump (each delivering to 4 cold legs with one leg spilling to the containment) is sufficient to maintain long term core cooling.

Accumulator Line Small Break LOCA - Due to the break location, the drain down time for an accumulator line small break LOCA is similar to that of a Large Break LOCA. The RHR pumps inject to the RCS through the accumulator line. With a break in the accumulator line, the RHR pumps spill directly to the containment even if the RCS pressure is above the RHR pump cut off head. This increases the RWST outflow and therefore reduces the time available for the operator to complete the recirculation alignment. To maintain long term core cooling, one charging pump and one SI pump (each delivering to 4 cold legs with one line spilling to the containment) is sufficient to maintain long term core cooling.

Each of these breaks has been evaluated with a limiting single failure to determine the minimum RWST drain down times. The limiting single failure for Salem Unit 1 is one RHR pump failing to stop on demand. This results in the failed (running) RHR pump continuing to draw down the RWST. The limiting single failure for Salem Unit 2 is the RWST common suction line to RHR pumps suction isolation valve (SJ69) failing to close during semi-automatic switchover. This results in draining of the RWST to the sump as the ECCS pumps continue to draw down the RWST during switchover. Additional assumptions used in the RWST drain down evaluation are as follows:

(1) The containment pressure is assumed to be 0 psig. This maximizes containment spray flow and RHR pump flow for the large break LOCA and accumulator line small break LOCA.

(2) The RCS pressure is assumed to be 0 psig for the large break LOCA. For the small break LOCA, the RCS pressure is assumed to be above the RHR pump cut in pressure. For the accumulator line small break LOCA, the RCS pressure is also assumed to be above the RHR cut-in pressure, but one injection line is spilling to the containment pressure of 0 psig.

(3) All pumps are operating in an alignment that maximizes outflow from the RWST.

(4) For Salem Unit 2, during the semi-automatic switchover, both the RWST/RHR isolation valve (RH4) and the containment sump isolation valve (SJ44) are open at the same time. This creates a direct path from the RWST to the containment sump. Valve RH4 has a maximum allowable stroke time of 87 seconds and valve SJ44 has a maximum allowable stroke time of 36 seconds. The valves are assumed to be fully open during the stroke time to conservatively maximize the RWST drain flow. With the failure of one RH4 valve to close, this drain path exists for the entire duration of the switchover for one of the containment sump lines. This drainage path also exists for the small break LOCA even though the RHR pumps are not injecting directly into the RCS. This results in reduced switchover times for Salem Unit 2 small break LOCA when compared to Salem Unit 1. Salem Unit 1 has an interlock between valves RH4 and SJ44 such that the containment sump isolation valve (SJ44) cannot be open unless the RWST/RHR isolation valve (RH4) is closed. This interlock precludes a direct drainage path from the RWST to the containment sump for Salem Unit 1.

Unit 1 Analysis of Manual Switchover

For Unit 1, manual switchover from the RWST to the containment sump is initiated at an RWST level of 15.2 feet (RWST low or low-backup alarm setpoint). Two significant operator actions are modeled in the RWST drain down evaluation. This simplified approach provides clearer training guidelines rather than modeling each specific operator action in the RWST to containment sump switchover. The first timed operator action is initiating a close on the RHR pump suction valves from RWST valves (RH4). As shown in Table 6.3-6, once the operator reaches this step, the RHR pumps have been stopped (or isolated) and one containment spray pump is about to be stopped (if two are running). The second significant time modeled is the time at which the RWST low-low level alarm is reached for the limiting large break LOCA. The Charging/SI and SI pumps must have their suctions re-aligned to the RHR pump(s) before reaching the RWST low-low level alarm.

The available times for operator actions to ensure that long term cooling will be maintained consistent with the 10CFR50.46 analysis for Unit 1 are:

Initiate Close RH4s	Complete Switchover
4 minutes	11.7 minutes

For the limiting design basis large break LOCA, the time to complete switchover is sufficient for the required operator actions.

Unit 2 - Semi-Automatic Switchover

For Salem Unit 2, the semi-automatic switchover is armed and the RHR pump suction is automatically switched from the RWST to the containment sump without stopping the RHR pumps. The timings of two operator actions are used as input to the RWST draindown evaluation, which determines the maximum time available for operators to complete the switchover to recirculation while maintaining RWST water level above the Low-Low level setpoint. First, operators must initiate shutting SJ69, the isolation valve for the common RWST suction line to the RHR pump suctions, within 3.7 minutes of receipt of the RWST Low level alarm. Second, operators must stop one containment spray pump within 5.5 minutes (if two are running). Based on this, the minimum time to reach the RWST Low-Low level setpoint is 11.2 minutes. Operators must complete switchover to recirculation within 11.2 minutes to ensure that adequate NPSH is available to the operating ECCS pumps.

For the design basis LOCA, one RHR pump provides adequate cooling flow. When the semi-automatic switchover is armed, the suction of the RHR pumps is automatically switched from the RWST to the containment sump, ensuring adequate cooling flow is available. This makes the design basis LOCA less limiting with respect to switchover time. Therefore, available operator action times are dictated by the small break LOCA. For small break LOCA (accumulator line small break LOCA is limiting), the time to complete switchover is sufficient for the required operator actions.

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The available times for operator actions to ensure that long term cooling will be maintained consistent with the 10CFR50.46 analysis for Unit 2 small break LOCA are:

Initiate SJ69 Closure	Stop One CS Pump	Complete Switchover
3.7 minutes	5.5 minutes	11.2 minutes

6.3.2.7 Pump Characteristics

Pump performance curves for the RHR are shown on Figure 6.3-4.

6.3.2.8 Heat Exchanger Characteristics

Residual heat exchanger characteristics are presented in Section 5.5.

6.3.2.9 Emergency Core Cooling System Flow Diagrams

An ECCS flow diagram is given on Plant Drawings 205234 and 205334.

6.3.2.10 Relief Valves

The ECCS relief valve capacities and leak rates are given in Section 6.3.2.2.

6.3.2.11 System Reliability

Specific design features of the ECCS to assure its ability to meet single failures include the following:

1. Inclusion of two charging pumps in the Injection System which deliver into the four cold legs through 1.5 inch diameter lines. Accumulator injection into the cold legs employ completely independent piping and connections from the charging pumps. The two charging pumps will supply recirculation flow from the containment sump (via the RHR pump discharge/charging pump suction cross tie) to the four cold legs through the same line.

2. (Deleted).

3. Inclusion of two safety injection pumps in the Injection System, which delivers to four cold leg injection points via the accumulator discharge lines during the injection phase and initial portion of the recirculation phase. Later in the recirculation phase of operation, flow from each of these pumps will be directed via a separate 4-inch header to two hot leg injection points in order that subcooling of the core can be completed. Redundant headers are provided for this phase of operation to assure at least one pump can deliver even in the case of a passive failure in one line. During recirculation operation, the safety injection pumps (as well as the charging pumps mentioned previously) take suction from

the recirculation sump via the RHR pump discharge or safety injection/centrifugal charging pump suction crosstie. This crosstie connection from the suction of the charging to the suction of the safety injection pumps assures that during recirculation with either a passive or an active failure, at least one charging and one safety injection pump or two safety injection or two charging pumps will deliver.

4. Inclusion of two RHR pumps in the Injection System which delivers to four cold leg injection points (one on each loop) via the accumulator discharge lines during the injection phase and initial portion of the recirculation phase of operation. To ensure each RHR pump can deliver to the four cold leg injection points, the discharge cross connect valves, RH-19's, are required to be open during the injection phase. During recirculation, the RHR pumps take suction from the recirculation sump and also provide flow to the suction of the charging and safety injection pumps. Later in the recirculation period, the injection flow provided by the RHR pumps via safety injection pumps will be redirected from the cold legs to two hot leg connections in order to complete subcooling of the core. In addition, one RHR pump will be providing flow directly to two cold legs.

   Thus, injection flow of borated water from the RWST is provided to all four RCS cold legs from the three pumping systems. During the recirculation phase of the accident all three pumping systems are capable of providing recirculation sump fluid flow to all four cold legs with the low head pumps (RHR) providing flow to the high head pumps (safety injection and charging pumps). The capability of long term recirculation flow to the RCS hot legs is provided from the safety injection pumps.

Failure Analysis

Separate single failure analyses were performed for both the injection and recirculation phases of an accident. Two basic types of failures were considered:

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1. Active failure, which is defined as the inability of any single dynamic component or instrument to perform its design function when called upon to do so by the proper actuation signal. Such functions include change of position of a valve or electrical breaker, operation of a pump, fan, or diesel-generator action of a relay contact, etc.

2. Passive failure which is defined as a failure affecting a device involved with the transport of fluid which limits its effectiveness in carrying out its design function. Most passive failures involve the development of abnormal leakage in valve stem packings, pump seals, etc, although passive failures concerned with abnormal flow restriction in lines are also considered.

Table 6.3-9 summarizes the results of the single failure analysis applied during the injection phase. All failures during this phase are assumed to be active failures. It is during this phase that the pumps are starting and automatic isolation valves are required to move. All credible active failures are considered, and are included in the accident analyses described in Section 15. A comprehensive failure analysis for post-accident electrical and control components is presented in Section 7.

The accumulators which are a principal factor of the Injection System are not subject to active failure. The only moving parts in the accumulator injection train are the two check valves. The working parts of the check valves are exposed to fluid of relatively low boric acid concentration. Even if some unforeseen deposition accumulated, calculations indicate that a reversed differential pressure of about 25 psi can shear any particles in the bearing surfaces that may tend to prevent valve functioning.

When the RCS is being pressurized during the normal plant heatup operation, the check valves are tested for leakage as soon as there is at least 100 psi differential across the valve. This test confirms the seating of the disc and provides a quantitative leakage rate measurement which can be compared with the results of earlier tests. When this test is completed, the discharge line test valves are opened and the RCS pressure increase continued. There should be no increase in leakage from this point on since increasing reactor coolant pressure increases the seating force and decreases the probability of leakage.

The accumulators can accept some leakage back from the RCS without compromising their availability. Table 6.3-10 indicates the frequency that the accumulator level would have to be readjusted as a function of leakage rate. It should also be noted that an accumulator can be isolated with a motor-operated valve if leakage becomes excessive.

Tables 6.3-9 and 6.3-11 summarize the single failure analyses of recirculation phase.

Leakage During Recirculation

Table 6.3-12 summarizes the potential leakage sources from the recirculation loop during the recirculation phase of an accident. The table lists the type of leakage control utilized for each leak source. A value of 50 gpm is employed as a design basis for sizing Auxiliary Building sump pumps which will be required to dispose of this leakage to the Waste Disposal System. The ECCS is separable into two complete subsystems during the long-term cooling period, either of which is capable of providing the minimum core cooling functions. Should a leak develop in either of these two subsystems, the only actions necessary to isolate it are the closing/opening of valves and the starting/stopping of pumps.

Leakage from the valve stem leakoffs is piped to the Equipment Drain System.

The total leakage resulting from all sources is administratively limited to 0.45 gpm as described in UFSAR Section 15.4.1

Recirculation loop leakage sources are summarized in Table 6.3-12. Leakage is monitored by procedure to ensure this leak rate is not exceeded.

With respect to piping and mechanical equipment outside the containment, considering the provisions for visual inspection and leak detection, leaks will be detected before they propagate to major proportions. A review of the equipment in the system indicates that the largest sudden leak potential would be the sudden failure of a pump shaft seal. Evaluation of the leak rate assuming only the presence of a seal retention ring around the pump shaft showed that flows less than 50 gpm would result. Piping leaks, valve packing leaks, or flange gasket leaks have been of a nature to build up slowly with time and are considered less severe than the pump seal failure.

Means are also provided to detect and isolate such leaks in the emergency core cooling flow path within 30 minutes. The RHR pumps and heat exchangers are located in individual compartments. Each compartment has a volume of 200 ft^3 to accommodate a 50-gpm leak for a period of 30 minutes.

Valving is provided to allow an operator to isolate, drain, and flush the RHR heat exchangers and pumps. The operation of the drain valves will be done by means of remote valve reach rod operators located in a shielded valve gallery. The radiation shielding criterion for this valve gallery will be the same as for manual containment isolation valves. Post-accident radiation levels around recirculation loop equipment are discussed in Section 15.

The layout permits the detection of a leaking recirculation loop component by means of a radiation monitor which samples the air in the plant vent. Alarms in the control room will alert the operator when the activity exceeds a preset level. Sump level and operation of sump pumps will be indicated in the control room as a backup for detection of water leaks.

Should a tube side to shell side leak develop in a residual heat exchanger, the operator will be warned by a component cooling water high radiation alarm. For large leaks the operator will also be warned by a component cooling water surge tank high level alarm. In the event that the leak cannot be isolated before the tank fills, the tank relief valve will pass the excess water to the waste holdup tank.

The operator actions required to detect, isolate, and realign a leaking component and, subsequently, realign the system, depend upon the location of the leak (i.e., which system, actual physical location). Depending on the location of the leak, the operator will carry out a series of actions. For each break location, a different set of actions will be required. The actions taken by the operator will be manual (e.g., starting or stopping pumps, opening or closing valves). These actions would be performed from the Control Room.

For the Service Water System, the rupture of a large pipe will be indicated to the operator by decreasing pump discharge header pressure. Low pump header pressure will cause a backup service water pump to start.

In the event that a pipe rupture occurs in a watertight pump compartment of the intake structure, which is larger than the capacity of the sump pump, high sump level for the affected compartment will be alarmed in the control room. The operator can remotely close the tie valves and header block valves at the intake structure to isolate the affected compartment.

In the event that a main yard supply header is ruptured, the affected header can be isolated and the tie valves at the Auxiliary Building opened. Rupture of a header pipe in the pipe tunnel can be detected by a pipe tunnel sump high level alarm. The operator can determine the affected header by remotely closing the intake tie valves and observing which pump header is affected by low pressure. Once the ruptured header is isolated, the intake tie valves can be reopened and all service water pumps made available.

In the event that a service water pipe rupture occurs inside the containment, the difference between flows entering and leaving the containment will be sensed and alarmed in the Control Room. High level alarms in the containment sump and fan cooler drain pots will also be indicated in the Control Room. The operator can remotely close the isolation valves to isolate the leaking fan cooler.

In the event that radiation is detected at one of the service water outlets from the containment, the condition is alarmed in the Control Room.

6.3.2.12 Protection Provisions

All four injection lines penetrate the containment adjacent to the Auxiliary Building.

One portion of the High Head Injection System within the containment is connected to the low head injection lines attached to each loop's accumulator injection piping. The other portion of the High Head Injection System within the containment is connected directly to the injection nozzles on the cold leg piping of the loops.

For most of the routing, these lines are outside the reactor and steam generator shielding, and hence they are protected from missiles originating within these areas.

The coolant loop supports are designed to restrict the motion to about one-tenth of an inch, whereas the attached safety injection piping can sustain a 3-inch displacement without exceeding the working stress range.

Hangers, stops, and anchors are designed in accordance with ANSI B31.1 Code for Pressure Piping, and ACI 318 Building Code Requirements for Reinforced Concrete, which provide minimum requirements on materials, design, and fabrication with ample safety margins for both dead and operational dynamic loads over the life of the equipment.

Materials used are in accordance with ASTM specifications which establish quality levels for the manufacturing process, minimum strength properties, and for test requirements which ensure compliance with the specifications; qualification of welding processes and welders for each class of material welded and for types and positions of welds.

Allowable stress values are established which provide an ample safety margin on yield strength for normal loads and ultimate strength for design basis accident or maximum hypothetical seismic loads.

6.3.2.13 Provisions for Performance Testing

The provisions incorporated to facilitate performance testing of components are discussed in Section 6.3.4.

6.3.2.14 Pump Net Positive Suction Head

Net positive suction head data for pumps which are required to operate post-accident are provided in Table 6.3-13.

6.3.2.15 Control of Motor-Operated Isolation Valves

Position indication and alarm circuits for the motor-operated valves, located between the accumulator tanks and the primary cooling system, are designed to provide assurance that these valves will be open when required. These valves are normally open and under administrative control with the motive power for the valves locked out during normal power operation. Redundant and independent information is provided in the Control Room to indicate when any one valve is not in the fully open position.

Valve status (fully open or fully closed) is indicated on the main control board via backlighted pushbuttons. These status lights are actuated by limit switches on the valve motor operator. In addition, an alarm is provided on the Overhead Annunciator System in the event the valve is not in the fully open position.

Another independent means of determining that the valve is not in its proper position is provided through the Auxiliary Alarm System which will initiate an audible signal and print out an alarm message indicating when the valve is not in the fully open position. This indication and alarm is derived from a separate valve stem limit switch and is energized from an independent power supply from that used for the overheat annunciator.

A safety injection signal also automatically initiates the opening of these valves.

6.3.2.16 Motor-Operated Valves and Controls

Remotely operated valves in the SIS which are in the "ready" position and which do not receive an "S" signal, are assured to be in the proper position for injection by means of the following:

1. Redundant indication of valve position in the control room for those valves in common, or non-redundant flow paths of an ECCS subsystem, or valves whose inadvertent operation could degrade the ECCS. The indication provided is identical to that of the accumulator discharge valves, described in Section 6.3.2.15.

2. Valves in redundant flow paths are provided with position indication on the main control console and "off-normal" indication in the Auxiliary Annunciator

System (i.e., 11RH4, 12RH4, 11SJ33, 12SJ33, 11SJ134, 12SJ134.

3. Manually-operated valves are under administrative control to assure that they are in the proper position. Additionally, the valves cited in Item 1 above are placed in the proper position for injection with the motive power removed from the valve.

Valves with redundant position indication (as described in Section 6.3.2.15) and power lockouts are:

1SJ30*	IISJ40*	11SJ54*
ISJ69*	12SJ40*	12SJ54*
ISJ135*	1RH26*	13SJ54*
11SJ49*	1CS14*	14SJ54*
12SJ49*	ISJ67*	11SJ44*
	1SJ68*	12SJ44*

Requirements for disconnecting ac power to these valves and for locking them in position are set forth in the plant Technical Specifications.

Valves marked with an asterisk (*) are provided with the capability to restore control power from the Control Room.

The safety injection initiation signal was removed from the centrifugal charging pump (CCP) miniflow isolation valves, CV139 and CV140, thus preventing automatic termination of miniflow. In addition, manual valve CV197, which directs reactor coolant pump sealwater return flow to the suction of the centrifugal charging pumps will be locked closed and manual valve CV130 will be locked open which will route reactor coolant pump sealwater return and centrifugal charging pump miniflow water to the volume control tank. This valve alignment will cause the volume control tank to fill solid during a safety injection initiation; the volume control tank relief valve, CV241, would then open, directing miniflow to the CVCS holdup tanks. Procedurally, when the high-head safety injection pumps are operating in the ECCS mode, the operator will be instructed by Emergency Operating Procedures to terminate miniflow below an RCS pressure of 1500 psig and to re-establish miniflow if RCS pressure rises again to 2000 psig.

6.3.2.17 Manual Actions

No manual actions are required of the operator for proper operation of the ECCS during the injection mode of operation. The only manual actions required to be taken by the operator are those necessary to complete the realignment of the system for its cold leg recirculation mode of operation and, subsequently, to realign the system for its hot leg recirculation mode of operation.

The transfer from the injection phase to the recirculation phase is described in Section 6.3.2.1 and in Table 6.3-6.

6.3.2.18 Process Instrumentation

Process instrumentation available to the operator in the control room to assist in assessing post-LOCA conditions are tabulated in Section 6.3.5 and Section 7.

6.3.2.19 Materials

Materials employed for components of the ECCS are given in Table 6.3-14. These materials are chosen based upon their ability to resist pyrolytic decomposition.

6.3.3 Design Evaluation

6.3.3.1 Evaluation Model

This information is provided in Section 15.

6.3.3.2 Small Break Analysis

This information is presented in Section 15.

6.3.3.3 Steam Line Rupture Analysis

This information is presented in Section 15.

6.3.3.4 Fuel Rod Perforations

Results for accidents that have acceptance criteria based on radiological consequences, metal-water reaction, or peak clad temperature are presented in Chapter 15.

6.3.3.5 Effects of Core Cooling System Operation on the Core

The effects of the ECCS on the reactor core are discussed in Section 4.

6.3.3.6 Use of Dual Function Components

The ECCS contains components which have no other operating function, as well as components which are shared with other systems and perform normal operating functions.

Components of the ECCS which perform no other operating functions are the following:

1. One accumulator for each loop which discharges borated water into its respective cold leg of the RCS.

2. One BIT.

3. Associated piping, valves, and instrumentation.

Components which also have a normal operating function are as follows:

1. The two RHR pumps and residual heat exchangers: These components are normally used during the latter stages of normal reactor cooldown and when the reactor is held at cold shutdown for core decay heat removal. However, during all other plant operating periods, they are aligned to perform the low head injection function.

2. The RWST: This tank is used to fill the refueling canal for refueling operations, provide a makeup source to the spent fuel pit as well as an emergency makeup source to the RCS via the CVCS charging pumps. These functions place no limitations on the function of the ECCS. During all plant operating periods, the RWST is aligned to the suction of the safety injection pumps, RHR pumps, and the containment spray pumps.

3. The two high head safety injection pumps: These pumps are normally aligned to perform their high head safety injection function. One of the two may be used to provide normal continuous charging during normal plant operation.

4. Two boric acid tanks.

An evaluation of all components required for ECCS operation demonstrates that either:

1. The component is not shared with other systems, or

2. If the component is shared with other systems, it is aligned during normal plant operation to perform its accident function.

Dependence on Other Systems

Other systems which operate in conjunction with the ECCS are as follows:

1. The Component Cooling System cools the residual heat exchangers during the recirculation mode of operation. It also supplies cooling water to the RHR pumps during the injection and recirculation modes of operation.

2. The Service Water System provides cooling water to the component cooling heat exchangers and to the safety injection pumps.

3. The Electrical System provides normal and emergency power sources for the ECCS.

4. The Engineered Safety Features Actuation System generates the initiation signal for emergency core cooling.

5. The Auxiliary Feedwater System supplies feedwater to the steam generators.

Limiting Conditions for Maintenance During Operation

The Technical Specifications establish limiting conditions governing the maintenance of ECCS components during plant operation with the core critical. It is expected that maintenance on a component will be permitted if the remaining components meet the minimum conditions for operation and the following conditions are also met:

Maintenance on an active component will be permitted if the remaining components meet the minimum conditions for operation and the following conditions are also met:

1. The remaining equipment has been demonstrated to be in operable condition, ready to function just before the initiation of the maintenance.

2. A suitable time limit is placed on the total time span of successful maintenance which returns the components to an operable condition, ready to function.

The design philosophy with respect to active components in the High Head/Low Head Injection System is to provide backup equipment so that maintenance is possible during operation without impairment of the safety function of the system. Routine servicing and maintenance of equipment of this type would generally be scheduled for periods of refueling and maintenance outages.

6.3.3.7 Lag Times

To provide protection for large area ruptures of the RCS, the ECCS must respond to rapidly 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. With three of the four available accumulators delivering their contents to the reactor vessel, the peak clad temperature is maintained below the cladding melting temperature as discussed in Section 15.

The function of the centrifugal charging, safety injection, and RHR pumps is to complete the refill of the vessel and ultimately return the core to a subcooled state. The starting sequence of the ECCS pumps and the related emergency power equipment will enable minimum required flows which are bounded by the delay times and associated flows assumed in the safety analyses.

The starting sequence is discussed in Section 7.

6.3.3.8 Thermal Shock Considerations

Thermal shock considerations are discussed in Section 15.

6.3.3.9 Limits on System Parameters

The limiting conditions for operation are detailed in the Technical Specifications. These conditions will apply to both active components and coolant storage components of the ECCS.

6.3.4 Tests and Inspections

All active and passive components of the ECCS are inspected periodically to demonstrate system readiness.

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

In addition, to the extent practical, the critical parts of the injection nozzles, pipes, valves, and safety injection pumps are inspected visually or by boroscopic examination for erosion, corrosion, and vibration wear evidence. A plan for periodic component and system testing and material examinations will be prepared prior to plant operation for use throughout plant life.

Environmental testing of ECCS components which are located inside the containment and are required to operate following a LOCA is discussed in Reference 1.

6.3.4.1 Component Testing

Preoperational performance tests of the components are performed in the manufacturer's shop. An initial system flow test demonstrates proper functioning of the system. Thereafter, periodic tests demonstrate that components are functioning properly.

Active components of the ECCS may be individually actuated on the normal power source during plant operation to demonstrate operability. The test of the safety injection pumps employs the minimum flow recirculation test line which connects back to the RWST. Remote operated valves are exercised and actuation circuits tested. The automatic actuation circuitry, valves, and pump breakers also may be checked during integrated system tests performed when the plant is cooled down and the RHR loop is in operation.

Containment sump isolation valves are normally closed. Inadvertent opening is prevented by using control power lockouts and electrical interlocks which prevent the opening of the valves whenever the corresponding RHR pump suction isolation valve is open. The valves will be exercised and tested after closing the appropriate RHR pump suction isolation valve during normal operation or refueling at a frequency specified in the Technical Specifications.

The containment sump valves will be tested only after closing the suction and discharge valves of the associated RHR pump. The isolated RHR line is located at Elevation 46 feet-10 inches and the center line of the sump valves at Elevation 53 feet-0 inch. Due to the elevation difference, no stagnant refueling water is expected to interfere with the sump valve tests.

If the necessity arises for the draining of this line, provisions have been provided to drain it through the RHR pump to the RHR sump. Sump interconnection with the Liquid Radwaste System provides satisfactory processing provisions for this drainage.

Inleakage through each of the check valves which isolate the SIS from the RCS can be tested by opening the remote test valves in the appropriate test line. Flow through the test line can be measured and the opening and closing of the discharge line stop valves can be verified by the flow instrumentation.

6.3.4.2 System Testing

Testing is conducted during plant shutdown to demonstrate proper automatic operation of the ECCS. A test signal is applied to initiate automatic action and verification made that the safety injection pumps attain required discharge heads. The test demonstrates the operation of the valves, pump circuit breakers, and automatic circuitry.

The operation of the RHR pumps is verified periodically. Performance of the centrifugal charging pumps is verified by their operation during normal plant operation and cooldown. Starting of these pumps by a safety injection signal is also verified during plant shutdown.

The test is considered satisfactory if control board indication and visual observations indicate all components have operated and sequenced properly.

The periodic testing of pumps in the RHR, SIS, and Containment Spray Systems requires recirculation of water from the RWST. Demonstration of proper operation of these pumps will also demonstrate the operability of the line from the RWST. The BIT and piping normally contain 0 to 2500 ppm boric acid solution. The concentration of boric acid in the BIT is tested periodically to detect the inadvertent introduction of any higher concentrated boron into the system. The pressure of the BIT is monitored routinely from the Control Room.

The accumulator pressure and level are continuously monitored during plant operation, and flow from the tanks can be checked at any time using test lines.

The accumulators and the injection piping up to the final isolation valve are charged with borated water while the plant is in operation. The accumulator boron concentration is checked periodically by sampling. The accumulators and injection lines are replenished with borated water as required by using the safety

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injection pumps to recirculate refueling water through the injection lines. A small test line is provided for this purpose in each injection header.

Flow in the centrifugal charging pumps' common discharge line, safety injection pumps' main flow lines, and in the main flow line for the RHR pumps is monitored by flow indicators in the Control Room. Pressure instrumentation is also provided for the main flow paths of the safety injection and RHR pumps and is located in the Control Room.

6.3.4.3 Operational Sequence Testing

The ECCS and the Containment Spray System were operationally tested prior to initial reactor fueling. The tests included individual pump performance tests, accumulator operation, and an integrated system test.

Each centrifugal charging, safety injection, RHR and containment spray pump were tested at rated flow capacity. The containment spray pumps discharged through a test line to the refueling canal, while the others discharged to the open RCS through the normal injection path. Additionally, the pumps were run for a minimum of 1 hour to ensure reliable operation. The purpose of these tests is to evaluate the hydraulic and mechanical performance of the pumps and to detect deficiencies which might occur during sustained operation.

Flow distribution tests will also be performed in which the pumps will deliver from the RWST to the RCS through the normal injection paths for emergency core cooling. Adjustments will be made where flow resistances are unacceptably low or high to limit pump runout and balance the flow between piping branches. Total flow and relative flows between branch lines will be compared with minimum acceptable flows as determined in the safety analysis.

The accumulators will be tested by charging them to between 67 and 70 psig, accumulator level between 96% and 100%, with the isolation valves closed. The isolation valves will be opened, discharging the accumulator into the open reactor vessel. Performance will be verified by extrapolating the data to normal accumulator pressure.

It is neither practical nor feasible to perform these tests at simulated reactor operating conditions. With the reactor at normal operating temperature and pressure, there are no means available to change the primary system parameters as rapidly as required to simulate a 100 percent LOCA, thereby allowing the ECCS to inject water into the system. The system will be tested during hot functional testing, however, to verify that the high pressure components (centrifugal charging pumps) can deliver water to the reactor through the normal injection path while the plant is at normal operating pressure. The test will be conducted by manually initiating the safety injection sequence.

A complete operational test will also be performed to demonstrate overall system performance. The purpose of this test is to demonstrate the proper functioning of actuation and instrumentation circuits, emergency power sources, and electrical load sequencing of the Integrated Safeguards System. Data obtained will be used to verify design operation and confirm various sequencing and operating times and logic.

The systems are accepted only after demonstration of proper actuation of all components and after demonstration of flow delivery of all components within design requirements.

6.3.4.4 Conformance with Regulatory Guide 1.79

The Salem preoperational testing program meets the requirements of Regulatory Guide 1.79, "Preoperational Testing of Emergency Core Cooling Systems for Pressurized Water Reactors." The scheduled tests, however, may deviate in part from certain specific test descriptions included in the Guide. These deviations are enumerated below.

Regulatory Position C.3.a.(2)

Not all injection pumps will be tested at operating conditions, nor will the Auxiliary Feedwater System be actuated by a safety injection signal. Check valves on the charging/safety injection cold leg injection path will be tested utilizing a charging/safety injection pump which will be started by manually initiating a safety injection signal. The check valves on the safety injection hot leg and cold leg injection paths will be tested by pressurizing the test line and throttling water through the check valves. A significant portion of safeguards equipment not directly involved in the delivery of emergency core cooling water to the RCS will be omitted from the test. Thermal shock is not expected, since the total quantity of water injected will be minimized. Branch line throttle valves will be initially shut and then slowly opened, one at a time, to demonstrate flow through the check valves.

Regulatory Position C.3.b.(2)

Adequate NPSH from the containment sump will be verified by taking a suction from a full sump with one RHR pump and discharging into the RCS. Duration of test run is estimated to be 45 seconds. Vortex control verification will be accomplished by visual observation at the sump. Pressure drop across sump screen will not be measured as it is considered negligible and will not compromise NPSH evaluation.

Regulatory Position C.3.c.(1)

The accumulators will be discharged, one at a time, into the open reactor vessel. With the RCS closed, pressurized, and solid, as the Guide infers, there is no convenient way to rapidly depressurize at a rate which would provide a meaningful accumulator discharge.

The discharge flow rate will be calculated from the measurement of accumulator pressure changes versus time vice level versus time.

Regulatory Position C.3.c(2)

Only the normal power supply will be used for this test. Emergency Power System capability will be demonstrated during other tests utilizing the emergency diesel-generators.

Regulatory Position C.3.c(3)

Flow through accumulator check valves will be demonstrated at normal operating temperature and pressure by pressurizing test lines with a charging/safety injection pump.

6.3.5 Instrumentation Application

Instrumentation and associated analog and logic channels employed for initiation of ECCS operation are discussed in Section 7. This section describes the instrumentation employed for monitoring ECCS components during normal plant operation and also ECCS post-accident operation. All alarms are annunciated in the Control Room.

6.3.5.1 Temperature Indication

Residual Heat Exchanger Outlet Temperature

The fluid temperature at the outlet of each residual heat exchanger is recorded in the Control Room.

6.3.5.2 Pressure Indication

Boron Injection Tank Pressure

Boron injection tank pressure is indicated in the Control Room. A high pressure alarm is provided.

Safety Injection Header Pressure

Safety injection pump discharge header pressure is indicated in the Control Room.

Accumulator Pressure

Duplicate pressure channels are installed on each accumulator. Pressure indication in the Control Room and high and low pressure alarms are provided by each channel.

Test Line Pressure

A local pressure indicator used to check for proper seating of the accumulator check valves between the injection lines and the RCS is installed on the leakage test line.

Residual Heat Removal Pump Discharge Pressure

Residual heat removal discharge pressure for each pump is indicated in the Control Room. A high pressure is actuated by each channel.

6.3.5.3 Flow Indication

Safety Injection Pump Header Flow

Flow through each safety injection pump header is indicated in the Control Room.

Test Line Flow

Local indication of the leakage test line flow is provided to check for proper seating of the accumulator check valves between the injection lines and the RCS.

Residual Heat Removal Pump Flow

The flow of reactor coolant through each RHR header during injection or recirculation is indicated in the Control Room.

Safety Injection Pump Minimum Flow

A flow indicator is installed in the safety injection pump minimum flow line.

6.3.5.4 Level Indication

Refueling Water Storage Tank Level

The level of water in the RWST is continuously measured by two separate instrument channels (Unit 2 is provided with four instrument channels) with readouts on the main control board. Alarms are set at the proper level to initiate the switchover from injection to cold leg recirculation.

Accumulator Water Level

Each accumulator tank has two level measuring instruments with readouts on the main control board. Each instrument is set to alarm if the tank level falls or rises by more than a set amount from the normal operating level.

Boric Acid Tanks Level

Two level indicators give indication and alarm in the Control Room.

Containment Building Sump Level

The Containment Building has two sumps - containment recirculation sump and Reactor Building sump. The containment recirculation sump has two redundant sump water level indicators on the console bezel in the main control room and two redundant sump water level switches that actuate separate console bezel lamps when the minimum required sump water level to support ECCS recirculation operation has been reached.

6.3.5.5 Valve Position Indication

Valve positions which are indicated on the control board are done so by a "normal off" system; i.e., should the valve not be in its proper position a bright white light will be lit and thus give a highly visible indication to the operator.

Accumulator Isolation Valve Position Indication

The accumulator motor-operated valves are provided with red (open) and green (closed) position indication lights located at the control switch for each valve. These lights are energized from independent power, other than the valve's control power, and actuated by valve motor-operator limit switches.

A monitor light that is on when the valve is not fully open is provided in an array of monitor lights that are all off when their respective valves are in proper position enabling safeguards operations. This light is energized from a separate monitor light supply and actuated by a valve motor-operated limit switch.

An alarm annunciator point is activated by both a valve motor-operator limit switch and by a valve position limit switch activated by stem travel whenever accumulator is not fully open for any reason with the system at pressure (the pressure at which the safety injection block is unblocked). A separate annunciator point is used for each accumulator valve. The motor-operator limit switch alarm will be recycled at approximately 1 hour intervals to remind the operator of the improper valve lineup.

6.3.6 References for Section 6.3

1. Igne, E. G. and Locante, J., "Environmental Testing of Engineered Safety Features Related Equipment (NSSS-Standard Scope)," WCAP-7410-L (Proprietary) December 1970 and WCAP-7744 (Non-Proprietary), Volume 1, August 1971, and Volume 2, September 1970.

2. Nystrom, J. B., "Experimental Evaluation of Flow Patterns in an RHR Sump With Simulation of Screen Blockage--Salem Nuclear Station," Alden Research Center, Worcester Polytechnic Institute, January 1981.

3. Nuclear Safety Advisory Letter NASL-93-016, Revision 1, "Containment Spray System Issues," Westinghouse, October 4, 1993.

4. Design Change Package 80068486 Rev. 0, "Relocation of RWST Pressure Boundary/Plugging of Weep Hole".

5. Design Change Packages 80080787 and 80080788, "Salem Units 1 & 2 Sump Upgrades".

6. S-C-RHR-MDC-2039, "Debris Generation due to LOCA within Containment for Resolution of GSI-191".

7. S-C-RHR-MDC-2056, "Post-LOCA Debris Transport to Containment Sump for Resolution of GSO-191".

TABLE 6.3-1

ECCS CODE REQUIREMENTS

Component	Code
Refueling Water Storage Tank	ASME Section III Class C (Note 2)
Residual Heat Exchanger	
Tube Side	ASME Section III Class C
Shell Side	ASME Section VIII
Accumulators	ASME Section III Class C
Valves	ANSI B16.5 or MSS-SP-66 or ASME Code, Section III, 1968
Piping	ANSI B31.1(Note 1)
Boron Injection Tank	ASME Section III Class C
Pumps	
Centrifugal Charging	ASME Section III
Safety Injection	ASME Section III
Residual Heat Removal	ASME Section III

(1) For piping not supplied by the NSSS supplier, material inspections, fabrication and quality control conform to ANSI B31.7. Where not possible to comply with ANSI B31.7, the requirements of ASME III-1971, which incorporated ANSI B31.7, were adhered to.

(2) The Unit 1 Refueling Water Storage Tank (in the area of the 20" suction line reinforcement plate) is ASME Section III Class III in accordance with the 1995 Edition/1996 Addenda per DCP 80068486.

TABLE 6.3-2

ACCUMULATOR DESIGN PARAMETERS

Number	4
Type	Stainless steel clad/carbon steel
Design pressure, psig	700
Design temperature, °F	300
Operating temperature, °F	50-150
Normal operating pressure, psig	650
Minimum operating pressure, psig	595.5
Total volume, ft^3	1350
Minimum operating water volume, ft^3	831.9
Volume N_2 gas, ft^3	500
Boron concentration (as boric acid)	
Nominal, ppm	2000
Minimum, ppm	1900
Code	ASME III Class C

TABLE 6.3-3

BORON INJECTION TANK DESIGN PARAMETERS

Number	1
Total volume, gal (also useable volume)	900
Design pressure, psig	2825
Design temperature, °F	150-180
Material	SS Clad Carbon Steel
Code	ASME III, Class C

TABLE 6.3-4

REFUELING WATER STORAGE TANK DESIGN PARAMETERS

Number	1
Tank capacity, gal.	400,000
Minimum volume, (solution) gal.	364,500
Operating pressure	atmospheric
Operating temperature, °F	40 - 100°F
Outside diameter, ft (approx.)	38
Straight side height, ft	48
Material	ASTM-A240 Type 304L Stainless steel
Design pressure	atmospheric
Design temperature, °F	120
Boron concentration,	
Nominal, ppm	2400
Minimum, ppm	2300
Maximum, ppm	2500

TABLE 6.3-5

DESIGN PARAMETERS - ECCS PUMPS

	Centrifugal Charging Pumps	Safety Injection Pumps	Residual Heat Removal Pumps
Number	2	2	2
Design pressure, psig	2800	1700	600
Design temperature, °F	300	300	400
Design flow rate, gpm	150	425	3000
Design head, ft.	5800	2500	350
Max. flow rate, gpm	560	675	4500*
Head at max. flow rate, ft	1300	1500	300
Discharge pressure at shutoff, psig	2670	1520	170
Motor horsepower	600	400	400
Type	Horizontal multi-stage centrifugal	Horizontal multi-stage centrifugal	Vertical single-stage centrifugal
Material	Stainless steel	Stainless steel	Stainless steel

* During the recirculation modes, higher flows can occur depending on system failure assumption.

TABLE 6.3-6

SEQUENCE OF CHANGEOVER OPERATION INJECTION TO RECIRCULATION

The following sequence of operations is used when terminating the injection mode and starting the recirculation mode when low level is reached in the RWST. Note: Because initiating events in MODES 3 and 4 may start at lower pressures and temperatures, the steps/sequences below may vary slightly:

	Unit 1	Unit 2
1.	Confirm Minimum Sump Level	Confirm Minimum Sump Level
1.a	N/A	Enable Semi-Automatic Switchover
2.	Reset SEC, SI and Motor Control Centers	N/A
2.a	N/A	Remove Lockouts for SJ69, 68 and 67
3.	Stop RHR Pumps 11 and 12	N/A
3.a	Close RHR Cross-tie Valves (RH19)	N/A
3.b	Ensure both RHR pumps have stopped	N/A
3.c	Remove Lockouts for SJ44, 69, 68, and 67 valves	N/A
3.d	If an RHR pump fails to stop, remove lockout for failed pumps' RHR Cold Leg Isolation valve (SJ49) and close valve	N/A
3.e	Close RWST/RHR Isolation Valves (RH4)	N/A
3.f	Stop one CS pump, if two are operating	N/A
3.g	N/A	Verify SJ44 Valves Open
3.h	N/A	Start RHR pumps 21 and 22
3.i	N/A	Close SJ69 Valve
3.j	N/A	Reset SI, SEC and Motor Control Centers
3.k	N/A	Stop one CS pump
3.l	N/A	Close RHR Cross-tie Valves (RH19)
4.	Determine Diesel Loading	Determine Diesel Loading
5.	Ensure that at least 2 CC pumps are operating	Ensure that at least 2 CC pumps are operating
5.a	Open CCW water supply to RHR Heat Exchanger valves (CC16)	Ensure CCW water supply to RHR Heat Exchanger valves (CC16) Open

6.	Open Containment Sump Isolation Valves (SJ44)	N/A
6.a	Open RHR Cold Leg Isolation Valves (SJ49)	N/A
6.b	Restart RHR pumps 11 and 12	N/A
7.	Close SI Pump miniflow isolation valves (SJ67 and SJ68)	Close SI Pump miniflow isolation valves (SJ67 and SJ68)
7.a	Open RHR pump discharge to Charging pumps and SI pumps isolation valves (SJ45)	Open RHR pump discharge to Charging pumps and SI pumps isolation valves (SJ45)
7.b	Open cross-tie between Charging pumps and SI pumps suction isolation valve (SJ113) Open	Ensure cross-tie between Charging pumps and SI pumps suction isolation valve (SJ113) Open
7.c	Start Charging pumps and SI pumps	Start Charging pumps and SI pumps

Note: Switchover for long-term core cooling flow is complete at this time.

8.	Isolate RWST from SI, C/SI and RHR pumps	Isolate RWST from SI, C/SI and RHR pumps
8.a	Remove lockout for RWST/SI pump isolation valve SJ30	Remove Lockout for RWST/SI pump isolation valve SJ30
8.b	Close RWST/Charging pump isolation valves (SJ1 and SJ2)	Close RWST/Charging pump isolation valves (SJ1 and SJ2)
8.c	Close RWST/Common Suction valve (SJ69)	N/A
8.d	Close RWST/SI pump isolation valve (SJ30)	Close RWST/SI pump isolation valves (SJ30)
8.e	Place RH29 valves in "Manual" and close valves	Place RH29 valves in "Manual" and close valves
9.	When the RWST low-low level is reached, perform the following:	When the RWST low-low level is reached, perform the following:
9.a	Stop the operating CS pump	Stop the operating CS pump
9.b	Close the RHR pump to RCS cold leg isolation valve (SJ49)	Close the RHR pump to RCS cold leg isolation valve (SJ49)
9.c	Open RHR supply to Containment Spray, valve (CS36)	Open RHR supply to Containment Spray, valve (CS36)

Note: The Emergency Core Cooling System is now aligned for cold leg recirculation with recirculation containment spray as follows:

1. RHR Pump 12 (22) is delivering from the recirculation sump directly to the spray header and to the suction of charging pumps through valve SJ45.

2. RHR Pump 11 (21) is delivering from the recirculation sump directly to the cold legs via valve SJ49 and to the suction of the safety injection pumps via valve SJ45.

3. Recirculation spray is established when RHR supply to Containment Spray, valve CS36, is open.

The sequence of operations for change-over from the cold leg recirculation phase to the hot leg recirculation phase is as follows:

Close the spray header isolation valve (12CS36).

Stop safety injection pump number 11.

Close the safety injection pump cross-tie isolation valve (11SJ134).

Open hot leg isolation valve (11SJ40).

Start safety injection pump number 11.

Stop safety injection pump number 12.

Close the cold leg isolation valve (1SJ135) and close the safety injection pump cross-tie isolation valve (12SJ134).

Open hot leg isolation valve (12SJ40)

Start safety injection pump number 12.

The emergency core cooling pumps are now aligned for the hot leg recirculation as follows:

a. The No. 12 RHR pump is delivering water from the containment recirculation sump to the following:

   (1) to the suction header of the centrifugal charging pumps via 12SJ45. The discharge from the centrifugal charging pumps is delivered the RCS cold legs via the BIT flow path.

   (2) to the suction header of the safety injection pumps via the SJ113 cross-over valves. The discharge from the safety injection pumps is delivered to the RCS hot legs.

b. The No. 11 RHR pump is delivering water from the containment recirculation sump to the following:

   (1) to the suction header of the safety injection pumps via 11SJ45. The discharge from the safety injection pumps is delivered to the RCS hot legs.

   (2) to the suction header of the centrifugal charging pumps via the SJ113 cross-over valves. The discharge from the centrifugal charging pumps is delivered the RCS cold legs via the BIT flow path.

   (3) directly to the RCS cold legs via the 11SJ49 valve and the RHR cold leg injection lines.

c. Number 11 and 12 safety injection pumps are delivering to the Reactor Coolant System through individual hot leg injection headers.

TABLE 6.3-7

All horsepower values are rated values. See ES-9.002 for actual values.

4-LOOP PUMP PARAMETERS

Pump	Normal Condition Parameters			Accident Condition Parameters			Motor Horsepower Selection			
	Head (Ft.)	Flow (GPM)	Brake Horsepower Required (HP)	Head (Ft.)	Flow (GPM)	Brake Horsepower Required (HP)	Specified Full Load Horsepower (HP)	Service Factor	Service Factor Rating (HP)(3)	Nema Temperature Limit for Service Factor Rating of 1.15
Centrifugal Charging	5800	150	500	1300(2)	560	625[6]	600	1.15	690	(7)
Safety Injection	---	---	---	2500(1) 1500(2)	425 675	360 390	400	1.15	460	(7)
Residual Heat Removal	350	3000	340	300	4500[4]	400[5]	400[5]	1.15	460	(7)

Note: (1) Design Flow Condition of Pump

(2) Runout Condition of Pump

(3) (Full Load HP) X (Service Factor) = Service Factor Rating

(4) During the recirculation modes, higher flows can occur depending on system failure assumption. [See Table 6.3-13]

(5) During the recirculation modes, a maximum 425 HP load can occur.

(6) Horsepowers range from 625 to approximately 650, depending on pump.

(7) Refer to NEMA MG1

TABLE 6.3-8

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TABLE 6.3-9

SINGLE ACTIVE FAILURE ANALYSIS EMERGENCY CORE COOLING SYSTEM
INJECTION PHASE

	Component	Malfunction	Comments
A.	Accumulator	Deliver to broken loop	Total passive system with one accumulator per loop. Evaluation based on three accumulators delivering to the core and one spilling from rupture loop.
B.	Pump:		
	1) Centrifugal Charging	Fails to start	Two provided. Evaluation based on operation of one
	2) Safety injection	Fails to start	Two provided. Evaluation based on operation of one
	3) Residual heat removal	Fails to start	Two provided. Evaluation based on operation of one
C.	Automatically operated valves:		
	1) Boron injection tank isolation		
	a) Inlet	Fails to open	Two parallel valves; one valve is required to open
	b) Outlet	Fails to open	Two parallel valves; one valve is required to open
	c) Recirculation to boric acid tank valve	Fails to close	Two valves in series; only one required to close

Component	Malfunction	Comments
2) Centrifugal charging		
a) Suction line to RWST isolation	Fails to open	Two parallel valves; one valve is required to open
b) Discharge line to the normal charging path* isolation	Fails to close	Two valves in series; only one valve required to close
c) Suction from volume control tank isolation	Fails to close	Two valves in series; only one valve required to close
D. Valves operated from control room		
Centrifugal charging pump recirculation line isolation	Fails to close	Two valves in series; only one valve required to close

* The reactor coolant pump seal water path is left open.

RECIRCULATION PHASE

Component	Malfunction	Comments
A. Valves operated from control room for recirculation:		
1. Containment sump recirculation isolation	Fails to open	Two lines parallel; only one valve in either line is required to open
2. Residual heat removal pumps suction line to RWST isolation	Fails to close	Two gate valves in series; operation of only one valve is required
3. Safety injection pumps suction line RWST	Fails to close	Check valve in series with gate valve; operation of only one valve required

TABLE 6.3-9 (Cont)

Component	Malfunction	Comments
4. Centrifugal charging pumps suction line to RWST isolation	Fails to close	Check valve in series with two parallel gate valves. Operation of either the check valve or the gate valves required
5. Safety injection pump suction line discharge of residual heat exchangers	Fails to open	Separate and independent high head injection path via the centrifugal charging pumps taking suction from discharge of alternate residual heat exchanger. A crossover line allows flow from one heat exchanger to reach both safety injection and charging pump if necessary.
B. Pumps:		
1) Component cooling	Fails to start	Three provided. Evaluation based on operation of one.
2) Service water	Fails to start	Six provided. Evaluation based on operation of two.
3) Residual heat removal pump	Fails to start	Two provided. Evaluation based on operation of one.
4) Charging pump	Fails to operate	Same as injection phase
5) Safety injection pumps	Fails to operate	Same as injection phase

TABLE 6.3-10

ACCUMULATOR INLEAKAGE

Observed Leak Rate cc/hr	Time Period Between Level Adjustments	Total Integrated Leakage ft^3 *+
2470	1 month	124.5
830	3 months	42.5
415	6 months	20.8
276	9 months	18.8
208	1 year	10.4

* A total of 163.4 cubic feet, added to the initial amount, can be accepted in each accumulator before an alarm is sounded.

+ Max. allowed leak rate for manufacturers acceptance test is 20cc/hr (Back leakage through check valves)

TABLE 6.3-11

SINGLE PASSIVE FAILURE ANALYSIS - EMERGENCY CORE COOLING SYSTEM
RECIRCULATION PHASE

Flow Path	Indication of Loss of Flow Path	Alternative Flow Path
Low Head Recirculation (Cold Leg)		
From containment sump to low head injection header via the residual heat removal pumps and the residual heat exchangers	Reduced flow in the discharge line from one of the residual heat exchangers (one flow monitor in each discharge line)	Via the independent, identical low head flow path utilizing the second residual heat exchanger
High Head Recirculation (Cold Leg)		
From containment sump to high head injection header via residual heat removal pump, residual heat exchanger to the safety injection pumps and charging pump (using cross-tie)	Reduced flow in the discharge lines from the safety injection pump and centrifugal charging pump (a flow monitor in the discharge lines of each set of pumps)	From containment sump to the high head cold leg injection headers via alternative residual heat removal pump, alternate residual heat exchanger and the centrifugal the charging/safety injection pumps. A cross-tie with two parallel valves is provided.

Flow Path	Indication of Loss of Flow Path	Alternative Flow Path
High Head Recirculation (Hot Leg)		
From containment sump to the high head hot leg injection headers via the residual heat removal pump residual heat exchanger to the safety injection pump.	Reduced flow in the discharge from safety injection pump.	From containment sump to the high head hot leg injection points via alternative residual heat removal pump, residual heat exchanger, and safety injection pump crossover line.

TABLE 6.3-12

RECIRCULATION LOOP LEAKAGE SOURCES

Items	Type of Leakage Control
Residual Heat Removal Pumps (Low Head Safety Injection)	Mechanical seal with leakoff
Centrifugal Charging Pump	Same as residual heat removal pump
Safety Injection Pumps	Same as residual heat removal pump
Flanges: a. Pump b. Valves: Bonnet, Body (larger than 2") c. Control Valves d. Other	Gasket - adjusted to zero leakage following any test
Valves - Stem Leakoffs - Seat Leakage	Backseated, double packing with leakoff
Misc. Small Valves	Flanged body packed stems

TABLE 6.3-13

NET POSITIVE SUCTION HEADS FOR
POST-ACCIDENT OPERATIONAL PUMPS

Pump	Elevation	Flow and Condition	Suction Source and Elevation	Minimum Available NPSH	Required NPSH	Maximum Water Temperature
Safety Injection	86'-3"	675 gpm runout	RWST 101'-8"	31.3'	24'	100°F
Centrifugal Charging	87'-5"	560 gpm runout	RWST 101-8"	38'	23'	100°F
Residual Heat Removal	46'-10"	1 pump operating 4500 gpm runout flow	RWST 101'-8"	63.3'	19.5'	100°F
Residual Heat Removal	46'-10"	2 pumps operating 3000 gpm/pump rated flow	RWST 101'-8"	53.2'	11'	100°F
Containment Spray	86'-3"	2600 gpm rated flow	RWST 101'-8"	29.9'	10'	100°F
Component Cooling	86'-0"	4600 gpm rated flow	Head Tank 128'	40'	14'	135°F
Service Water	Impeller Suction 72'-3" Pump Dis. 94'-0"	14,400 gpm runout flow	Plant Intake Water Level 76'	32.1'	31.7'	90°F

Pump	Elevation	Flow and Condition	Suction Source and Elevation	Minimum Available NPSH	Required NPSH	Maximum Water Temperature
Residual Heat Removal (one pump Operation)	46' - 10"	See Below	Containment Sump	See Below	See Below	Saturation
- Cold Leg Recirculation (Unit 1)		5110 gpm (maximum)	80' - 10"	26.6'	25'	
- Cold Leg Recirculation (Unit 2)		4900 gpm (maximum)	80' - 10"	24.8'	23.1'	
- Hot Leg Recirculation (Unit 1)		4980 gpm (maximum)	81' - 8"	27.8'	24'	
- Hot Leg Recirculation (Unit 2)		4980 gpm (maximum)	81' - 8"	25.4'	24.4'	

The available NPSH was calculated for the pumps indicated above using the following conservative assumptions:

1. All calculations assume an empty refueling water storage tank.
2. No credit is taken for RWST fluid temperature below 100°F.
3. No credit is taken for increased containment pressures following the LOCA.

TABLE 6.3-14

MATERIALS EMPLOYED FOR
EMERGENCY CORE COOLING SYSTEM COMPONENTS

Component	Material
Accumulators	Carbon steel, clad with Austenitic stainless steel
Boron injection tank	Carbon steel, clad with Austenitic stainless steel
Pumps	
Safety injection	Austenitic stainless steel
Residual heat removal	Austenitic stainless steel
Boron injection tank	
Recirculation pump	Austenitic stainless steel
Residual heat exchangers	
Shell	Carbon steel
Shell end cap	Carbon steel
Tubes	Austenitic stainless steel
Channel	Austenitic stainless steel
Tube sheet	Austenitic stainless steel
Valves	
Motor operated valves	
Containing radioactive fluid	
Pressure Containing parts	Austenitic stainless steel or equivalent

TABLE 6.3-14 (Cont)

Component	Material
Body-to-bonnet	
Bolting and nuts	Low alloy steel
Seating surfaces	Stellite No. 6 or equivalent
Stems	Austenitic stainless steel or, 17-4PH stainless
Motor-operated valves Containing nonradioactive, Boron - free fluids	
Body, bonnet and flange	Carbon steel
Stems	Corrosion resistant steel
Diaphragm valves	Austenitic stainless steel
Accumulator check valves	
Parts contacting borated water	Austenitic stainless steel
Clapper arm shaft	17-4PH stainless
Relief valves	
Stainless steel bodies	Stainless steel
Carbon steel bodies	Carbon steel
All nozzles, discs, spindles and guides	Austenitic stainless steel

TABLE 6.3-14 (Cont)

Component	Material
Bonnets for stainless steel valves without a balancing bellows	Stainless steel or Plated carbon steel
All other bonnets	Carbon steel
Piping	
All piping in contact with borated water	Austenitic stainless steel

Figure F6.3-1A Sheets 1, 2, 3 & 4 of 4 intentionally deleted.

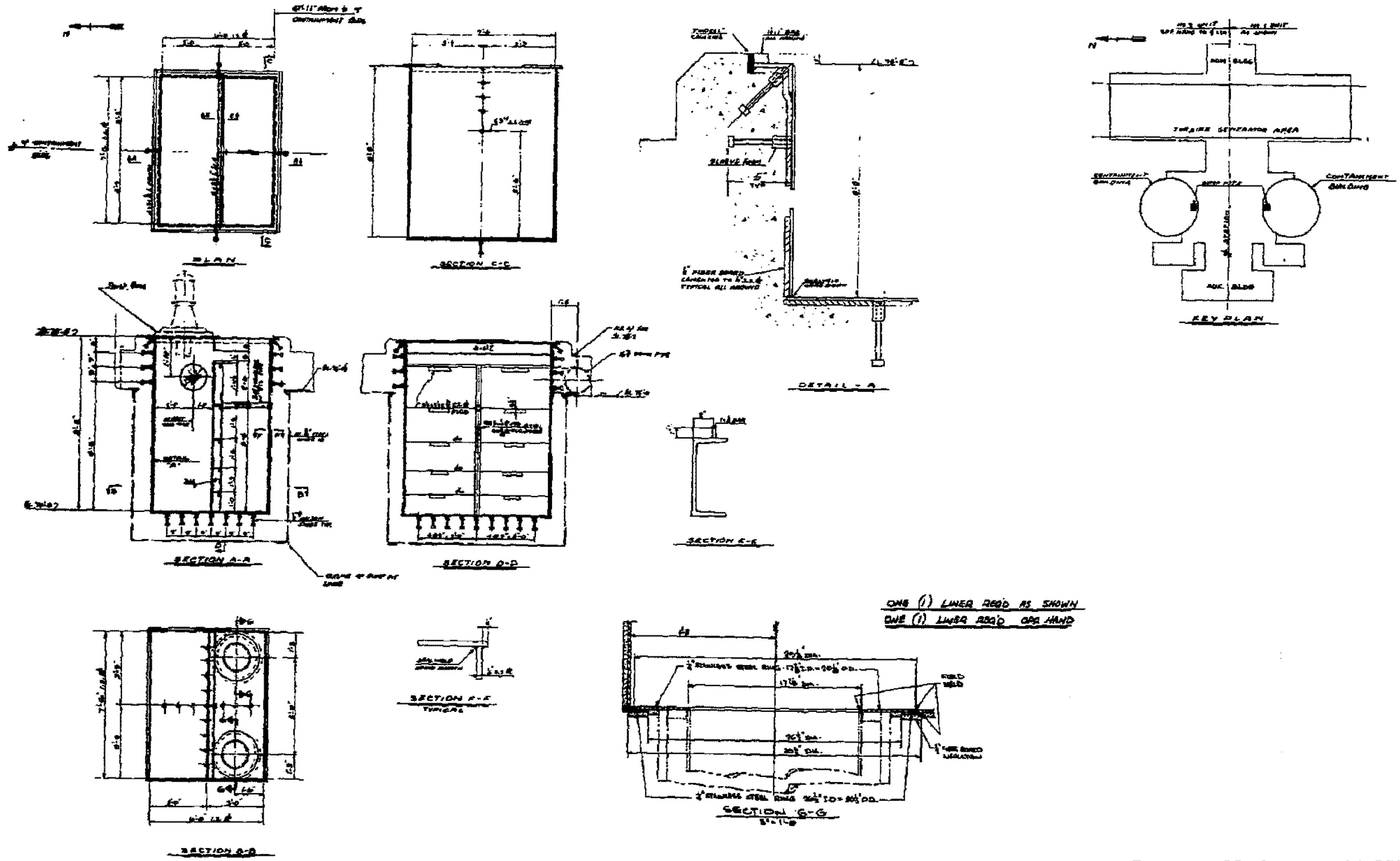
Refer to plant drawing 205234 in DCRMS

Figure F6.3-1B Sheets 1, 2, 3 & 4 of 4 intentionally deleted.

Refer to plant drawing 205334 in DCRMS

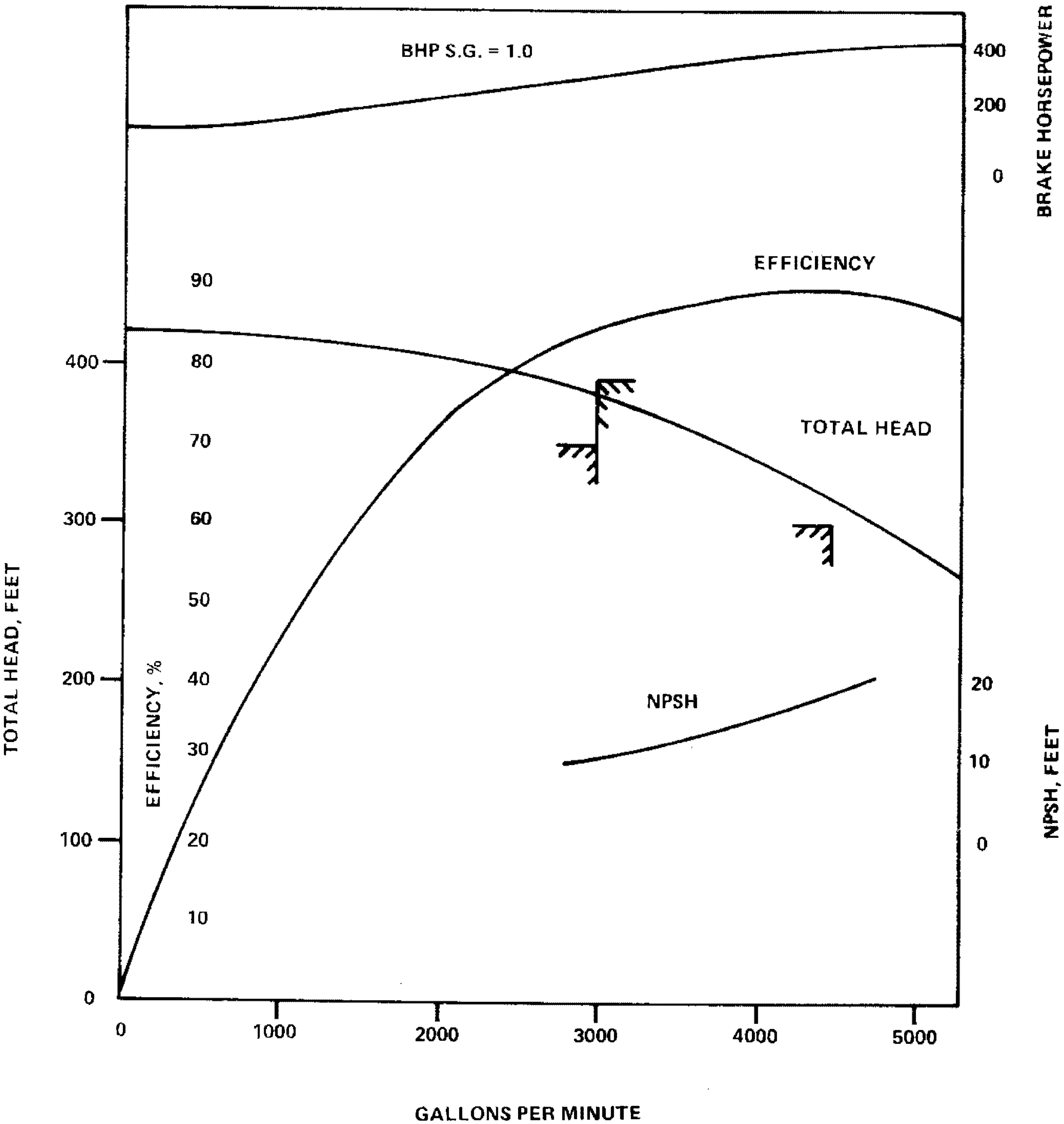
Figure F6.3-2 intentionally deleted.

Refer to plant drawing 208915 in DCRMS



Revision 23, October 17, 2007

PSEG Nuclear, LLC SALEM NUCLEAR GENERATING STATION	Salem Nuclear Generating Station Containment Sump Pit
	Updated FSAR — Figure 6.3-3



REVISION 6
FEBRUARY 15, 1987

PUBLIC SERVICE ELECTRIC AND GAS COMPANY SALEM NUCLEAR GENERATING STATION	Pump Head Characteristic Curve RHR Pump
	Updated FSAR — Figure 6.3-4

6.4 HABITABILITY SYSTEMS

The Salem Generating Station (SGS) Units 1 and 2 control rooms are located within a common Control Room Envelope (CRE). The control rooms are located at elevation 122 feet of the Auxiliary Building and contains those controls and instrumentation necessary for operation of the units under normal and abnormal conditions. The CRE is continuously occupied by operating personnel under all operating conditions.

The facilities located within the CRE are designed to be habitable throughout the course of a design basis accident (DBA) and the resulting radiological condition.

6.4.1 Design Bases

The control room habitability system provides for the access and occupancy of the CRE during normal conditions, radiological emergencies, hazardous chemical emergencies and fire emergencies. The system design conforms with the intent of AEC General Design Criterion (GDC) 19 (1971) as described in Section 3.1. To this end, administrative procedures, shielding, ventilation system, radiological monitoring and the fire protection system are used.

The control room habitability system also conforms with the intent of AEC GDC 5 (1971), Sharing of systems, as described in section 3.1.

The control room habitability systems functional design was evaluated for each of the following conditions:

(1) Normal operating conditions

(2) Radiological conditions resulting from a DBA

(3) Hazardous chemical release

(4) Fire or smoke inside or outside the common control room

6.4.2 System Design

6.4.2.1 Control Room Envelope

The Control Room Envelope (CRE) consists of the following rooms and facilities that the control room habitability system provides for continuous occupancy to support personnel during normal operating conditions and for the duration of an accident:

(1) Units 1 and 2 Control Rooms

(2) Units 1 and 2 Data Logging Rooms

(3) Conference Room

(4) Operations Superintendent Office

(5) Units 1 and 2 Control Room Supervisor Platform Area

(6) Operator Ready Room

The walls, recorder panels, doors, floors and ceiling for the rooms in the CRE make up the physical boundary between the adjacent rooms (relay, controls equipment, work control center, and HVAC equipment) and the outside environment.

As described in Section 12, the control room shielding consists of concrete walls, floor and roof. The control room shielding design ensures that the radiation exposure to the occupants in the control room is consistent with the GDC 19 limits.

6.4.2.2 Ventilation System Design

The control room ventilation system is designed to support personnel during normal operating conditions and during an accident. The design of the control room emergency air conditioning system conforms with the intent of the Regulatory Guide 1.52, Rev. 1 (1976) with the variances described in Section 3, Appendix 3A. Section 9.4 describes the control room ventilation and air conditioning systems.

As described in Section 11.4, the control room ventilation system consists of redundant radiation monitors that are shared by Units 1 and 2. Each unit consists of a digital microprocessor-based radiation monitor with two detection channels; one channel monitors the Unit 1 normal intake air and the other channel monitors the Unit 2 normal intake air. Each monitor provides actuation signals, based on high radiation, to the Unit 1 and 2 control area ventilation controls system.

Plant systems affecting control room habitability are the following:

(1)	Control Area Air Conditioning System (CAACS)	(Section 9.4.1)
(2)	Fire Protection	(Section 9.5.1)
(3)	Communications	(Section 9.5.2)
(4)	Lighting	(Section 9.5.3)
(5)	Radiological Monitoring	(Section 11.4)
(6)	Shielding	(Section 12.1)

Air Conditioning Units (per unit):

Three 50% capacity package chiller units are provided, each consisting of a reciprocating compressor, cooling coil or chiller unit and water cooled condenser. Two 100% capacity chilled water pumps (one standby) circulate water through all three chiller units. During emergency conditions, two chillers are available per unit supplying 48°F (nominal) chilled water to each units CAACS and CREACS coils for the removal of the design heat loads based on summer design conditions.

6.4.2.3 Leak Tightness

The control room habitability system is designed to pressurize the CRE to equal to or greater than a combination of 1/8 inwc and 1/16 inwc differential to adjacent rooms and the environment. The dp is 1/8 inwc for all areas except the control room boundary with the relay rooms, which is 1/16 inwc. The positive CRE pressure is achieved by providing a total of 2000 (nominal) scfm of filtered makeup air from a preferred selected intake. The Unit 1 and 2 control area ventilation system during an accident normally operates with two emergency filter trains providing 1000 (nominal) scfm of makeup air each, or during an abnormal alignment with a single emergency filter train providing 2000 (nominal) scfm makeup air. The pressurized CRE ensures contamination flow out of the CRE, thereby reducing operator dose.

In order to assess the amount of unfiltered air in-leakage into the CRE, tracer gas air in-leakage tests were performed in 2003. Air in-leakage rates of the CRE and associated ventilation boundary were determined with the ventilation in various pressurization modes. In all cases the nominal in-leakage rate was determined to be less than 100 cfm. A rate of 275 cfm is assumed in design basis accident radiological consequence analyses to provide margin with respect to the in-leakage rates determined by the tests and to account for additional in-leakage due to ingress and egress.

6.4.3 System Operational Procedures

Procedures are provided for operating the control room habitability system in the required modes of operation to protect operating personnel in the control room during an emergency condition. During emergency conditions, the control room habitability system is initiated automatically upon an SI or high radiation signal or manually to place the system in the preferred mode of operation to protect operating personnel in the control room.

6.4.4 Design Evaluation

Table 6.4-3 summarizes data in the control area ventilation, which is described in detail in Section 9.4.1.

6.4.4.1 Normal Operation

During normal operating conditions, the control area ventilation system operates to supply cool filtered air to maintain ambient room temperatures for personnel comfort and instrumentation accuracy. The ventilation system is set to maintain room temperatures at a nominal temperature of 76°F. The normal operating limits and equipment design temperature limitations in the control room and adjoining control equipment room are described in Section 3.11.1.3. In this mode, most of the air inside the control room areas is recirculated with some quantity of makeup air introduced to maintain the CRE and control room areas at a positive pressure to minimize the infiltration of dust, smoke and other airborne contaminants. Both Units 1 and 2 control area ventilation systems provide cool filtered air to the CRE during normal operation.

6.4.4.2 Radiological Protection

The adequacy of the control room shielding is evaluated for normal operating and accident conditions in Section 12.

The adequacy of the control area ventilation system is evaluated for radiological emergencies in Section 9.4.1 and 15.4.1.9.

Radiological consequences within the SGS control room envelope, which includes the control rooms for both units, are evaluated for the following design basis accidents at either unit:

- Loss of Offsite Power (Section 15.2.9)
- Small Line Break Outside Containment (Section 15.3.1)
- Volume Control Tank Rupture (Section 15.3.6.2)
- Waste Gas Decay Tank Rupture (Section 15.3.6.3)
- Loss-of-Coolant (Section 15.4.1)
- Main Steam Line Break (Section 15.4.2)
- Steam Generator Tube Rupture (Section 15.4.4)
- Locked RC Pump Rotor (Section 15.4.5)
- Fuel Handling Accident Inside Containment (Section 15.4.6)
- Fuel Handling Accident Inside Fuel Building (Section 15.4.6)
- Rod Ejection Accident (Section 15.4.7)

The parameters/assumptions used to evaluate offsite dose consequences following the above DBA's are discussed in the referenced Sections. These parameters/assumptions are also used to estimate the associated control room doses. The parameters associated with the control room design as used in the control room habitability analyses are provided in Table 6.4-3. The atmospheric dispersion factors used in the control room habitability analyses are based on the ARCON96 model described in NUREG/CR-6331, Revision 1. The atmospheric dispersion factors that are generally used in the control room habitability analyses are provided in Table 15.4-5D.

With the exception of the LOCA, which utilizes the safety injection signal, the redundant control room in-duct monitors initiate the control room emergency ventilation system. The design ensures that initiating instrumentation can select the less contaminated intake, i.e., the radiation monitors select the less contaminated intake based on a comparison of the radiation readings at either intake, whereas the SI signal selects the intake associated with the non-accident unit.

The 30 day accident dose in the control room is within 10 CFR 50.67 dose limits and is summarized in Table 15.4-5C. This dose value represents the post-LOCA dose in the control room.

6.4.4.3 Toxic and Chemical Gas Protection

Regulatory Guide 1.78 requires that hazardous chemicals, such as those indicated in Table C-1 of the Guide, be considered in an analysis of Control Room habitability if they are frequently shipped within a 5-mile radius of the station. The Guide also defines frequent shipments as being 50 or more trips per year for barge traffic and 10 or more trips per year for truck traffic. Chemicals stored or situated at distances greater than 5 miles from the facility need not be considered in the habitability analysis.

The Salem site is located in a rural area with no major manufacturing or chemical plants located with 5 miles of the site. The only major transportation route within 5 miles of the station is the Delaware River, with the intra-coastal waterway passing 1 mile west of the site.

The Salem Generating Station uses a hypochlorite biocide system, thus eliminating an onsite chlorine hazard. The Control Room area fresh air intake ducts are equipped with redundant radiation monitoring systems which provide annunciation, automatically isolate the Control Room, and switch the ventilation system to the accident pressurized mode on high radiation detection. Sections 2.2.3.2 and 2.2.3.3 discuss and conclude that a release of any of the hazardous chemicals stored onsite or shipped past the site will not impact control room habitability.

Hazardous chemicals shipped past the Salem site occur infrequently. The frequencies of the deliveries are listed in Table 2.2-4. Regulatory Guide 1.78 requires a control room habitability evaluation for shipments of hazardous chemicals that are considered "frequent" shipments. The frequent criteria for river barges are 50 per year. As seen from Table 2.2-4, none of the hazardous chemicals shipped past the site exceed this criteria, therefore, a control room habitability evaluation is not required.

As previously mentioned, several chemicals are stored onsite that are considered hazardous. Sulfuric acid is stored in 4,000 and 2,250 gallon tanks in the SGS Turbine Buildings and it is stored in 16,000 gallon tanks at the HCGS. Calculations indicated that the toxicity limit found in Regulatory Guide 1.78 will not be exceeded in the control rooms during a postulated release at any of the sources.

Liquid nitrogen and nitrogen stored as a compressed gas is stored at various locations onsite. According to the criteria contained in Regulatory Guide 1.78, the largest single source should be evaluated for its impact on control room habitability. The sources evaluated at the SGS are the portable nitrogen tube trailers located in various areas throughout the SGS yard area and the (2) liquid nitrogen tanks located behind Unit No. 1 & 2 Auxiliary Buildings which can contain up to 7500 gallons of liquid nitrogen. In addition to these sources, liquid nitrogen is also stored in 9,000 gallon tanks at the HCGS. Calculations indicated that the oxygen depletion is negligible in the control rooms during a postulated release at any of the significant sources.

Chemicals used as fire-fighting agents were evaluated. Carbon dioxide is stored on the 84 foot elevation of each of the Auxiliary Buildings.

It is also stored at HCGS. Calculations indicated that the toxicity limit established in Regulatory Guide 1.78 as well as asphyxiation levels would not be exceeded during postulated releases at the significant sources. The Halon storage vessels are relatively small and do not contain the volume of Halon required to cause asphyxiation in the control rooms, therefore, a postulated release will not pose a danger to the control rooms.

Ammonium hydroxide is stored in two 350 gallon vessel totes that are connected in series in the SGS Unit No. 1 and SGS Unit No. 2 Turbine Buildings. Evaluations concluded that the control rooms would remain habitable during a postulated release at either of the storage tank locations. The shipments to the site are considered "frequent" and are discussed in Section 2.2.3.3.

Ethanolamine is stored in two 350 gallon totes that are connected in series in the SGS Unit 2. The effective volume is 700 gallons. Evaluations concluded that the control rooms would remain habitable during a postulated release at the storage totes. The shipments to the site are considered "frequent" and are discussed in Section 2.2.3.3.

Hydrazine is stored in a 300 gallon vessel also in the Unit No. 1 side of the SGS Turbine Building. The calculations indicated that the control room concentrations will not exceed toxicity limits established in 29CFR Part 1910.1000, Subpart Z during a postulated release.

Aqueous sodium hydroxide is stored in various quantities and vessels at both the SGS and HCGS. Upon a release, sodium hydroxide vapors may form locally at the spill, but the physical properties of this chemical preclude the formation of a plume that will travel in the control room air intakes. The vapor pressure of aqueous sodium hydroxide is very low, especially as the concentration is increased. During a postulated release, mostly water will evaporate from the liquid pool, leaving the solid sodium hydroxide behind. The solid form of sodium hydroxide poses no danger to the control room due to its physical properties.

Helium is stored in 150 lb cylinders at both the SGS and HCGS. It is much lighter than air and upon a postulated failure of one of the cylinders, the helium would disperse rapidly into the atmosphere and not form a continuing plume.

It is concluded that Control Room personnel are adequately protected against the effects of accidental release of toxic and radioactive gases and that the plant can be safely operated or shut down under design basis accident conditions. Due to the use of sodium hypochlorite, there is no chlorine hazard.

6.4.4.4 Smoke and Fire Protection

The adequacy of the control room fire protection system is evaluated in Section 9.5.1. Smoke infiltration inside and outside the CRE is evaluated in Section 9.4.1.

6.4.4.5 Conclusion

The control room habitability systems are capable of performing their functions reliably during normal operating periods and under emergency conditions. It is concluded that control room personnel are adequately protected against the effects of accidental release of toxic and radioactive gases and that the plant can be safely operated or shutdown under design accident conditions.

6.4.5 Testing and Inspection

Surveillance requirements for inspection and testing of the control room ventilation system are contained in Technical Specifications. These requirements ensure that performance capability is maintained throughout the plant's lifetime.

6.4.6 References for Section 6.4

1. "Waterborne Commerce of the United States," U.S. Army Corps of Engineers Annual Publication.

2. Commodity traffic data for imports and exports collected by the Philadelphia Maritime Exchange.

3. Foreign trade cargo movements collected by the Delaware River Port Authority.

4. U.S. Dept. of Commerce, Census Bureau (handling foreign trade data for customs purposes).

5. Interstate Oil Transport, Inc. (which handles most of the barge operations of the Delaware River).

6. U.S. Coast Guard, Captain of the Port, Philadelphia (cognizant of all hazardous materials shipments in the Delaware River).

7. NUREG/CR-6331, "Atmospheric Relative Concentrations in Building Wakes", May 1, 1995

8. NUREG/CR-6331, Revision 1, "Atmospheric Relative Concentrations in Building Wakes", May 1, 1995

TABLE 6.4-1

TABLE DELETED

TABLE 6.4-2

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TABLE 6.4-3

CONTROL AREA VENTILATION SYSTEM PARAMETERS

1. Volume of Control Room Envelope (CRE)	81,420 cu ft
2. Normal Operation per unit for Units 1 & 2 (CAACS[1]):	
- total system outside makeup airflow (design)	2200 cfm
- outside makeup airflow to the CRE (design)	600 cfm
- outside makeup airflow to the CRE (used in dose analysis)	600 cfm
- total system recirculated airflow (design)	30,400 scfm
3. Emergency Operation (2 train alignment, CREACS[1]):	
- unfiltered in-leakage	≤275 cfm[3]
- outside filtered makeup airflow (total for 2 trains)	≤2100 cfm[2]
- total filtered airflow per train (design)	8000 cfm[2]
4. Emergency Operation (1 train alignment, CREACS[1]):	
- unfiltered in-leakage (used in dose analysis)	275 cfm[3]
- filtered makeup air (used in dose analysis)	2100 cfm[2]
- total filtered airflow per train (design)	8000 cfm[2]
- recirculated airflow per train (used in dose analysis)	5100 cfm[2]
- filtered airflow per train (used in dose analysis)	7200 cfm[2]
5. Time Required to Isolate the Control Envelope:	
- automatic damper operation	20 seconds[4]
- including pressurization (used in dose analysis)	1 minute

NOTES:

(1) CAACS = Control Area Air Conditioning System
CREACS = Control Room Emergency Air Conditioning System

(2) Air is filtered through roughing, HEPA and charcoal filter units

(3) This parameter was used in the dose analyses and bounds actual control room in-leakage test results, which include limited ingress/egress and other potential leakage paths such as the CREACS filter housing and the CAA14 isolation dampers.

(4) Time to open or close damper upon receipt of an actuation signal