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6.0 ENGINEERED SAFETY FEATURES

6.1 ENGINEERED SAFETY FEATURE MATERIALS

6.1.1 Metallic Materials

6.1.1.1 Materials Selection and Fabrication

Typical material specifications used for the principal pressure retaining applications in components in the Engineered Safety Features (ESF) are listed in Table 5.2-9. All materials utilized are procured in accordance with the material specification requirements of the ASME Boiler and Pressure Vessel Code, Section III, plus applicable and appropriate Addenda and Code Cases.

The welding materials used for joining the ferritic base materials of the ESF conform to, or are equivalent to, ASME Material Specifications SFA 5.1, 5.2, 5.5, 5.17, 5.18, and 5.20. The welding materials used for joining nickel-chromium-iron alloy in similar base material combination and in dissimilar ferritic or austenitic base material combination conform to ASME Material Specifications SFA 5.11 and 5.14. The welding materials used for Joining the austenitic stainless steel base materials conform to ASME Material Specifications SFA 5.4 and 5.9. These materials are tested and qualified to the requirements of the ASME Code, Section III and Section IX rules and are used in procedures which have been qualified to these same rules. The methods utilized to control delta ferrite content in austenitic stainless steel weldments are discussed in Section 5.2.5.7.

The parts of components in contact with borated water are fabricated of or clad with austenitic stainless steel or equivalent corrosion resistant material.[4] The integrity of the safety-related components of the ESF is maintained during all stages of component manufacture. Austenitic stainless steel is utilized in the final heat treated condition as required by the respective ASME Code Section II material specification for the particular type or grade of alloy. Furthermore, it is required that austenitic stainless steel materials used in the ESF components be handled, protected, stored, and cleaned according to recognized and accepted methods which are designed to minimize contamination which could lead to stress corrosion cracking. The rules covering these controls are stipulated in Westinghouse process specifications, which are discussed in Section 5.2.5.1. Additional information concerning austenitic stainless steel, including the avoidance of sensitization and the prevention of intergranular attack, can be found in Section 5.2.5. No cold worked austenitic stainless steels having yield strengths greater than 90,000 psi are used for components of the ESF within the Westinghouse standard scope.

Westinghouse supplied components within the containment that would be exposed to core cooling water and containment sprays in the event of a loss-of-coolant accident utilize materials listed in Table 5.2-9. These components are manufactured primarily of stainless steel or other corrosion resistant, high temperature material.

The integrity of the materials of construction for ESF equipment when exposed to post design basis accident (DBA) conditions has been evaluated. Post-DBA conditions were conservatively represented by test conditions. The test program[1] performed by Westinghouse considered spray and core cooling solutions of the design chemical compositions, as well as the design chemical compositions contaminated with corrosion and deterioration products which may be transferred to the solution during recirculation. The effects of sodium (free caustic), chlorine (chloride), and fluorine (fluoride) on austenitic stainless steels were considered. Based on the results of this investigation, as well as testing by ORNL and others, the behavior of austenitic stainless steels in the post-DBA environment will be very acceptable. No cracking is anticipated on any equipment even in the presence of postulated levels of contaminants, provided the core cooling and spray solution pH is maintained at an adequate level. The inhibitive properties of alkalinity (hydroxyl ion) against chloride cracking and the inhibitive characteristic of boric acid on fluoride cracking have been demonstrated. Note that qualified coatings inside primary containment located within the zone of influence are assumed to fail for the analysis in the event of a loss-of-coolant accident. The zone of influence for qualified coatings is defined as a spherical zone with a radius of 10 times the break diameter. Coatings on exposed surfaces outside the zone of influence within the containment are not subject to breakdown under exposure to the spray solution and can withstand the temperature and pressure expected in the event of a loss-of-coolant accident.

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

The vessels used for storing ESF coolants include the accumulators and the refueling water storage tank.

The accumulators are carbon steel clad with austenitic stainless steel.[4] Because of the corrosion resistance of these materials, significant corrosive attack on the storage vessels is not expected.

The accumulators are vessels filled with borated water and pressurized with nitrogen gas. The nominal boron concentration, as boric acid, is 3150 ppm. Samples of the solution in the accumulators are taken periodically for checks of boron concentration. Principal design parameters of the accumulators are listed in Table 6.3-1.

The refueling water storage tank is a source of borated cooling water for injection. The nominal boron concentration, as boric acid, is 3200 ppm. The temperature of the refueling water is maintained above the solubility limit for the maximum boron concentration. Principal design parameters of the refueling water storage tank are given in Section 9.2.7.

The ice in the ice condenser is borated by adding sodium tetraborate to the ice. The aqueous solution resulting from the melted ice has a nominal boron concentration of ≥1800 ppm. In the event of an accident, this solution would be delivered to the containment sump. Containment sump pH is also controlled by the sodium tetraborate in the ice. The pH of the ice is maintained between 9.0 and 9.5, which results in a sump pH of approximately 7.5.

Information concerning hydrogen release by the corrosion of containment metals and the control of the hydrogen and combustible gas concentrations within the containment following a LOCA is discussed in Section 6.2.5.

6.1.2 Organic Materials

For paints and coatings inside containment, the conformance with Regulatory Guide 1.54 is described in Section 6.1.4.

Organic materials within the primary containment are identified and quantified according to the following categories: electrical insulation, surface coatings, miscellaneous ALARA (catch basin) containment and shielding (lead blankets), ice condenser equipment, and identification tags for valves and instruments. There is no asphalt inside the containment. There is a small amount of wood (roughly 215 lbs) located inside the ice condenser. The effects of elastomers and plastics on hydrogen generation have been evaluated and determined to be inconsequential. Therefore, the quantities identified below are considered historical and need not be revised due to design changes.

The information in this section is based on a single reactor unit.

6.1.2.1 Electrical Insulation

The typical types of electrical cable insulation/jacket material that are utilized within the primary containment are: silicon rubber, polyethylene, ethylene rubber, chlorosulfonated polyethylene, polyolefin, cross linked polyethylene, kapton. These materials are not significant contributors to hydrogen generation during a design basis accident (approximately 28,000 lbs).

6.1.2.2 Surface Coatings

Material	Mass, lbs
Concrete Surfaces	
Epoxy	2070
Phenolic-epoxy	300
Steel Surfaces:	
Phenolic epoxy	1810

Steel surfaces are undercoated with approximately 85% zinc in a silicate binder (carbozinc 11) or epoxy such as Amerlock 400.

Protective coatings for use in the reactor containment have been evaluated as to their suitability in post-DBA conditions. Tests have shown that the epoxy and modified phenolic systems are the most desirable of the generic types evaluated for in-containment use. This evaluation considered resistance to high temperature and chemical conditions anticipated following a LOCA, as well as high radiation resistance.[2] However, other coating systems meeting DBA testing requirements may be used. Coating systems qualified as CSL-1, for one plant may be used by the other plants provided the applicable DBA requirements for the area where it is to be used are enveloped by the qualification testing system. These requirements include but are not limited to temperature, pressure, radiation and spray solution.

6.1.2.3 Ice Condenser Equipment

Material	Mass, lbs
Lower Door Seals	530
(Styrene butadiene)	
Equipment Access Door Seals	5
(Natural rubber)	
Vent curtain	5
(Laminated mylar)	
Ice Condenser Seal:	
Natural Rubber	600
Nylon	360
Miscellaneous Washers:	
Noryl SEIOO (phenylene oxide)	50
Gasketing Material:	
Neoprene	5060
Drain Line Expansion Joint	

6.1.2.4 Identification Tags

Material	Mass, lbs
Valves:	
ABS (acrylonitrile-butadiene-styrene)	50
Instruments:	
ABS (acrylonitrile-butadiene-styrene)	30

6.1.2.5 Valves and Instruments within Containment

Material	Mass, lbs
Diaphragms, O-Rings, Solenoid Seals:	
Buna-N (acrylonitrile-butadiene)	130

6.1.2.6 Heating and Ventilating Door Seals

Material	Mass, lbs
Neoprene (chloroprene)	100

6.1.2.7 Miscellaneous

Material	Mass, lbs
Catch Basins (Polyethylene)	100
Lead Shielding Blankets (Hypalon, Vinyl, Methylpolysiloxanes)	1200

6.1.3 Post-Accident Chemistry

Following a LOCA, the emergency core cooling solution recirculated in containment is composed of boric acid (H_3BO_3) from the reactor coolant, refueling water storage tank (RWST), cold leg accumulators and affected injection piping, lithium hydroxide (LiOH) from the reactor coolant and sodium tetraborate ($Na_2B_4O_7$) from the ice in the ice condenser.

6.1.3.1 Boric Acid, H_3BO_3

Boric acid up to a maximum concentration of 3300 ppm boron, can be found in the reactor coolant loop (4 loops, reactor vessel, pressurizer), and boric acid at a maximum concentration of 3300 ppm boron is found in the cold leg injection accumulators, the refueling water storage tank, and in associated piping. This limit may be exceeded during Mode 6 operation. These subsystems, when at maximum volume, represent a total mass of boric acid in the amount of 93,928 pounds.

6.1.3.2 Lithium Hydroxide

Lithium Hydroxide at a maximum concentration of 7.6 ppm lithium is found in the reactor coolant system for pH control.

6.1.3.3 Sodium Tetraborate

Sodium tetraborate is an additive in the ice stored in the ice condenser for the purpose of maintaining containment sump pH of at least 7.5 after all the ice has melted.

The minimum amount of ice assumed in the post-LOCA sump pH analysis is 2.26 x 10^6 lbm.[3] Boric acid and NaOH are formed during ice melt following a LOCA according to the following equation:

$$Na_2 B_4 O_7 + 7 H_2 O \rightarrow 2NaOH + 4 H_3 BO_3$$

6.1.3.4 Final Post-Accident Chemistry

The final post-accident sump pH is greater than 7.5. The estimated sump pH versus time calculation indicates that the post-LOCA sump pH remains within the allowable range of 7.5 to 10.0 for the duration of the event.

6.1.4 Degree of Compliance with Regulatory Guide 1.54 for Paints and Coatings Inside Containment

TVA is committed to adhere to Appendix B of 10 CFR 50 and ANSI N45.2 as required to produce a quality end product. Basically, it is TVA's position that the Quality Assurance Program (QA) for protective coatings inside the containment should control four activities in the coating program. The four major areas to be controlled are:

(1) The coating material itself, by extending requirements on the manufacturing process and qualification of coating systems through the use of applicable portions of ANSI Standards N101.2 and N512.

(2) The preparation of the surface to which coatings are to be applied.

(3) The inspection process.

(4) The application of the coating systems.

All four of these controlled activities have appropriate documentation and records to meet Appendix B requirements.

TVA agrees with Regulatory Guide 1.54, except the endorsement to ANSI N101.4 in paragraph C.1.

TVA's protective coating application program within the containment is in conformance with Appendix B to 10 CFR 50 and ANSI N45.2. In addition, applicable provisions found in ANSI N101.4 have been incorporated into TVA surface preparation, coating application/inspection specifications, and coating QA procedures.

Unqualified/uncontrolled coatings are accounted for and maintained within the limits specified in the analysis for containment coatings and in the transport analysis for the zone of influence. The zone of influence is defined as that area at the water surface into which a falling paint particle does not settle to the bottom, but rather, is transported to the sump strainer assembly by the flow of water.

REFERENCES

1. WCAP-7803, "Behavior of Austenitic Stainless Steel in Post Hypothetical Loss of Coolant Environment."

2. WCAP-7825, "Evaluation of Protective Coatings for Use in Reactor Containment."

3. WCAP-15699, “Tennessee Valley Authority Watts Bar Nuclear Plant Unit 1 Containment Integrity Analyses for Ice Weight Optimization Engineering Report,” Revision 1, dated August 2001.

4. TVA’s letter to NRC dated July 30, 2003, Petition Pursuant to 10 CFR 2.206 – Reactor Coolant System Stainless Steel Cladding.

5. WBT-D-1413, “Response to Comments on FSAR Mark-ups,” dated January 7, 2010.

6.2 CONTAINMENT SYSTEMS

6.2.1 Containment Functional Design - Unit 1

6.2.1.1 Design Bases

6.2.1.1.1 Primary Containment Design Bases

The containment is designed to assure that an acceptable upper limit of leakage of radioactive material is not exceeded under design basis accident conditions. For purposes of integrity, the containment may be considered as the containment vessel and containment isolation system. This structure and system are directly relied upon to maintain containment integrity. The emergency gas treatment system and Reactor Building function to keep out-leakage minimal (the Reactor Building also serves as a protective structure), but are not factors in determining the design leak rate.

The containment is specifically designed to meet the intent of the applicable General Design Criteria listed in Section 3.1. This section, Chapter 3, and other portions of Chapter 6 present information showing conformance of design of the containment and related systems to these criteria.

The ice condenser is designed to limit the containment pressure below the design pressure for all reactor coolant pipe break sizes up to and including a double-ended severance. Characterizing the performance of the ice condenser requires consideration of the rate of addition of mass and energy to the containment as well as the total amounts of mass and energy added. Analyses have shown that the accident which produces the highest blowdown rate into a condenser containment will result in the maximum containment pressure rise; that accident is the double-ended guillotine or split severance of a reactor coolant pipe. The design basis accident for containment analysis based on sensitivity studies is therefore the double-ended guillotine severance of a reactor coolant pipe at the reactor coolant pump suction. Post-blowdown energy releases can also be accommodated without exceeding containment design pressure.

The functional design of the containment is based upon the following accident input source term assumptions and conditions:

1. The design basis blowdown energy of 353.9×10^6 Btu and mass of 586.6×10^3 lb put into the containment (See Section 6.2.1.3.6).

2. A core power of 3459 MWt (plus 0.6% allowance for calorimetric error) (See Section 6.2.1.3.6).

3. The minimum engineered safety features are (i.e., the single failure criterion applied to each safety system) comprised of the following:

   a. The ice condenser which condenses steam generated during a LOCA, thereby limiting the pressure peak inside the containment (see Section 6.7).

   b. The containment isolation system which closes those fluid penetrations not serving accident-consequence limiting purposes (see Section 6.2.4).

   c. The containment spray system which sprays cool water into the containment atmosphere, thereby limiting the pressure peak (particularly in the long term - see Section 6.2.2).

   d. The emergency gas treatment system (EGTS) which produces a slightly negative pressure within the annulus, thereby precluding out-leakage and relieving the post-accident thermal expansion of air in the annulus (see Section 6.5.1).

   e. The air return fans which return air to the lower compartment (See Section 6.8).

Consideration is given to subcompartment differential pressure resulting from a design basis accident discussed in Sections 3.8.3.3, 6.2.1.3.9, and 6.2.1.3.4. If a design basis accident were to occur due to a pipe rupture in these relatively small volumes, the pressure would build up at a faster rate than in the containment, thus imposing a differential pressure across the wall of these structures.

Parameters affecting the assumed capability for post-accident pressure reduction are discussed in Section 6.2.1.3.3.

Three events that may result in an external pressure on the containment vessel have been considered:

1. Rupture of a process pipe where it passes through the annulus.

2. Inadvertent air return fan operation during normal operation.

3. Inadvertent containment spray system initiation during normal operation.

The design of the guard pipe portion of hot penetrations is such that any process pipe leakage in the annulus is returned to the containment. All process piping which has potential for annulus pressurization upon rupture is routed through hot penetrations. Section 6.2.4 discusses hot penetrations.

Inadvertent air return fan operation during normal operation opens the ice condenser lower inlet doors, which in turn, results in sounding an alarm in the MCR.

The logic and control circuits of the containment spray system are such that inadvertent containment spray would not take place with a single failure. The spray pump must start and the isolation valve must open before there can be any spray. In addition, the Watts Bar containment is so designed that even if an inadvertent spray occurs, containment integrity is preserved without the use of a vacuum relief.

The containment spray system is automatically actuated by a hi-hi containment pressure signal from the solid state protection system (SSPS). To prevent inadvertent automatic actuation, four comparator outputs, one from each protection set are processed through two coincidence gates. Both coincidence gates are required to have at least two high inputs before the output relays, which actuate the containment spray system, are energized. Separate output relays are provided for the pump start logic and discharge valve open logic. Additional protection is provided by an interlock between the pump and discharge valve, which requires the pump to be running before the discharge valve will automatically open.

Section 3.8.2 describes the structural design of the containment vessel. The containment vessel is designed to withstand a net external pressure of 2.0 psi. The containment vessel is designed to withstand the maximum expected net external pressure in accordance with ASME Boiler and Pressure and Vessel Code Section III, paragraph NE-7116.

6.2.1.2 Primary Containment System Design

The containment consists of a containment vessel and a separate Shield Building enclosing an annulus. The containment vessel is a freestanding, welded steel structure with a vertical cylinder, hemispherical dome, and a flat circular base. The Shield Building is a reinforced concrete structure similar in shape to the containment vessel. The design of these structures is described in Section 3.8.

The design internal pressure for the containment is 13.5 psig, and the design temperature is 250°F. The design basis leakage rate is 0.25 weight percent/24 hr. The design methods to assure integrity of the containment internal structures and sub-compartments from accident pressure pulses are described in Section 3.8.

6.2.1.3 Design Evaluation

6.2.1.3.1 Primary Containment Evaluation

1. The leaktightness aspect of the secondary containment is discussed in Section 6.2.3. The primary containment's leaktightness does not depend on the operation of any continuous monitoring or compressor system. The leak testing of the primary containment and its isolation system is discussed in Section 6.2.6.

2. The acceptance criteria for the leaktightness of the primary containment are such that at containment design pressure, there is a 25% margin between the acceptable maximum leakage rate and the maximum permissible leakage rate.

6.2.1.3.2 General Description of Containment Pressure Analysis

The time history of conditions within an ice condenser containment during a postulated loss of coolant accident can be divided into two periods for calculation purposes:

1. The initial reactor coolant blowdown, which for the largest assumed pipe break occurs in approximately 10 seconds.

2. The post blowdown phase of the accident which begins following the blowdown and extends several hours after the start of the accident.

During the first few seconds of the blowdown period of the reactor coolant system, containment conditions are characterized by rapid pressure and temperature transients. It is during this period that the peak transient pressures, differential pressures, temperature and blowdown loads occur. To calculate these transients a detailed spatial and short time increment analysis was necessary. This analysis was performed with the Transient Mass Distribution (TMD) computer code with the calculation time of interest extending up to a few seconds following the accident initiation (See Section 6.2.1.3.4).

Physically, tests at the ice condenser Waltz Mill test facility have shown that the blowdown phase represents that period of time in which the lower compartment air and a portion of the ice condenser air are displaced and compressed into the upper compartment and the remainder of the ice condenser. The containment pressure at or near the end of blowdown is governed by this air compression process. The containment compression ratio calculation is described in Section 6.2.1.3.4.

Containment pressure during the post blowdown phase of the accident is calculated with the LOTIC code which models the containment structural heat sinks and containment safeguards systems.

6.2.1.3.3 Long-Term Containment Pressure Analysis

Early in the ice condenser development program it was recognized that there was a need for modeling of long-term ice condenser containment performance. It was realized that the model would have to have capabilities comparable to those of the dry containment (COCO) model. These capabilities would permit the model to be used to solve problems of containment design and optimize the containment and safeguards systems. This has been accomplished in the development of the LOTIC code[1].

The model of the containment consists of five distinct control volumes; the upper compartment, the lower compartment, the portion of the ice bed from which the ice has melted, the portion of the ice bed containing unmelted ice, and the dead ended compartments. The ice condenser control volume with unmelted ice is further subdivided into six subcompartments to allow for maldistribution of break flow to the ice bed.

The conditions in these compartments are obtained as a function of time by the use of fundamental equations solved through numerical techniques. These equations are solved for three distinct phases in time. Each phase corresponds to a distinct physical characteristic of the problem. Each of these phases has a unique set of simplifying assumptions based on test results from the ice condenser test facility. These phases are the blowdown period, the depressurization period, and the long term.

The most significant simplification of the problem is the assumption that the total pressure in the containment is uniform. This assumption is justified by the fact that after the initial blowdown of the reactor coolant system, the remaining mass and energy released from this system into the containment are small and very slowly changing. The resulting flow rates between the control volumes will also be relatively small. These small flow rates then are unable to maintain significant pressure differences between the compartments.

In the control volumes, which are always assumed to be saturated, steam and air are assumed to be uniformly mixed and at the control volume temperature. The air is considered a perfect gas, and the thermodynamic properties of steam are taken from the ASME steam table.

For the purpose of calculation, the condensation of steam is assumed to take place in a condensing node located between the two control volumes in the ice storage compartment.

Containment Pressure Calculation

The following are the major input assumptions used in the LOTIC analysis for the pump suction pipe rupture case with the steam generators considered as an active heat source for the Watts Bar Nuclear Plant containment:

1. Minimum safeguards are employed in all calculations, e.g., one of two spray pumps and one of two spray heat exchangers; one of two RHR pumps and one of two RHR heat exchangers providing flow to the core; one of two safety injection pumps and one of two centrifugal charging pumps; and one of two air return fans.

2. 2.26×10^6 lbs. of ice initially in the ice condenser which is at 15°F. (This is less than the Technical Specification limit.)

3. The blowdown, reflood, and post reflood mass and energy releases described in Section 6.2.1.3.6 were used.

4. Blowdown and post-blowdown ice condenser drain temperatures of 190°F and 130°F are used.[5]

5. Nitrogen from the accumulators in the amount of 2973.5 lbs. included in the calculations.

6. Essential raw cooling water temperature of 88°F is used on the spray heat exchanger and the component cooling heat exchanger.

7. The air return fan is effective 10 minutes after the transient is initiated. The actual air return fan initiation can take place in 9 ± 1 minutes, with initiation as early as 8 minutes not adversely affecting the analysis results.

8. No maldistribution of steam flow to the ice bed is assumed.

9. No ice condenser bypass is assumed. (This assumption depletes the ice in the shortest time and is thus conservative.)

10. The initial conditions in the containment are a temperature of 100°F in the lower and dead-ended volumes, 80°F in the upper volume, and 27°F in the ice condenser. (Note: The 80°F temperature in the upper compartment is a reduction from the 85°F lower technical specification limit to account for the upper plenum volume of the ice condenser which is included in upper compartment volume for the analysis. The volume is adjusted to maximize air mass and the compression ratio.) All volumes are at a pressure of 0.3 psig and a 10-percent relative humidity, except the ice condenser which is at 100-percent relative humidity.

11. A containment spray pump flow of 4000 gpm is used in the upper compartment. The analyzed diesel loading sequence for the containment sprays to energize and come up to full flow and head in 234 seconds is tabulated in Table 6.2.1-25. It is also noted that the calculated CSS flow rate is 4550 gpm, which bounds the 4000 gpm flow rate used in the analysis.

12. Though the capability to divert flow from the RHR system to a spray header in the upper compartment exists, this capability was not modeled in this containment integrity analysis. (Unit 1 only)

13. Containment structural heat sink data is found in Table 6.2.1-1. (Note that the dead-ended compartment structural heat sinks were conservatively neglected.)

14. The operation of one containment spray heat exchanger (UA = 2.44 x 10^6 Btu/hr-°F incorporating a 10% tube plugging margin) for containment cooling and the operation of one RHR heat exchanger (UA = 1.496 x 10^6 Btu/hr-°F) for core cooling. The component cooling heat exchanger UA was modeled at 3.17 x 10^6 Btu/hr-°F.

15. The air return fan returns air at a rate of 40,000 cfm from the upper to lower compartment.

16. An active sump volume of 51000 ft^3 is used.

17. The pump flowrates vs. time given in Table 6.2.1-2 were used to calculate conservative RWST draindown times. (These flow values reflect ECCS pumps at runout against the design containment pressure, using the minimum composite pump curves shown in Figures 6.3-2, 6.3-3, and 6.3-4, which are degraded by 5% and bound what is achievable in the plant. Switchover times from injection to recirculation that are achievable in the plant for each ECCS pump are also conservative in the analysis.)

18. A core power of 3459 MWt (plus 0.6% allowance for calorimetric error). [See Section 6.2.1.3.6.]

19. Hydrogen gas was added to the containment in the amount of 25,230.2 Standard Cubic Feet (SCF) over 24 hours. Sources accounted for were radiolysis in the core and sump post-LOCA, corrosion of plant materials (Aluminum, Zinc, and painted surfaces found in containment), reaction of 1% of the Zirconium fuel rod cladding in the core, and hydrogen gas assumed to be dissolved in the reactor coolant system water. (This bounds tritium producing core designs).

20. The containment compartment volumes were based on the following: upper compartment 645,818 ft^3; lower compartment 221,074 ft^3; and dead-ended compartment 146,600 ft^3. (Note: These volumes represent TMD volumes. For Containment Integrity Analysis, the volumes are adjusted to maximize air mass and the compression ratio.)

21. Subcooling of emergency core cooling (ECC) water from the RHR heat exchanger is assumed.

22. Essential raw cooling water flow to the containment spray heat exchanger was modeled as 5,200 gpm. Also the essential raw cooling water flow to the component cooling heat exchanger was modeled as 3,500 gpm.

23. ANS 5.1-1979 + 2sigma decay heat was modeled in the mass and energy release analysis. After moderator feedback effects dissipate during blowdown, the UFSAR subsection 6.2.1.3.6 curve was used.

With these assumptions, the heat removal capability of the containment is sufficient to absorb the energy releases and still keep the maximum calculated pressure well below design.

The following plots are provided:

Figure 6.2.1-1, Containment Pressure Transient,

Figure 6.2.1-2a, Upper Compartment Temperature Transients,

Figure 6.2.1-2b, Lower Compartment Temperature Transient

Figure 6.2.1-3, Active and Inactive Sump Temperature Transient,

Figure 6.2.1-4, Ice Melt Transient.

Figure 6.2.1-4a, Comparison of Containment Pressure versus Ice Melt Transients

Tables 6.2.1-3 and 6.2.1-4 give energy accountings at various points in the transient.

As can be seen from Figure 6.2.1-1 the maximum calculated Containment pressure is 9.36 psig, occurring at approximately 8,959 seconds. The transient shown does account for hydrogen partial pressure as a result of the post DBA LOCA hydrogen production discussed in Sections 6.2.5 and 15.4.1.2. Accumulation of hydrogen prior to recombiner operation can account for approximately 0.25 psig at the time of containment peak pressure.

Structural Heat Removal

Provision is made in the containment pressure analysis for heat storage in interior and exterior walls. Each wall is divided into a number of nodes. For each node, a conservation of energy equation expressed in finite difference forms accounts for transient conduction into and out of the node and temperature rise of the node. Table 6.2.1-1 is a summary of the containment structural heat sinks used in the analysis. The material property data used is found in Table 6.2.1-5.

The heat transfer coefficient to the containment structures is based primarily on the work of Tagami.[21] An explanation of the manner of application is given in Reference [3].

When applying the Tagami correlations a conservative limit was placed on the lower compartment stagnant heat transfer coefficients. They were limited to 72 Btu/hr-ft^2. This corresponds to a steam-air ratio of 1.4 according to the Tagami correlation. The imposition of this limitation is to restrict the use of the Tagami correlation within the test range of steam-air ratios where the correlation was derived.

6.2.1.3.4 Short-Term Blowdown Analysis

TMD Code - Short-Term Analysis

1. Introduction

The basic performance of the ice condenser reactor containment system has been demonstrated for a wide range of conditions by the Waltz Mill Ice Condenser Test Program. These results have clearly shown the capability and reliability of the ice condenser concept to limit the Containment pressure rise subsequent to a hypothetical loss-of-coolant accident.

To supplement this experimental proof of performance, a mathematical model has been developed to simulate the ice condenser pressure transients. This model, encoded as computer program TMD (Transient Mass Distribution), provides a means for computing pressures, temperatures, heat transfer rates, and mass flow rates as a function of time and location throughout the containment. This model is used to compute pressure differences on various structures within the containment as well as the distribution of steam flow as the air is displaced from the lower compartment. Although the TMD code can calculate the entire blowdown transient, the peak pressure differences on various structures occur within the first few seconds of the transient.

2. Analytical Models (No Entrainment)

The mathematical modeling in TMD is similar to that of the SATAN blowdown code in that the analytical solution is developed by considering the conservation equations of mass, momentum and energy and the equation of state, together with the control volume technique for simulating spatial variation. The governing equations for TMD are given in Reference [4].

The moisture entrainment modifications to the TMD code are discussed, in detail, in Reference [4]. These modifications comprise incorporating the additional entrainment effects into the momentum and energy equations.

As part of the review of the TMD code, additional effects are considered. Changes to the analytical model required for these studies are described in Reference [4].

These studies consist of:

a. Spatial acceleration effects in ice bed
b. Liquid entrainment in ice beds
c. Upper limit on sonic velocity
d. Variable ice bed loss coefficient
e. Variable door response
f. Wave propagation effects

Additionally the TMD code has been modified to account for fluid compressibility effects in the high Mach number subsonic flow regime.

<u>Experimental Verification</u>

The performance of the TMD code was verified against the 1/24 scale air tests and the 1968 Waltz Mill tests. For the 1/24 scale model the TMD code was utilized to calculate flow rates to compare against experimental results. The effect of increased nodalization was also evaluated. The Waltz Mill test comparisons involved a reexamination of test data. In conducting the reanalyses, representation of the 1968 Waltz Mill test was reviewed with regard to parameters such as loss coefficients and blowdown time history. The details of this information are given in Reference [4].

The Waltz Mill Ice Condenser Blowdown Test Facility was reactivated in 1973 to verify the ice condenser performance with the following redesigned plant hardware scaled to the test configuration:

1. Perforated metal ice baskets and new design couplings.

2. Lattice frames sized to provide the correct loss coefficient relative to plant design.

3. Lower support beamed structure and turning vanes sized to provide the correct turning loss relative to the plant design.

4. No ice baskets in the lower ice condenser plenum opposite the inlet doors.

The result of these tests was to confirm that conclusions derived from previous Waltz Mill tests have not been significantly changed by the redesign of plant hardware. The TMD Code has, as a result of the 1973 test series, been modified to match ice bed heat transfer performance. Detailed information on the 1973 Waltz Mill test series is found in Reference [5].

<u>Application to Plant Design (General Description)</u>

As described in Reference [4], the control volume technique is used to spatially represent the containment. The containment is divided into 50 elements to give a detailed representation of the local pressure transient on the containment shell and internal concrete structures. This division of the containment is similar for all ice condenser plants.

The Watts Bar plant containment has been divided into 50 elements or compartments as shown in Figures 6.2.1-5, 6.2.1-6, 6.2.1-7, and 6.2.1-8. The interconnections between containment elements in the TMD code is shown schematically in Figure 6.2.1-9. Flow resistance and inertia are lumped together in the flow paths connecting the elements shown. The division of the lower compartments into 6 volumes occurs at the points of greatest flow resistance, i.e., the four steam generators, pressurizer and refueling cavity.

Each of these lower compartment sections delivers flow through doors into a section behind the doors and below the ice bed. Each vertical section of the ice bed is, in turn, divided into three elements. The upper plenum between the top of the ice bed and the upper doors is represented by an element. Thus, a total of thirty elements (Elements 7 through 24 and 38 through 49 are used to simulate the ice condenser). The six elements at the top of the ice bed between bed and upper doors deliver to element number 25 the upper compartment. Note that cross flow in the ice bed is not accounted for in the analysis; this yields the most conservative results for the particular calculations described herein. The upper reactor cavity (Element 33) is connected to the lower compartment volumes and provides cross flow for pressure equalization of the lower compartments. The less active compartments, called dead-ended compartments (Elements 26 through 32 and 34 through 37) outside the crane wall are pressurized by ventilation openings through the crane wall into the fan compartments.

For each element in the TMD network the volume, initial pressure and initial temperature conditions are specified. The ice condenser elements have additional inputs of mass of ice, heat transfer area and condensate layer length. For each flow path between elements flow resistance is specified as a loss coefficient "K" or a fraction loss "L/D" or a combination of the two based on the flow area specified between elements. Friction factor, friction factor length and hydraulic diameter are specified for the friction loss.

Additionally, input for each flow path includes the area ratio (minimum area/maximum area) which is used to account for compressibility effects across flow path contractions. The code input for each flow path is the flow path length used in the momentum equation. The ice condenser loss coefficients have been based on the ¼-scale tests representative of the current ice condenser geometry. The test loss coefficient was increased to include basket roughness effects and to include intermediate and top deck pressure losses. The loss coefficient is based on removal of door port flow restrictors.

To better represent short term transients effects, the opening characteristics of the lower, intermediate, and top deck ice condenser doors have been modeled in the TMD code. The containment geometric data for the elements and flow paths used in the TMD code is confirmed to agree with the actual design by TVA and Westinghouse. An initial containment pressure of 0.3 psig was assumed in the analysis. Initial containment pressure variation about the assumed 0.3 psig value has only a slight affect on the initial pressure peak and the compression ratio pressure peak. TMD input data is given in Tables 6.2.1-6 and 6.2.1-7.

The reactor coolant blowdown rates used in these cases are based on the SATAN analysis of a double-ended rupture of either a hot or a cold leg reactor coolant pipe utilizing a discharge coefficient of 1.0. The models and assumptions used to calculate the short-term mass and energy releases are described in Reference [9]. Tables 6.2.1-23 and 6.2.1-24 present the mass and energy release data used for this analysis.

A number of analyses have been performed to determine the various pressure transients resulting from hot and cold leg reactor coolant pipe breaks in any one of the six lower compartment elements. The analyses were performed using the following assumptions and correlations:

1. Flow was limited by the unaugmented critical flow correlation.

2. The TMD variable volume door model, which accounts for changes in the volumes of TMD elements as the door opens, was implemented.

3. The heat transfer calculation used was based on performance during the 1973-1974 Waltz Mill test series. A higher value of the ELJAC parameter has been used and an upper bound on calculated heat transfer coefficients has been imposed[5].

4. One hundred percent moisture entrainment was assumed.

5. Compressibility effects due to flow area contractions were modeled.

6. A 15% reduction in flow area through the ice condenser was assumed to account for the effect of the flow area reduction due to frost and ice accumulation during the course of an operating cycle.

Figures 6.2.1-10 and 6.2.1-11 are representative of the typical upper and lower compartment pressure transients that result from a hypothetical double-ended rupture of a reactor coolant pipe for the worst possible location in the lower compartment of the containment; i.e., hot leg and cold leg breaks in Element 1.

Initial Pressures

Results of the analysis for the Watts Bar Plant are presented in Tables 6.2.1-8 through 6.2.1-11. The peak pressures and peak differential pressures resulting from hot and cold leg reactor coolant pipe breaks in each of the six lower compartment control volumes were calculated.

Table 6.2.1-8 presents the maximum calculated pressure peak for the lower compartment elements resulting from hot and cold leg double ended pipe breaks. Generally, the maximum peak pressure within a lower compartment element results when the pipe break occurs in that element. A hot leg break in Element 1 creates the highest pressure peak, also in Element 1, of 16.7 psig.

Table 6.2.1-9 presents the maximum calculated peak pressure in each of the ice condenser sections resulting from any pipe break location. The maximum peak pressure in each of the ice condenser sections is found in the lower plenum element of the section. The peak pressure was calculated to be 12.1 psig in Element 40.

Table 6.2.1-10 presents the maximum calculated differential pressures across the operating deck (divider barrier) between the lower compartment elements and the upper compartment. These values are approximately the same as the maximum calculated differential pressure across the lower crane wall between the lower compartment elements and the dead ended volumes surrounding the lower compartment. The peak differential pressure of 16.3 psi was calculated to be between Elements 1 and 25 for a hot leg break.

Table 6.2.1-11 presents the maximum calculated differential pressures across the upper crane wall between the upper ice condenser elements and the upper compartment. The peak differential of 8.8 psi pressure was calculated to be between Element 7-8-9 and 25 for a hot leg pipe break.

An evaluation of this subcompartment was conducted as part of the 1.4% Uprate Program. The previously discussed mass and energy releases and the corresponding subcompartment pressurization analysis were determined to remain bounding.

Consideration is given to the calculation of subcompartment pressures (and pressure differentials) for cases other than the design basis double ended reactor coolant pipe rupture in the lower compartment. Discussion of these analyses is treated in Section 6.2.1.3.9.

Sensitivity Studies

A series of TMD runs for D. C. Cook investigated the sensitivity of peak pressures to variations in individual input parameters for the design basis blowdown rate and 100 percent entrainment. This analysis used a DEHL break in Element 6 of D. C. Cook. Table 6.2.1-12 presents the results of this sensitivity study.

As part of the short-term containment pressure analysis of ice condenser units, the pressure response to both DEHL and DECL breaks are routinely considered for each of the loop compartments.

Choked Flow Characteristics

The data in Figure 6.2.1-12 illustrate the behavior of mass flow rate as a function of upstream and downstream pressures, including the effects of flow choking. The upper plot shows mass flow rate as a function of upstream pressure for various assumed values of downstream pressure. For zero back pressure (P_d = 0), the entire curve represents choked flow conditions with the flow rate approximately proportional to upstream pressure P_u. For higher back pressure, the flow rates are lower until the upstream pressure is high enough to provide choked flow. After the increase in upstream pressure is sufficient to provide flow chokings further increases in upstream pressure cause increases in mass flow rate along the curve for P_d = 0. The key point in this illustration is that flow rate continues to increase with increasing upstream pressure, even after flow choking conditions have been reached. Thus, choking does not represent a threshold beyond which dramatically sharper increases in compartment pressures could be expected because of limitations on flow relief to adjacent compartments.

The phenomenon of flow choking is more frequently explained by assuming a fixed upstream pressure and examining the dependence of flow rate with respect to decreasing downstream pressure. This approach is illustrated for an assumed upstream pressure of 30 psia as shown in the upper plot with the results plotted vs. downstream pressure in the lower plot. For fixed upstream conditions, flow choking represents an upper limit flow rate beyond which further decreases in back pressure do not produce any increase in mass flow rate.

Compression Ratio Analysis

As blowdown continues following the initial pressure peak from a double-ended cold leg break, the pressure in the lower compartment again increases, reaching a peak at or before the end of blowdown. The pressure in the upper compartment continues to rise from beginning of blowdown and reaches a peak which is approximately equal to the lower compartment pressure. After blowdown is complete, the steam in the lower compartment continues to flow through the doors into the ice bed compartment and is condensed.

The primary factor in producing this upper containment pressure peak and, therefore, in determining design pressure, is the displacement of air from the lower compartment into the upper containment. The ice condenser quite effectively performs its function of condensing virtually all the steam that enters the ice beds. Essentially, the only source of steam entering the upper containment is from leakage through the drain holes and other leakage around crack openings in hatches in the operating deck separating the lower and upper portions of the containment building.

A method of analysis of the compression peak pressure was developed based on the results of full-scale section tests. This method consists of the calculation of the air mass compression ratio, the polytropic exponent for the compression process, and the effect of steam bypass through the operating deck on this compression.

The compression peak pressure in the upper containment for the Watts Bar plant design is calculated to be 7.81 psig (for an initial air pressure of 0.3 psig). This compression pressure includes the effect of a pressure increase of 0.4 psi from steam bypass and also for the effects of the dead-ended volumes. The nitrogen partial pressure from the accumulators is not included since this nitrogen is not added to the containment until after the compression peak pressure has been reduced, which is after blowdown is completed. This nitrogen is considered in the analysis of pressure decay following blowdown as presented in the long term performance analysis using the LOTIC code. The following sections discuss the major parameters affecting the compression peak. Specifically they are: air compression, steam bypass, blowdown rate, and blowdown energy.

Air Compression Process Description

The volumes of the various containment compartments determine directly the air volume compression ratio. This is basically the ratio of the total active containment air volume to the compressed air volume during blowdown. During blowdown air is displaced from the lower compartment and compressed into the ice condenser beds and into the upper containment above the operating deck. It is this air compression process which primarily determines the peak in containment pressure, following the initial blowdown release. A peak compression pressure of 7.81 psig is based on the Watts Bar Plant design compartment volumes shown in Table 6.2.1-13.

Figure 6.2.1-13 shows the sensitivity of the compression peak pressure with different air compression ratios.

Methods of Calculation and Results

Full-Scale Section Tests

The actual Waltz Mill test compression ratios were found by performing air mass balances before the blowdown and at the time of the compression peak pressure, using the results of three full-scale special section tests. These three tests were conducted with an energy input representative of the plant design.

In the calculation of the mass balance for the ice condenser, the compartment is divided into two sub-volumes; one volume representing the flow channels and one volume representing the ice baskets. The flow channel volume is further divided into four sub-volumes. The partial air pressure and mass in each sub-volume is found from thermocouple readings by assuming that the air is saturated with steam at the measured temperature. From these results, the average temperature of the air in the ice condenser compartment is found, and the volume occupied by the air at the total condenser pressure is found from the equation of state as follows:

$$V_{a2} = \frac{M_{a2} \; R_a \; T_{a2}}{P_2} \qquad (1)$$

where:

V_{a2} = Volume of ice condenser occupied by air (ft^3)

M_{a2} = Mass of air in ice condenser compartment (lb)

T_{a2} = Average temperature of air in ice condenser (°F)

P_2 = Total ice condenser pressure (lb/ft^2)

R_a = Ideal gas constant

The partial pressure and mass of air in the lower compartment are found by averaging the temperatures indicated by the thermocouples located in that compartment and assuming saturation conditions. For these three tests, it was found that the partial pressure, and hence the mass of air in the lower compartment, was zero at the time of the compression peak pressure.

The actual Waltz Mill test compression ratio is then found from the following:

$$C = \frac{V_1 + V_2 + V_3}{V_3 + V_{a2}} \qquad (2)$$

where:

V_1 = Lower compartment volume (ft^3)
V_2 = Ice condenser compartment volume (ft^3)
V_3 = Upper compartment volume (ft^3)

The polytropic exponent for these tests is then found from the measured compression pressure and the compression ratio calculated above. Also considered is the pressure increase that results from the leakage of steam through the deck into the upper compartment.

The compression peak pressure in the upper compartment for the tests or containment design is then given by:

$$P = P_o (C_r)^n + \Delta P_{deck} \qquad (3)$$

where:

P_O	= Initial pressure (psia)
P	= Compression peak pressure (psia)
C_r	= Volume compression ratio
n	= Polytropic exponent
ΔP_{deck}	= Pressure increase caused by deck leakage (psi)

Using the method of calculation described above, the compression ratio is calculated for the three full-scale section tests. From the results of the air mass balances, it was found that air occupied 0.645 of the ice condenser compartment volume at the time of peak compression, or

$$V_{a2} = 0.645 V_2 \qquad (4)$$

The final compression volume includes the volume of the upper compartment as well as part of the volume of air in the ice condenser. The results of the full-scale section tests (Figure 6.2.1-14) show a variation in steam partial pressure from 100% near the bottom of the ice condenser to essentially zero near the top. The thermocouples and pressure detectors confirm that at the time when the compression peak pressure is reached steam occupies less than half of the volume of the ice condenser. The analytical model used in defining the containment pressure peak uses upper compartment volume plus 64.5% of the ice condenser air volumes as the final volume. This 64.5% value was determined from appropriate test results.

The calculated volume compression ratios are shown in Figure 6.2.1-15, along with the compression peak pressures for these tests. The compression peak pressure is determined from the measured pressure, after accounting for the deck leakage contribution. From the results shown in Figure 6.2.1-15, the polytropic exponent for these tests is found to be 1.13.

Plant Case

For the Watts Bar design, the volume compression ratio is calculated using Equation 2, modeling the upper plenum as part of the upper compartment, and Table 6.2.1-13 as:

$$C_r = \frac{1,077,012}{692,818 + [0.645 \text{ x } 110,520]} \qquad (5)$$

C_r = 1.4095

The peak compression pressure, based on an initial containment pressure of 15.0 psia (0.3 psig), is then given by Equation 3 as:

$P_3 = 15.0\ (1.4095)^{1.13} + 0.4$

P_3 = 22.507 psia or 7.81 psig

This peak compression pressure includes a pressure increase of 0.4 psi from steam bypass through the deck (see Section 6.2.1.3.5).

Sensitivity to Blowdown Energy

The sensitivity of the upper and lower compartment peak pressure versus blowdown rate as measured from the 1974 Waltz Mill Tests is shown in Figure 6.2.1-16. This figure shows the magnitude of the peak pressure versus the amount of energy released in terms of percentage of RCS energy release rate.

Percent energy blowdown rate was selected for the plot because energy flow rate more directly relates to volume flow rate and therefore pressure. There are two important effects to note from the peak upper compartment pressure versus blowdown rate: (1) the magnitude of the final peak pressure in the upper compartment is low (about 9 psig) for the plant design DECL blowdown rate; (2) even an increase in this rate up to 141% of the blowdown energy rate produces only a small increase in the magnitude of this peak pressure (about 1 psi).

The major factor setting the peak pressure reached in the upper compartment is the compression of air displaced by steam from the lower compartment into the upper compartment. The lower compartment initial peak pressure shows a relatively low peak pressure of 12.9 psig for the design basis DECL blowdown rate, and even a substantial increase in blowdown energy rate (141% reference initial DECL) would cause an increase in initial peak pressure of only 3 psi. The peak pressure in the lower compartment is due mainly to flow resistance caused by displacement of air from the lower compartment into the upper compartment.

6.2.1.3.5 Effect of Steam Bypass

The sensitivity of the compression peak pressure to deck bypass is shown in Figure 6.2.1-17, which shows that an increase in deck bypass area of 50% would cause an increase of about 0.2 psi in final peak compression pressure. Also, it is important to note that the plant final peak compression pressure of 7.81 psig already includes a contribution of 0.4 psi from the plant deck bypass area of 5 ft^2.

This effect of deck leakage on upper containment pressure has been verified by a series of four special, full-scale section tests. These tests were all identical except different size deck leakage areas were used.

The results of these tests are given in Figure 6.2.1-18 which includes two curves of test results. Each curve shows the difference in upper compartment pressure between one test and another resulting from a difference in deck leakage area. One curve shows the increase in upper compartment pressure at the end of the boiler blowdown (after the compression peak pressure, at about 50 seconds in these tests), and the second curve shows the increase in upper compartment peak pressure (at about 10 seconds in these tests). It should be noted that the pressure at the end of the blowdown is less than the peak compression ratio pressure occurring at about 10 seconds for reference blowdown test.

The containment pressure increase due to deck leakage is directly proportional to the total amount of steam leakage into the upper compartment, and the amount of this steam leakage is, in turn, proportional to the amount of steam released from the boiler, less the inventory of steam remaining in the lower compartment. Notably, the increase in upper compartment compression peak pressure is substantially less than the upper compartment pressure increase at the end of blowdown, because the peak compression pressure occurs before the boiler has released all of its energy.

The calculated maximum pressure rise due to deck leakage (when all of the boiler energy release has occurred) is also shown in Figure 6.2.1-18. The slope of this curve is 0.095 psi/ft^2 for the tests and is equivalent to 0.107 psi/ft^2 for the plant design. The difference between the two coefficients is due to a small difference in upper compartment volume between the plant design and these tests.

As shown in Figure 6.2.1-18, the calculated curve for maximum pressure increase at the end of blowdown agrees closely with the measured curve at small deck leakage areas but deviates at larger leakage areas. This deviation apparently results from the condensation of upper compartment steam by the walls of the upper compartment and by the ice at the top of the condenser during the tests. Pressure would also be reduced by heat losses in a plant; however, for conservatism, no credit is taken for this effect. As demonstrated by tests, the compression peak pressure in the upper compartment occurs before the boiler releases all of its energy, and the measured increase in peak compression pressure due to increased deck leakage, is proportionately reduced. For the case of the plant design, the final peak compression pressure is conservatively assumed to occur when the reactor coolant system release is 75% of its total energy. This value is selected as a reference value, based on the results of a number of tests conducted with different blowdown rates and total energy releases, as shown in Figure 6.2.1-19. The actual deck leakage coefficient is therefore:

$$\frac{\Delta P_3}{A_{deck}} = 0.107 \text{ x } 0.75 = 0.080 \text{ psi}/\text{ft}^2$$

The divider barrier including the enclosures over the pressurizer, steam generators and reactor vessel, is designed to provide a reasonably tight seal against leakage. Holes are purposely provided in the bottom of the refueling cavity to allow water from sprays in the upper compartment to drain to the sump in the lower compartment. Potential leakage paths exist at all the joints between the operating deck and the pump access hatches and reactor vessel enclosure slabs. The total of all deck leakage flow areas is approximately 5 ft^2. The effect of this potential leakage path is small and is found to be:

$$\Delta P_{deck} = 5 \text{ x } 0.080 = 0.4 \text{ psi}$$

In the event that the reactor coolant system break flow is so small that it would leak through these flow paths without developing sufficient differential pressure (1 lb/ft^2) to open the ice condenser doors, steam from the break would slowly pressurize the containment. The containment spray system has sufficient capacity to maintain pressure well below design for this case.

The Watts Bar Nuclear Plant and the Sequoyah Nuclear Plant are geometrically very similar. Some differences between the two plants are the design pressure, spray flow rates, and a slight difference in thermal ratings. The fact that the spray flow rate is higher for the Sequoyah plant (4750 gpm versus 4000 gpm) is offset by Watts Bar's higher maximum internal pressure (15 psig versus 12 psig). The following discussion presents the deck leakage analysis performed for the Sequoyah plant. The purpose of this analysis is only to show the substantial margin which exists between the design deck leakage of 5 ft^2 and the tolerable deck leakage. The Sequoyah analysis which shows conservatism by a factor of 7, is more than sufficient for this purpose.

The method of analysis used to obtain the maximum allowable deck leakage capacity as a function of the primary system break size is as follows.

During the blowdown transient, steam and air flow through the ice condenser doors and also through the deck bypass area into the upper compartment. For the containment, this bypass, area is composed of two parts, a known leakage area of 2.2 ft^2 with a geometric loss coefficient of 1.5 through the deck drainage holes location at the bottom of the refueling canal and an undefined deck leakage area with a conservatively small loss coefficient of 2.5.

A resistance network similar to that used to TMD is used to represent 6 lower compartment volumes each with a representative portion of the deck leakage, and the lower inlet door flow resistance and flow area is calculated for small breaks that would only partially open these doors. The coolant blowdown rate as a function of time is used with this flow network to calculate the differential pressures on the lower inlet doors and across the operating deck.

The resultant deck leakage rate and integrated steam leakage into the upper compartment is then calculated. The lower inlet doors are initially held shut by the cold head of air behind the doors (approximately one pound per square foot). The initial blowdown from a small break opens the doors and removes the cold head on the doors. With the door differential removed, the door position is slightly open. An additional pressure differential of one pound per square foot is then sufficient to fully open the doors. The nominal door opening characteristics are based on test results.

One analysis conservatively assumed that flow through the postulated leakage paths is pure steam. During the actual blowdown transient, steam and air representative of the lower compartment mixture leak through the holes, thus less steam would enter the upper compartment. If flow were considered to be a mixture of liquid and vapor, the total leakage mass would increase, but the steam flow rate would decrease. The analysis also assumed that no condensing of the flow occurs due to structural heat sinks. The peak air compression in the upper compartment for the various break sizes is assumed with steam mass added to this value to obtain the total containment pressure. Air compression for the various break sizes is obtained from previous full-scale section tests conducted at Waltz Mill.

The allowable leakage area for the following reactor coolant system (RCS) break sizes was determined: DE, 0.6 DE, 3 ft^2, 10 inch diameter, 6 inch diameter, 2.5 inch diameter, and 0.5 inch diameter. The allowable deck leakage area for the DE break was based on the test results previously discussed. For break sizes of 3 ft^2 and 0.6 DE, a series of deck leakage sensitivity studies were made to establish the total steam leakage to the upper compartment over the blowdown transient. This steam was added to the peak compression air mass in the upper compartment to calculate a peak pressure.

Air and steam were assumed to be in thermal equilibrium, with the air partial pressure increased over the air compression value to account for heating effects. For these breaks, sprays were neglected. Reduction in compression ratio by return of air to the lower compartment was conservatively neglected. The results of this analysis are shown in Table 6.2.1-14. This analysis is confirmed by Waltz Mill tests conducted with various deck leaks equivalent to over 50 ft^2 feet of deck leakage for the double-ended blowdown rate and is shown in Figure 6.2.1-20.

For breaks of 10 inch diameter and smaller, the effect of containment sprays was included. The method used calculates, for each time step of the blowdown, the amount of steam leaking into the upper compartment to obtain the steam mass in the upper compartment. This steam was mixed with the air in the upper compartment, assuming thermal equilibrium with air. The air partial pressure was increased to account for air heating effects. After sprays were initiated, the pressure was calculated based on the rate of accumulation of steam in the upper compartment.

This analysis was conducted for the 10 inch, 6 inch, and 2 inch break sizes, assuming one spray pump operated (4750 gpm at 100°F). As shown in Table 6.2.1-14, the 10-inch break is the limiting case for the given range of break sizes.

A second, more realistic, method was used to analyze the 10-inch, 6-inch, and 2-inch breaks. This analysis assumed a 30% air and 70% steam mix flowing through the deck leakage area. This is conservative considering the amount of air in the lower compartment during this portion of the transient. Operation of the deck fan increases the air content of the lower compartment, thus increasing the allowable deck leakage area. Based on the LOTIC code analysis, a structural heat removal rate of over 6000 Btu/sec from the upper compartment is indicated. Therefore, a steam condensation rate of 6 lbs/sec was used for the upper compartment. The results indicate that with one spray pump operating and a deck leakage area of 50 ft^2, the peak containment pressure is below design pressure.

The ½-inch diameter break is not sufficient to open the ice condenser inlet doors. For this break, the upper compartment spray is sufficient to condense the break steam flow.

In conclusion, it is apparent that there is a substantial margin between the design deck leakage area of 5 ft^2 and that which can be tolerated without exceeding containment design pressure. A preoperational visual inspection has been performed to ensure that the seals between the upper and lower containment have been properly installed.

6.2.1.3.6 Mass and Energy Release Data

Long-Term Loss-of-Coolant Accident Mass and Energy Releases

The evaluation model used for the long-term LOCA mass and energy release calculations is the WCOBRA/TRAC mass and energy release model described in Reference [20]. This evaluation model has been reviewed and the methodology has received the final safety evaluation report from the NRC in August 2015.

The time history of conditions within an ice condenser containment during a postulated loss-of-coolant accident (LOCA) can be divided into two periods:

1. The initial reactor coolant blowdown, which for the largest assumed pipe break occurs within approximately 30 seconds.

2. The post blowdown phase of the accident which begins following the blowdown and extends several hours after the start of the accident.

LOCA Mass and Energy Release Phases

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

1. Blowdown: The period of time from accident initiation (When the reactor is at steady state operation) to the time that the lower plenum begins to re-pressurize after initial coolant evacuation.

2. Refill: The period of time when the reactor vessel lower plenum is being filled by accumulator and ECCS water. The WCOBRA/TRAC code mechanistically calculates this phase.

3. Reflood: The period of time that begins when the water from the reactor vessel lower plenum enters the core and ends when the core is completely quenched.

4. Post-Reflood: The period of time following the reflood phase. It is during this portion of the transient that (for the DECL and DEPS breaks) a two phase mixture exiting the core enters the steam generators, resulting in reverse heat transfer from the secondary side to the primary side. heat transfer from the steam generator secondary metal to the fluid, and then from the fluid to the tubes, is accounted for in a mechanistic fashion.

The WCAP-17721-P [20] methodology uses a single code for all phases of the LOCA transient through the time of peak containment pressure.

Break Size and Location

Three distinct locations in the RCS loop can be postulated for pipe rupture:

1. Hot leg (between vessel and steam generator),

2. Cold leg (between pump and vessel),

3. Pump suction (Between SG and pump).

Using the WCAP-17721-P WCOBRA/TRAC methodology [20], full double ended ruptures were analyzed in the cold leg and the pump suction leg, each with 1 and 2 trains of safety injection, to cover the spectrum of possible limiting break locations for Watts Bar Unit 1 in the context of the new methodology.

Full double ended ruptures of the cold leg and pump suction leg behave similarly, or more specifically, both behave dissimilarly from a double ended rupture of the hot leg. This is because the hot leg break provides a direct vent path to containment for the core exit flow during the post-blowdown phases. This has the effect of allowing the two phase mixture to bypass the steam generators, yielding significantly reduces integrated energy release in the long term. While the blowdown has the potential to be more severe for the hot leg break, ice condenser plants are inherently limited by the long term mass and energy releases after the ice bed is depleted. Therefore, the hot leg break has been excluded from the spectrum of runs. The double ended pump suction and double ended cold let breaks both provide a mechanism to transport a two phase mixture to both the intact and broken steam generators, and so both have been analyzed with WCOBRA/TRAC.

Application of Single-Failure Criteria

An analysis of the effects of the single-failure criteria has been performed on the mass and energy release rates for the pump suction (DEPSG) break. An inherent assumption in the generation of the mass and energy release is that offsite power is lost. This results in the actuation of the emergency diesel generators, required to power the Safety Injection System. This is not an issue for the blowdown period, which is limited by the compression peak pressure.

The limiting minimum safety injection case has been analyzed for the effects of a single failure. In the case of minimum safeguards, the single failure postulated to occur is the loss of an emergency diesel generator. This results in the loss of one pumped safety injection train, that is, ECCS pumps and heat exchangers.

Basis of the Analysis

I. Significant Modeling Assumptions.

The following summarized assumptions were employed to ensure that the mass and energy releases were conservatively calculated, thereby maximizing energy release to containment:

1. Maximum expected operating temperature of the RCS at 100-percent full-power conditions; (619.1^{o}F).

2. An allowance in temperature for instrument error and dead band was assumed on the vessel/core inlet temperature (+6.0^{o}F)

3. Margin in volume of 3 percent (which is composed of a 1.6-percent allowance for thermal expansion, and a 1.4 percent allowance for uncertainty)

4. Core rated power of 3,459 MWt

5. Allowance for calorimetric error (+0.6 percent of power)

6. Steam generator secondary side mass was biased conservatively high.

7. Initial fuel temperatures, and thus core stored energy, were based on late in life conditions that included the effects of fuel pellet thermal conductivity degradation.

8. An allowance for RCS initial pressure uncertainty (+70 psi)

9. A maximum containment backpressure equal to design pressure

10. Steam generator tube plugging leveling (0-percent uniform)

    a. Maximizes reactor coolant volume and fluid release

    b. Maximizes heat transfer area across the steam generators tubes

    c. Reduces coolant loop resistance, which reduces the delta-p upstream of the break and increases break flow.

II. Initial Conditions

Table 6.2.1-15 presents the System Parameters Initial Conditions utilized.

Thus, based on the previously noted conditions and assumptions, a bounding analysis of Watts Bar Nuclear Plant Unit 1 is made for the release of mass and energy from the RCS in the event of a LOCA.

Mass and Energy Release Data

WCOBRA/TRAC consists of two primary source codes, COBR-TF and TRAC-PD2. COBRA-TF is a three dimensional thermal hydraulic code that is used to model the vessel in detail, and TRAC-PD2 is a one dimensional code that is used to model loop piping, pumps, steam generators, and various boundary conditions. The WOCBRA/TRAC computer code is currently used as the PWR ECCS evaluation mode by Westinghouse; it is fully capable of calculating the thermal/hydraulic RCS response to a large pipe rupture. The use of WCOBRA/TRAC for this application has been qualified by comparison with scalable test data covering the expected range of conditions and important phenomena. Modifications to WCOBRA/TRAC were required to model the specifics of importance to the LOCA M&E analysis, and these updates are discussed below.

The WCOBRA/TRAC Steam generator, modeled in accordance with the ECCS evaluation model, was shown to over predict the reverse heat transfer from the steam generators. Updates were made to more accurately calculate the steam generator cool down, this was validated by comparison the FLECHT-SEASET steam generator separate effects tests. The results is a computer code capable of modeling phenomena associated with the large break LOCA conditions that provides significant margin relative to the current WCAP-10325-P-A methodology because the stored RCS and SG energy is leased at a mechanistically calculated rate instead of forced out over a conservatively short duration.

Beginning with peak clad temperature Watts Bar Unit 1 model, specific updates were made to the model to bias it for containment integrity purposes. These key updates included:

1. Updated the PCT noding structure to include safety injection and accumulator injection in all loops

2. Maximized the RWST Temperature (105°F, technical specification maximum)

3. Applied accumulator upper limit pressure (704.7 psia), lower limit liquid volume (1,005 ft^3), and maximum temperature (130°F)

4. RCS volume was increased by 3% for thermal expansion and measurements uncertainty

5. SG tube plugging level was reduced to 0% (assumption maximizes RCS volume and flow/ heat transfer area of SG tubes)

6. Increases the RCS temperatures to the high end of the operating range band and included uncertainty for a tart T_{avg} of 594.2°F.

7. Increased the RCS initial pressures to 2320 psia (including uncertainties)

8. Increased the pressurizer liquid level. targeting maximum water volume of 1232 ft^3

9. Increased core power to account for uncertainty at full power, targeting 3479.75 MWt

10. Applied ANS 1979 + 2σ decay heat

11. Turned off fuel rod swelling model

12. Used Baker-Just correlation for metal/water reaction

13. Updated the steam generator noding structure per WCOBRA/TRAC M&E methodology

14. Biased the steam generator secondary side volumes high by 3% to account for uncertainty

15. Updated material properties to be consistent with the American Society of Mechanical Engineers (ASME) data including any applicable uncertainties

The resulting model was used to calculate the mass and energy releases for the limiting break scenario, a double ended pump suction break with minimum safeguards. The WCOBRA/TRAC tool calculates all phases of the peak pressure LOCA Transient, including blowdown, refill, reflood, and post reflood long term. The resulting blowdown and post blowdown mass and energy releases are found in Table 6.2.1-16 and Table 6.2.1-19, respectively.

Decay Heat Model

ANS Standard 5.1[25] was used in the LOCA mass and energy release model for Watts Bar Unit 1 for the determination of decay heat energy. This standard was balloted by the Nuclear Power Plant Standards Committee (NUPPSCO) in October 1978 and subsequently approved. The official standard, Reference [25], was issued in August 1979.

Significant assumptions in the generation of the decay heat curve are the following:

1. The highest decay heat release rates come from the fission of U-238 nuclei. Thus, to maximize the decay heat rate a maximum value (8%) has been assumed for the U-238 fission fraction.

2. The second highest decay heat release rate comes from the fission of U-235 nuclei. Therefore, the remaining fission fraction (92%) has been assumed for U-235.

3. The factor which accounts for neutron capture in fission products has been taken directly from Table 10 of the standard.

4. The number of atoms of Pu-239 produced per second has been assumed to be 70% of the fission rate.

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

6. The fuel has been assumed to at full power for 10^8 seconds.

7. 2σ uncertainty has been applied to the fission product decay. This accounts for a 98% confidence level.

Short-Term Mass and Energy Releases

The short-term mass and energy release models and assumptions are described in Reference [9]. The LOCA short-term mass and energy release data used to perform the containment analysis given in Sections 6.2.1.3.4 and 6.2.1.3.9 are listed below:

Section	Break Size and Location	Table
6.2.1.3.4	Double-Ended Cold Leg Guillotine Break Outside the Biological Shield	6.2.1-23
6.2.1.3.4	Double-Ended Hot Leg Guillotine Break Outside the Biological Shield	6.2.1-24
6.2.1.3.9	Double-Ended Pressurizer Spray Line Break	6.2.1-28
6.2.1.3.9	127 in^2 Cold Leg Break at the Reactor Vessel	6.2.1-30 (historical information)

6.2.1.3.7 Accident Chronology

For a double-ended pump suction loss-of-coolant accident, the major events and their time of occurrence are shown in Table 6.2.1-25 for the minimum safeguards case.

6.2.1.3.8 Mass and Energy Balance Tables

Sources of Mass and Energy

The sources of mass and energy in the WCOBRA/TRAC M&E analysis are the following:

1. RCS water

2. Accumulator water

3. Pumped safety injection water

4. Decay heat

5. Core stored energy

6. RCS metal (including the SG tube metal)

7. Steam generator metal (including shell, wrapper, and internals)
8. Steam generator secondary fluid energy (liquid and steam)

All sources of mass and energy listed above are considered in the WCOBRA/TRAC portion of the analysis. The water in the RCS, accumulators, safety injection boundary conditions, and SG secondary is explicitly modeled. Core decay heat (including feedback effects during blowdown) is included in the WCOBRA/TRAC fuel rod model. The fuel rod model also includes an energy term to represent core stored energy. The RCS and SG metal, and the associated heat transfer from these sources, is modeled in the WCOBRA/TRAC analysis.

The methods and assumptions used to release various energy sources are given in Reference [20].

The consideration of the various energy sources in the mass and energy release analysis provides assurance that all available sources of energy have been included in this analysis. Therefore, the review guidelines presented in Standard Review Plan Section 6.2.1.3 have been satisfied.

6.2.1.3.9 Containment Pressure Differentials

Consideration is given in the design of the containment internal structures to localized pressure pulses that could occur following a loss-of-coolant accident or a main steam line break. If either were to occur due to a pipe rupture in these relatively small volumes, the pressure would build up at a rate faster than the overall containment, thus imposing a differential pressure across the walls of the structures.

These subcompartments include the steam generator enclosure, pressurizer enclosure, and upper and lower reactor cavity. Each compartment is designed for the largest blowdown flow resulting from the severance of the largest connecting pipe within the enclosure or the blowdown flow into the enclosure from a break in an adjacent region.

The following paragraphs summarize the design basis calculations:

Steam Generator Enclosure

The worst break possible in the steam generator enclosure is a double-ended rupture of the steamline pipe at no load conditions. Based on an investigation of postulated break locations, the rupture is assumed to occur at the point where the steamline exits the steam generator. The blowdown for this break is given in Table 6.2.1-27a. The TMD computer code using the compressibility factor and assuming unaugmented critical flow is used to calculate the short-term pressure transients. The nodalization of the steam generator enclosure where the break occurs is shown in Figure 6.2.1-81. Node 51 is the break element and has a flow path to the adjacent steam generator enclosure which is a mirror image of the enclosure where the break occurs. Both enclosures are nodalized in the same manner; their nodal network is shown in Figure 6.2.1-82 and their input data is given in Tables 6.2.1-27b and 6.2.1-27c. This input data assumes that the insulation remains intact. The loss coefficients were computed using Reference [12]. The maximum number of nodes used is based on the geometry of the system. The steam generator compartment is essentially symmetrical with no major obstructions to flow which would introduce asymmetric pressures. In addition, the flow path to the adjacent steam generator is at the top of the enclosure. Therefore, a significant differential pressure will not occur across the steam generator vessel. The balance of plant data is similar to that presented in Section 6.2.1.3.4.

The peak pressure differentials across the steam generator enclosure, the steam generator vessel, and the steam generator separator wall are given in Table 6.2.1-27d. Figure 6.2.1-83 shows the differential pressure transient between the break element and the upper compartment (Node 25). Figures 6.2.1-84 and 6.2.1-85 illustrate the differential pressure transient across the steam generator vessel. As Figures 6.2.1-84 and 6.2.1-85 show, the pressure differentials across the vessel are low and are due solely to inertial effects. The pressure versus time curve for the break element is given in Figure 6.2.1-86 and for the upper compartment (Node 25) in Figure 6.2.1-86a (Refer to Section 3.8.3.4.8 for steam generator compartment structural design description).

Pressurizer Enclosure

The worst break possible in the pressurizer enclosure is a double-ended rupture of the six-inch spray line. The rupture is assumed to occur at the top of the enclosure. The blowdown for this break is given in Table 6.2.1-28. The TMD computer code using the compressibility factor and assuming unaugmented critical flow is used to calculate the short-term pressure transient. The nodalization of the enclosure is shown in Figure 6.2.1-87. Node 51 is the break element. The input data is given in Table 6.2.1-29. This input data assumes that the insulation remains intact. The loss coefficients were computed using Reference [12]. The maximum number of nodes used was based on the geometry of the system. The pressurizer compartment is essentially symmetrical with no major obstructions to flow which would introduce asymmetric pressures on the pressurizer vessel. The balance of plant data is similar to that presented in Section 6.2.1.3.4.

The peak pressure differentials across the pressurizer enclosure's walls, and across the pressurizer vessel are given in Table 6.2.1-29a. Figure 6.2.1-88 shows the pressure transient between the break element and the upper compartment (Node 25). As Figures 6.2.1-89 through 6.2.1-91 show, the significant pressure differential across the vessel are low, occur early, and are due solely to inertial effects. The pressure vs. time curve for the break element is given in Figure 6.2.1-92 (Refer to Section 3.8.3.4.9 for pressurizer compartment structural design description).

An evaluation of this subcompartment was conducted as part of the 1.4% Uprate Program. The previously discussed mass and energy releases and the corresponding subcompartment pressurization analysis were determined to remain bounding.

Reactor Cavity *Historical Information*

The TMD computer code with the unaugmented homogeneous critical flow correlation and the isentropic compressible subsonic flow correlation was used to calculate pressure transients in the reactor cavity region.

Nodalization sensitivity studies were performed before the analysis was begun. The total number of nodes used varied from 6 to 68. In the 6-element model, no detail of the reactor vessel annulus was involved, and for that reason the model was discarded.

Subsequent model changes primarily involved greater detail in the reactor vessel annulus. First, the annulus was divided into two vertical and eight circumferential regions. Next, some additional detail was added to the region of the broken nozzle. The next changes were effected by increasing the model to three vertical and eight circumferential regions. The total integrated pressure in the reactor cavity changed only slightly because of the last change. The next change, to 68 elements, produced the model shown with detailed modeling around the nozzle sustaining the break. The additional elements from 48 to 52 are external to the reactor cavity (ice condenser). Additional elements were added to account for all real area changes in the immediate vicinity of the break (i.e., Elements 53 and 54 were added to model the broken loop pipe annulus and the broken loop inspection port, respectively).

The nodal scheme around the reactor vessel produces a very accurate post accident pressure profile because of its design. Element 3 is a small element inside the primary shield. It would contain internal flow losses due to turning and thus contain a pressure gradient if it were made larger. The four elements numbered 33, 34, 45, and 46 are made small to minimize internal pressure variation, and the elements farther from the break are made larger because pressure gradients are low in those regions.

Figure 6.2.1-27 illustrates the positions of some of the compartments. Figure 6.2.1-28 shows the flow path connections for the 68 element model. Figure 6.2.1-29 illustrates the general configuration of the reactor vessel annulus nodalization. In the model, the lower containment is divided into four loop compartments (21 to 24). The upper containment is represented by Compartment 32. The ice condenser is modeled as five elements (48 to 52), neglecting any flow distribution effects. The break simultaneously occurs in Elements 1 and 25, immediately surrounding the nozzle. The corresponding broken loop pipe annulus is represented by Element 53. The lower reactor cavity is modeled by Element 2, the upper reactor cavity by Element 47, and the remainder of the elements, as shown in Figure 6.2.1-29, model the reactor vessel annulus. Compartments 15, 42, and 16 are really adjoining Compartments 17, 43, and 18, respectively, and Compartment 13 is on the opposite side of the vessel from the assumed break. Element 54 represents the inspection port volume above the break.

A break limiting restraint restricts the break size. A 127 in^2 cold leg break is the limiting case break for the reactor cavity analysis. The mass and energy release rates are presented in Table 6.2.1-30. Tables 6.2.1-31 and 6.2.1-32 provide the volumes, flow paths, lengths, diameters, flow areas, resistance factors, and area ratios for the elements and their connections.

The inspection port plugs were assumed to be removed at the start of the accident. All insulation is assumed in place and uncrushed during the entire transient except for the insulation between the break and the reactor vessel annulus. This insulation was conservatively assumed to crush to zero thickness.

The loss of coefficient (k) values were determined by changes in flow area and by turns the flow makes in traveling from the centroid of the upstream node to the centroid of the downstream node. The k and f factors for each path were determined using methods from such references as "Flow of Fluids through Valves, Fittings, and Pipes" by the crane company and "Chemical Engineering" by J. M. Coulson and J. A. Richardson.

Figures 6.2.1-30 through 6.2.1-68 show representative pressure transients for the break compartments, the upper and lower reactor cavities, the inspection port volume and pipe annulus near the break, the upper containment and the reactor vessel annulus. These plots demonstrate that the pressure gradient is steep near the break location and is very gradual farther away from the break. This indicates that the model must be very detailed close to the break location, but less detail is required with increasing distance (Refer to Section 3.8.3.5.3 for reactor cavity structural design description).

An evaluation of this subcompartment was conducted as part of the 1.4% Uprate Program. The previously discussed mass and energy releases and the corresponding subcompartment pressurization analysis were determined to remain bounding.

6.2.1.3.10 Steam Line Break Inside Containment

Pipe Break Blowdowns - Spectra and Assumptions

A series of steam line breaks was analyzed to determine the most severe break condition for containment temperature and pressure response. The following assumptions were used in these analysis.

1. The following break types were evaluated.

   a. Double-ended 4.6 ft^2 ruptures occurring at the nozzle on one steam generator. Steam line flow restrictors in the stream generators limit the effective break area of a full double-ended pipe rupture to a maximum of 1.4 ft^2 per steam generator.

   b. The largest split break which will not generate the low steamline pressure signal for steamline isolation.

   c. Small split breaks of 0.6 and 0.35 ft^2.

2. Steam line isolation signals and feedwater line isolation signals are generated by either a low steam line pressure signal, high or high-high containment pressure signal, or high steam line pressure rate signal. An allowance of 8 seconds is assumed for steam line isolation including generation, processing, and delay of the isolation signal and valve closure. An allowance of 8 seconds is assumed for feedwater line isolation including generation, processing, and delay of the isolation signal and valve closure.

3. Failure of a diesel generator is assumed in all cases. This results in the loss of one containment safeguards train resulting in minimum heat removal capability.

4. Blowdown from the broken steam line is assumed to be dry saturated steam.

5. Plant power levels of 100.6% and zero of nominal full-load power for DER, and split pipe ruptures at 30% of nominal full-load power.

6. Failure of a main steamline isolation valve (MSIV), failure of a feedwater isolation valve (FIV), failure of auxiliary feedwater runout control protection, and failure of a safety injection train are considered.

7. Up to three cases for each double-ended rupture and power level scenario are evaluated. One case each models the main steamline isolation valve failure, feedwater isolation valve failure, and auxiliary feedwater runout control protection failure, individually. At zero power, the feedwater isolation valve failure is not analyzed since the main feedwater is not hot enough to flash once the system depressurizes.

8. The auxiliary feedwater system is manually realigned by the operator after 10 minutes to terminate AFW to the faulted steam generator.

9. For the full double-ended ruptures, the main feedwater flow to the steam generator with the broken steam line was calculated based on an initial flow of 100% of nominal full power flow and a conservatively rapid steam generator depressurization. The peak value of this flow occurring just prior to isolation is 328% of nominal.

<u>Break Flow Calculations</u>

1. Steam Generator Blowdown

   Break flows and enthalpies from the steam generators are calculated using the Westinghouse LOFTRAN code.[14] Blowdown mass and energy release are determined using the LOFTRAN code which includes effects of core power generation, main and auxiliary feedwater additions, engineered safeguards systems, reactor coolant system thick metal heat storage, and reverse steam generator heat transfer.

2. Steam Plant Piping Blowdown

   The contribution to the mass and energy releases from the secondary plant steam piping is included with the mass and energy release rates presented in Table 6.2.1-39. For all ruptures, the steam piping volume blowdown begins at the time of the break and continues at a uniform rate until the entire piping inventory is released. The flowrate is determined using the Moody correlation, the pipe cross-sectional area, and the initial steam pressure. Following the piping blowdown, reverse flow from the intact steam generators continues to simulate the reverse steam generator flow until steam line isolation.

Single-Failure Effects

1. Failure of the main steam isolation valve (located outside of containment) in the steam line with the break allows steam from all four main steam lines (downstream of the other main steam isolation valves which close) to flow out the break. The analysis accounts for this effect by including an allowance for additional mass and energy released through the break due to the volume of steam contained in the main steam lines. No additional steam is released through the break if the postulated single failure is a main steam isolation valve in another steam line not closing. In this case, the main steam isolation valve in the broken steam line does close and there is no backflow from the downstream piping to the break.

2. Failure of a feedwater isolation valve could only result in additional inventory in the feedwater line which would not be isolated from the steam generator. The mass in this volume can flash into the steam generator and exit through the break. The feedwater regulating valve closes in no more than 6.5 seconds precluding any additional feedwater from being pumped into the steam generator. The additional line volume available to flash into the steam generator is that between the feedwater isolation valve and the feedwater regulating valve, including all headers and connecting lines.

3. Failure of the auxiliary feedwater runout control equipment would result in higher auxiliary feedwater flows entering the steam generator prior to realignment of the auxiliary feedwater system. For cases where the runout control operates properly, a constant auxiliary feed flow of approximately 1,607 gpm was assumed. This value was increased to 2,810 gpm to simulate a failure of the turbine driven auxiliary feedwater pump runout control. With installation of cavitating venturis on the auxiliary motor driven feedwater pumps, runout of these pumps is no longer a consideration for mass energy release. The 2810 gpm used for simulation of runout control failure is a maximum flow from runout of a turbine driven auxiliary feedwater pump and is still the maximum flow, therefore the current analysis is unchanged.

4. Failure of a safety injection train results in less SI flow and will result in a greater return to power. For consistency, the steam line break core response analysis, in all cases, conservatively assumes failure of a safety injection train.

Worst-Case Mass and Energy Releases

The following steam line break cases were determined to represent the worst case steamline break results.

1. Full double-ended rupture at 100.6% of nominal full power with a failure of a FIV. This represents the limiting DER case in terms of calculated peak temperature.

2. A 0.6 ft^2 split break at 30% of nominal full power with a failure of the AFW runout control system. This represents the limiting SB case in terms of calculated peak temperature.

3. A 0.35 ft^2 split break at 30% at nominal full power with a failure of the AFW runout control system. This represents the limiting case SB case in terms of superheat temperature duration.

Mass and energy releases for these cases are listed in Table 6.2.1-39.

Maximum Containment Temperature Analysis for Steam Line Break

Following a steam line break in the lower compartment of an ice condenser plant, two distinct analyses must be performed. The first analysis, a short-term pressure analysis, has been performed with the TMD computer code (see Section 6.2.1.3.9). The second analysis, a long-term analysis, does not require the large number of nodes which the TMD analysis requires. The computer code which performs this analysis is the LOTIC computer code.

The LOTIC-3 computer code was developed to analyze steamline breaks in an ice condenser plant. Details of the LOTIC-3 computer code are given in Reference [3]. It includes the capability to calculate superheat conditions, and has the ability to begin calculations from time zero.[3] The LOTIC-3 computer code has been found to be acceptable for the analysis of steam line breaks[3] with the following restrictions.

1. Mass and energy release rates are calculated with an approved model.

2. Complete break spectrums are analyzed.

3. Convective heat flux calculations, as described in References [2] and [28] are performed for all break sizes.

One condensation model was used by the LOTIC-3 computer code. For all breaks, the conservative 0% condensate reevaporization and convective heat flux model is used.

Containment Transient Calculations

The following are the major input assumptions used in the LOTIC-3 steam line break analysis for the Watts Bar Nuclear Plant.

1. Minimum safeguards are employed, e.g., one of two spray pumps, and one of two air return fans.

2. A quantity of 2.125×10^6 lbs of ice is assumed for the steam line break cases to be initially in the ice condenser. The LOCA ice mass of 2.26×10^6 lbs would still be acceptable for main steam line breaks, since these breaks only melt about 1.77×10^5 lbs of ice.

3. The boron injection tank remains installed without heat tracing, and the boric acid concentration is reduced to zero ppm (Table 6.2.1-40).

4. The air return fan is effective 10 minutes after the transient is initiated. Actual air return fan initiation can take place in 9 ± 1 minutes. Initiation as early as 8 minutes does not adversely affect the outcome of the analysis.

5. A uniform distribution of steam flow into the ice bed is assumed.

6. The initial conditions in the containment are a temperature of 120°F in the lower compartment, 120°F in the dead-ended compartment, a temperature of 85°F in the upper compartment, and a temperature of 15°F in the ice condenser. All volumes are at a pressure of 0.3 psig (see Table 6.2.1-13).

7. A containment spray pump flow of 4,030 gpm is conservatively used in the upper compartment. A diesel loading sequence for the containment sprays to energize and come up to full flow and head in 234 seconds was used in the analysis.

8. Containment structural heat sinks as presented in Table 6.2.1-1 were used. The material properties are given in Table 6.2.1-5.

9. The air return fan empties air at a rate of 40,000 ft^3/min from the upper to the lower compartments.

10. A series of large break cases (1.4 ft^2 double-ended ruptures) was run to determine the limiting large break case (Table 6.2.1-41). In addition, a series of small breaks was analyzed with LOTIC at the 30% power level (Table 6.2.1-42).

11. The mass and energy releases for the limiting breaks are given in Table 6.2.1-39. Since these rates are considerably less than the RCS double-ended breaks and their total integrated energy is not sufficient to cause ice bed melt out, the containment pressure transients generated for the RCS breaks will be more severe. However, since the steam line break blowdowns are superheated, the lower compartment temperature transients calculated in this analysis will be limited. These temperature transients are given in Figures 6.2.1-69 through 6.2.1-74.

12. The heat transfer coefficients to the containment structures are based on the work of Tagami. An explanation of their manner of application is given in Reference [3]. The stagnant heat transfer coefficients were limited to 72 Btu/hr-ft^2. This corresponds to a steam-air ratio of 1.4 (according to the Tagami correlation). The imposition of this limitation is to restrict the use of the Tagami correlation within the range of steam-air ratios from which the correlation was derived.

The containment responses presented identify the limiting and most severe cases for the large double-ended ruptures and small split breaks.

<u>Large Break</u>

The limiting case among the double-ended ruptures, which yielded a calculated peak temperature of 324.5°F and a peak pressure of 10.4 psig, is the 1.4 ft^2 loop break at 100.6% of nominal full power with a failure in a main feedwater isolation valve. Figure 6.2.1-69 provides the upper and lower compartment temperature transients, and Figure 6.2.1-70 illustrates the lower compartment pressure transients. Table 6.2.1-39 contains the mass and energy release rates for the above case.

Small Break

The most severe transient in terms of superheat temperature duration for the small break spectrum is the 0.35 ft^2, 30% nominal full power, with AFW pump runout protection failure. The temperature transient with a peak temperature of 325.1°F and peak pressure of 6.57 psig for the case is presented in Figure 6.2.1-71, and the pressure transient is provided in Figure 6.2.1-72. Table 6.2.1-39 provides the mass and energy release rates for this case.

The most limiting case in terms of peak calculated temperature is the 0.6 ft^2, 30% power, with AFW pump runout protection failure case. This case resulted in a calculated peak temperature of 325.9°F and peak pressure of 6.9 psig. Figure 6.2.1-73 presents the temperature transient, and Figure 6.2.1-74 shows the pressure transient of the lower compartment. The mass and energy releases are provided in Table 6.2.1-39.

Tables 6.2.1-43 and 6.2.1-44 provide the overall results of the calculated peak temperatures for the large and small break spectrums, respectively.

6.2.1.3.11 Maximum Reverse Pressure Differentials

Following a postulated pipe break accident, the occurrence inside the ice condenser containment may be characterized by two distinct periods:

1. The initial blowdown, which occurs in approximately 10 seconds. During this period, the air initially in the lower compartment is swept into the upper compartment and the dead-ended compartment by the blowdown mass. Large mass and pressure gradients occur throughout the containment.

2. The depressurization and post-blowdown period which occurs after the end of the initial blowdown. During this period the pressure gradient within the four compartments (upper, lower, ice condenser, and dead-ended) is almost nonexistent. The shape of the pressure transient resembles that of the mass and energy releases. Pressure decreases as blowdown diminishes, followed by a slow increase sometime during the reflood.

The analysis for the first period will usually require the modeling of the containment into many nodes so that the non-uniformity of pressure and mass distribution may be properly represented. This has been done in the TMD code.

On the other hand, the analysis for the second period will only require the modeling of the containment by a four-compartment system. These calculations are performed by the LOTIC code.[1],[28]

The code options and features discussed are used in calculating ECCS back-pressure and reverse pressure differentials across the operating deck.

Basic Assumptions

1. The containment is assumed to be physically divided into four compartments: upper, lower, ice condenser, and dead-ended compartments. Each compartment is a control volume of uniform temperature, pressure and mass distribution. Steam is also assumed to be saturated in each control volume.

2. Flow between compartments is related to the pressure differential between the compartments by a flow resistance factor.

3. A two-sump model is assumed. Temperature is considered to be uniform in each sump.

Conservation Equations

For each control volume or compartment, the conservation equations of mass, energy, momentum, and volume, an ideal gas law for air, and the equation of state for saturated steam may be written:

1. Energy equation:

$$\frac{d}{dt}\left(M_a h_a + M_s h_s + M_c h_c\right) - \frac{(V_{as} + V_c)}{J}\frac{d(P_s + P_a)}{dt} + (mh)_{out} - (mh)_{in} = R_e$$

For the lower compartment:

R_e = [Rate of energy out of break]

+ [Rate of flow energy from accumulator in the form of steam, water, and nitrogen]

- [Rate of structural heat removal]

- [Rate of flow energy of sprays if applicable]

- [Rate of heat transfer to the sump]

- [Rate of heat removal by the ice condenser drain flow, if acting as a spray]

- [Rate of energy associated with the loss on condensate from atmosphere falling to floor]

+ [Net rate of flow energy from the dead-ended compartment]

For the upper compartment:

R_e = [Flow energy of the entering spray]

- [Structure heat removal rate]

- [Energy rate associated with condensate falling from atmosphere]

For the ice condenser:

R_e = [Structure heat removal rate]

- [Rate of heat transfer to the ice]

- [Energy rate associated with ice melt and steam condensate falling from atmosphere]

2. Conservation of steam and water masses:

$$\frac{dM_s}{dt} + \frac{dM_c}{dt} + (M_s)_{out} - (M_s)_{in} = R_s$$

For the lower compartment:

R_s = [Rate of flow out of the RCS]

+ [Rate of flow out of the accumulator in the form of steam and water]

+ [Flow rate of the entering spray if applicable]

- [Rate of condensate falling to the floor]

+ [Rate of steam flow from the dead-ended compartment]

For the upper compartment:

R_s = [Flow rate of the entering spray]

- [Rate of condensate falling to the floor]

For the ice condenser:

R_s = - [Rate of condensate falling to the floor]

3. Conservation of air mass:

$$\frac{dM_a}{dt} + (M_a)_{out} - (M_a)_{in} = R_a$$

For the lower compartment:

R_a = [Rate of nitrogen flow out of the accumulator]

\+ [Rate of air flow out of the dead-ended compartment]

For the upper compartment and the ice condenser:

R_a = 0

4. Conservation of momentum:

$$P_I - P_J + \frac{1}{2}\left(\frac{K_{ij}}{A^2}\right)\frac{M_{IJ}^2}{\rho g_C}$$

5. Volume conservation:

$$\frac{dV_{as}}{dt} + \frac{dV_c}{dt} = R_v$$

For the lower compartment:

R_v = [Rate of increase in sump water volume]

For the upper compartment:

R_v = 0

For the ice condenser:

R_v = [Rate of increase in free volume due to ice melting]

6. Ideal gas law for air:

$$P_a V_{as} = M_a R_a T$$

7. Equations of state for saturated steam:

$$P_s = f_1(T),\ h_s = f_2(T),\ v_s = f_3(T)$$

For the dead-ended compartment, the structure heat removal is assumed to be negligible, and the conservation equations of energy and mass simplified to:

$$\frac{d}{dt}(M_a h_a + M_s h_s) - \frac{v}{J}\frac{d(P_s + P_a)}{dt} = \text{[Rate of energy flow from the lower compartment]}$$

$$\frac{dM_a}{dt} = \text{[Rate of air flow from the lower compartment]}$$

$$\frac{dM_s}{dt} = \text{[Rate of steam flow from the lower compartment]}$$

Method of Solution

The preceding equations were linearized and programmed for simultaneous solutions using the standard Gauss-Jordan reduction method. For each time step, the solutions are the rates of increase of mass and pressure for each constituent in each compartment, and the flow rates between the compartments. These rates are used to control the time step so that total change of the compartment conditions in each time step can be controlled. This assures more accurate and stable solutions.

Structure Heat Transfer

The standard Westinghouse ECCS containment structural heat transfer model is applied to this code. This model assumes one dimensional conduction heat transfer in the structure and uses film heat transfer coefficient based primarily on the work of Tagami. The Tagami correlation for the film heat transfer may be written as:

$$H_{max} = 75\left[\frac{E}{t_p V}\right]^{0\ 6} \qquad (1)$$

$$H = H_{max}\frac{\sqrt{t}}{\sqrt{t_p}} \quad \text{for } 0 \leq t \leq t_p \qquad (2)$$

$$H = H_{stag} + [H_{max} - H_{stag}]\ e^{-0\ 5[t - t_p]} \quad \text{for } t_p \leq t \qquad (3)$$

where:

$$H_{stag} = 2 + 50\ X \qquad (4)$$

For this application, we have found it is useful to relate the "coolant energy transfer", (E/t_pV), to containment conditions. This may be done by writing:

$$\frac{E}{t_p V} = \frac{M_s \bar{h}_s + M_f \bar{h}_f}{t_p V} = \frac{1}{t_p V_s}\left(\bar{h}_s + \frac{M_f}{M_s}\bar{h}_f\right) \qquad (5)$$

where h_s, h_f, and ν_s are respectively the enthalpies of saturated steam and water, and the specific volume of steam, at t_p, the time when the peak containment pressure is reached.

Equations (1) through (5) are used for the lower compartment structure calculations. For the upper compartment, only the stagnant heat transfer correlation of Equation (4) is used because of little steam penetration into the upper compartment even during the initial blowdown period.

Ice Condenser Heat Transfer

The transfer of heat from steam to ice which results in the simultaneous occurrence of steam condensation and ice melting is a complex mechanism.

During the initial blowdown period when high temperature blowdown steam and water hits the bottom of ice columns, and then flows over the ice surface, turbulent condensation results. During this period the heat transfer rate is strongly dependent on the thickness of the liquid film which separates the high temperature blowdown masses from the ice. This liquid film is composed of steam condensate and ice melt. On the macroscopic scale, this is the only heat transfer resistance and the effectiveness of the ice condenser is determined by the rate which this liquid film may be withdrawn. A semi-empirical model for the ice condenser heat transfer during this period is available and has been used successfully in the TMD code. The LOTIC code is not intended to duplicate this effort. Instead mass and energy balances are used to calculate the total ice melting during this period. Following the initial blowdown period, there is a transition period when the blowdown mass and energy rates are decreasing rapidly and the containment atmosphere as a whole is losing internal energy. Depressurization and decreasing compartment temperature generally characterize this transition period. As the containment conditions lapse into a much more stable and slowly changing pace after the transition period, the blowdown from the broken pipe is almost drawing to an end. Flow in the ice condenser is now at a rate which is almost negligible compared to that in the initial blowdown period. Temperature in the ice condenser atmosphere has also decreased. Thus, heat transfer is governed by combining natural convection and steam diffusion through an almost stagnant atmosphere. Due to the large air content, the resistance to diffusion is large. Therefore, most of the temperature difference between the free-steam steam-air mixture and the ice occurs between the free-steam and the free surface of the liquid film. Temperature difference across the liquid film is now comparatively small.

Due to the loss of dominance for the liquid film resistance in the overall heat transfer mechanism, it is not surprising for Yen, Zender, Zavohik, and Tien[11] to conclude that ice melting has very little effect on the overall heat transfer coefficient for condensation-melting heat transfer in the presence of a substantial air concentration. From this, we may therefore treat the ice as if it were simply a cold structure and use Equation (4) to calculate the heat transfer coefficient after the transition period.

During the transition period, it is plausible to assume that the ice condenser is capable of maintaining its internal energy by condensing any excess energy which flow into the ice condenser.

Special Code Capabilities in Response to Previous NRC Concerns

1. Heat removal from the lower compartment by the ice condenser drain may be accounted for by input of a spray-like efficiency.

2. Heat transfer between the lower compartment atmosphere and the sump surface can also be taken into account.

Drains are provided at the bottom of the ice condenser compartment to allow the melt/condensate water to flow out of the compartment during a loss of coolant accident. In the modified LOTIC code, a calculation of the flow rate at which water leaves these ice condenser drains is included. The solution was reached by using the hydraulic incompressible flow equations commonly found in the literature for both filled pipe flow and fall (weir) flow conditions and at any point in time using the minimum flow rate calculated by the two methods. The filled pipe flow equation employed was a simplified Bernoulli balance:

$$Z_1 = \frac{V_2^2}{2g} + h_f + Z_2$$

where:

Z = Elevation

V = Velocity

g = Gravitational constant

D = Density

$$h_f = \frac{fl}{d} \bullet \frac{V_2^2}{2} g$$

Subscripts 1 and 2 represent conditions at the inlet and outlet of the drain, respectively.

The area of the ice condenser sump was taken to be 3170 ft^2, and the height of the door sill to be 8.75 inches. After calculating the velocity from the previous equation, the mass flow rate can be calculated from

$$\dot{m} = \rho V_2 A_2$$

Since a filled pipe flow condition may not exist during the entire post accident transient, a calculation of the draining rate based on the existence of a fall flow phenomena was included. The corresponding equations are outlined below,

$$Q = \frac{2}{3} 2g \, (H_e^{3/2} - h_1^{3/2})$$

where:

Q = Discharge per foot of width (ft^3/sec-ft)

H_e = Energy of fluid upstream of the fall

h_1 = Energy of the fluid at the fall edge minus the flowing height

$D_1 = 0.643\ D_c$

$h_1 = H_e - D_1$

By assuming the approach velocity equals zero and through substitution, we arrive at the simplified equation:

$$Q = \frac{2}{3} 2g \, [D_c^{3/2} - (0.357\ D_c)^{3/2}] \quad 1$$

or

$$Q = 4.2088\ D_c^{3/2} \quad 2$$

where:

D_c = ft.

Calculation of Maximum Reverse Pressure Differential

The computer model previously described was used to calculate the reverse differential pressure across the operating deck. In order to calculate a maximum reverse differential pressure the following assumptions were made:

1. The dead-ended compartment volumes adjacent to the lower compartment (fan and accumulator rooms, pipe trenches, etc.) were assumed to be swept of air during the initial blowdown. This is a very conservative assumption, since this will maximize the air mass forced into the upper ice bed and upper compartment thus raising the compression pressure. In addition, it will minimize the mass of the noncondensables in the lower compartment. With this modeling the dead-ended volume is included with that of the lower compartment (see Figure 6.2.1-75), resulting in a 3-volume simulation of the containment.

2. The minimum containment temperatures are assumed in the various subcompartments. This will maximize the air mass forced into the upper containment. It will also increase the heat removal capability of the cold lower compartment structures.

3. A maximum RWST temperature of 100°F was assumed in the original Westinghouse analysis. The current maximum RWST temperature is 105°F. TVA has performed an analysis to evaluate the effects of containment spray system changes on the maximum reverse pressure differential analysis. A Westinghouse review of the TVA analysis concluded that the non-conservative effect of the 5°F increase is largely offset by the effects of conservatism in assumptions 4 and 5 below (See Table 6.2.1-37).

4. The upper containment spray flowrates used were runout flows.

5. Containment spray to the upper compartment was assumed to start at 25 seconds in the original Westinghouse analysis. Containment spray initiation to the upper containment is tabulated in Table 6.2.1-25. TVA has performed an analysis to evaluate the effects of containment spray system changes on the maximum reverse pressure differential analysis. A Westinghouse review of the TVA analysis concluded that increased delay should have no negative effect on the maximum reverse pressure differential.

6. The containment geometry is the same as that used in the minimum pressure analysis for ECCS purposes. (See Tables 6.2.1-33 through 6.2.1-36.)

7. The Westinghouse ECCS model (see WCAP-8339) was used for heat transfer to the structure.

8. The mass and energy releases used are based on the analysis presented in WCAP-8479.

9. Ice condenser doors are assumed to act as check valves, allowing flow only into the ice condenser.

10. The loss coefficient (k/A^2) of the deck fins for air flow from the upper to the lower compartment was taken to be 0.0072 ft^{-4}. This value was based on the capabilities of the fans while running. With the fans not running the loss coefficient would be 0.0278 $ft.^{-4}$

With these assumptions the maximum reverse pressure differential across the operating deck was calculated to be 0.65 psi. The following plots have been provided:

Figure 6.2.1-76 which shows upper and lower compartment pressures.

Figure 6.2.1-77 which shows upper and lower containment temperatures.

Figure 6.2.1-78 which shows upper to lower containment flowrates.

Parametric studies have been made with this model. Various effects have been investigated to determine changes in the maximum reverse pressure differential. Table 6.2.1-37 gives some of these studies with their results. For Case 6, Figures 6.2.1-79 and 6.2.1-80 give plots similar to Figures 6.2.1-76 and 6.2.1-77. Presented in Table 6.2.1-38, also for Case 6, are the sump temperature and the steam exit flow from the ice condenser, both as a function of time.

Significant margin exists between the design reverse differential pressures across the operating deck and the ice condenser lower inlet doors, and those calculated pressures presented in Table 6.2.1-37.

An evaluation of this event was conducted as part of the 1.4% Uprate Program. The previously discussed analysis was determined to remain bounding.

6.2.1.3.12 Tritium Production Core Evaluation

The effects of converting WBN to a Tritium Production Core (TPC) on the primary containment functional design have been evaluated. The evaluation includes the effect of adding a leading edge flow meter (LEFM) into the main feedwater piping and the effect of uprating the plant nominal power level by 1.4% and also for the replacement steam generator program.

The effect on changes to the primary containment design evaluation are controlled by changes to the mass and energy release for the various events for which the primary containment is evaluated as a results of the up rating and LEFM. The important aspects of the change to a TPC are the amount of steam generator tube plugging, the core stored energy and engineering judgment. The single effect of the reactor power does not need to be evaluated since the 1.4% increase is accommodated by reducing the calorimetric error from 2% to 0.6% as a result of adding the LEFM into the main feedwater piping which results in a more accurate calculation of the actual power.

For the long-term LOCA mass and energy analysis described in Section 6.2.1.3.3, the pertinent parameters are the initial energy content of the RCS metal mass, the core-stored energy, the pressure drop through the core, and the decay heat. The vessel average temperature for the TPBAR program is 588.2ºF, which is the same as for the 1.4% Power Uprate and Replacement Steam Generator (RSG) Programs. Therefore, there is no change in the initial energy content of the RCS fluid, and any deviations in RCS metal energy content due to a different temperature distribution in the RCS is judged to be insignificant. The presence of TPBARs has been shown to have an insignificant effect on the core-stored energy. The fuel assembly hydraulic characteristics with the TPBARs are the same as with thimble plugs inserted, therefore, the pressure drop through the core will not change with the addition of TPBARs. In addition, the WBN LOCA mass and energy analysis currently utilizes the ANS Standard-79 decay heat model which is unaffected by and applicable to cores with TPBARs. Therefore, the tritium production core does not impact the decay heat. In summary, the tritium production core will not adversely impact the long-term LOCA mass and energy related analyses, methodology, or the corresponding containment analysis. The conclusions of the 1.4% Power Uprate Program remains applicable for the RSG program.

For the short-term LOCA mass and energy analysis described in Section 6.2.1.3.4, the analysis input that has the potential to change the mass and energy release are the initial RCS fluid temperatures and the pressure drop through the core. Since this event lasts for approximately 3 seconds, the effects of the fuel type core stored energy, and other fuel-related parameters, are not significant. A vessel outlet temperature of 617.3ºF and a vessel core inlet temperature of 557.7ºF, both providing a conservatively lower bound for short-term considerations, were evaluated for the SGTP program. The TPC value of 619.1ºF for the vessel outlet temperature is, therefore, bounded by the SGTP evaluation. However, the TPC value of 557.3ºF for the vessel/core inlet temperature is slightly lower by 0.4ºF. This is a very small difference that would not invalidate the conclusions reached in the SGTP, and the evaluation therefore, applies to the TPC. In addition, the fuel assembly hydraulic characteristics with the TPBARs are the same as with thimble plugs inserted, therefore the pressure drop through the core will not change with the addition of TPBARs. In summary, the tritium production core will not adversely impact the short-term LOCA mass and energy, nor will it affect the loop subcompartment, reactor cavity, or pressurizer enclosure evaluations described in Section 6.2.1.3.9. The short-term blowdown transients are characterized by a peak mass and energy (M&E) release rate that occurs during a subcooled condition. The Zaloudek correlation, which models this condition, is currently used in the short-term LOCA M&E release analyses[9]. This correlation was used to conservatively evaluate the impact of the deviations in the RCS inlet and outlet temperatures due to the reduced T_{avg} and RSG program. The use of the lower temperatures maximizes the critical mass flux in the Zaloudek correlation. Subsequently, as part of the 1.4% power uprate program a more limiting set of RCS conditions were evaluated. A vessel outlet temperature of 607.3^{o}F and a vessel core inlet temperature of 549.1^{o}F, both conservatively bounded low for short term LOCA mass and energy release application were evaluated. These values are also lower than the vessel outlet temperature of 617.2^{o}F and a vessel core inlet temperature of 555.2^{o}F proposed for the reduced T_{avg} and replacement steam generator program. The current licensing basis analyses remain valid and the conclusions of the 1.4% Power Uprate Program are applicable for the RSG program.

For the steam line break analysis described in Section 6.2.1.3.10, mass and energy releases in non-LOCA accidents due to high energy secondary side steamline and feedwater line breaks are sensitive to changes in core reactivity coefficients; specifically, moderator density coefficients and shutdown margin. In general, bounding values of reactivity coefficients are used in the analyses to bound the accident over a wide range of core conditions.

A comparison of key safety analysis parameters used in the standard Westinghouse reload safety evaluation process was made with the respect to the high energy secondary side line break analyses of record. All of the key safety analysis parameters which comprise the reference plant safety analyses for steamline break and feedwater line break mass and energy releases bound the core reload design values of the TPC. In addition, the effects of the fuel temperatures associated with ZIRLO® fuel rod cladding have been evaluated. In general, small changes in the fuel temperatures have little or no effect on non-LOCA safety analyses using the LOFTRAN code. In particular, secondary side line break transients for which mass and energy releases are calculated by LOFTRAN are dominated by assumptions related to the main steam system design and protection and are not sensitive to core-related inputs such as fuel temperatures.

Since none of the key safety analysis parameters has been exceeded for the core reload design using the TPC, the licensing basis analyses of record for the high energy secondary side line breaks remain valid. Therefore, the analysis results with respect to mass and energy releases and the associated pressure and/or temperature response analyses also remain valid.

For the maximum reverse pressure differential analysis described in Section 6.2.1.3.11, there are no changes associated with this program which will have an impact on the analysis. The variables that change do no adversely affect the normal plant operating parameters, system actuations, accident mitigating capabilities or assumptions important to this analysis, or create conditions more limiting that those assumed in this analysis.

Nomenclature

SYMBOL	DESCRIPTION
A	Flow area
E	Total energy
J	Conversion constant, 778 ft-lbf/Btu
K	Flow resistance factor
M	Mass
P	Pressure
R	Gas constant
R	Rate
T	Temperature
V	Volume

g_c	Conversion constant, 32.2 ft-lbm/lbf-sec^2
h	Enthalpy
m	Mass flow rate between two compartments
t	Time
x	Steam-air ratio
v	Specific volume
D	Density

SUBSCRIPT

a	Air
as	Air and steam
c	Suspended or entrained water
e	Energy
i	i-th compartment
j	j-th compartment
ij	from i-th compartment to j-th compartment
s	Steam

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6.2 CONTAINMENT SYSTEMS

6.2.1 Containment Functional Design

6.2.1.1 Design Bases

6.2.1.1.1 Primary Containment Design Bases

The containment is designed to assure that an acceptable upper limit of leakage of radioactive material is not exceeded under design basis accident conditions. For purposes of integrity, the containment may be considered as the containment vessel and containment isolation system. This structure and system are directly relied upon to maintain containment integrity. The emergency gas treatment system and Reactor Building function to keep out-leakage minimal (the Reactor Building also serves as a protective structure), but are not factors in determining the design leak rate.

The containment is specifically designed to meet the intent of the applicable General Design Criteria listed in Section 3.1. This section, Chapter 3, and other portions of Chapter 6 present information showing conformance of design of the containment and related systems to these criteria.

The ice condenser is designed to limit the containment pressure below the design pressure for all reactor coolant pipe break sizes up to and including a double-ended severance. Characterizing the performance of the ice condenser requires consideration of the rate of addition of mass and energy to the containment as well as the total amounts of mass and energy added. Analyses have shown that the accident which produces the highest blowdown rate into a condenser containment will result in the maximum containment pressure rise; that accident is the double-ended guillotine or split severance of a reactor coolant pipe. The design basis accident for containment analysis based on sensitivity studies is therefore the double-ended guillotine severance of a reactor coolant pipe at the reactor coolant pump suction. Post-blowdown energy releases can also be accommodated without exceeding containment design pressure.

The functional design of the containment is based upon the following accident input source term assumptions and conditions:

1. The design basis blowdown energy of 353.9 x 10^6 Btu and mass of 586.6 x 10^3 lb put into the containment (See Section 6.2.1.3.6).

2. A core power of 3411 MWt (plus 2% allowance for calorimetric error) (See Section 6.2.1.3.6).

3. The minimum engineered safety features are (i.e., the single failure criterion applied to each safety system) comprised of the following:

    a. The ice condenser which condenses steam generated during a LOCA, thereby limiting the pressure peak inside the containment (see Section 6.7).

    b. The containment isolation system which closes those fluid penetrations not serving accident-consequence limiting purposes (see Section 6.2.4).

    c. The containment spray system which sprays cool water into the containment atmosphere, thereby limiting the pressure peak (particularly in the long term - see Section 6.2.2).

    d. The emergency gas treatment system (EGTS) which produces a slightly negative pressure within the annulus, thereby precluding out-leakage and relieving the post-accident thermal expansion of air in the annulus (see Section 6.5.1).

    e. The air return fans which return air to the lower compartment (See Section 6.8).

Consideration is given to subcompartment differential pressure resulting from a design basis accident discussed in Sections 3.8.3.3, 6.2.1.3.9, and 6.2.1.3.4. If a design basis accident were to occur due to a pipe rupture in these relatively small volumes, the pressure would build up at a faster rate than in the containment, thus imposing a differential pressure across the wall of these structures.

Parameters affecting the assumed capability for post-accident pressure reduction are discussed in Section 6.2.1.3.3.

Three events that may result in an external pressure on the containment vessel have been considered:

1. Rupture of a process pipe where it passes through the annulus.

2. Inadvertent air return fan operation during normal operation.

3. Inadvertent containment spray system initiation during normal operation.

The design of the guard pipe portion of hot penetrations is such that any process pipe leakage in the annulus is returned to the containment. All process piping which has potential for annulus pressurization upon rupture is routed through hot penetrations. Section 6.2.4 discusses hot penetrations.

Inadvertent air return fan operation during normal operation opens the ice condenser lower inlet doors, which in turn, results in sounding an alarm in the MCR. Sufficient time exists for operator action to terminate fan operation prior to exceeding the containment design external pressure.

The logic and control circuits of the containment spray system are such that inadvertent containment spray would not take place with a single failure. The spray pump must start and the isolation valve must open before there can be any spray. In addition, the Watts Bar containment is so designed that even if an inadvertent spray occurs, containment integrity is preserved without the use of a vacuum relief.

The containment spray system is automatically actuated by a hi-hi containment pressure signal from the solid state protection system (SSPS). To prevent inadvertent automatic actuation, four comparator outputs, one from each protection set are processed through two coincidence gates. Both coincidence gates are required to have at least two high inputs before the output relays, which actuate the containment spray system, are energized. Separate output relays are provided for the pump start logic and discharge valve open logic. Additional protection is provided by an interlock between the pump and discharge valve, which requires the pump to be running before the discharge valve will automatically open.

Section 3.8.2 describes the structural design of the containment vessel. The containment vessel is designed to withstand a net external pressure of 2.0 psi. The containment vessel is designed to withstand the maximum expected net external pressure in accordance with ASME Boiler and Pressure and Vessel Code Section III, paragraph NE-7116.

6.2.1.2 Primary Containment System Design

The containment consists of a containment vessel and a separate Shield Building enclosing an annulus. The containment vessel is a freestanding, welded steel structure with a vertical cylinder, hemispherical dome, and a flat circular base. The Shield Building is a reinforced concrete structure similar in shape to the containment vessel. The design of these structures is described in Section 3.8.

The design internal pressure for the containment is 13.5 psig, and the design temperature is 250°F. The design basis leakage rate is 0.25 weight percent/24 hr. The design methods to assure integrity of the containment internal structures and sub-compartments from accident pressure pulses are described in Section 3.8.

6.2.1.3 Design Evaluation

6.2.1.3.1 Primary Containment Evaluation

1. The leaktightness aspect of the secondary containment is discussed in Section 6.2.3. The primary containment's leaktightness does not depend on the operation of any continuous monitoring or compressor system. The leak testing of the primary containment and its isolation system is discussed in Section 6.2.6.

2. The acceptance criteria for the leaktightness of the primary containment are such that at containment design pressure, there is a 25% margin between the acceptable maximum leakage rate and the maximum permissible leakage rate.

6.2.1.3.2 General Description of Containment Pressure Analysis

The time history of conditions within an ice condenser containment during a postulated loss of coolant accident can be divided into two periods for calculation purposes:

1. The initial reactor coolant blowdown, which for the largest assumed pipe break occurs in approximately 10 seconds.

2. The post blowdown phase of the accident which begins following the blowdown and extends several hours after the start of the accident.

During the first few seconds of the blowdown period of the reactor coolant system, containment conditions are characterized by rapid pressure and temperature transients. It is during this period that the peak transient pressures, differential pressures, temperature and blowdown loads occur. To calculate these transients a detailed spatial and short time increment analysis was necessary. This analysis was performed with the Transient Mass Distribution (TMD) computer code [4] with the calculation time of interest extending up to a few seconds following the accident initiation (See Section 6.2.1.3.4).

Physically, tests at the ice condenser Waltz Mill test facility have shown that the blowdown phase represents that period of time in which the lower compartment air and a portion of the ice condenser air are displaced and compressed into the upper compartment and the remainder of the ice condenser. The containment pressure at or near the end of blowdown is governed by this air compression process. The containment compression ratio calculation is described in Section 6.2.1.3.4.

Containment pressure during the post blowdown phase of the accident is calculated with the LOTIC1 code which models the containment structural heat sinks and containment safeguards systems.

6.2.1.3.3 Long-Term Containment Pressure Analysis

Early in the ice condenser development program it was recognized that there was a need for modeling of long-term ice condenser containment performance. It was realized that the model would have to have capabilities comparable to those of the dry containment (COCO) model. These capabilities would permit the model to be used to solve problems of containment design and optimize the containment and safeguards systems. This has been accomplished in the development of the LOTIC1 code[1].

The model of the containment consists of five distinct control volumes; the upper compartment, the lower compartment, the portion of the ice bed from which the ice has melted, the portion of the ice bed containing unmelted ice, and the dead ended compartments. The ice condenser control volume with unmelted and melted ice is further subdivided into six subcompartments to allow for maldistribution of break flow to the ice bed.

The conditions in these compartments are obtained as a function of time by the use of fundamental equations solved through numerical techniques. These equations are solved for three distinct phases in time. Each phase corresponds to a distinct physical characteristic of the problem. Each of these phases has a unique set of simplifying assumptions based on test results from the ice condenser test facility. These phases are the blowdown period, the depressurization period, and the long term period.

The most significant simplification of the problem is the assumption that the total pressure in the containment is uniform. This assumption is justified by the fact that after the initial blowdown of the reactor coolant system, the remaining mass and energy released from this system into the containment are small and very slowly changing. The resulting flow rates between the control volumes will also be relatively small. These small flow rates then are unable to maintain significant pressure differences between the compartments.

In the control volumes, which are always assumed to be saturated, steam and air are assumed to be uniformly mixed and at the control volume temperature. The air is considered a perfect gas, and the thermodynamic properties of steam are taken from the ASME steam table.

For the purpose of calculation, the condensation of steam is assumed to take place in a condensing node located between the two control volumes in the ice storage compartment.

Containment Pressure Calculation

The following are the major input assumptions used in the LOTIC1 analysis for the pump suction pipe rupture case with the steam generators considered as an active heat source for the Watts Bar Nuclear Plant containment:

1. Minimum safeguards are employed in all calculations, e.g., one of two spray pumps and one of two spray heat exchangers; one of two RHR pumps and one of two RHR heat exchangers providing flow to the core; one of two safety injection pumps and one of two centrifugal charging pumps; and one of two air return fans.

2. 2.26×10^6 lbs. of ice initially in the ice condenser which is at 27°F.

3. The blowdown, reflood, and post reflood mass and energy releases described in Section 6.2.1.3.6 were used.

4. Blowdown and post-blowdown ice condenser drain temperatures of 190°F and 130°F are used.[5]

5. Nitrogen from the accumulators in the amount of 3961 lbs. is included in the calculations.

6. Essential raw cooling water temperature of 88°F is used on the spray heat exchanger and the component cooling heat exchanger.

7. The air return fan is effective 10 minutes after the transient is initiated.

8. No maldistribution of steam flow to the ice bed is assumed.

9. No ice condenser bypass is assumed. (This assumption depletes the ice in the shortest time and is thus conservative.)

10. The initial conditions in the containment are a temperature of 100°F in the lower and dead-ended volumes, 80°F in the upper compartment, and 27°F in the ice condenser. (Note: The 80°F temperature in the upper compartment is a reduction from the 85°F lower technical specification limit to account for the upper plenum volume of the ice condenser which is included in upper compartment volume for the analysis. The volume is adjusted to maximize air mass and the compression ratio.) All volumes are at a pressure of 0.3 psig and a 10-percent relative humidity, except the ice condenser which is at 100-percent relative humidity.

11. A containment spray pump flow of 4000 gpm is used in the upper compartment. The analyzed diesel loading sequence for the containment sprays to energize and come up to full flow and head in 234 seconds is tabulated in Table 6.2.1-25.

12. The capability to divert flow from the RHR system to a spray header in upper containment exists, but has not been modeled into the WCOBRA/TRAC containment integrity analysis.

13. Containment structural heat sink data is found in Table 6.2.1-1. (Note that the dead-ended compartment structural heat sinks were conservatively neglected.)

14. The operation of one containment spray heat exchanger (UA = 2.44×10^6 Btu/hr-°F incorporating a 10% tube plugging margin) for containment cooling and the operation of one RHR heat exchanger (UA = 1.496×10^6 Btu/hr-°F) for core cooling. The component cooling heat exchanger UA was modeled at 3.17×10^6 Btu/hr-°F.

15. The air return fan returns air at a rate of 40,000 cfm from the upper to lower compartment.

16. An active sump volume of 51000 ft^3 is used.

17. Deleted

18. 100.6% of 3459 MWt power is used in the WCOBRA/TRAC mass and energy release analysis.

19. Hydrogen gas was added to the containment in the amount of 25,230.2 Standard Cubic Feet (SCF) over 24 hours. Sources accounted for were radiolysis in the core and sump post-LOCA, corrosion of plant materials (Aluminum, Zinc, and painted surfaces found in containment), reaction of 1% of the Zirconium fuel rod, and hydrogen gas dissolved in the reactor coolant system water. (This bounds tritium producing core designs).

20. The containment compartment volumes were based on the following: upper compartment 645,818 ft^3; lower compartment 221,074 ft^3; and dead-ended compartment 146,600 ft^3. (Note: These volumes represent TMD volumes. For Containment Integrity Analysis, the volumes are adjusted to maximize air mass and the compression ratio.)

21. Subcooling of emergency core cooling (ECC) water from the RHR heat exchanger is assumed.

22. DEPS minimum safeguards mass and energy releases are used.

23. The RWST is assumed to be at a temperature of 105°F.

24. ANS 5.1-1979 + 2 sigma heat model was used in the WCOBRA/TRAC mass and energy release analysis.

With these assumptions, the heat removal capability of the containment is sufficient to absorb the energy releases and still keep the maximum calculated pressure well below design.

The following plots are provided:

Figure 6.2.1-1, Containment Pressure Transient,

Figure 6.2.1-2a, Upper Compartment Temperature Transients,

Figure 6.2.1-2b, Lower Compartment Temperature Transient

Figure 6.2.1-3, Active and Inactive Sump Temperature Transient,

Figure 6.2.1-4, Ice Melt Transient.

Figure 6.2.1-4a, Comparison of Containment Pressure versus Ice Melt Transients

Tables 6.2.1-3 and 6.2.1-4 give energy accountings at various points in the transient.

As can be seen from Figure 6.2.1-1 the maximum calculated Containment pressure is 9.36 psig, occurring at approximately 8959 seconds.

Structural Heat Removal

Provision is made in the containment pressure analysis for heat storage in interior and exterior walls. Each wall is divided into a number of nodes. For each node, a conservation of energy equation expressed in finite difference forms accounts for transient conduction into and out of the node and temperature rise of the node. Table 6.2.1-1 is a summary of the containment structural heat sinks used in the analysis. The material property data used is found in Table 6.2.1-5.

The heat transfer coefficient to the containment structures is based primarily on the work of Tagami.[21] An explanation of the manner of application is given in Reference [3].

When applying the Tagami correlations a conservative limit was placed on the lower compartment stagnant heat transfer coefficients. This corresponds to a steam-air ratio of 1.4 according to the Tagami correlation. The imposition of this limitation is to restrict the use of the Tagami correlation within the test range of steam-air ratios where the correlation was derived.

6.2.1.3.4 Short-Term Blowdown Analysis

TMD Code - Short-Term Analysis

1. Introduction

The basic performance of the ice condenser reactor containment system has been demonstrated for a wide range of conditions by the Waltz Mill Ice Condenser Test Program. These results have clearly shown the capability and reliability of the ice condenser concept to limit the Containment pressure rise subsequent to a hypothetical loss-of-coolant accident.

To supplement this experimental proof of performance, a mathematical model has been developed to simulate the ice condenser pressure transients. This model, encoded as computer program TMD (Transient Mass Distribution), provides a means for computing pressures, temperatures, heat transfer rates, and mass flow rates as a function of time and location throughout the containment. This model is used to compute pressure differences on various structures within the containment as well as the distribution of steam flow as the air is displaced from the lower compartment. Although the TMD code can calculate the entire blowdown transient, the peak pressure differences on various structures occur within the first few seconds of the transient.

2. Analytical Models (No Entrainment)

The mathematical modeling in TMD is similar to that of the SATAN blowdown code in that the analytical solution is developed by considering the conservation equations of mass, momentum and energy and the equation of state, together with the control volume technique for simulating spatial variation. The governing equations for TMD are given in Reference [4].

The moisture entrainment modifications to the TMD code are discussed, in detail, in Reference [4]. These modifications comprise incorporating the additional entrainment effects into the momentum and energy equations.

As part of the review of the TMD code, additional effects are considered. Changes to the analytical model required for these studies are described in Reference [4].

These studies consist of:

a. Spatial acceleration effects in ice bed
b. Liquid entrainment in ice beds
c. Upper limit on sonic velocity
d. Variable ice bed loss coefficient
e. Variable door response
f. Wave propagation effects

Additionally the TMD code has been modified to account for fluid compressibility effects in the high Mach number subsonic flow regime.

<u>Experimental Verification</u>

The performance of the TMD code was verified against the 1/24 scale air tests and the 1968 Waltz Mill tests. For the 1/24 scale model the TMD code was utilized to calculate flow rates to compare against experimental results. The effect of increased nodalization was also evaluated. The Waltz Mill test comparisons involved a reexamination of test data. In conducting the reanalyses, representation of the 1968 Waltz Mill test was reviewed with regard to parameters such as loss coefficients and blowdown time history. The details of this information are given in Reference [4].

The Waltz Mill Ice Condenser Blowdown Test Facility was reactivated in 1973 to verify the ice condenser performance with the following redesigned plant hardware scaled to the test configuration:

1. Perforated metal ice baskets and new design couplings.

2. Lattice frames sized to provide the correct loss coefficient relative to plant design.

3. Lower support beamed structure and turning vanes sized to provide the correct turning loss relative to the plant design.

4. No ice baskets in the lower ice condenser plenum opposite the inlet doors.

The result of these tests was to confirm that conclusions derived from previous Waltz Mill tests have not been significantly changed by the redesign of plant hardware. The TMD Code has, as a result of the 1973 test series, been modified to match ice bed heat transfer performance. Detailed information on the 1973 Waltz Mill test series is found in Reference [5].

Application to Plant Design (General Description)

As described in Reference [4], the control volume technique is used to spatially represent the containment. The containment is divided into 50 elements to give a detailed representation of the local pressure transient on the containment shell and internal concrete structures. This division of the containment is similar for all ice condenser plants.

The Watts Bar plant containment has been divided into 50 elements or compartments as shown in Figures 6.2.1-5, 6.2.1-6, 6.2.1-7, and 6.2.1-8. The interconnections between containment elements in the TMD code is shown schematically in Figure 6.2.1-9. Flow resistance and inertia are lumped together in the flow paths connecting the elements shown. The division of the lower compartments into 6 volumes occurs at the points of greatest flow resistance, i.e., the four steam generators, pressurizer and refueling cavity.

Each of these lower compartment sections delivers flow through doors into a section behind the doors and below the ice bed. Each vertical section of the ice bed is, in turn, divided into three elements. The upper plenum between the top of the ice bed and the upper doors is represented by an element. Thus, a total of thirty elements (Elements 7 through 24 and 38 through 49 are used to simulate the ice condenser). The six elements at the top of the ice bed between bed and upper doors deliver to element number 25 the upper compartment. Note that cross flow in the ice bed is not accounted for in the analysis; this yields the most conservative results for the particular calculations described herein. The upper reactor cavity (Element 33) is connected to the lower compartment volumes and provides cross flow for pressure equalization of the lower compartments. The less active compartments, called dead-ended compartments (Elements 26 through 32 and 34 through 37) outside the crane wall are pressurized by ventilation openings through the crane wall into the fan compartments.

For each element in the TMD network the volume, initial pressure and initial temperature conditions are specified. The ice condenser elements have additional inputs of mass of ice, heat transfer area and condensate layer length. For each flow path between elements flow resistance is specified as a loss coefficient "K" or a fraction loss "L/D" or a combination of the two based on the flow area specified between elements. Friction factor, friction factor length and hydraulic diameter are specified for the friction loss.

Additionally, input for each flow path includes the area ratio (minimum area/maximum area) which is used to account for compressibility effects across flow path contractions. The code input for each flow path is the flow path length used in the momentum equation. The ice condenser loss coefficients have been based on the ¼-scale tests representative of the current ice condenser geometry. The test loss coefficient was increased to include basket roughness effects and to include intermediate and top deck pressure losses. The loss coefficient is based on removal of door port flow restrictors.

To better represent short term transients effects, the opening characteristics of the lower, intermediate, and top deck ice condenser doors have been modeled in the TMD code. The containment geometric data for the elements and flow paths used in the TMD code is confirmed to agree with the actual design by TVA and Westinghouse. An initial containment pressure of 0.3 psig was assumed in the analysis. Initial containment pressure variation about the assumed 0.3 psig value has only a slight affect on the initial pressure peak and the compression ratio pressure peak. TMD input data is given in Tables 6.2.1-6 and 6.2.1-7.

The reactor coolant blowdown rates used in these cases are based on the SATAN analysis of a double-ended rupture of either a hot or a cold leg reactor coolant pipe utilizing a discharge coefficient of 1.0. The models and assumptions used to calculate the short-term mass and energy releases are described in Reference [9]. Tables 6.2.1-23 and 6.2.1-24 present the mass and energy release data used for this analysis.

A number of analyses have been performed to determine the various pressure transients resulting from hot and cold leg reactor coolant pipe breaks in any one of the six lower compartment elements. The analyses were performed using the following assumptions and correlations:

1. Flow was limited by the unaugmented critical flow correlation.

2. The TMD variable volume door model, which accounts for changes in the volumes of TMD elements as the door opens, was implemented.

3. The heat transfer calculation used was based on performance during the 1973-1974 Waltz Mill test series. A higher value of the ELJAC parameter has been used and an upper bound on calculated heat transfer coefficients has been imposed[5].

4. One hundred percent moisture entrainment was assumed.

5. Compressibility effects due to flow area contractions were modeled.

6. A 15% reduction in flow area through the ice condenser was assumed to account for the effect of the flow area reduction due to frost and ice accumulation during the course of an operating cycle.

Figures 6.2.1-10 and 6.2.1-11 are representative of the typical upper and lower compartment pressure transients that result from a hypothetical double-ended rupture of a reactor coolant pipe for the worst possible location in the lower compartment of the containment; i.e., hot leg and cold leg breaks in Element 1.

Initial Pressures

Results of the analysis for the Watts Bar Plant are presented in Tables 6.2.1-8 through 6.2.1-11. The peak pressures and peak differential pressures resulting from hot and cold leg reactor coolant pipe breaks in each of the six lower compartment control volumes were calculated.

Table 6.2.1-8 presents the maximum calculated pressure peak for the lower compartment elements resulting from hot and cold leg double ended pipe breaks. Generally, the maximum peak pressure within a lower compartment element results when the pipe break occurs in that element. A hot leg break in Element 1 creates the highest pressure peak, also in Element 1, of 16.7 psig.

Table 6.2.1-9 presents the maximum calculated peak pressure in each of the ice condenser sections resulting from any pipe break location. The maximum peak pressure in each of the ice condenser sections is found in the lower plenum element of the section. The peak pressure was calculated to be 12.1 psig in Element 40.

Table 6.2.1-10 presents the maximum calculated differential pressures across the operating deck (divider barrier) between the lower compartment elements and the upper compartment. These values are approximately the same as the maximum calculated differential pressure across the lower crane wall between the lower compartment elements and the dead ended volumes surrounding the lower compartment. The peak differential pressure of 16.3 psi was calculated to be between Elements 1 and 25 for a hot leg break.

Table 6.2.1-11 presents the maximum calculated differential pressures across the upper crane wall between the upper ice condenser elements and the upper compartment. The peak differential of 8.8 psi pressure was calculated to be between Element 7-8-9 and 25 for a hot leg pipe break.

Consideration is given to the calculation of subcompartment pressures (and pressure differentials) for cases other than the design basis double ended reactor coolant pipe rupture in the lower compartment. Discussion of these analyses is treated in Section 6.2.1.3.9.

Sensitivity Studies

A series of TMD runs for D. C. Cook investigated the sensitivity of peak pressures to variations in individual input parameters for the design basis blowdown rate and 100 percent entrainment. This analysis used a DEHL break in Element 6 of D. C. Cook. Table 6.2.1-12 presents the results of this sensitivity study.

As part of the short-term containment pressure analysis of ice condenser units, the pressure response to both DEHL and DECL breaks are routinely considered for each of the loop compartments.

Choked Flow Characteristics

The data in Figure 6.2.1-12 illustrate the behavior of mass flow rate as a function of upstream and downstream pressures, including the effects of flow choking. The upper plot shows mass flow rate as a function of upstream pressure for various assumed values of downstream pressure. For zero back pressure (P_d = 0), the entire curve represents choked flow conditions with the flow rate approximately proportional to upstream pressure P_u. For higher back pressure, the flow rates are lower until the upstream pressure is high enough to provide choked flow. After the increase in upstream pressure is sufficient to provide flow chokings further increases in upstream pressure cause increases in mass flow rate along the curve for P_d = 0. The key point in this illustration is that flow rate continues to increase with increasing upstream pressure, even after flow choking conditions have been reached. Thus, choking does not represent a threshold beyond which dramatically sharper increases in compartment pressures could be expected because of limitations on flow relief to adjacent compartments.

The phenomenon of flow choking is more frequently explained by assuming a fixed upstream pressure and examining the dependence of flow rate with respect to decreasing downstream pressure. This approach is illustrated for an assumed upstream pressure of 30 psia as shown in the upper plot with the results plotted vs. downstream pressure in the lower plot. For fixed upstream conditions, flow choking represents an upper limit flow rate beyond which further decreases in back pressure do not produce any increase in mass flow rate.

Compression Ratio Analysis

As blowdown continues following the initial pressure peak from a double-ended cold leg break, the pressure in the lower compartment again increases, reaching a peak at or before the end of blowdown. The pressure in the upper compartment continues to rise from beginning of blowdown and reaches a peak which is approximately equal to the lower compartment pressure. After blowdown is complete, the steam in the lower compartment continues to flow through the doors into the ice bed compartment and is condensed.

The primary factor in producing this upper containment pressure peak and, therefore, in determining design pressure, is the displacement of air from the lower compartment into the upper containment. The ice condenser quite effectively performs its function of condensing virtually all the steam that enters the ice beds. Essentially, the only source of steam entering the upper containment is from leakage through the drain holes and other leakage around crack openings in hatches in the operating deck separating the lower and upper portions of the containment building.

A method of analysis of the compression peak pressure was developed based on the results of full-scale section tests. This method consists of the calculation of the air mass compression ratio, the polytropic exponent for the compression process, and the effect of steam bypass through the operating deck on this compression.

The compression peak pressure in the upper containment for the Watts Bar plant design is calculated to be 7.81 psig (for an initial air pressure of 0.3 psig). This compression pressure includes the effect of a pressure increase of 0.4 psi from steam bypass and also for the effects of the dead-ended volumes. The nitrogen partial pressure from the accumulators is not included since this nitrogen is not added to the containment until after the compression peak pressure has been reduced, which is after blowdown is completed. This nitrogen is considered in the analysis of pressure decay following blowdown as presented in the long term performance analysis using the LOTIC1 code. The following sections discuss the major parameters affecting the compression peak. Specifically they are: air compression, steam bypass, blowdown rate, and blowdown energy.

Air Compression Process Description

The volumes of the various containment compartments determine directly the air volume compression ratio. This is basically the ratio of the total active containment air volume to the compressed air volume during blowdown. During blowdown air is displaced from the lower compartment and compressed into the ice condenser beds and into the upper containment above the operating deck. It is this air compression process which primarily determines the peak in containment pressure, following the initial blowdown release. A peak compression pressure of 7.81 psig is based on the Watts Bar Plant design compartment volumes shown in Table 6.2.1-13.

Figure 6.2.1-13 shows the sensitivity of the compression peak pressure with different air compression ratios.

Methods of Calculation and Results

Full-Scale Section Tests

The actual Waltz Mill test compression ratios were found by performing air mass balances before the blowdown and at the time of the compression peak pressure, using the results of three full-scale special section tests. These three tests were conducted with an energy input representative of the plant design.

In the calculation of the mass balance for the ice condenser, the compartment is divided into two sub-volumes; one volume representing the flow channels and one volume representing the ice baskets. The flow channel volume is further divided into four sub-volumes. The partial air pressure and mass in each sub-volume is found from thermocouple readings by assuming that the air is saturated with steam at the measured temperature. From these results, the average temperature of the air in the ice condenser compartment is found, and the volume occupied by the air at the total condenser pressure is found from the equation of state as follows:

$$V_{a2} = \frac{M_{a2} \; R_a \; T_{a2}}{P_2} \qquad (1)$$

where:

V_{a2} = Volume of ice condenser occupied by air (ft^3)

M_{a2} = Mass of air in ice condenser compartment (lb)

T_{a2} = Average temperature of air in ice condenser (°F)

P_2 = Total ice condenser pressure (lb/ft^2)

R_a = Ideal gas constant

The partial pressure and mass of air in the lower compartment are found by averaging the temperatures indicated by the thermocouples located in that compartment and assuming saturation conditions. For these three tests, it was found that the partial pressure, and hence the mass of air in the lower compartment, was zero at the time of the compression peak pressure.

The actual Waltz Mill test compression ratio is then found from the following:

$$C = \frac{V_1 + V_2 + V_3}{V_3 + V_{a2}} \qquad (2)$$

where:

V_1 = Lower compartment volume (ft^3)
V_2 = Ice condenser compartment volume (ft^3)
V_3 = Upper compartment volume (ft^3)

The polytropic exponent for these tests is then found from the measured compression pressure and the compression ratio calculated above. Also considered is the pressure increase that results from the leakage of steam through the deck into the upper compartment.

The compression peak pressure in the upper compartment for the tests or containment design is then given by:

$$P = P_o \left(C_r\right)^n + \Delta P_{deck} \qquad (3)$$

where:

P_O	= Initial pressure (psia)
P	= Compression peak pressure (psia)
C_r	= Volume compression ratio
n	= Polytropic exponent
ΔP_{deck}	= Pressure increase caused by deck leakage (psi)

Using the method of calculation described above, the compression ratio is calculated for the three full-scale section tests. From the results of the air mass balances, it was found that air occupied 0.645 of the ice condenser compartment volume at the time of peak compression, or

$$V_{a2} = 0.645V_2 \qquad (4)$$

The final compression volume includes the volume of the upper compartment as well as part of the volume of air in the ice condenser. The results of the full-scale section tests (Figure 6.2.1-14) show a variation in steam partial pressure from 100% near the bottom of the ice condenser to essentially zero near the top. The thermocouples and pressure detectors confirm that at the time when the compression peak pressure is reached steam occupies less than half of the volume of the ice condenser. The analytical model used in defining the containment pressure peak uses upper compartment volume plus 64.5% of the ice condenser air volumes as the final volume. This 64.5% value was determined from appropriate test results.

The calculated volume compression ratios are shown in Figure 6.2.1-15, along with the compression peak pressures for these tests. The compression peak pressure is determined from the measured pressure, after accounting for the deck leakage contribution. From the results shown in Figure 6.2.1-15, the polytropic exponent for these tests is found to be 1.13.

Plant Case

For the Watts Bar design, the volume compression ratio is calculated using Equation 2, modeling the upper plenum as part of the upper compartment, and Table 6.2.1-13 as:

$$C_r = \frac{1,077,012}{692,818 + [0.645 \text{ x } 110,520]} \qquad (5)$$

$C_r = 1.4095$

The peak compression pressure, based on an initial containment pressure of 15.0 psia (0.3 psig), is then given by Equation 3 as:

$P_3 = 15.0\ (1.4095)^{1.13} + 0.4$

$P_3 = 22.507$ psia or 7.81 psig

This peak compression pressure includes a pressure increase of 0.4 psi from steam bypass through the deck (see Section 6.2.1.3.5).

Sensitivity to Blowdown Energy

The sensitivity of the upper and lower compartment peak pressure versus blowdown rate as measured from the 1974 Waltz Mill Tests is shown in Figure 6.2.1-16. This figure shows the magnitude of the peak pressure versus the amount of energy released in terms of percentage of RCS energy release rate.

Percent energy blowdown rate was selected for the plot because energy flow rate more directly relates to volume flow rate and therefore pressure. There are two important effects to note from the peak upper compartment pressure versus blowdown rate: (1) the magnitude of the final peak pressure in the upper compartment is low (about 9 psig) for the plant design DECL blowdown rate; (2) even an increase in this rate up to 141% of the blowdown energy rate produces only a small increase in the magnitude of this peak pressure (about 1 psi).

The major factor setting the peak pressure reached in the upper compartment is the compression of air displaced by steam from the lower compartment into the upper compartment. The lower compartment initial peak pressure shows a relatively low peak pressure of 12.9 psig for the design basis DECL blowdown rate, and even a substantial increase in blowdown energy rate (141% reference initial DECL) would cause an increase in initial peak pressure of only 3 psi. The peak pressure in the lower compartment is due mainly to flow resistance caused by displacement of air from the lower compartment into the upper compartment.

6.2.1.3.5 Effect of Steam Bypass

The sensitivity of the compression peak pressure to deck bypass is shown in Figure 6.2.1-17, which shows that an increase in deck bypass area of 50% would cause an increase of about 0.2 psi in final peak compression pressure. Also, it is important to note that the plant final peak compression pressure of 7.81 psig already includes a contribution of 0.4 psi from the plant deck bypass area of 5 ft^2.

This effect of deck leakage on upper containment pressure has been verified by a series of four special, full-scale section tests. These tests were all identical except different size deck leakage areas were used.

The results of these tests are given in Figure 6.2.1-18 which includes two curves of test results. Each curve shows the difference in upper compartment pressure between one test and another resulting from a difference in deck leakage area. One curve shows the increase in upper compartment pressure at the end of the boiler blowdown (after the compression peak pressure, at about 50 seconds in these tests), and the second curve shows the increase in upper compartment peak pressure (at about 10 seconds in these tests). It should be noted that the pressure at the end of the blowdown is less than the peak compression ratio pressure occurring at about 10 seconds for reference blowdown test.

The containment pressure increase due to deck leakage is directly proportional to the total amount of steam leakage into the upper compartment, and the amount of this steam leakage is, in turn, proportional to the amount of steam released from the boiler, less the inventory of steam remaining in the lower compartment. Notably, the increase in upper compartment compression peak pressure is substantially less than the upper compartment pressure increase at the end of blowdown, because the peak compression pressure occurs before the boiler has released all of its energy.

The calculated maximum pressure rise due to deck leakage (when all of the boiler energy release has occurred) is also shown in Figure 6.2.1-18. The slope of this curve is 0.095 psi/ft^2 for the tests and is equivalent to 0.107 psi/ft^2 for the plant design. The difference between the two coefficients is due to a small difference in upper compartment volume between the plant design and these tests.

As shown in Figure 6.2.1-18, the calculated curve for maximum pressure increase at the end of blowdown agrees closely with the measured curve at small deck leakage areas but deviates at larger leakage areas. This deviation apparently results from the condensation of upper compartment steam by the walls of the upper compartment and by the ice at the top of the condenser during the tests. Pressure would also be reduced by heat losses in a plant; however, for conservatism, no credit is taken for this effect. As demonstrated by tests, the compression peak pressure in the upper compartment occurs before the boiler releases all of its energy, and the measured increase in peak compression pressure due to increased deck leakage, is proportionately reduced. For the case of the plant design, the final peak compression pressure is conservatively assumed to occur when the reactor coolant system release is 75% of its total energy. This value is selected as a reference value, based on the results of a number of tests conducted with different blowdown rates and total energy releases, as shown in Figure 6.2.1-19. The actual deck leakage coefficient is therefore:

$$\frac{\Delta P_3}{A_{deck}} = 0.107 \text{ x } 0.75 = 0.080 \text{ psi}/\text{ft}^2$$

The divider barrier including the enclosures over the pressurizer, steam generators and reactor vessel, is designed to provide a reasonably tight seal against leakage. Holes are purposely provided in the bottom of the refueling cavity to allow water from sprays in the upper compartment to drain to the sump in the lower compartment. Potential leakage paths exist at all the joints between the operating deck and the pump access hatches and reactor vessel enclosure slabs. The total of all deck leakage flow areas is approximately 5 ft^2. The effect of this potential leakage path is small and is found to be:

$$\Delta P_{deck} = 5 \text{ x } 0.080 = 0.4 \text{ psi}$$

In the event that the reactor coolant system break flow is so small that it would leak through these flow paths without developing sufficient differential pressure (1 lb/ft^2) to open the ice condenser doors, steam from the break would slowly pressurize the containment. The containment spray system has sufficient capacity to maintain pressure well below design for this case.

The Watts Bar Nuclear Plant and the Sequoyah Nuclear Plant are geometrically very similar. Some differences between the two plants are the design pressure, spray flow rates, and a slight difference in thermal ratings. The fact that the spray flow rate is higher for the Sequoyah plant (4750 gpm versus 4000 gpm) is offset by Watts Bar's higher maximum internal pressure (15 psig versus 12 psig). The following discussion presents the deck leakage analysis performed for the Sequoyah plant. The purpose of this analysis is only to show the substantial margin which exists between the design deck leakage of 5 ft^2 and the tolerable deck leakage. The Sequoyah analysis which shows conservatism by a factor of 7, is more than sufficient for this purpose.

The method of analysis used to obtain the maximum allowable deck leakage capacity as a function of the primary system break size is as follows.

During the blowdown transient, steam and air flow through the ice condenser doors and also through the deck bypass area into the upper compartment. For the containment, this bypass, area is composed of two parts, a known leakage area of 2.2 ft^2 with a geometric loss coefficient of 1.5 through the deck drainage holes location at the bottom of the refueling canal and an undefined deck leakage area with a conservatively small loss coefficient of 2.5.

A resistance network similar to that used to TMD is used to represent 6 lower compartment volumes each with a representative portion of the deck leakage, and the lower inlet door flow resistance and flow area is calculated for small breaks that would only partially open these doors. The coolant blowdown rate as a function of time is used with this flow network to calculate the differential pressures on the lower inlet doors and across the operating deck.

The resultant deck leakage rate and integrated steam leakage into the upper compartment is then calculated. The lower inlet doors are initially held shut by the cold head of air behind the doors (approximately one pound per square foot). The initial blowdown from a small break opens the doors and removes the cold head on the doors. With the door differential removed, the door position is slightly open. An additional pressure differential of one pound per square foot is then sufficient to fully open the doors. The nominal door opening characteristics are based on test results.

One analysis conservatively assumed that flow through the postulated leakage paths is pure steam. During the actual blowdown transient, steam and air representative of the lower compartment mixture leak through the holes, thus less steam would enter the upper compartment. If flow were considered to be a mixture of liquid and vapor, the total leakage mass would increase, but the steam flow rate would decrease. The analysis also assumed that no condensing of the flow occurs due to structural heat sinks. The peak air compression in the upper compartment for the various break sizes is assumed with steam mass added to this value to obtain the total containment pressure. Air compression for the various break sizes is obtained from previous full-scale section tests conducted at Waltz Mill.

The allowable leakage area for the following reactor coolant system (RCS) break sizes was determined: DE, 0.6 DE, 3 ft^2, 10 inch diameter, 6 inch diameter, and 2 inch diameter, and 0.5 inch diameter. The allowable deck leakage area for the DE break was based on the test results previously discussed. For break sizes of 3 ft^2 and 0.6 DE, a series of deck leakage sensitivity studies were made to establish the total steam leakage to the upper compartment over the blowdown transient. This steam was added to the peak compression air mass in the upper compartment to calculate a peak pressure.

Air and steam were assumed to be in thermal equilibrium, with the air partial pressure increased over the air compression value to account for heating effects. For these breaks, sprays were neglected. Reduction in compression ratio by return of air to the lower compartment was conservatively neglected. The results of this analysis are shown in Table 6.2.1-14. This analysis is confirmed by Waltz Mill tests conducted with various deck leaks equivalent to over 50 ft^2 feet of deck leakage for the double-ended blowdown rate and is shown in Figure 6.2.1-20.

For breaks of 10 inch diameter and smaller, the effect of containment sprays was included. The method used calculates, for each time step of the blowdown, the amount of steam leaking into the upper compartment to obtain the steam mass in the upper compartment. This steam was mixed with the air in the upper compartment, assuming thermal equilibrium with air. The air partial pressure was increased to account for air heating effects. After sprays were initiated, the pressure was calculated based on the rate of accumulation of steam in the upper compartment.

This analysis was conducted for the 10 inch, 6 inch, and 2 inch break sizes, assuming one spray pump operated (4750 gpm at 100°F). As shown in Table 6.2.1-14, the 10-inch break is the limiting case for the given range of break sizes.

A second, more realistic, method was used to analyze the 10-inch, 6-inch, and 2-inch breaks. This analysis assumed a 30% air and 70% steam mix flowing through the deck leakage area. This is conservative considering the amount of air in the lower compartment during this portion of the transient. Operation of the deck fan increases the air content of the lower compartment, thus increasing the allowable deck leakage area. Based on the LOTIC code analysis, a structural heat removal rate of over 6000 Btu/sec from the upper compartment is indicated. Therefore, a steam condensation rate of 6 lbs/sec was used for the upper compartment. The results indicate that with one spray pump operating and a deck leakage area of 50 ft^2, the peak containment pressure is below design pressure.

The ½-inch diameter break is not sufficient to open the ice condenser inlet doors. For this break, the upper compartment spray is sufficient to condense the break steam flow.

In conclusion, it is apparent that there is a substantial margin between the design deck leakage area of 5 ft^2 and that which can be tolerated without exceeding containment design pressure. A preoperational visual inspection has been performed to ensure that the seals between the upper and lower containment have been properly installed.

6.2.1.3.6 Mass and Energy Release Data

Long-Term Loss-of-Coolant Accident Mass and Energy Releases

The evaluation model used for the long-term LOCA mass and energy release calculations is the WCOBRA/TRAC mass and energy release model described in Reference [20]. This methodology received the Final Safety Evaluation report from the NRC in August 2015.

The time history of conditions within an ice condenser containment during a postulated loss-of-coolant accident (LOCA) can be divided into two periods:

1. The initial reactor coolant blowdown, which for the largest assumed pipe break occurs within approximately 30 seconds.

2. The post blowdown phase of the accident which begins following the blowdown and extends several hours after the start of the accident.

LOCA Mass and Energy Release Phases

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

1. Blowdown: The period of time from accident initiation (when the reactor is at steady state operation) to the time that the lower plenum begins to re-pressurize after initial coolant evacuation.

2. Refill: The period of time when the reactor vessel lower plenum is being filled by accumulator and Emergency Core Cooling System (ECCS) water. The WCOBRA/TRAC code mechanistically calculates this phase.

3. Reflood: The period of time that begins when the water from the reactor vessel lower plenum enters the core and ends when the core is completely quenched.

4. Post-reflood: The period of time following the reflood phase. It is during this portion of the transient that (for the DECL and DEPS breaks) a two phase mixture exiting the core enters the steam generators, resulting in reverse heat transfer from the secondary side to the primary side. Heat transfer from the steam generator secondary metal to the fluid, and then from the fluid to the tubes, is accounted for in a mechanistic fashion.

   The WCOBRA/TRAC (WCAP-18204-P) methodology uses a single code for all phases of the LOCA transient trough the time of peak containment pressure.

Break Size and Location

Three distinct locations in the RCS loop can be postulated for pipe rupture:

1. Hot leg (between vessel and steam generator),

2. Cold leg (between pump and vessel), or

3. Pump suction (between SG and pump).

The WCAP-18204-P WCOBRA/TRAC methodology was used to apply the applicability of WCAP-17721-P to unit 2. Full double ended ruptures were analyzed in the cold leg and the pump suction leg, each with 1 and 2 trains of safety injection, to cover the spectrum of possible limiting break locations for Watts Bar Unit 2 in the context of the WCOBRA/TRAC methodology.

Full double ended double ruptures of the cold leg and pump suction leg behave similarly, or more specifically, both behave dissimilarly from a double ended rupture of the hot leg. This is because the hot leg break provides a direct vent path to containment for the core exit flow during the post blowdown phases. This has the effect of allowing the two phase mixture to bypass the steam generators, yielding significantly reduced integrated energy release in the long term. While the blowdown has the potential to be more severe for the hot leg break, ice condenser plants are inherently limited by the long term mass and energy releases after the ice bed is depleted. Therefore, the hot leg break has been excluded from the spectrum of runs. The double ended pump suction and double ended cold leg breaks both provide a mechanism to transport a two phase mixture to both the intact and broken steam generators, and so both have been analyzed with WCOBRA/TRAC.

Application of Single-Failure Criteria

An analysis of the effects of the single-failure criteria has been performed on the mass and energy release rates for the pump suction (DEPSG) break. An inherent assumption in the generation of the mass and energy release is that offsite power is lost. This results in the actuation of the emergency diesel generators, required to power the Safety Injection System. This is not an issue for the blowdown period, which is limited by the compression peak pressure.

The limiting minimum safety injection case has been analyzed for the effects of a single failure. In the case of minimum safeguards, the single failure postulated to occur is the loss of an emergency diesel generator. This results in the loss of one pumped safety injection train, that is, ECCS pumps and heat exchangers.

Basis of the Analysis

I. Significant Modeling Assumptions.

The following summarized assumptions were employed to ensure that the mass and energy releases were conservatively calculated, thereby maximizing energy release to containment:

1. Maximum expected operating temperature of the RCS at 100-percent full-power conditions; (619.1°F).

2. An allowance in temperature for instrument error and dead band was assumed on the vessel/core inlet temperature (+6.0°F)

3. Margin in volume of 3 percent (which is composed of a 1.6-percent allowance for thermal expansion, and a 1.4 percent allowance for uncertainty)

4. Core rated power of 3,411 MWt

5. Allowance for calorimetric error (+2.0 percent of power)

6. Steam Generator secondary side mass was biased conservatively high.

7. Initial fuel temperatures and thus core stored energy were based on late in life conditions that included the effects of fuel pellet thermal conductivity degradation.

8. An allowance for RCS initial pressure uncertainty (+70 psi)

9. A maximum containment backpressure equal to design pressure

10. Steam generator tube plugging leveling (0-percent uniform)

    a. Maximizes reactor coolant volume and fluid release

    b. Maximizes heat transfer area across the steam generators tubes

    c. Reduces coolant loop resistance, which reduces the delta-p upstream of the break and increases break flow

11. Decay heat was based on the ANS 1979 +2σ model.

II. Initial Conditions

Table 6.2.1-15 presents the System Parameters Initial Conditions utilized.

Thus, based on the previously noted conditions and assumptions, a bounding analysis of Watts Bar Nuclear Plant Unit 2 is made for the release of mass and energy from the RCS in the event of a LOCA.

<u>Mass and Energy Release Data</u>

<u>W</u>COBRA/TRAC consists of two primary source codes, COBRA-TF and TRAC-PD2. COBRA-TF is a three dimensional thermal hydraulic code that is used to model the vessel in detail, and TRAC-PD2 is a one dimensional code that is used to model loop piping, pumps, steam generators, and various boundary conditions. The <u>W</u>COBRA/TRAC computer code is currently used as the PWR ECCS evaluation model by Westinghouse; it is fully capable of calculating the thermal/hydraulic RCS response to a large pipe rupture. The use of <u>W</u>COBRA/TRAC for this application has been qualified by comparison with scalable test data covering the expected range of conditions and important phenomena. Modifications to <u>W</u>COBRA/TRAC were required to model the specifics of importance to the LOCA M&E analysis, and these updates are discussed below.

The WCOBRA/TRAC steam generator, modeled in accordance with the ECCS evaluation model, was shown to over predict the reverse heat transfer from the steam generators. Updates were made to more accurately calculate the steam generator cool down, this was validated by comparison to FLECHT-SEASET steam generator separate effects tests. The result is a computer code capable of modeling phenomena associated with the large break LOCA conditions that provides significant margin relative to the current WCAP10325PA methodology because the stored RCS and SG energy is released at a mechanistically calculated rate instead of forced out over a conservatively short duration.

Key updates include:

1. Updated the PCT nodding structure to include safety injection and accumulator injection in all loops.

2. Maximized the RWST temperature (105°F, technical specification maximum).

3. Applied accumulator upper limit pressure (704.7 psia), lower limit liquid volume (1005 ft^3), and maximum temperature (130°F).

4. RCS volume was increased by 3% for thermal expansion and measurement uncertainty.

5. SG tube plugging level was reduced to 0% (assumption maximizes RCS volume and flow/heat transfer area of SG tubes).

6. Increased the RCS temperatures to the high end of the operating range band and included uncertainty for a target Tavg of 594.2°F.

7. Increased the RCS initial pressure to 2320 psia (including uncertainties).

8. Increased the pressurizer liquid level, targeting maximum water volume of 1232 ft^3.

9. Increased core power to account for uncertainty at full power, targeting 3479.75 MWt.

10. Applied ANS 1979 +2σ decay heat.

11. Turned off fuel rod swelling model.

12. Used Baker-Just correlation for metal/water reaction.

13. Updated the steam generator noding structure per WCOBRA/TRAC M&E methodology.

14. Biased the steam generator secondary side volumes high by 3% to account for uncertainty.

15. Updated material properties to be consistent with the American Society of Mechanical Engineers (ASME) data including any applicable uncertainties.

The resulting model was used to calculate the mass and energy releases for the limiting break scenario, a double ended pump suction break with minimum safeguards. The WCOBRA/TRAC tool calculates all phases of the peak pressure LOCA transient, including blowdown, refill, reflood, and post reflood long term. The resulting blowdown and post blowdown mass and

energy releases are found in Table 6.2.1-16 and Table 6.2.1-19, respectively.

Decay Heat Model

ANS Standard 5.1[25] was used in the LOCA mass and energy release model for Watts Bar Unit 1 for the determination of decay heat energy. This standard was balloted by the Nuclear Power Plant Standards Committee (NUPPSCO) in October 1978 and subsequently approved. The official standard, Reference [25], was issued in August 1979.

The Table 6.2.1-20 lists the decay heat curve used.

Significant assumptions in the generation of the decay heat curve are the following:

1. The highest decay heat release rates come from the fission of U-238 nuclei. Thus, to maximize the decay heat rate a maximum value (8%) has been assumed for the U-238 fission fraction.

2. The second highest decay heat release rate comes from the fission of U-235 nuclei. Therefore, the remaining fission fraction (92%) has been assumed for U-235.

3. The factor which accounts for neutron capture in fission products has been taken directly from Table 10 of the standard.

4. The number of atoms of Pu-239 produced per second has been assumed to be 70% of the fission rate.

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

6. The fuel has been assumed to be at full power for 10^8 seconds.

7. 2σ uncertainty has been applied to the fission product decay. This accounts for a 98% confidence level.

Short-Term Mass and Energy Releases

The short-term mass and energy release models and assumptions are described in Reference [9]. The LOCA short-term mass and energy release data used to perform the containment analysis given in Sections 6.2.1.3.4 and 6.2.1.3.9 are listed below:

Section	Break Size and Location	Table
6.2.1.3.4	Double-Ended Cold Leg Guillotine Break Outside the Biological Shield	6.2.1-23
6.2.1.3.4	Double-Ended Hot Leg Guillotine Break Outside the Biological Shield	6.2.1-24

6.2.1.3.7 Accident Chronology

For a double-ended pump suction loss-of-coolant accident, the major events and their time of occurrence are shown in Table 6.2.1-25 for the minimum safeguards case.

6.2.1.3.8 Mass and Energy Balance Tables

Sources of Mass and Energy

The sources of mass and energy in the WCOBRA/TRAC M&E analysis are the following:

1. RCS water
2. Accumulator water
3. Pumped safety injection water
4. Decay heat
5. Core stored energy
6. RCS metal (including the SG tube metal)
7. Steam generator metal (including shell, wrapper, and internals)
8. Steam generator secondary fluid energy (liquid and steam)

All sources of mass and energy listed above are considered in the WCOBRA/TRAC portion of the analysis. The water in the RCS, accumulators, safety injection boundary conditions, and SG secondary is explicitly modeled. Core decay heat (including feedback effects during blowdown) is included in the WCOBRA/TRAC fuel rod model. The fuel rod model also includes an energy term to represent core stored energy. The RCS and SG metal, and the associated heat transfer from these sources, is modeled in the WCOBRA/TRAC analysis.

The methods and assumptions used to release various energy sources are given in Reference [20].

The consideration of the various energy sources in the mass and energy release analysis provides assurance that all available sources of energy have been included in this analysis. Therefore, the review guidelines presented in Standard Review Plan Section 6.2.1.3 have been satisfied.

6.2.1.3.9 Containment Pressure Differentials

Consideration is given in the design of the containment internal structures to localized pressure pulses that could occur following a loss-of-coolant accident or a main steam line break. If either type of pipe rupture were to occur in these relatively small volumes, the pressure would build up at a rate faster than the overall containment, thus imposing a differential pressure across the walls of the structures.

These subcompartments include the steam generator enclosure, pressurizer enclosure, and upper and lower reactor cavity. Each compartment is designed for the largest blowdown flow resulting from the severance of the largest connecting pipe within the enclosure or the blowdown flow into the enclosure from a break in an adjacent region.

The following paragraphs summarize the design basis calculations:

Steam Generator Enclosure

The worst break possible in the steam generator enclosure is a double-ended rupture of the steamline pipe at no load conditions. Based on an investigation of postulated break locations, the rupture is assumed to occur at the point where the steamline exits the steam generator. The blowdown for this break is given in Table 6.2.1-27a. The TMD computer code [4] using the compressibility factor and assuming unaugmented critical flow is used to calculate the short-term pressure transients. The nodalization of the steam generator enclosure where the break occurs is shown in Figure 6.2.1-81. Node 51 is the break element and has a flow path to the adjacent steam generator enclosure which is a mirror image of the enclosure where the break occurs. Both enclosures are nodalized in the same manner; their nodal network is shown in Figure 6.2.1-82 and their input data is given in Tables 6.2.1-27b and 6.2.1-27c. This input data assumes that the insulation remains intact. The loss coefficients were computed using Reference [12]. The maximum number of nodes used is based on the geometry of the system. The steam generator compartment is essentially symmetrical with no major obstructions to flow which would introduce asymmetric pressures. In addition, the flow path to the adjacent steam generator is at the top of the enclosure. Therefore, a significant differential pressure will not occur across the steam generator vessel. The balance of plant data is similar to that presented in Section 6.2.1.3.4.

The peak pressure differentials across the steam generator enclosure, the steam generator vessel, and the steam generator separator wall are given in Table 6.2.1-27d. Figure 6.2.1-83 shows the differential pressure transient between the break element and the upper compartment (Node 25). Figures 6.2.1-84 and 6.2.1-85 illustrate the differential pressure transient across the steam generator vessel. As Figures 6.2.1-84 and 6.2.1-85 show, the pressure differentials across the vessel are low and are due solely to inertial effects. The pressure versus time curve for the break element is given in Figure 6.2.1-86 and for the upper compartment (Node 25) in Figure 6.2.1-86a (Refer to Section 3.8.3.4.8 for steam generator compartment structural design description).

Pressurizer Enclosure

The worst break possible in the pressurizer enclosure is a double-ended rupture of the six-inch spray line. The rupture is assumed to occur at the top of the enclosure. The blowdown for this break is given in Table 6.2.1-28. The TMD computer code using the compressibility factor and assuming unaugmented critical flow is used to calculate the short-term pressure transient. The nodalization of the enclosure is shown in Figure 6.2.1-87. Node 51 is the break element. The input data is given in Table 6.2.1-29a. This input data assumes that the insulation remains intact. The loss coefficients were computed using Reference [12]. The maximum number of nodes used was based on the geometry of the system. The pressurizer compartment is essentially symmetrical with no major obstructions to flow which would introduce asymmetric pressures on the pressurizer vessel. The balance of plant data is similar to that presented in Section 6.2.1.3.4.

The peak pressure differentials across the pressurizer enclosure's walls, and across the pressurizer vessel are given in Table 6.2.1-29b. Figure 6.2.1-88 shows the pressure transient between the break element and the upper compartment (Node 25). As Figures 6.2.1-89 through 6.2.1-91 show, the significant pressure differential across the vessel are low, occur early, and are due solely to inertial effects. The pressure vs. time curve for the break element is given in Figure 6.2.1-92 (Refer to Section 3.8.3.4.9 for pressurizer compartment structural design description).

Reactor Cavity

The TMD computer code with the unaugmented homogeneous critical flow correlation and the isentropic compressible subsonic flow correlation was used to calculate pressure transients in the reactor cavity region.

Nodalization sensitivity studies were performed before the analysis was begun. The total number of nodes used varied from 6 to 68. In the 6-element model, no detail of the reactor vessel annulus was involved, and for that reason the model was discarded.

Subsequent model changes primarily involved greater detail in the reactor vessel annulus. First, the annulus was divided into two vertical and eight circumferential regions. Next, some additional detail was added to the region of the broken nozzle. The next changes were effected by increasing the model to three vertical and eight circumferential regions. The total integrated pressure in the reactor cavity changed only slightly because of the last change. The next change, to 68 elements, produced the model shown with detailed modeling around the nozzle sustaining the break. The additional elements from 48 to 52 are external to the reactor cavity (ice condenser). Additional elements were added to account for all real area changes in the immediate vicinity of the break (i.e., Elements 53 and 54 were added to model the broken loop pipe annulus and the broken loop inspection port, respectively).

The nodal scheme around the reactor vessel produces a very accurate post accident pressure profile because of its design. Element 3 is a small element inside the primary shield. It would contain internal flow losses due to turning and thus contain a pressure gradient if it were made larger. The four elements numbered 33, 34, 45, and 46 are made small to minimize internal pressure variation, and the elements farther from the break are made larger because pressure gradients are low in those regions.

Figure 6.2.1-27 illustrates the positions of some of the compartments. Figure 6.2.1-28 shows the flow path connections for the 68 element model. Figure 6.2.1-29 illustrates the general configuration of the reactor vessel annulus nodalization. In the model, the lower containment is divided into four loop compartments (21 to 24). The upper containment is represented by Compartment 32. The ice condenser is modeled as five elements (48 to 52), neglecting any flow distribution effects. The break simultaneously occurs in Elements 1 and 25, immediately surrounding the nozzle. The corresponding broken loop pipe annulus is represented by Element 53. The lower reactor cavity is modeled by Element 2, the upper reactor cavity by Element 47, and the remainder of the elements, as shown in Figure 6.2.1-29, model the reactor vessel annulus. Compartments 15, 42, and 16 are really adjoining Compartments 17, 43, and 18, respectively, and Compartment 13 is on the opposite side of the vessel from the assumed break. Element 54 represents the inspection port volume above the break.

A break limiting restraint restricts the break size. A 127 in^2 cold leg break is the limiting case break for the reactor cavity analysis. The mass and energy release rates are presented in Table 6.2.1-30. Tables 6.2.1-31 and 6.2.1-32 provide the volumes, flow paths, lengths, diameters, flow areas, resistance factors, and area ratios for the elements and their connections.

The inspection port plugs were assumed to be removed at the start of the accident. All insulation is assumed in place and uncrushed during the entire transient except for the insulation between the break and the reactor vessel annulus. This insulation was conservatively assumed to crush to zero thickness.

The loss of coefficient (k) values were determined by changes in flow area and by turns the flow makes in traveling from the centroid of the upstream node to the centroid of the downstream node. The k and f factors for each path were determined using methods from such references as "Flow of Fluids through Valves, Fittings, and Pipes" by the crane company and "Chemical Engineering" by J. M. Coulson and J. A. Richardson.

Figures 6.2.1-30 through 6.2.1-68 show representative pressure transients for the break compartments, the upper and lower reactor cavities, the inspection port volume and pipe annulus near the break, the upper containment and the reactor vessel annulus. These plots demonstrate that the pressure gradient is steep near the break location and is very gradual farther away from the break. This indicates that the model must be very detailed close to the break location, but less detail is required with increasing distance (Refer to Section 3.8.3.5.3 for reactor cavity structural design description).

6.2.1.3.10 Steam Line Break Inside Containment

Pipe Break Blowdowns - Spectra and Assumptions

A series of steam line breaks was analyzed to determine the most severe break condition for containment temperature and pressure response. The following assumptions were used in these analysis.

1. The following break types were evaluated.

   a. Double-ended 4.6 ft^2 ruptures occurring at the nozzle on one steam generator. Steam line flow restrictors in the steam generators limit the effective break area of a full double-ended pipe rupture to a maximum of 1.4 ft^2 per steam generator.

   b. The largest split break which will not generate the low steamline pressure signal for steamline isolation.

   c. Small split breaks of 0.6 and 0.35 ft^2.

2. Steam line isolation signals and feedwater line isolation signals are generated by either a low steam line pressure signal, high or high-high containment pressure signal, or high steam line pressure rate signal. An allowance of 8 seconds is assumed for steam line isolation including generation, processing, and delay of the isolation signal and valve closure. An allowance of 8.5 seconds is assumed for feedwater line isolation by the air-operated feedwater regulation valve or 13.5 seconds by the motor- operated main feedwater isolation valve including generation, processing, and delay of the isolation signal and valve closure.

3. Failure of a diesel generator is assumed in all cases. This results in the loss of one containment safeguards train resulting in minimum heat removal capability.

4. Blowdown from the broken steam line is assumed to be dry saturated steam.

5. Plant power levels of 100.6% and zero of nominal full-load power for DER, and split pipe ruptures at 100.6% and 30% of nominal full-load power.

6. Failure of a main steamline isolation valve (MSIV), failure of a feedwater regulation valve (FRV), failure of auxiliary feedwater runout control protection, and failure of a safety injection train are considered.

7. The auxiliary feedwater system is manually realigned by the operator after 10 minutes to terminate AFW to the faulted steam generator.

8. For the full double-ended ruptures, the main feedwater flow to the steam generator with the broken steam line was calculated based on an initial flow of 100% of nominal full power flow and a conservatively rapid steam generator depressurization.

<u>Break Flow Calculations</u>

1. Steam Generator Blowdown

   Break flows and enthalpies from the steam generators are calculated using the Westinghouse LOFTRAN code.[14] Blowdown mass and energy release are determined using the LOFTRAN code which includes effects of core power generation, main and auxiliary feedwater additions, engineered safeguards systems, reactor coolant system thick metal heat storage, and reverse steam generator heat transfer.

2. Steam Plant Piping Blowdown

   The contribution to the mass and energy releases from the secondary plant steam piping is included with the mass and energy release rates presented in Table 6.2.1-39. For all ruptures, the steam piping volume blowdown begins at the time of the break and continues at a uniform rate until the entire piping inventory is released. The flowrate is determined using the Moody correlation, the pipe cross-sectional area, and the initial steam pressure. Following the piping blowdown, reverse flow from the intact steam generators continues to simulate the reverse steam generator flow until steam line isolation.

Single-Failure Effects

1. Failure of the main steam isolation valve (located outside of containment) in the steam line with the break allows steam from all four main steam lines (downstream of the other main steam isolation valves which close) to flow out the break. The analysis accounts for this effect by including an allowance for additional mass and energy released through the break due to the volume of steam contained in the main steam lines. No additional steam is released through the break if the postulated single failure is a main steam isolation valve in another steam line not closing. In this case, the main steam isolation valve in the broken steam line does close and there is no backflow from the downstream piping to the break.

2. The additional mass inventory in the feedwater line that would not be isolated from the steam generator following main feedwater isolation can flash into steam and pass through the steam generator and exit out of the break. Failure of the air-operated feedwater regulation valve results in an increase of 5 seconds for main feedwater isolation by the motor-operated feedwater isolation valve, but also results in a decrease in the feedwater piping volume that is not isolated from the steam generator. The feedwater regulation valve closes in no more that 6.5 seconds and the feedwater isolation valve closes in no more than 11.5 seconds precluding any additional feedwater from being pumped into the steam generator.

3. Failure of the auxiliary feedwater runout control equipment would result in higher auxiliary feedwater flows entering the steam generator prior to realignment of the auxiliary feedwater system. For cases where the runout control operates properly, a constant auxiliary feed flow of approximately 1,500 gpm was assumed. This value was increased to 2,250 gpm for the 100% and 0% power cases and 2040 gpm for the 30% power cases to simulate a failure of the runout control.

4. Failure of a safety injection train results in less SI flow and will result in a greater return to power. For consistency, with the steam line break core response analysis, the steam line break mass and energy model conservatively assumes failure of a safety injection train.

Worst-Case Mass and Energy Releases

The following steam line break cases were determined to represent the worst case steamline break results.

1. Full double-ended rupture at 100.6% of nominal full power with a failure of a MSIV. This represents the limiting DER case in terms of calculated peak temperature.

2. A 0.6 ft^2 split break at 30% of nominal full power with a failure of the AFW runout control system. This represents the limiting SB case in terms of calculated peak temperature.

3. A 0.35 ft^2 split break at 30% at nominal full power with a failure of the AFW runout control system. This represents the limiting case SB case in terms of superheat temperature duration.

Mass and energy releases for these cases are listed in Table 6.2.1-39.

Maximum Containment Temperature Analysis for Steam Line Break

Following a steam line break in the lower compartment of an ice condenser plant, two distinct analyses must be performed. The first analysis, a short-term pressure analysis, has been performed with the TMD computer code (see Section 6.2.1.3.9). The second analysis, a long-term analysis, does not require the large number of nodes which the TMD analysis requires. The computer code which performs this analysis is the LOTIC-3 computer code.

The LOTIC-3 computer code was developed to analyze steamline breaks in an ice condenser plant. Details of the LOTIC-3 computer code are given in Reference [3]. It includes the capability to calculate superheat conditions, and has the ability to begin calculations from time zero. The LOTIC-3 computer code has been found to be acceptable for the analysis of steam line breaks[3] with the following restrictions.

1. Mass and energy release rates are calculated with an approved model.

2. Complete break spectrums are analyzed.

3. Convective heat flux calculations, as described in References [2] and [27] are performed for all break sizes.

One condensation model is used by the LOTIC-3 computer code. For all breaks, the conservative 0% condensate reevaporization and convective heat flux model is used.

Containment Transient Calculations

The following are the major input assumptions used in the LOTIC-3 steam line break analysis for the Watts Bar Nuclear Plant.

1. Minimum safeguards are employed, e.g., one of two spray pumps, and one of two air return fans.

2. A quantity of 2.125×10^6 lbs of ice is assumed for the steam line break cases to be initially in the ice condenser.

3. The boron injection tank remains installed without heat tracing, and the boric acid concentration is reduced to zero ppm (Table 6.2.1-40).

4. The air return fan is effective 10 minutes after the transient is initiated. Actual air return fan initiation can take place in 9 ± 1 minutes. Initiation as early as 8 minutes does not adversely affect the outcome of the analysis.

5. A uniform distribution of steam flow into the ice bed is assumed.

6. The initial conditions in the containment are a temperature of 120°F in the lower compartment, 120°F in the dead-ended compartment, a temperature of 85°F in the upper compartment, and a temperature of 32°F in the ice condenser. All volumes are at a pressure of 0.3 psig (see Table 6.2.1-13).

7. A containment spray pump flow of 4,000 gpm is conservatively used in the upper compartment. A diesel loading sequence for the containment sprays to energize and come up to full flow and head in 234 seconds was used in the analysis.

8. Containment structural heat sinks as presented in Table 6.2.1-1 were used. The material properties are given in Table 6.2.1-5.

9. The air return fan empties air at a rate of 40,000 ft^3/min from the upper to the lower compartments.

10. A series of large break cases (1.4 ft^2 double-ended ruptures) was run to determine the limiting large break case (Table 6.2.1-41). In addition, a series of small breaks was analyzed with LOTIC-3 at the full power and 30% power level (Table 6.2.1-42).

11. The mass and energy releases for the limiting breaks are given in Table 6.2.1-39. Since these rates are considerably less than the RCS double-ended breaks and their total integrated energy is not sufficient to cause ice bed melt out, the containment pressure transients generated for the RCS breaks will be more severe. However, since the steam line break blowdowns are superheated, the lower compartment temperature transients calculated in this analysis will be limiting. These temperature transients are given in Figures 6.2.1-69 through 6.2.1-74.

12. The heat transfer coefficients to the containment structures are based on the work of Tagami. An explanation of their manner of application is given in Reference [3]. The stagnant heat transfer coefficients were limited to 72 Btu/hr-ft^2. This corresponds to a steam-air ratio of 1.4 (according to the Tagami correlation). The imposition of this limitation is to restrict the use of the Tagami correlation within the range of steam-air ratios from which the correlation was derived.

The containment responses presented identify the limiting and most severe cases for the large double-ended ruptures and small split breaks.

Large Break

The limiting case among the double-ended ruptures, which yielded a calculated peak temperature of 324.4°F and a peak pressure of 10.3 psig, is the 1.4 ft^2 loop break at 100.6% of nominal full power with a failure of a main steamline isolation valve. Figure 6.2.1-69 provides the upper and lower compartment temperature transients, and Figure 6.2.1-70 illustrates the lower compartment pressure transient. Table 6.2.1-39 contains the mass and energy release rates for the above case.

Small Break

The most severe transient in terms of superheat temperature duration for the small break spectrum is the 0.35 ft^2, 30% nominal full power, with AFW pump runout protection failure. The temperature transient with a peak temperature of 325.0°F and peak pressure of 6.59 psig for the case is presented in Figure 6.2.1-71, and the pressure transient is provided in Figure 6.2.1-72. Table 6.2.1-39 provides the mass and energy release rates for this case.

The most limiting case in terms of peak calculated temperature is the 0.6 ft^2, 30% power, with AFW pump runout protection failure. This case resulted in a calculated peak temperature of 325.9°F and peak pressure of 6.84 psig. Figure 6.2.1-73 presents the temperature transient, and Figure 6.2.1-74 shows the pressure transient of the lower compartment. The mass and energy releases are provided in Table 6.2.1-39.

Tables 6.2.1-43 and 6.2.1-44 provide the overall results of the calculated peak temperatures for the large and small break spectrums, respectively.

6.2.1.3.11 Maximum Reverse Pressure Differentials

Following a postulated pipe break accident, the occurrence inside the ice condenser containment may be characterized by two distinct periods:

1. The initial blowdown, which occurs in approximately 10 seconds. During this period, the air initially in the lower compartment is swept into the upper compartment and the dead-ended compartment by the blowdown mass. Large mass and pressure gradients occur throughout the containment.

2. The depressurization and post-blowdown period which occurs after the end of the initial blowdown. During this period the pressure gradient within the four compartments (upper, lower, ice condenser, and dead-ended) is almost nonexistent. The shape of the pressure transient resembles that of the mass and energy releases. Pressure decreases as blowdown diminishes, followed by a slow increase sometime during the reflood.

The analysis for the first period will usually require the modeling of the containment into many nodes so that the non-uniformity of pressure and mass distribution may be properly represented. This has been done in the TMD code.

On the other hand, the analysis for the second period will only require the modeling of the containment by a four-compartment system. These calculations are performed by the LOTIC code.[1],[28]

The code options and features discussed are used in calculating ECCS back-pressure and reverse pressure differentials across the operating deck.

Basic Assumptions

1. The containment is assumed to be physically divided into four compartments: upper, lower, ice condenser, and dead-ended compartments. Each compartment is a control volume of uniform temperature, pressure and mass distribution. Steam is also assumed to be saturated in each control volume.

2. Flow between compartments is related to the pressure differential between the compartments by a flow resistance factor.

3. A two-sump model is assumed. Temperature is considered to be uniform in each sump.

Conservation Equations

For each control volume or compartment, the conservation equations of mass, energy, momentum, and volume, an ideal gas law for air, and the equation of state for saturated steam may be written:

1. Energy equation:

$$\frac{d}{dt}(M_a h_a + M_s h_s + M_c h_c) - \frac{(V_{as} + V_c)}{J}\frac{d(P_s + P_a)}{dt} + (mh)_{out} - (mh)_{in} = R_e$$

For the lower compartment:

R_e = [Rate of energy out of break]

+ [Rate of flow energy from accumulator in the form of steam, water, and nitrogen]

- [Rate of structural heat removal]

- [Rate of flow energy of sprays if applicable]

- [Rate of heat transfer to the sump]

- [Rate of heat removal by the ice condenser drain flow, if acting as a spray]

- [Rate of energy associated with the loss on condensate from atmosphere falling to floor]

+ [Net rate of flow energy from the dead-ended compartment]

For the upper compartment:

R_e = [Flow energy of the entering spray]

- [Structure heat removal rate]

- [Energy rate associated with condensate falling from atmosphere]

For the ice condenser:

R_e = [Structure heat removal rate]

- [Rate of heat transfer to the ice]

- [Energy rate associated with ice melt and steam condensate falling from atmosphere]

2. Conservation of steam and water masses:

$$\frac{dM_s}{dt} + \frac{dM_c}{dt} + (M_s)_{out} - (M_s)_{in} = R_s$$

For the lower compartment:

R_s = [Rate of flow out of the RCS]

+ [Rate of flow out of the accumulator in the form of steam and water]

+ [Flow rate of the entering spray if applicable]

- [Rate of condensate falling to the floor]

+ [Rate of steam flow from the dead-ended compartment]

For the upper compartment:

R_s = [Flow rate of the entering spray]

- [Rate of condensate falling to the floor]

For the ice condenser:

R_s = - [Rate of condensate falling to the floor]

3. Conservation of air mass:

$$\frac{dM_a}{dt} + (M_a)_{out} - (M_a)_{in} = R_a$$

For the lower compartment:

R_a = [Rate of nitrogen flow out of the accumulator]

+ [Rate of air flow out of the dead-ended compartment]

For the upper compartment and the ice condenser:

R_a = 0

4. Conservation of momentum:

$$P_I - P_J + \frac{1}{2}\left(\frac{K_{ij}}{A^2}\right)\frac{M_{IJ}^2}{\rho g_C}$$

5. Volume conservation:

$$\frac{dV_{as}}{dt} + \frac{dV_c}{dt} = R_v$$

For the lower compartment:

R_v = [Rate of increase in sump water volume]

For the upper compartment:

R_v = 0

For the ice condenser:

R_v = [Rate of increase in free volume due to ice melting]

6. Ideal gas law for air:

$$P_a V_{as} = M_a R_a T$$

7. Equations of state for saturated steam:

$$P_s = f_1(T),\ h_s = f_2(T),\ v_s = f_3(T)$$

For the dead-ended compartment, the structure heat removal is assumed to be negligible, and the conservation equations of energy and mass simplified to:

$$\frac{d}{dt}(M_a h_a + M_s h_s) - \frac{v}{J}\frac{d(P_s + P_a)}{dt} = \text{[Rate of energy flow from the lower compartment]}$$

$$\frac{dM_a}{dt} = \text{[Rate of air flow from the lower compartment]}$$

$$\frac{dM_s}{dt} = \text{[Rate of steam flow from the lower compartment]}$$

Method of Solution

The preceding equations were linearized and programmed for simultaneous solutions using the standard Gauss-Jordan reduction method. For each time step, the solutions are the rates of increase of mass and pressure for each constituent in each compartment, and the flow rates between the compartments. These rates are used to control the time step so that total change of the compartment conditions in each time step can be controlled. This assures more accurate and stable solutions.

Structure Heat Transfer

The standard Westinghouse ECCS containment structural heat transfer model is applied to this code. This model assumes one dimensional conduction heat transfer in the structure and uses film heat transfer coefficient based primarily on the work of Tagami. The Tagami correlation for the film heat transfer may be written as:

$$H_{max} = 75\left[\frac{E}{t_p V}\right]^{0.6} \tag{1}$$

$$H = H_{max} \frac{\sqrt{t}}{\sqrt{t_p}} \quad \text{for } 0 \leq t \leq t_p \tag{2}$$

$$H = H_{stag} + [H_{max} - H_{stag}]\ e^{-0.5[t - t_p]} \quad \text{for } t_p \leq t \tag{3}$$

where:

$$H_{stag} = 2 + 50\ X \tag{4}$$

For this application, we have found it is useful to relate the "coolant energy transfer", (E/t_pV), to containment conditions. This may be done by writing:

$$\frac{E}{t_p V} = \frac{M_s \bar{h}_s + M_f \bar{h}_f}{t_p V} = \frac{1}{t_p V_s}\left(\bar{h}_s + \frac{M_f}{M_s}\bar{h}_f\right) \qquad (5)$$

where h_s, h_f, and ν_s are respectively the enthalpies of saturated steam and water, and the specific volume of steam, at t_p, the time when the peak containment pressure is reached.

Equations (1) through (5) are used for the lower compartment structure calculations. For the upper compartment, only the stagnant heat transfer correlation of Equation (4) is used because of little steam penetration into the upper compartment even during the initial blowdown period.

Ice Condenser Heat Transfer

The transfer of heat from steam to ice which results in the simultaneous occurrence of steam condensation and ice melting is a complex mechanism.

During the initial blowdown period when high temperature blowdown steam and water hits the bottom of ice columns, and then flows over the ice surface, turbulent condensation results. During this period the heat transfer rate is strongly dependent on the thickness of the liquid film which separates the high temperature blowdown masses from the ice. This liquid film is composed of steam condensate and ice melt. On the macroscopic scale, this is the only heat transfer resistance and the effectiveness of the ice condenser is determined by the rate which this liquid film may be withdrawn. A semi-empirical model for the ice condenser heat transfer during this period is available and has been used successfully in the TMD code. The LOTIC code is not intended to duplicate this effort. Instead mass and energy balances are used to calculate the total ice melting during this period. Following the initial blowdown period, there is a transition period when the blowdown mass and energy rates are decreasing rapidly and the containment atmosphere as a whole is losing internal energy. Depressurization and decreasing compartment temperature generally characterize this transition period. As the containment conditions lapse into a much more stable and slowly changing pace after the transition period, the blowdown from the broken pipe is almost drawing to an end. Flow in the ice condenser is now at a rate which is almost negligible compared to that in the initial blowdown period. Temperature in the ice condenser atmosphere has also decreased. Thus, heat transfer is governed by combining natural convection and steam diffusion through an almost stagnant atmosphere. Due to the large air content, the resistance to diffusion is large. Therefore, most of the temperature difference between the free-steam steam-air mixture and the ice occurs between the free-steam and the free surface of the liquid film. Temperature difference across the liquid film is now comparatively small.

Due to the loss of dominance for the liquid film resistance in the overall heat transfer mechanism, it is not surprising for Yen, Zender, Zavohik, and Tien[11] to conclude that ice melting has very little effect on the overall heat transfer coefficient for condensation-melting heat transfer in the presence of a substantial air concentration. From this, we may therefore treat the ice as if it were simply a cold structure and use Equation (4) to calculate the heat transfer coefficient after the transition period.

During the transition period, it is plausible to assume that the ice condenser is capable of maintaining its internal energy by condensing any excess energy which flow into the ice condenser.

Special Code Capabilities in Response to Previous NRC Concerns

1. Heat removal from the lower compartment by the ice condenser drain may be accounted for by input of a spray-like efficiency.

2. Heat transfer between the lower compartment atmosphere and the sump surface can also be taken into account.

Drains are provided at the bottom of the ice condenser compartment to allow the melt/condensate water to flow out of the compartment during a loss of coolant accident. In the modified LOTIC code, a calculation of the flow rate at which water leaves these ice condenser drains is included. The solution was reached by using the hydraulic incompressible flow equations commonly found in the literature for both filled pipe flow and fall (weir) flow conditions and at any point in time using the minimum flow rate calculated by the two methods. The filled pipe flow equation employed was a simplified Bernoulli balance:

$$Z_1 = \frac{V_2^2}{2g} + h_f + Z_2$$

where:

Z = Elevation

V = Velocity

g = Gravitational constant

ρ = Density

$$h_f = \frac{fl}{sd} \bullet \frac{V_2^2}{2} g$$

Subscripts 1 and 2 represent conditions at the inlet and outlet of the drain, respectively.

The area of the ice condenser sump was taken to be 3170 ft^2, and the height of the door sill to be 8.75 inches. After calculating the velocity from the previous equation, the mass flow rate can be calculated from

$$\dot{m} = \rho V_2 A_2$$

Since a filled pipe flow condition may not exist during the entire post accident transient, a calculation of the draining rate based on the existence of a fall flow phenomena was included. The corresponding equations are outlined below,

$$Q = \frac{2}{3} \, 2g \, (H_e^{3/2} - h_1^{3/2})$$

where:

Q = Discharge per foot of width (ft^3/sec-ft)

H_e = Energy of fluid upstream of the fall

h_1 = Energy of the fluid at the fall edge minus the flowing height

$D_1 = 0.643 \, D_c$

$h_1 = H_e - D_1$

By assuming the approach velocity equals zero and through substitution, we arrive at the simplified equation:

$$Q = \frac{2}{3} 2g \, [D_c^{3/2} - (0.357 \, D_c)^{3/2}]$$

or

$$Q = 4.2088 \, D_c^{3/2}$$

where:

D_c = ft.

Calculation of Maximum Reverse Pressure Differential

The computer model previously described was used to calculate the reverse differential pressure across the operating deck. In order to calculate a maximum reverse differential pressure the following assumptions were made:

1. The dead-ended compartment volumes adjacent to the lower compartment (fan and accumulator rooms, pipe trenches, etc.) were assumed to be swept of air during the initial blowdown. This is a very conservative assumption, since this will maximize the air mass forced into the upper ice bed and upper compartment thus raising the compression pressure. In addition, it will minimize the mass of the noncondensables in the lower compartment. With this modeling the dead-ended volume is included with that of the lower compartment (see Figure 6.2.1-75), resulting in a 3-volume simulation of the containment.

2. The minimum containment temperatures are assumed in the various subcompartments. This will maximize the air mass forced into the upper containment. It will also increase the heat removal capability of the cold lower compartment structures.

3. An RWST temperature of 100°F is assumed. This will help raise the upper containment temperature and pressure higher for a longer period of time. The current maximum RWST temperature is 105°F (See Table 6.2.1-37).

4. The upper containment spray flowrates used were runout flows.

5. Containment spray to the upper compartment was assumed to start at 25 seconds. An early start time is conservative in that it raises the upper compartment temperature and pressure when the air mass in the upper compartment is at its highest value. Containment spray initiation to the upper containment is tabulated in Table 6.2.1-25. An increased delay should have no negative effect on the maximum reverse pressure differential.

6. The containment geometry is the same as that used in the minimum pressure analysis for ECCS purposes. (See Tables 6.2.1-33 through 6.2.1-36.)

7. The Westinghouse ECCS model (see WCAP-8339) was used for heat transfer to the structure.

8. The mass and energy releases used are based on the analysis presented in WCAP-8479.

9. Ice condenser doors are assumed to act as check valves, allowing flow only into the ice condenser.

10. The loss coefficient (k/A^2) of the deck fins for air flow from the upper to the lower compartment was taken to be 0.0072 ft^{-4}. This value was based on the capabilities of the fans while running. With the fans not running the loss coefficient would be 0.0278 $ft.^{-4}$

With these assumptions the maximum reverse pressure differential across the operating deck was calculated to be 0.65 psi. The following plots have been provided:

Figure 6.2.1-76 which shows upper and lower compartment pressures.

Figure 6.2.1-77 which shows upper and lower compartment temperatures.

Figure 6.2.1-78 which shows upper to lower compartment flowrates.

Parametric studies have been made with this model. Various effects have been investigated to determine changes in the maximum reverse pressure differential. Table 6.2.1-37 gives some of these studies with their results. For Case 6, Figures 6.2.1-79 and 6.2.1-80 give plots similar to Figures 6.2.1-76 and 6.2.1-77. Presented in Table 6.2.1-38, also for Case 6, are the sump temperature and the steam exit flow from the ice condenser, both as a function of time.

Significant margin exists between the design reverse differential pressures across the operating deck and the ice condenser lower inlet doors, and those calculated pressures presented in Table 6.2.1-37.

Nomenclature

SYMBOL	DESCRIPTION
A	Flow area
E	Total energy
J	Conversion constant, 778 ft-lbf/Btu
K	Flow resistance factor
M	Mass
P	Pressure
R	Gas constant
R	Rate
T	Temperature
V	Volume

g_c	Conversion constant, 32.2 ft-lbm/lbf-sec^2
h	Enthalpy
m	Mass flow rate between two compartments
t	Time
x	Steam-air ratio
v	Specific volume
p	Density

SUBSCRIPT

a	Air
as	Air and steam
c	Suspended or entrained water
e	Energy
i	i-th compartment
j	j-th compartment
ij	from i-th compartment to j-th compartment
s	Steam

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TABLE 6.2.1-1 (Sheet 1 of 2)

STRUCTURAL HEAT SINKS

	Area(ft^2)	Thickness (ft)	Material
A. Upper Compartment			
1. Operating Deck			
Slab 1	4,880	1.066	Concrete
Slab 2	18,280	0.0055	Paint
		1.4	Concrete
Slab 3	760	0.0055	Paint
		1.5	Concrete
Slab 4	3,840	0.0208	Stainless Steel
		1.5	Concrete
2. Shell and Misc			
Slab 5	56,331	0.001	Paint
		0.079	Carbon Steel
B. Lower Compartment			
1. Operating Deck, Crane Wall, and Interior Concrete			
Slab 6	31,963	1.43	Concrete
2. Operating Deck			
Slab 7	2,830	0.0055	Paint
		1.1	Concrete
Slab 8	760	0.0055	Paint
		1.75	Concrete
3. Interior Concrete and Stainless Steel			
Slab 9	2,270	0.0208	Stainless Steel
		2.0	Concrete
4. Floor*			
Slab 10	15,921	0.0055	Paint
		1.6	Concrete
5. Misc. Steel			
Slab 11	28,500	0.001	Paint
		0.0656	Carbon Steel

TABLE 6.2.1-1 (Sheet 2 of 2)

STRUCTURAL HEAT SINKS

	Area (Ft^2)	Thickness (Ft)	Material
C. Ice Condenser			
1. Ice Baskets			
Slab 12	149,600	0.00663	Carbon Steel
2. Lattice Frames			
Slab 13	75,865	0.0217	Carbon Steel
3. Lower Support Structure			
Slab 14	28,670	0.0587	Carbon Steel
4. Ice Condenser Floor			
Slab 15	3,336	0.0055 0.333	Paint Concrete
5. Containment Wall Panels and Containment Shell			
Slab 16	19,100	1.0 0.0625	Steel & Insulation Steel Shell
6. Crane Wall Panels and Crane Wall			
Slab 17	13,055	1.0 1.0	Steel & Insulation Concrete

*In contact with sump.

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TABLE 6.2.1-2
UNIT 1

PUMP FLOW RATES VS. TIME

Time After Safeguards Initiation (Sec)	ECCS Flow To core (RWST) (gpm)	Spray (Flow) (gpm)	RHR Spray (Flow) (gpm	ECCS Flow To core (Sump) (gpm)	Comments
0	0	0	0	0	“S” - Signal
11.9	0	0	0	0	
12.0	358.9	0	0	0	CC Pump Start
16.9	359.9	0	0	0	
17.0	942.3	0	0	0	SI Pump Start
21.9	942.3	0	0	0	
22.0	4699.8	0	0	0	RHR Pump Start
233.9	4699.8	0	0	0	
234.0	4699.8	4000	0	0	Containment Spray Start
1631.2	4699.8	4000	0	0	
1631.3	4699.8	4000	0	3757.5	RHR Switchover to Sump
1707.9	4699.8	4000	0	3757.5	
1708.0	0	4000	0	3757.5	CCP/SI Pump Switchover
3339.9	0	4000	0	3757.5	
3340.0	0	0	0	3757.5	CS Pump Stopped
3459.9	0	0	0	3757.5	
3460.0	0	4000 (Sump)	0	3757.5	CS Pump Switchover
3848.2	0	4000 (Sump)	0	3757.5	
3848.3	0	4000 (Sump)	1475	1855	RHR alignment for Auxiliary CS
End of Transient	0	4000 (Sump)	1475	1855	

WBN

TABLE 6.2.1-2 (Sheet 1 of 2)
UNIT 2

PUMP FLOW RATES VS. TIME

	Time After Safeguards Initiation (sec)	CCP Flow (gpm)	HHSI Spray Flow (gpm)	RHR Spray (Flow) (gpm	CSS Pump Flow (gpm)	RHR Spray Flow (gpm)	Sump Flowrate (gpm)	RWST Flowrate (gpm)
SI Signal Initiated	0	0	0	0	0	0	0	0
CCP Starts and Reaches Rated Flow	12	560	0	0	0	0	0	560
HHSI Pump Starts and Reaches Rated Flow	17	560	675	0	0	0	0	1235
RHR Pump Starts and Reaches Rated Flow	22	560	675	5500	0	0	0	6735
CSS Pump Starts and Reaches Rated Flow	234	560	675	5500	4950	0	0	11685
RWST Low Level Alarm, Automatic Realignment of RHR	1207.3	560	675	5500	4950	0	0	11685

TABLE 6.2.1-2 (Sheet 2 of 2)
UNIT 2

PUMP FLOW RATES VS. TIME

	Time After Safeguards Initiation (sec)	CCP Flow (gpm)	HHSI Spray Flow (gpm)	RHR Spray (Flow) (gpm	CSS Pump Flow (gpm)	RHR Spray Flow (gpm)	Sump Flowrate (gpm)	RWST Flowrate (gpm)
Sump Valves Open, RWST to RHR Valves Close	1267.3	560	675	2750	4950	0	2750	6185
HHSI & CCP Suction Valves from RHR Discharge Open	1344.3	560	675	2750	4950	0	2750	4950
Low-Low RWST Level Alarm	2588.7	560	675	2750	4950	0	2750	4950
Shut Off CSS Pump	2598.7	560	675	2750	0	0	2750	0
Restart CSS Pump	2718.7	560	675	2750	0	0	2750	0
Start RHR Spray	≥3600	560	675	2750	4950	2000	2750	0

TABLE 6.2.1-3

ENERGY BALANCES

Sink	Approx. End of Blowdown
	(t=10.0 sec)
	(In Millions of BTUs)
Ice Heat Removal*	208.42
Structural Heat Sinks*	17.41
RHR Heat Exchanger Heat Removal*	0
Spray Heat Exchanger Heat Removal*	0
Energy Content of Sump**	198.42
Ice Melted (Pounds) (10^6)	0.686

* Integrated Energies

** Energy Content of Sump Includes Active and Inactive Regions

TABLE 6.2.1-4

ENERGY BALANCES

Sink	Approx. Time of Ice Melt Out	Approx. Time of Peak Pressure
	(t~5875 sec)	(t~8959 sec)
	(In Millions of BTUs)	
Ice Heat Removal*	592.3	592.3
Structural Heat Sinks*	98.50	123.33
RHR Heat Exchanger Heat Removal*	73.47	124.15
Spray Heat Exchanger Heat Removal*	108.18	206.34
Energy Content of Sump**	612.09	631.38
Ice Melted (pounds) (10^6)	2.26	2.26

* Integrated Energies

** Energy Content of Sump Includes Active and Inactive Regions

TABLE 6.2.1-5

MATERIAL PROPERTY DATA

UNIT 1

Material	Thermal Conductivity Btu/hr-ft -°F	Volumetric Heat Btu/ft^3-°F
Paint on Steel	0.21	19.9
Paint on Concrete	0.083	39.9
Concrete	0.8	31.9
Stainless Steel	9.4	53.68
Carbon Steel	26.0	53.9
Carbon Steel (located in ice condenser compartment)	26.0	56.4
Insulation on Steel	0.15	2.75

UNIT 2
LOTIC1 MODEL

Material	Thermal Conductivity Btu/hr-ft -°F	Volumetric Heat Btu/ft^3-°F
Paint on Steel	0.21	19.9
Paint on Concrete	0.083	39.9
Concrete	0.8	31.9
Stainless Steel	9.4	53.68
Carbon Steel	26.0	53.9
Carbon Steel *	26.0	56.4
Concrete*	0.8	28.8
Insulation on Steel (containment walls)*	0.15	2.75
Insulation of steel (crane walls)*	0.20	3.663

* Located in Ice Condenser Compartment

TABLE 6.2.1-6

TMD INPUT FOR WATTS BAR

Element	Volume (ft^3)	P_{Steam} (psia)	P_{Air} (psia)	Initial Temperature (°F)
1	28700	0.3	14.7	120
2	36800	0.3	14.7	120
3	70200	0.3	14.7	120
4	38800	0.3	14.7	120
5	36800	0.3	14.7	120
6	25114	0.3	14.7	120
25	651000	0.3	14.7	120
26	11700	0.3	14.7	120
27	17900	0.3	14.7	120
28	11200	0.3	14.7	120
29	18700	0.3	14.7	120
30	11200	0.3	14.7	120
31	18000	0.3	14.7	120
32	10100	0.3	14.7	120
33	15300	0.3	14.7	120
34	13000	0.3	14.7	120
35	4400	0.3	14.7	120
36	4400	0.3	14.7	120
37	9300	0.3	14.7	120
50	1400	0.3	14.7	120

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TABLE 6.2.1-7 (Sheet 1 of 2)

TMD FLOW INPUT DATA FOR WATTS BAR

Flow Path Element to Element	Flow Path Length (ft)	Flow Area (ft^2)	Loss Coefficient K	Area Ratio a/A
1 to 33	6.5	22	1.5	0.048
2 to 27	3.5	48	4.2	0.027
3 to 33	10.2	64	1.5	0.048
4 to 33	7.9	44	1.5	0.048
5 to 31	3.5	42	4.2	0.027
6 to 33	5.7	16	1.5	0.048
26 to 27	9	23	2.7	0.067
27 to 3	9.3	46	4.2	0.027
28 to 27	9	23	2.7	0.067
29 to 36	3.7	15	3	0.044
30 to 31	9	23	2.7	0.067
31 to 6	11	58	4.2	0.027
32 to 31	9	23	2.7	0.067
33 to 5	7.8	36	1.5	0.048
34 to 26	6.6	59	1.5	0.171
35 to 28	2.8	17	1.5	0.049
36 to 30	2.8	17	1.5	0.049
37 to 32	3.2	23	1.5	0.067
50 to 4	3.6	1.6	1.5	0.002
50 to 4	3.9	2.5	1.5	0.002
50 to 30	3.8	6.8	1.5	0.067

TABLE 6.2.1-7 (Sheet 2 of 2)

TMD FLOW INPUT DATA FOR WATTS BAR

Flow Path Element to Element	Flow Path Length (ft)	Flow Area (ft^2)	Loss Coefficient K	Area Ratio a/A
1 to 2	17.5	550	0.33	0.43
2 to 3	24.2	550	0.33	0.43
3 to 4	22.3	600	0.33	0.47
4 to 5	19.7	550	0.33	0.43
5 to 6	17.2	550	0.33	0.43
6 to 1	29.4	140	1.32	0.09
26 to 32	71	126	1.6	0.843
27 to 1	6.9	60	4.2	0.027
28 to 26	80	146	0.5	0.977
29 to 35	3.8	15	3	0.044
30 to 28	51	81	1.6	0.542
31 to 4	9.3	44	4.2	0.027
32 to 30	80	146	0.5	0.977
33 to 2	8.1	38	1.5	0.048
34 to 27	4.5	17	3	0.049
35 to 27	3.7	15	3	0.044
36 to 31	3.1	10	3	0.029
37 to 31	3.4	10	3	0.029
40 to 1	10.36	121.9		
41 to 2	10.36	144.0		
42 to 3	10.36	288.0		
43 to 4	10.36	199.4		
44 to 5	10.36	155.1		
45 to 6	10.36	155.1		

TABLE 6.2.1-8

CALCULATED MAXIMUM PEAK PRESSURES IN LOWER
COMPARTMENT ELEMENTS ASSUMING UNAUGMENTED FLOW AND 15% ICE CONDENSER FLOW BLOCKAGE

Element	1	2	3	4	5	6	
Peak Pressure (psig)	14.9	12.1	11.1	11.0	11.8	13.5	DECL - 100% Ent.
Peak Pressure (psig)	16.7	12.5	11.3	11.3	12.1	14.6	DEHL - 100% Ent.

TABLE 6.2.1-9

CALCULATED MAXIMUM PEAK PRESSURES IN THE ICE CONDENSER COMPARTMENT ASSUMING UNAUGMENTED FLOW AND 15% ICE CONDENSER FLOW BLOCKAGE

Element	40	41	42	43	44	45	
Peak Pressure (psig)	10.8	8.7	8.1	8.0	8.2	9.8	DECL - 100% Ent.
Peak Pressure (psig)	12.1	9.3	8.1	8.1	9.3	10.9	DEHL -100% Ent.

TABLE 6.2.1-10

CALCULATED MAXIMUM DIFFERENTIAL PRESSURES ACROSS THE
OPERATING DECK OR LOWER CRANE WALL (ELEMENT 25) ASSUMING UNAUGMENTED FLOW
AND 15% ICE CONDENSER FLOW BLOCKAGE

Element	1	2	3	4	5	6	
Peak P (psi)	14.2	10.0	8.3	8.6	9.9	12.7	DECL - 100% Ent.
Peak P (psi)	16.3	12.1	9.5	9.4	11.8	14.3	DEHL - 100% Ent.

TABLE 6.2.1-11

CALCULATED MAXIMUM DIFFERENTIAL PRESSURES
ACROSS THE UPPER CRANE WALL (ELEMENT 25) ASSUMING UNAUGMENTED FLOW
AND 15% ICE CONDENSER FLOW BLOCKAGE

Element	7-8-9	10-11-12	13-14-15	16-17-18	19-20-21	22-23-24	
Peak P (psi)	7.2	5.7	5.2	4.8	6.0	7.2	DECL - 100% Ent.
Peak P (psi)	8.8	6.7	6.0	5.1	6.9	8.4	DEHL - 100% Ent.

TABLE 6.2.1-12 (Sheet 1 of 2)

SENSITIVITY STUDIES FOR D. C. COOK PLANT

PARAMETER	CHANGE MADE FROM BASE VALUE(1)	CHANGE IN OPERATING DECK ΔP(1)	CHANGE IN PEAK PRESSURE AGAINST THE SHELL(1)
Blowdown	+ 10%	+ 11%	+ 12%
Blowdown	- 10%	- 10%	- 12%
Blowdown	- 20%	- 20%	- 23%
Blowdown	- 50%	- 50%	- 53%
Break Compartment Inertial Length	+ 10%	+ 4%	+ 1%
Break Compartment Inertial Length	- 10%	- 4%	- 1%
Break Compartment Volume	+ 10%	- 2%	- 1%
Break Compartment Volume	- 10%	+ 2%	+ 1%
Break Compartment Vent Areas	+ 10%	- 6%	- 5%
Break Compartment Vent Areas	- 10%	+ 8%	+ 5%
Door Port Failure in Break Compartment	one door port fails to open	+ 1%	- 1%
Ice Mass	+ 10%	0	0
Ice Mass	- 10%	0	0
Door Inertia	+ 10%	+ 1%	0
Door Inertia	- 10%	- 1%	0
All Inertial Lengths	+ 10%	+ 5%	+ 4%
All Inertial Lengths	- 10%	- 5%	- 3%
Ice Bed Loss Coefficients	+ 10%	0	0

TABLE 6.2.1-12 (Sheet 2 of 2)

SENSITIVITY STUDIES FOR D. C. COOK PLANT

PARAMETER	CHANGE MADE FROM BASE VALUE[1]	CHANGE IN OPERATING DECK ΔP[1]	CHANGE IN PEAK PRESSURE AGAINST THE SHELL[1]
Ice Bed Loss Coefficients	- 10%	0	0
Entrainment Level	0% Ent	- 27%	- 11%
Entrainment Level	30% Ent	- 19%	- 15%
Entrainmnet Level	50% Ent	- 13%	- 12%
Entrainment Level	75% Ent	- 6%	- 6%
Lower Compartment Loss Coefficients	+ 10%	0	0
Lower Compartment Loss Coefficients	- 10%	0	0
Cross Flow in Lower Plenum	low estimate of resistance	0	- 7%
Cross Flow in Lower Plenum	high estimate of resistance	0	- 3%
Ice Condenser Flow Area	+ 10%	0	- 3%
Ice Condenser Flow Area	- 10%	0	+ 4%
Ice Condenser Flow Area	+ 20%	0	- 6%
Ice Condenser Flow Area	- 20%	0	+ 8%
Initial Pressure in Containment	+ 0.3 psi	+ 2%	+ 2%
Initial Pressure in Containment	- 0.3 psi	- 2%	- 2%
Initial Ice Bed Temperature	+ 15°F	0	0
Initial Ice Bed Temperature	- 15°F	0	+ 1%

[1] All values shown are to the nearest percent

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TABLE 6.2.1-13

WATTS BAR ICE CONDENSER DESIGN PARAMETERS

Reactor Containment Volume (net free volume, ft^3)	
Upper Compartment	692,818
Ice Compartment	110,520
Lower Compartment	273,674
Lower Compartment (dead-ended)	94,000
Total Containment Volume	1,171,012

NSSS	
Fraction of Nominal (FON) based on NSSS Power of, MWt	3,474.21[1] - Unit 1 3,475 [2] - Unit 2
Analysis weight of ice in condenser, lbs (all main steam line breaks)	2.125×10^6*
Core Nuclear Power - % FON:	
100% power cases	1.006
30% power cases	0.30
0% power cases	Critical at 0.0

1. Includes RCP power (15.21 MWt) which is the net heat input (NHI) power value and is calculated by subtracting the total RCS heat losses from the total RCS heat inputs (gains).
2. Includes RCP power (16 MWt)

* The LOCA value of 2.26×10^6 is bounding for main steam line breaks (MSLB), since MSLB only melts approximately half of the available ice.

TABLE 6.2.1-14

ALLOWABLE LEAKAGE AREA FOR VARIOUS

REACTOR COOLANT SYSTEM BREAK SIZES

Break Size	5 ft^2 Deck Leak Air Compression Peak (psig)	Deck Leakage Area (ft^2)	Resultant Peak Containment Pressure (psig)
Double-ended	7.8	54	12.0
0.6 Double-ended	6.6	40	12.0
3 ft^2	6.25	46	12.0
10 inch diameter	5.75	38	12.0
10 inch diameter*	5.75	50	10.7*
6 inch diameter	5.5	41	12.0
6 inch diameter*	5.5	50	10.0*
2 inch diameter	5.0	50	5.0
2 inch diameter*	4.0	50	4.2*
1/2 inch diameter	3.0	>50	3.0

* This case assumes an upper compartment structural heat sink steam condensation of 6 lb/sec and 30% of deck leakage is air.

Note: One spray pump at 4750 gpm at 100°F was assumed for all breaks smaller than the 3 ft^2 break.

TABLE 6.2.1-15
UNIT 1 & UNIT 2[6]

SYSTEM PARAMETERS INITIAL CONDITIONS

Parameters	Values
Core Thermal Power [5] (MWt)	3,459
Reactor Coolant System Flow Rate, per Loop (gpm)	93,100
Vessel Outlet Temperature[1] (^oF)	619.1
Core Inlet Temperature[1] (^oF)	557.3
Vessel Average Temperature[1] [2] (^oF)	588.2
Reactor Coolant System Pressure [3] (psia)	2,250
Core Design Rod Array	17x17 RFA (w/IFMs)
Initial Steam Generator Steam Pressure[3] (psia)	1,112
Steam Generator Design	(Unit 1) Model 68AXP - (Unit 2) Model D3-2
Steam Generator Tube Plugging (%)	0
Initial Steam Generator Secondary Side Mass (lbm)	~170,000
Accumulator	
Water Volume (ft^3)	1,005/tank
Total Mass (lbm)	247,962
N_2 Cover Gas Pressure (psig)	690
Temperature (^oF)	130
Safety Injection Delay (sec) (includes time to reach pressure setpoint)	36.76
Auxiliary Feedwater Flow (gpm/steam generator)	0.0
Assumed Containment Reference Pressure (psia)	28.2
Pumped Injection [4]	See Table 6.2.1-45
Recirculation Time (assumed) (sec)	1272.1

Note:

1. Analysis value includes additional +6.0^oF allowance for instrument error and dead band.
2. This temperature assumes that the plant is not operating at the reduced 2^oF T_{avg} (586.2^oF). Therefore, margin exists in this analysis input if the plant operates at a reduced average temperature.
3. Analysis value includes an additional allowance for RCS initial pressure uncertainty (+70 psi).
4. The assumed safety injection flows consider the effects due to EDG frequency variation (+/- 0.2 Hz).
5. Analysis value includes 0.6% allowance for calorimetric error.
6. Ref 20 is performed with Unit 1 initial conditions and justified to bound the initial conditions of Unit 2. Therefore, Unit 2 utilizes the same initial conditions.

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TABLE 6.2.1-16

DOUBLE-ENDED PUMP SUCTION GUILLOTINE
BLOWDOWN MASS AND ENERGY RELEASE

	BREAK PATH NO. 1		BREAK PATH NO. 2	
Time (sec)	Flow (lbm/s)	Energy (BTU/s)	Flow (lbm/s)	Energy (BTU/s)
0.00	0.00	0.00	0.00	0.00
1.00	22693.76	12673098.14	39366.67	22430588.59
2.00	22267.79	12410711.77	34273.14	21183160.00
3.00	20881.99	11647021.21	21745.41	15432103.64
4.00	19455.19	10878508.01	15908.78	12075764.11
5.00	19170.05	10771429.70	14389.75	10818061.79
6.00	20264.65	11480460.28	14361.57	10453199.40
7.00	18512.90	10556378.40	14127.49	10162587.93
8.00	16796.34	9684582.13	13457.98	9720215.06
9.00	15313.92	8925658.19	13104.72	9360214.21
10.00	13943.70	8228808.13	12787.34	9063271.92
11.00	12667.01	7559701.13	11958.22	8595138.62
12.00	11357.41	6908455.17	10755.15	7957857.43
13.00	9799.83	6110595.06	9416.31	7243160.22
14.00	8164.31	5128189.40	8143.93	6526740.52
15.00	7371.41	4323614.16	7057.02	5868326.61
16.00	8231.82	4052862.18	5580.86	5009217.53
17.00	9509.06	3838723.41	4368.64	4173434.54
18.00	8789.36	3126852.12	3543.74	3453821.89
19.00	8633.08	2733703.11	2762.42	2838005.36
20.00	8273.64	2372672.62	2067.50	2252195.93
21.00	7088.07	1900348.53	1504.56	1735184.89
22.00	6403.46	1547046.46	1105.02	1374108.73
23.00	5027.54	1084998.87	804.21	1012413.82
24.00	4710.66	976742.17	617.21	779708.55
25.00	4908.73	921673.97	485.99	614968.43
26.00	3351.35	536572.67	387.72	491324.10
27.00	2392.00	362315.52	323.87	411076.09
28.00	2102.33	304238.28	273.79	348042.24
29.00	1489.65	201021.66	239.22	304516.39
30.00	872.63	112266.88	207.60	264612.49
31.00	376.73	48169.37	198.17	252730.24
32.00	255.48	32583.10	186.12	237523.12

TABLE 6.2.1-17

DELETED

TABLE 6.2.1-18

DELETED

TABLE 6.2.1-19 (Sheet 1 of 7)

DOUBLE-ENDED PUMP SUCTION GUILLOTINE BREAK MINIMUM
SAFETY INJECTION POST-BLOWDOWN MASS AND ENERGY RELEASE

	BREAK PATH NO. 1		BREAK PATH NO. 2	
Time	Flow (lbm/s)	Energy (BTU/s)	Flow (lbm/s)	Energy (BTU/s)
32.00	0.00	0.0	0.00	0.0
32.01	250.75	31983.8	154.20	196935.2
32.99	250.75	31983.8	154.20	196935.2
33.00	241.64	31671.5	162.90	208092.2
33.99	241.64	31671.5	162.90	208092.2
34.00	226.03	29868.2	155.26	198363.1
34.99	226.03	29868.2	155.26	198363.1
35.00	356.40	46817.6	221.28	282432.6
35.99	356.40	46817.6	221.28	282432.6
36.00	1605.11	192826.8	317.38	404419.8
36.99	1605.11	192826.8	317.38	404419.8
37.00	5210.26	555169.0	400.86	509600.0
37.99	5210.26	555169.0	400.86	509600.0
38.00	5241.91	588033.8	470.81	595214.6
38.99	5241.91	588033.8	470.81	595214.6
39.00	3170.16	444944.2	435.83	550997.0
39.99	3170.16	444944.2	435.83	550997.0
40.00	4275.16	597498.8	405.45	512950.0
40.99	4275.16	597498.8	405.45	512950.0
41.00	6347.39	892228.9	377.91	479410.3
41.99	6347.39	892228.9	377.91	479410.3
42.00	3522.14	524681.2	395.71	502149.6
42.99	3522.14	524681.2	395.71	502149.6
43.00	3161.78	490701.6	426.70	539983.0
43.99	3161.78	490701.6	426.70	539983.0
44.00	5107.77	770879.2	404.78	511930.9
44.99	5107.77	770879.2	404.78	511930.9
45.00	4984.47	746846.6	373.13	472986.6
45.99	4984.47	746846.6	373.13	472986.6
46.00	2852.69	463989.6	384.30	487071.7
46.99	2852.69	463989.6	384.30	487071.7
47.00	3274.30	516700.2	403.39	510144.8
47.99	3274.30	516700.2	403.39	510144.8
48.00	3731.52	571016.1	359.26	454549.8
48.99	3731.52	571016.1	359.26	454549.8
49.00	1865.96	385076.1	423.22	487351.6
68.99	1865.96	385076.1	423.22	487351.6
69.00	454.17	201612.9	520.48	533934.3
88.99	454.17	201612.9	520.48	533934.3
89.00	407.11	178114.0	483.14	467891.3
108.99	407.11	178114.0	483.14	467891.3
109.00	323.93	146794.5	567.14	483049.1
128.99	323.93	146794.5	567.14	483049.1
129.00	419.12	159063.6	441.24	406673.2
148.99	419.12	159063.6	441.24	406673.2

TABLE 6.2.1-19 (Sheet 2 of 7)

DOUBLE-ENDED PUMP SUCTION GUILLOTINE BREAK MINIMUM
SAFETY INJECTION POST-BLOWDOWN MASS AND ENERGY RELEASE

	BREAK PATH NO. 1		BREAK PATH NO. 2	
Time	Flow (lbm/s)	Energy (BTU/s)	Flow (lbm/s)	Energy (BTU/s)
149.00	537.37	174102.0	685.72	489401.0
168.99	537.37	174102.0	685.72	489401.0
169.00	464.00	149357.6	602.16	393147.2
188.99	464.00	149357.6	602.16	393147.2
189.00	316.42	112790.9	453.93	318973.2
208.99	316.42	112790.9	453.93	318973.2
209.00	243.04	84739.3	285.54	233766.0
228.99	243.04	84739.3	285.54	233766.0
229.00	216.91	71855.7	216.39	188184.3
248.99	216.91	71855.7	216.39	188184.3
249.00	213.42	67396.7	257.88	189840.6
268.99	213.42	67396.7	257.88	189840.6
269.00	210.93	66835.6	288.59	188494.9
288.99	210.93	66835.6	288.59	188494.9
289.00	215.84	65741.7	292.25	182807.1
308.99	215.84	65741.7	292.25	182807.1
309.00	210.36	61696.6	254.58	166617.3
328.99	210.36	61696.6	254.58	166617.3
329.00	210.68	63013.8	294.12	173087.6
348.99	210.68	63013.8	294.12	173087.6
349.00	208.94	62116.7	316.40	175261.2
368.99	208.94	62116.7	316.40	175261.2
369.00	221.04	66933.9	359.30	188133.4
388.99	221.04	66933.9	359.30	188133.4
389.00	221.34	68736.6	371.44	193602.0
408.99	221.34	68736.6	371.44	193602.0
409.00	221.02	68543.3	366.52	196121.3
428.99	221.02	68543.3	366.52	196121.3
429.00	222.90	69025.4	381.93	195153.6
448.99	222.90	69025.4	381.93	195153.6
449.00	220.99	68971.6	388.93	194729.6
468.99	220.99	68971.6	388.93	194729.6
469.00	218.48	67057.9	370.73	187391.9
488.99	218.48	67057.9	370.73	187391.9
489.00	216.18	66552.8	379.26	187182.9
538.99	216.18	66552.8	379.26	187182.9
539.00	207.56	62894.4	365.33	184478.9
588.99	207.56	62894.4	365.33	184478.9
589.00	217.25	65588.7	386.54	189357.6
638.99	217.25	65588.7	386.54	189357.6
639.00	219.44	67327.2	413.25	181519.9
688.99	219.44	67327.2	413.25	181519.9
689.00	215.30	64291.8	387.11	171275.2
738.99	215.30	64291.8	387.11	171275.2
739.00	215.03	57260.4	372.99	160849.8
788.99	215.03	57260.4	372.99	160849.8

TABLE 6.2.1-19 (Sheet 3 of 7)

DOUBLE-ENDED PUMP SUCTION GUILLOTINE BREAK MINIMUM
SAFETY INJECTION POST-BLOWDOWN MASS AND ENERGY RELEASE

	BREAK PATH NO. 1		BREAK PATH NO. 2	
Time	Flow (lbm/s)	Energy (BTU/s)	Flow (lbm/s)	Energy (BTU/s)
789.00	210.46	53549.1	382.44	157244.5
838.99	210.46	53549.1	382.44	157244.5
839.00	203.03	54428.1	389.23	159184.9
888.99	203.03	54428.1	389.23	159184.9
889.00	201.58	50907.9	364.80	155490.1
938.99	201.58	50907.9	364.80	155490.1
939.00	202.05	54486.6	407.77	159996.0
988.99	202.05	54486.6	407.77	159996.0
989.00	189.64	45397.0	381.30	147912.2
1188.99	189.64	45397.0	381.30	147912.2
1189.00	167.52	42999.6	386.18	145016.7
1388.99	167.52	42999.6	386.18	145016.7
1389.00	134.15	35876.5	278.51	119467.9
1588.99	134.15	35876.5	278.51	119467.9
1589.00	122.79	28141.9	287.58	117817.6
1788.99	122.79	28141.9	287.58	117817.6
1789.00	116.68	21740.2	300.84	118331.6
1988.99	116.68	21740.2	300.84	118331.6
1989.00	117.23	20509.5	301.74	116764.9
2188.99	117.23	20509.5	301.74	116764.9
2189.00	110.96	19113.6	289.76	111567.1
2388.99	110.96	19113.6	289.76	111567.1
2389.00	114.91	18936.8	297.84	111129.0
2588.99	114.91	18936.8	297.84	111129.0
2589.00	117.60	19263.9	291.22	107096.0
2788.99	117.60	19263.9	291.22	107096.0
2789.00	114.04	19075.2	264.08	98236.2
2988.99	114.04	19075.2	264.08	98236.2
2989.00	120.72	20722.5	300.50	102276.0
3188.99	120.72	20722.5	300.50	102276.0
3189.00	101.90	20603.9	327.16	100476.9
3388.99	101.90	20603.9	327.16	100476.9
3389.00	108.03	19361.9	316.76	102242.9
3588.99	108.03	19361.9	316.76	102242.9
3589.00	128.20	21479.3	254.19	87068.8
3788.99	128.20	21479.3	254.19	87068.8
3789.00	149.10	24336.1	241.74	79537.6
3988.99	149.10	24336.1	241.74	79537.6
3989.00	192.53	30381.7	265.97	82180.5
4188.99	192.53	30381.7	265.97	82180.5
4189.00	196.85	29622.2	229.59	73293.0
4388.99	196.85	29622.2	229.59	73293.0
4389.00	197.05	28976.2	231.22	72864.2
4588.99	197.05	28976.2	231.22	72864.2
4589.00	205.86	30018.5	224.45	71097.6
4788.99	205.86	30018.5	224.45	71097.6

TABLE 6.2.1-19 (Sheet 4 of 7)

DOUBLE-ENDED PUMP SUCTION GUILLOTINE BREAK MINIMUM
SAFETY INJECTION POST-BLOWDOWN MASS AND ENERGY RELEASE

	BREAK PATH NO. 1		BREAK PATH NO. 2	
Time	Flow (lbm/s)	Energy (BTU/s)	Flow (lbm/s)	Energy (BTU/s)
4789.00	202.93	28541.0	224.03	70943.9
4988.99	202.93	28541.0	224.03	70943.9
4989.00	195.70	30984.0	238.44	73228.3
5188.99	195.70	30984.0	238.44	73228.3
5189.00	254.80	40469.9	210.69	65509.3
5388.99	254.80	40469.9	210.69	65509.3
5389.00	213.87	32363.9	238.16	73709.8
5588.99	213.87	32363.9	238.16	73709.8
5589.00	221.98	33731.0	189.95	64488.0
5788.99	221.98	33731.0	189.95	64488.0
5789.00	164.50	25033.7	235.96	74269.1
5988.99	164.50	25033.7	235.96	74269.1
5989.00	178.05	26612.9	197.39	65521.6
6188.99	178.05	26612.9	197.39	65521.6
6189.00	188.87	27223.2	190.80	63356.9
6388.99	188.87	27223.2	190.80	63356.9
6389.00	232.56	32321.8	178.09	59385.4
6588.99	232.56	32321.8	178.09	59385.4
6589.00	260.99	36111.5	181.73	59575.6
6788.99	260.99	36111.5	181.73	59575.6
6789.00	272.64	37297.7	177.28	57962.9
6988.99	272.64	37297.7	177.28	57962.9
6989.00	252.19	34492.5	161.42	54039.0
7188.99	252.19	34492.5	161.42	54039.0
7189.00	289.26	39272.8	168.25	56205.7
7388.99	289.26	39272.8	168.25	56205.7
7389.00	276.80	37512.5	155.58	51867.1
7588.99	276.80	37512.5	155.58	51867.1
7589.00	272.90	36672.2	168.80	55426.8
7788.99	272.90	36672.2	168.80	55426.8
7789.00	267.22	36059.1	156.59	51491.3
7988.99	267.22	36059.1	156.59	51491.3
7989.00	279.77	36986.5	166.53	54311.1
8188.99	279.77	36986.5	166.53	54311.1
8189.00	282.91	37595.3	142.86	48715.3
8288.99	282.91	37595.3	142.86	48715.3
8389.00	299.87	39769.6	152.35	51083.3
8588.99	299.87	39769.6	152.35	51083.3
8589.00	284.51	38027.8	145.70	49189.6
8788.99	284.51	38027.8	145.70	49189.6
8789.00	287.00	38213.5	147.60	50358.3
8788.99	287.00	38213.5	147.60	50358.3
8989.00	309.70	40776.1	141.87	48092.4
9188.99	309.70	40776.1	141.87	48092.4
9189.00	291.97	38976.1	139.87	48039.4
9388.99	291.97	38976.1	139.87	48039.4

TABLE 6.2.1-19 (Sheet 5 of 7)

DOUBLE-ENDED PUMP SUCTION GUILLOTINE BREAK MINIMUM
SAFETY INJECTION POST-BLOWDOWN MASS AND ENERGY RELEASE

	BREAK PATH NO. 1		BREAK PATH NO. 2	
Time	Flow (lbm/s)	Energy (BTU/s)	Flow (lbm/s)	Energy (BTU/s)
9389.00	303.68	39962.9	136.61	47221.9
9588.99	303.68	39962.9	136.61	47221.9
9589.00	319.86	42986.2	143.02	47745.8
9788.99	319.86	42986.2	143.02	47745.8
9789.00	346.39	50234.3	136.01	43868.9
9988.99	346.39	50234.3	136.01	43868.9
9989.00	313.60	43505.3	153.75	48100.7
10188.99	313.60	43505.3	153.75	48100.7
10189.00	279.51	42428.9	154.10	47850.0
10388.99	279.51	42428.9	154.10	47850.0
10389.00	257.67	37044.0	166.97	52271.6
10588.99	257.67	37044.0	166.97	52271.6
10589.00	241.64	34303.2	154.65	51165.7
10788.99	241.64	34303.2	154.65	51165.7
10789.00	271.21	37437.2	134.95	46809.2
10988.99	271.21	37437.2	134.95	46809.2
10989.00	303.30	40708.0	127.46	44316.9
11188.99	303.30	40708.0	127.46	44316.9
11189.00	302.32	40760.3	123.52	42549.1
11388.99	302.32	40760.3	123.52	42549.1
11389.00	364.95	50552.1	107.22	36307.1
11588.99	364.95	50552.1	107.22	36307.1
11589.00	298.72	40642.0	138.67	46243.3
11788.99	298.72	40642.0	138.67	46243.3
11789.00	252.29	33707.0	128.32	45040.4
11988.99	252.29	33707.0	128.32	45040.4
11989.00	312.65	41646.3	117.36	41339.6
12188.99	312.65	41646.3	117.36	41339.6
12189.00	337.57	45256.2	109.13	39638.3
12388.99	337.57	45256.2	109.13	39638.3
12389.00	283.52	38090.3	118.02	40893.9
12588.99	283.52	38090.3	118.02	40893.9
12589.00	350.83	46258.5	123.62	42146.8
12788.99	350.83	46258.5	123.62	42146.8
12789.00	281.66	37658.9	104.77	38191.3
12988.99	281.66	37658.9	104.77	38191.3
12989.00	369.01	48514.6	109.17	39759.5
13188.99	369.01	48514.6	109.17	39759.5
13189.00	328.64	43415.4	108.60	38701.8
13388.99	328.64	43415.4	108.60	38701.8
13389.00	318.61	42696.2	104.78	38314.5
13588.99	318.61	42696.2	104.78	38314.5
12589.00	330.36	44165.8	107.35	37547.5
13788.99	330.36	44165.8	107.35	37547.5
13789.00	337.53	44345.0	105.48	38954.5
13988.99	337.53	44345.0	105.48	38954.5

TABLE 6.2.1-19 (Sheet 6 of 7)

DOUBLE-ENDED PUMP SUCTION GUILLOTINE BREAK MINIMUM
SAFETY INJECTION POST-BLOWDOWN MASS AND ENERGY RELEASE

	BREAK PATH NO. 1		BREAK PATH NO. 2	
Time	Flow (lbm/s)	Energy (BTU/s)	Flow (lbm/s)	Energy (BTU/s)
13989.00	319.87	42738.9	106.86	38248.4
14188.99	319.87	42738.9	106.86	38248.4
14189.00	350.39	46118.2	101.60	37441.6
14388.99	350.39	46118.2	101.60	37441.6
14389.00	327.15	42633.2	97.63	36557.6
14588.99	327.15	42633.2	97.63	36557.6
14589.00	320.39	42547.0	105.55	38619.4
14788.99	320.39	42547.0	105.55	38619.4
14789.00	355.29	46328.0	93.30	35171.8
14988.99	355.29	46328.0	93.30	35171.8
14989.00	347.57	46267.6	95.36	35019.3
15188.99	347.57	46267.6	95.36	35019.3
15189.00	321.82	42665.8	105.92	38661.2
15388.99	321.82	42665.8	105.92	38661.2
15389.00	363.12	46875.9	103.57	37478.6
15588.99	363.12	46875.9	103.57	37476.6
15589.00	339.55	44517.1	94.52	35246.2
15788.99	339.55	44517.1	94.52	35246.2
15789.00	321.97	42808.7	96.70	36020.4
15988.99	321.97	42808.7	96.70	36020.4
15989.00	317.88	41266.4	99.72	37256.5
16188.99	317.88	41266.4	99.72	37256.5
16189.00	346.60	44056.1	95.31	36268.8
16388.99	346.60	44056.1	95.31	36268.8
16389.00	345.81	45124.2	97.59	36224.9
16588.99	345.81	45124.2	97.59	36224.9
16589.00	347.34	45287.3	94.52	35011.2
16788.99	347.34	45287.3	94.52	35011.2
16789.00	326.24	42876.1	96.12	35659.1
16988.99	326.24	42876.1	96.12	35659.1
16989.00	338.95	44256.7	93.52	34626.7
17188.99	338.95	44256.7	93.52	34626.7
17189.00	327.53	42990.0	93.80	35459.8
17388.99	327.53	42990.0	93.80	35459.8
17389.00	374.15	47515.6	94.26	36037.6
17588.99	374.15	47515.6	94.26	36037.6
17589.00	343.77	44545.2	87.98	33946.4
17788.99	343.77	44545.2	87.98	33946.4
17789.00	355.41	45178.5	89.94	34418.6
17988.99	353.41	45178.5	89.94	34418.6
17989.00	326.36	41616.5	94.50	35792.2
18188.99	326.36	41616.5	94.50	35792.2
18189.00	359.06	45875.1	91.57	34794.2
18388.99	359.06	45875.1	91.57	34794.2
18389.00	337.46	43818.2	102.50	36992.0
18588.99	337.46	43818.2	102.50	36992.0

TABLE 6.2.1-19 (Sheet 7 of 7)

DOUBLE-ENDED PUMP SUCTION GUILLOTINE BREAK MINIMUM
SAFETY INJECTION POST-BLOWDOWN MASS AND ENERGY RELEASE

	BREAK PATH NO. 1		BREAK PATH NO. 2	
Time	Flow (lbm/s)	Energy (BTU/s)	Flow (lbm/s)	Energy (BTU/s)
18589.00	326.67	42276.8	81.10	32073.5
18788.99	326.67	42276.8	81.10	32073.5
18789.00	375.17	48010.6	86.76	33575.8
18988.99	375.17	48010.6	86.76	33575.8
18989.00	322.70	40915.8	91.12	34919.1
19188.99	322.70	40915.8	91.12	34919.1
19189.00	366.80	46391.7	86.74	33494.8
19388.99	366.80	46391.7	86.74	33494.8
19389.00	347.83	44278.8	91.96	34896.6
19588.99	347.83	44278.8	91.96	34896.6
19589.00	334.60	42808.1	84.27	32639.3
19788.99	334.60	42808.1	84.27	32639.3
19789.00	367.36	47144.9	87.81	33600.2
19989.00	367.36	47144.9	87.81	33600.2

TABLE 6.2.1-20

DECAY HEAT CURVE

Time (sec)	Decay Heat (Btu/Btu)
10	0.053876
15	0.050401
20	0.048018
40	0.042401
60	0.039244
80	0.037065
100	0.035466
150	0.032724
200	0.030936
400	0.027078
600	0.024931
800	0.023389
1000	0.022156
1500	0.019921
2000	0.018315
4000	0.014781
6000	0.013040
8000	0.012000
10000	0.011262
15000	0.010097
20000	0.009350
40000	0.007778
60000	0.006958
80000	0.006424
100000	0.006021
150000	0.005323
200000	0.004847
400000	0.003770
600000	0.003201
800000	0.002834
1000000	0.002580
1500000	0.0022448
2000000	0.0019090
4000000	0.0013350

TABLE 6.2.1-21

and

TABLE 6.2.1-22

DELETED

TABLE 6.2.1-23 (Sheet 1 of 6)

BREAK MASS AND ENERGY FLOW
FROM A DOUBLE-ENDED COLD LEG BREAK

Time (sec)	Mass Flow (lbm/sec)	Energy Flow (Btu/sec)	Avg. Enthalpy (Btu/lbm)
0.00000	9.5330000E+03	5.2779455E+06	553.65
0.00102	4.4011217E+04	2.4170167E+07	549.18
0.00201	5.6275939E+04	3.0911554E+07	549.29
0.00302	6.0837595E+04	3.3416041E+07	549.27
0.00400	6.1885829E+04	3.3984937E+07	549.16
0.00505	6.1419488E+04	3.3717570E+07	548.97
0.00605	6.0217483E+04	3.3043687E+07	548.74
0.00702	5.8724381E+04	3.2209645E+07	548.49
0.00800	5.7204434E+04	3.1364336E+07	548.29
0.00904	5.5699994E+04	3.0530804E+07	548.13
0.01002	5.4524101E+04	2.9883079E+07	548.07
0.01103	5.3682953E+04	2.9423518E+07	548.10
0.01207	5.3225044E+04	2.9178055E+07	548.20
0.01307	5.3157556E+04	2.9148416E+07	548.34
0.01402	5.3381584E+04	2.9277878E+07	548.46
0.01508	5.3738390E+04	2.9478400E+07	548.55
0.01604	5.4090450E+04	2.9674432E+07	548.61
0.01706	5.4438424E+04	2.9867024E+07	548.64
0.01809	5.4728511E+04	3.0027384E+07	548.66
0.01908	5.4986231E+04	3.0169058E+07	548.67
0.02008	5.5188710E+04	3.0280438E+07	548.67
0.02101	5.5349263E+04	3.0369345E+07	548.69
0.02205	5.5516077E+04	3.0462092E+07	548.71
0.02306	5.5671777E+04	3.0548967E+07	548.73
0.02410	5.5820651E+04	3.0632164E+07	548.76
0.02505	5.5956351E+04	3.0708220E+07	548.79
0.02613	5.6109519E+04	3.0794010E+07	548.82
0.02708	5.6239831E+04	3.0866937E+07	548.84
0.02810	5.6375569E+04	3.0942938E+07	548.87
0.02902	5.6494360E+04	3.1009460E+07	548.89
0.03000	5.6617494E+04	3.1078459E+07	548.92
0.03115	5.6762269E+04	3.1159579E+07	548.95
0.03208	5.6878356E+04	3.1224552E+07	548.97
0.03303	5.7001748E+04	3.1293564E+07	548.99
0.03406	5.7132497E+04	3.1366442E+07	549.01
0.03500	5.7250439E+04	3.1432092E+07	549.03
0.03601	5.7369309E+04	3.1498340E+07	549.05
0.03701	5.7487079E+04	3.1564093E+07	549.06

TABLE 6.2.1-23 (Sheet 2 of 6)

BREAK MASS AND ENERGY FLOW
FROM A DOUBLE-ENDED COLD LEG BREAK

Time (sec)	Mass Flow (lbm/sec)	Energy Flow (Btu/sec)	Avg. Enthalpy (Btu/lbm)
0.03801	5.7606430E+04	3.1631013E+07	549.09
0.03901	5.7727994E+04	3.1701164E+07	549.15
0.04001	5.7859629E+04	3.1777517E+07	549.22
0.04103	5.8008186E+04	3.1863467E+07	549.29
0.04200	5.8164500E+04	3.1953463E+07	549.36
0.04302	5.8338190E+04	3.2052951E+07	549.43
0.04402	5.8516911E+04	3.2154814E+07	549.50
0.04502	5.8695158E+04	3.2255963E+07	549.55
0.04600	5.8914698E+04	3.2395139E+07	549.87
0.04701	5.9381194E+04	3.2662505E+07	550.05
0.04801	5.9930049E+04	3.2969806E+07	550.14
0.04900	6.0521617E+04	3.3298385E+07	550.19
0.05002	6.1079340E+04	3.3607032E+07	550.22
0.05101	6.1553470E+04	3.3868709E+07	550.23
0.05203	6.1986669E+04	3.4106951E+07	550.23
0.05302	7.0805693E+04	3.9192162E+07	553.52
0.05401	7.5356753E+04	4.1480610E+07	550.46
0.05500	7.4757589E+04	4.1211546E+07	551.27
0.05601	7.7480747E+04	4.2655030E+07	550.52
0.05701	7.5957665E+04	4.1830676E+07	550.71
0.05800	7.8472525E+04	4.3211894E+07	550.66
0.05902	7.8988165E+04	4.3501189E+07	550.73
0.06004	7.9162655E+04	4.3579636E+07	550.51
0.06102	7.8663312E+04	4.3304530E+07	550.50
0.06206	7.9469059E+04	4.3744319E+07	550.46
0.06305	7.9885116E+04	4.3976304E+07	550.49
0.06404	7.9704566E+04	4.3866081E+07	550.36
0.06502	8.0093155E+04	4.4088720E+07	550.47
0.06605	8.1038196E+04	4.4616351E+07	550.56
0.06702	8.2171459E+04	4.5249175E+07	550.67
0.06803	8.3010104E+04	4.5712072E+07	550.68
0.06901	8.3420144E+04	4.5939141E+07	550.70
0.07009	8.3834008E+04	4.6168257E+07	550.71
0.07108	8.4083444E+04	4.6305013E+07	550.70
0.07203	8.4176648E+04	4.6355157E+07	550.69
0.07310	8.4366766E+04	4.6461996E+07	550.71
0.07406	8.4687890E+04	4.6642651E+07	550.76
0.07509	8.5069460E+04	4.6855898E+07	550.80
0.07605	8.5291359E+04	4.6979786E+07	550.82

TABLE 6.2.1-23 (Sheet 3 of 6)

BREAK MASS AND ENERGY FLOW
FROM A DOUBLE-ENDED COLD LEG BREAK

Time (sec)	Mass Flow (lbm/sec)	Energy Flow (Btu/sec)	Avg. Enthalpy (Btu/lbm)
0.07700	8.5560677E+04	4.7130797E+07	550.85
0.07808	8.5975380E+04	4.7364574E+07	550.91
0.07904	8.6396125E+04	4.7601471E+07	550.97
0.08014	8.6831938E+04	4.7845218E+07	551.01
0.08112	8.7169273E+04	4.8033244E+07	551.03
0.08202	8.7408028E+04	4.8166073E+07	551.05
0.08311	8.7566758E+04	4.8252799E+07	551.04
0.08408	8.7629069E+04	4.8286096E+07	551.03
0.08510	8.7697243E+04	4.8323926E+07	551.03
0.08602	8.7834628E+04	4.8401150E+07	551.05
0.08713	8.8072408E+04	4.8534975E+07	551.08
0.08815	8.8326367E+04	4.8677777E+07	551.11
0.08903	8.8559803E+04	4.8808881E+07	551.14
0.09011	8.8835499E+04	4.8963278E+07	551.17
0.09106	8.9027430E+04	4.9069945E+07	551.18
0.09203	8.9162845E+04	4.9144366E+07	551.18
0.09318	8.9258085E+04	4.9195890E+07	551.16
0.09414	8.9320618E+04	4.9229491E+07	551.15
0.09509	8.9390681E+04	4.9267660E+07	551.15
0.09602	8.9483513E+04	4.9319001E+07	551.15
0.09704	8.9628921E+04	4.9399907E+07	551.16
0.09808	8.9815926E+04	4.9504099E+07	551.17
0.09901	8.9996236E+04	4.9604432E+07	551.18
0.10017	9.0186831E+04	4.9709676E+07	551.19
0.10515	9.0484261E+04	4.9865175E+07	551.09
0.11004	9.0446715E+04	4.9833828E+07	550.97
0.11512	8.9671795E+04	4.9390031E+07	550.79
0.12012	8.8502729E+04	4.8736499E+07	550.68
0.12513	8.7301816E+04	4.8077220E+07	550.70
0.13013	8.6705615E+04	4.7765491E+07	550.89
0.13506	8.6600239E+04	4.7721242E+07	551.05
0.14014	8.6643187E+04	4.7753143E+07	551.15
0.14510	8.6640729E+04	4.7755648E+07	551.19
0.15002	8.6616399E+04	4.7745567E+07	551.23
0.15503	8.6619555E+04	4.7751027E+07	551.27
0.16017	8.6675812E+04	4.7786162E+07	551.32
0.16511	8.6746002E+04	4.7827686E+07	551.35
0.17005	8.6757050E+04	4.7834309E+07	551.36
0.17502	8.6689168E+04	4.7794900E+07	551.34

TABLE 6.2.1-23 (Sheet 4 of 6)

BREAK MASS AND ENERGY FLOW
FROM A DOUBLE-ENDED COLD LEG BREAK

Time (sec)	Mass Flow (lbm/sec)	Energy Flow (Btu/sec)	Avg. Enthalpy (Btu/lbm)
0.18008	8.6530182E+04	4.7702879E+07	551.29
0.18505	8.6261412E+04	4.7548196E+07	551.21
0.19007	8.5839547E+04	4.7307970E+07	551.12
0.19511	8.5290235E+04	4.6999355E+07	551.05
0.20007	8.4760468E+04	4.6704790E+07	551.02
0.21007	8.3839594E+04	4.6204662E+07	551.11
0.22001	8.3317105E+04	4.5921991E+07	551.17
0.23022	8.2782598E+04	4.5626118E+07	551.16
0.24012	8.2665978E+04	4.5566162E+07	551.21
0.25010	8.2791643E+04	4.5640088E+07	551.26
0.26011	8.2857336E+04	4.5678559E+07	551.29
0.27003	8.2682137E+04	4.5581845E+07	551.29
0.28009	8.2441046E+04	4.5449098E+07	551.29
0.29021	8.2254193E+04	4.5345947E+07	551.29
0.30009	8.2142447E+04	4.5283453E+07	551.28
0.31011	8.1981643E+04	4.5192793E+07	551.26
0.32001	8.1744907E+04	4.5060004E+07	551.23
0.33004	8.1490282E+04	4.4918684E+07	551.22
0.34014	8.1318832E+04	4.4824748E+07	551.22
0.35021	8.1213820E+04	4.4767770E+07	551.23
0.36010	8.1124659E+04	4.4718961E+07	551.24
0.37013	8.1059546E+04	4.4683345E+07	551.24
0.38006	8.1024864E+04	4.4664531E+07	551.24
0.39001	8.0991761E+04	4.4646277E+07	551.24
0.40007	8.0908141E+04	4.4599641E+07	551.24
0.41001	8.0782300E+04	4.4529699E+07	551.23
0.42008	8.0669241E+04	4.4467676E+07	551.23
0.43001	8.0601164E+04	4.4430759E+07	551.24
0.44000	8.0520276E+04	4.4386506E+07	551.25
0.45007	8.0371565E+04	4.4304181E+07	551.24
0.46003	8.0178969E+04	4.4197512E+07	551.24
0.47004	7.9990836E+04	4.4093579E+07	551.23
0.48027	7.9835359E+04	4.4008308E+07	551.24
0.49001	7.9696318E+04	4.3932619E+07	551.25
0.50011	7.9554322E+04	4.3855174E+07	551.26
0.51005	7.9408803E+04	4.3775330E+07	551.27
0.52000	7.9271304E+04	4.3699938E+07	551.27
0.53007	7.9137883E+04	4.3626882E+07	551.28
0.54022	7.8988311E+04	4.3544749E+07	551.28

TABLE 6.2.1-23 (Sheet 5 of 6)

BREAK MASS AND ENERGY FLOW
FROM A DOUBLE-ENDED COLD LEG BREAK

Time (sec)	Mass Flow (lbm/sec)	Energy Flow (Btu/sec)	Avg. Enthalpy (Btu/lbm)
0.55013	7.8814628E+04	4.3449230E+07	551.28
0.56009	7.8626396E+04	4.3345934E+07	551.29
0.57004	7.8446330E+04	4.3247570E+07	551.30
0.58031	7.8270000E+04	4.3151255E+07	551.31
0.59004	7.8087544E+04	4.3051342E+07	551.32
0.60001	7.7890710E+04	4.2943636E+07	551.33
0.61033	7.7688420E+04	4.2832747E+07	551.34
0.62005	7.7486767E+04	4.2722260E+07	551.35
0.63018	7.7279992E+04	4.2608927E+07	551.36
0.64009	7.7065920E+04	4.2491642E+07	551.37
0.65006	7.6849999E+04	4.2373346E+07	551.38
0.66007	7.6635344E+04	4.2255925E+07	551.39
0.67010	7.6415894E+04	4.2135791E+07	551.40
0.68000	7.6183243E+04	4.2008626E+07	551.42
0.69016	7.5942008E+04	4.1877113E+07	551.44
0.70022	7.5717402E+04	4.1754932E+07	551.46
0.71006	7.5500218E+04	4.1636496E+07	551.48
0.72003	7.5275023E+04	4.1513352E+07	551.49
0.73003	7.5036534E+04	4.1382780E+07	551.50
0.74028	7.4781489E+04	4.1242838E+07	551.51
0.75011	7.4519098E+04	4.1098901E+07	551.52
0.76012	7.4246847E+04	4.0949791E+07	551.54
0.77011	7.3974375E+04	4.0800775E+07	551.55
0.78030	7.3851738E+04	4.0736186E+07	551.59
0.79005	7.3805928E+04	4.0713881E+07	551.63
0.80011	7.3766415E+04	4.0694802E+07	551.67
0.81011	7.3730208E+04	4.0677453E+07	551.71
0.82037	7.3656561E+04	4.0639177E+07	551.74
0.83003	7.3560651E+04	4.0588611E+07	551.77
0.84001	7.3455476E+04	4.0532864E+07	551.80
0.85040	7.3320551E+04	4.0460479E+07	551.83
0.86005	7.3165192E+04	4.0376933E+07	551.86
0.87003	7.3010723E+04	4.0294005E+07	551.89
0.88004	7.2868326E+04	4.0218274E+07	551.93
0.89009	7.2758070E+04	4.0160966E+07	551.98
0.90013	7.2677348E+04	4.0119787E+07	552.03
0.91013	7.2606855E+04	4.0083902E+07	552.07
0.92028	7.2537221E+04	4.0048371E+07	552.11
0.93010	7.2476868E+04	4.0017943E+07	552.15

TABLE 6.2.1-23 (Sheet 6 of 6)

BREAK MASS AND ENERGY FLOW
FROM A DOUBLE-ENDED COLD LEG BREAK

Time (sec)	Mass Flow (lbm/sec)	Energy Flow (Btu/sec)	Avg. Enthalpy (Btu/lbm)
0.94005	7.2425266E+04	3.9992496E+07	552.19
0.95006	7.2359180E+04	3.9959174E+07	552.23
0.96002	7.2265109E+04	3.9910379E+07	552.28
0.97011	7.2153580E+04	3.9852144E+07	552.32
0.98009	7.2033376E+04	3.9789083E+07	552.37
0.99001	7.1903822E+04	3.9720819E+07	552.42
1.00019	7.1763521E+04	3.9646556E+07	552.46
1.10002	7.0499419E+04	3.8987248E+07	553.02
1.20002	6.8949377E+04	3.8168890E+07	553.58
1.30001	6.7071184E+04	3.7166953E+07	554.14
1.40014	6.4946301E+04	3.6022949E+07	554.66
1.50001	6.2996916E+04	3.4970203E+07	555.11
1.60012	6.1127042E+04	3.3954600E+07	555.48
1.70001	5.9053076E+04	3.2820746E+07	555.78
1.80002	5.4888698E+04	3.0514663E+07	555.94
1.90002	5.3754178E+04	2.9897616E+07	556.19
2.00003	5.2877172E+04	2.9411454E+07	556.22
2.10006	5.0817329E+04	2.8273312E+07	556.37
2.20010	5.0463763E+04	2.8085654E+07	556.55
2.30031	5.0199059E+04	2.7948182E+07	556.75
2.40000	4.9849919E+04	2.7768266E+07	557.04
2.50012	4.9365700E+04	2.7513531E+07	557.34
2.60001	4.8513476E+04	2.7050245E+07	557.58
2.70019	4.7560714E+04	2.6530990E+07	557.83
2.80002	4.6680573E+04	2.6052964E+07	558.11
2.90009	4.5792188E+04	2.5571836E+07	558.43
3.00050	4.5120083E+04	2.5209147E+07	558.71

TABLE 6.2.1-24 (Sheet 1 of 6)

BREAK MASS AND ENERGY FLOW FROM A DOUBLE-ENDED HOT LEG BREAK

Time (sec)	Mass Flow (lbm/sec)	Energy Flow (Btu/sec)	Avg. Enthalpy (Btu/lbm)
0.00000	9.5330000E+03	6.0868205E+06	638.50
0.00100	6.1416131E+04	3.9109426E+07	636.79
0.00203	8.1135961E+04	5.1666478E+07	636.79
0.00300	8.2509207E+04	5.2490620E+07	636.18
0.00403	7.8835251E+04	5.0077196E+07	635.21
0.00504	7.4070465E+04	4.6987340E+07	634.36
0.00600	7.0382525E+04	4.4625410E+07	634.04
0.00701	6.8279591E+04	4.3305904E+07	634.24
0.00803	6.7936777E+04	4.3114738E+07	634.63
0.00902	6.8553213E+04	4.3521447E+07	634.86
0.01004	6.9266019E+04	4.3980261E+07	634.95
0.01105	6.9707969E+04	4.4264704E+07	635.00
0.01203	6.9943640E+04	4.4422762E+07	635.12
0.01303	7.0135464E+04	4.4557799E+07	635.31
0.01404	7.0391783E+04	4.4735078E+07	635.52
0.01502	7.0719643E+04	4.4955217E+07	635.68
0.01604	7.1080759E+04	4.5193750E+07	635.81
0.01703	7.1421408E+04	4.5416805E+07	635.90
0.01806	7.1739117E+04	4.5624213E+07	635.97
0.01901	7.2013730E+04	4.5803622E+07	636.04
0.02002	7.2289247E+04	4.5983322E+07	636.10
0.02104	7.2546018E+04	4.6150629E+07	636.16
0.02203	7.2779223E+04	4.6302581E+07	636.21
0.02300	7.2990380E+04	4.6440427E+07	636.25
0.02405	7.3209529E+04	4.6583431E+07	636.30
0.02506	7.3408737E+04	4.6713371E+07	636.35
0.02601	7.3585136E+04	4.6828459E+07	636.38
0.02704	7.3766636E+04	4.6946927E+07	636.42
0.02803	7.3932879E+04	4.7055438E+07	636.46
0.02904	7.4096998E+04	4.7162333E+07	636.49
0.03002	7.4246747E+04	4.7259817E+07	636.52
0.03105	7.4394934E+04	4.7356369E+07	636.55
0.03202	7.4527359E+04	4.7442789E+07	636.58
0.03305	7.4663136E+04	4.7531444E+07	636.61
0.03404	7.4784977E+04	4.7611125E+07	636.64
0.03507	7.4904151E+04	4.7689293E+07	636.67
0.03603	7.5009924E+04	4.7758969E+07	636.70
0.03704	7.5118554E+04	4.7830746E+07	636.74
0.03803	7.5223125E+04	4.7899938E+07	636.77
0.03905	7.5325886E+04	4.7968146E+07	636.81

TABLE 6.2.1-24 (Sheet 2 of 6)

BREAK MASS AND ENERGY FLOW FROM A DOUBLE-ENDED HOT LEG BREAK

Time (sec)	Mass Flow (lbm/sec)	Energy Flow (Btu/sec)	Avg. Enthalpy (Btu/lbm)
0.04005	7.5423967E+04	4.8033558E+07	636.85
0.04105	7.5520039E+04	4.8097883E+07	636.89
0.04201	7.5612528E+04	4.8160069E+07	636.93
0.04305	7.5710411E+04	4.8226204E+07	636.98
0.04404	7.5803775E+04	4.8289641E+07	637.03
0.04509	7.5902935E+04	4.8357434E+07	637.10
0.04608	7.5996750E+04	4.8421991E+07	637.16
0.04713	7.6097497E+04	4.8491736E+07	637.23
0.04810	7.6193516E+04	4.8558502E+07	637.30
0.04913	7.6299369E+04	4.8632275E+07	637.39
0.05005	7.6399525E+04	4.8702123E+07	637.47
0.05102	7.6508259E+04	4.8777983E+07	637.55
0.05204	7.6625357E+04	4.8859723E+07	637.64
0.05313	7.6751258E+04	4.8947636E+07	637.74
0.05406	7.6861837E+04	4.9024782E+07	637.83
0.05501	7.6976014E+04	4.9104304E+07	637.92
0.05617	7.7117581E+04	4.9202677E+07	638.02
0.05702	7.7243272E+04	4.9289099E+07	638.10
0.05804	7.7375268E+04	4.9378024E+07	638.16
0.05901	7.7493447E+04	4.9458162E+07	638.22
0.06006	7.7623204E+04	4.9545858E+07	638.29
0.06100	7.7744384E+04	4.9627745E+07	638.35
0.06203	7.7873689E+04	4.9714354E+07	638.40
0.06301	7.7997061E+04	4.9796912E+07	638.45
0.06401	7.8122079E+04	4.9880358E+07	638.49
0.06502	7.8252156E+04	4.9966914E+07	638.54
0.06603	7.8386090E+04	5.0055568E+07	638.58
0.06702	7.8520445E+04	5.0143912E+07	638.61
0.06801	7.8650968E+04	5.0229542E+07	638.64
0.06901	7.8784460E+04	5.0316970E+07	638.67
0.07001	7.8921805E+04	5.0406602E+07	638.69
0.07102	7.9061454E+04	5.0497330E+07	638.71
0.07201	7.9198409E+04	5.0586204E+07	638.73
0.07302	7.9337532E+04	5.0676474E+07	638.75
0.07402	7.9475413E+04	5.0765948E+07	638.76
0.07502	7.9624618E+04	5.0862488E+07	638.78
0.07602	7.9766342E+04	5.0954267E+07	638.79
0.07704	7.9912739E+04	5.1049271E+07	638.81
0.07802	8.0061918E+04	5.1146208E+07	638.83
0.07903	8.0212265E+04	5.1244326E+07	638.86

TABLE 6.2.1-24 (Sheet 3 of 6)

BREAK MASS AND ENERGY FLOW FROM A DOUBLE-ENDED HOT LEG BREAK

Time (sec)	Mass Flow (lbm/sec)	Energy Flow (Btu/sec)	Avg. Enthalpy (Btu/lbm)
0.08002	8.0366041E+04	5.1345066E+07	638.89
0.08102	8.0530553E+04	5.1453313E+07	638.93
0.08202	8.0705882E+04	5.1569477E+07	638.98
0.08303	8.0897216E+04	5.1695370E+07	639.03
0.08400	8.1094981E+04	5.1824479E+07	639.06
0.08504	8.1313084E+04	5.1965537E+07	639.08
0.08603	8.1535666E+04	5.2112280E+07	639.13
0.08703	8.1867681E+04	5.2328417E+07	639.18
0.08802	8.2201170E+04	5.2542124E+07	639.19
0.08901	8.2525222E+04	5.2749607E+07	639.19
0.09002	8.2879860E+04	5.2974999E+07	639.18
0.09104	8.3214935E+04	5.3186954E+07	639.15
0.09204	8.3508187E+04	5.3371121E+07	639.11
0.09301	8.3750105E+04	5.3521590E+07	639.06
0.09401	8.3955512E+04	5.3648335E+07	639.01
0.09501	8.6534575E+04	5.5315873E+07	639.23
0.09600	8.6568326E+04	5.5333831E+07	639.19
0.09702	8.7505146E+04	5.5918938E+07	639.04
0.09804	8.7371901E+04	5.5828792E+07	638.98
0.09904	8.7195239E+04	5.5703590E+07	638.84
0.10005	8.6885716E+04	5.5498627E+07	638.75
0.10503	8.7824524E+04	5.6084551E+07	638.60
0.11001	8.6299840E+04	5.5088491E+07	638.34
0.11504	8.5489602E+04	5.4555478E+07	638.15
0.12009	8.5323183E+04	5.4452511E+07	638.19
0.12503	8.4154565E+04	5.3705223E+07	638.17
0.13008	8.3487352E+04	5.3289754E+07	638.30
0.13503	8.3056991E+04	5.3018759E+07	638.34
0.14001	8.2247243E+04	5.2507146E+07	638.41
0.14503	8.1674428E+04	5.2144826E+07	638.45
0.15021	8.1246116E+04	5.1875860E+07	638.50
0.15504	8.0856786E+04	5.1631527E+07	638.56
0.16011	8.0421881E+04	5.1359022E+07	638.62
0.16513	7.9977134E+04	5.1080731E+07	638.69
0.17009	7.9562053E+04	5.0822660E+07	638.78
0.17512	7.9181660E+04	5.0586832E+07	638.87
0.18009	7.8846393E+04	5.0377806E+07	638.94
0.18506	7.8524393E+04	5.0174834E+07	638.97
0.19014	7.8198508E+04	4.9968023E+07	638.99
0.19504	7.7892517E+04	4.9772826E+07	638.99

TABLE 6.2.1-24 (Sheet 4 of 6)

BREAK MASS AND ENERGY FLOW FROM A DOUBLE-ENDED HOT LEG BREAK

Time (sec)	Mass Flow (lbm/sec)	Energy Flow (Btu/sec)	Avg. Enthalpy (Btu/lbm)
0.20018	7.7603146E+04	4.9587752E+07	638.99
0.21033	7.7188942E+04	4.9317843E+07	638.92
0.22029	7.6960586E+04	4.9164371E+07	638.83
0.23006	7.6570783E+04	4.8898361E+07	638.60
0.24033	7.5942171E+04	4.8480151E+07	638.38
0.25028	7.5205719E+04	4.7997095E+07	638.21
0.26027	7.4401346E+04	4.7473344E+07	638.07
0.27009	7.3562227E+04	4.6928884E+07	637.95
0.28016	7.2716634E+04	4.6379848E+07	637.82
0.29001	7.1952386E+04	4.5881509E+07	637.66
0.30000	7.1309925E+04	4.5457981E+07	637.47
0.31009	7.0824030E+04	4.5130368E+07	637.22
0.32035	7.0437980E+04	4.4861891E+07	636.90
0.33019	7.0099449E+04	4.4621966E+07	636.55
0.34018	6.9752251E+04	4.4376203E+07	636.20
0.35003	6.9391810E+04	4.4124323E+07	635.87
0.36018	6.8990825E+04	4.3849551E+07	635.59
0.37018	6.8558773E+04	4.3554306E+07	635.28
0.38024	6.8133031E+04	4.3264630E+07	635.00
0.39027	6.7752047E+04	4.3006105E+07	634.76
0.40025	6.7439196E+04	4.2792900E+07	634.54
0.41036	6.7201924E+04	4.2628528E+07	634.33
0.42009	6.7050351E+04	4.2520006E+07	634.15
0.43003	6.6963694E+04	4.2453027E+07	633.97
0.44018	6.6909325E+04	4.2404485E+07	633.76
0.45009	6.6863174E+04	4.2361959E+07	633.56
0.46008	6.6800552E+04	4.2308862E+07	633.36
0.47045	6.6712372E+04	4.2240606E+07	633.17
0.48011	6.6590510E+04	4.2151291E+07	632.99
0.49011	6.6442775E+04	4.2046634E+07	632.82
0.50023	6.6267878E+04	4.1925148E+07	632.66
0.51019	6.6084078E+04	4.1797635E+07	632.49
0.52012	6.5890021E+04	4.1664079E+07	632.33
0.53015	6.5684598E+04	4.1521276E+07	632.13
0.54012	6.5478193E+04	4.1378401E+07	631.94
0.55010	6.5252544E+04	4.1221239E+07	631.72
0.56020	6.5019090E+04	4.1059483E+07	631.50
0.57017	6.4763871E+04	4.0882936E+07	631.26
0.58038	6.4497439E+04	4.0699955E+07	631.03
0.59001	6.4228177E+04	4.0515515E+07	630.81

TABLE 6.2.1-24 (Sheet 5 of 6)

BREAK MASS AND ENERGY FLOW FROM A DOUBLE-ENDED HOT LEG BREAK

Time (sec)	Mass Flow (lbm/sec)	Energy Flow (Btu/sec)	Avg. Enthalpy (Btu/lbm)
0.60018	6.3943308E+04	4.0321649E+07	630.58
0.61007	6.3662027E+04	4.0130655E+07	630.37
0.62037	6.3378776E+04	3.9939341E+07	630.17
0.63017	6.3114024E+04	3.9760618E+07	629.98
0.64025	6.2860710E+04	3.9590085E+07	629.81
0.65002	6.2631549E+04	3.9435594E+07	629.64
0.66020	6.2412126E+04	3.9288074E+07	629.49
0.67010	6.2221984E+04	3.9160312E+07	629.36
0.68009	6.2036864E+04	3.9035805E+07	629.24
0.69025	6.1869142E+04	3.8923279E+07	629.12
0.70011	6.1717564E+04	3.8822316E+07	629.03
0.71015	6.1571474E+04	3.8726410E+07	628.97
0.72004	6.1444624E+04	3.8642516E+07	628.90
0.73018	6.1293788E+04	3.8542341E+07	628.81
0.74003	6.1150277E+04	3.8450573E+07	628.79
0.75039	6.1039173E+04	3.8380512E+07	628.78
0.76021	6.0940905E+04	3.8319086E+07	628.79
0.77031	6.0843813E+04	3.8258999E+07	628.81
0.78018	6.0744549E+04	3.8197390E+07	628.82
0.79021	6.0631277E+04	3.8127609E+07	628.84
0.80021	6.0525662E+04	3.8063165E+07	628.88
0.81006	6.0416020E+04	3.7996106E+07	628.91
0.82039	6.0289904E+04	3.7919165E+07	628.95
0.83021	6.0157177E+04	3.7837457E+07	628.98
0.84022	6.0008502E+04	3.7745642E+07	629.00
0.85024	5.9848118E+04	3.7646320E+07	629.03
0.86024	5.9673042E+04	3.7537318E+07	629.05
0.87044	5.9489421E+04	3.7423320E+07	629.08
0.88009	5.9304730E+04	3.7308046E+07	629.09
0.89018	5.9111585E+04	3.7188044E+07	629.12
0.90017	5.8922187E+04	3.7070155E+07	629.14
0.91003	5.8736367E+04	3.6954737E+07	629.16
0.92033	5.8550528E+04	3.6839612E+07	629.19
0.93024	5.8371038E+04	3.6728512E+07	629.22
0.94005	5.8204119E+04	3.6625537E+07	629.26
0.95019	5.8037641E+04	3.6522990E+07	629.30
0.96015	5.7881454E+04	3.6427227E+07	629.34
0.97022	5.7732213E+04	3.6335855E+07	629.39
0.98017	5.7585885E+04	3.6246392E+07	629.43
0.99036	5.7445062E+04	3.6160485E+07	629.48

TABLE 6.2.1-24 (Sheet 6 of 6)

BREAK MASS AND ENERGY FLOW FROM A DOUBLE-ENDED HOT LEG BREAK

Time (sec)	Mass Flow (lbm/sec)	Energy Flow (Btu/sec)	Avg. Enthalpy (Btu/lbm)
1.00019	5.7307975E+04	3.6076659E+07	629.52
1.10002	5.5849374E+04	3.5171962E+07	629.76
1.20041	5.4225806E+04	3.4151100E+07	629.79
1.30004	5.2608302E+04	3.3145390E+07	630.04
1.40018	5.0848604E+04	3.2053813E+07	630.38
1.50026	4.9049838E+04	3.0938572E+07	630.76
1.60002	4.7255066E+04	2.9813654E+07	630.91
1.70024	4.5520651E+04	2.8708354E+07	630.67
1.80019	4.4026912E+04	2.7742987E+07	630.14
1.90028	4.2682946E+04	2.6853311E+07	629.13
2.00015	4.1657401E+04	2.6153907E+07	627.83
2.10007	4.1198541E+04	2.5767900E+07	625.46
2.20038	4.1224277E+04	2.5661759E+07	622.49
2.30027	4.1353980E+04	2.5618336E+07	619.49
2.40019	4.1639010E+04	2.5668416E+07	616.45
2.50037	4.1802614E+04	2.5655840E+07	613.74
2.60053	4.2022491E+04	2.5707189E+07	611.75
2.70042	4.1917437E+04	2.5588035E+07	610.44
2.80044	4.1786800E+04	2.5489282E+07	609.98
2.90049	4.1412662E+04	2.5263056E+07	610.03
3.00045	4.0922594E+04	2.4985878E+07	610.56

TABLE 6.2.1-25

DOUBLE-ENDED PUMP SUCTION LOCA

Event	Time (sec)
Rupture	0
Accumulator flow starts	~13
End of blowdown	~32
Assumed initiation of ECCS	36.76
Accumulators empty	~64
Assumed initiation of spray system	234.0
End of reflood	--
Low level alarm of refueling water storage tank	1212.1
Beginning of recirculation phase of safeguards operation	1272.1

TABLE 6.2.1-26a

DELETED

TABLE 6.2.1-26b

DELETED

WBN

TABLE 6.2.1-27a

STEAM LINE BREAK BLOWDOWN

UNIT 1

Time (sec)	Mass Flow Rate, m (lbm/sec)	Energy Flow Rate, e (10^6 Btu/sec)
0	14184	16.862
1.05	14184	16.862
1.65	15220	17.228
1.96	16525	17.929
1.97	36531	23.338
2.55	38961	23.894
3.55	40527	24.392
5.05	41057	24.549
10.05	40729	24.453
13.05	39822	24.173
17.05	38179	23.636

UNIT 2

Time (sec)	Mass Flow Rate, m (lbm/sec)	Energy Flow Rate, e (10^6 Btu/sec)
0	14214	16.898
1.34	14214	16.898
1.94	15260	17.267
2.25	16577	17.975
2.26	36873	23.556
2.84	39326	24.118
3.84	40907	24.621
5.34	41441	24.779
10.34	41111	24.682
13.34	40195	24.400
17.34	38536	23.858

WBN

TABLE 6.2-1-27b

STEAM GENERATOR ENCLOSURE GEOMETRY

Nodes	Volume (ft^3)
51, 56	5551
52, 57	1688
53, 58	1695
54, 59	1826
55, 60	1836

WBN

TABLE 6.2.1-27c

STEAM GENERATOR ENCLOSURE FLOW PATH DATA

Path	k	F	L_I (ft)	D_H (ft)	A (ft^2)	L_{EQ} (ft)	a/A
H51	1.50	0.02	13.30	8.70	170.90	4.00	0.58
H52, H57	0.86	0.02	5.20	4.50	33.20	1.20	0.23
H53, H58	0.96	0.02	4.10	3.20	23.90	0.30	0.17
H54, H59	0.86	0.02	5.20	5.60	41.80	1.20	0.28
H55, H60	0.96	0.02	4.10	5.80	43.80	0.30	0.32
R51, R56	0.24	0.02	9.40	6.40	114.00	7.60	0.27
R52, R57	0.00	0.02	14.80	6.40	114.00	14.80	1.00
R53, R58	0.00	0.02	14.80	7.70	115.00	14.80	1.00
R54, R59	1.50	0.02	5.00	5.30	87.00	3.00	0.70
R55, R60	1.50	0.02	5.50	6.30	93.90	3.40	0.75
A51, A56	0.23	0.02	9.40	7.70	115.00	7.60	0.27

WBN

TABLE 6.2.1-27d
UNIT 1

PEAK DIFFERENTIAL PRESSURE - STEAM GENERATOR ENCLOSURE

Across Enclosure Walls

Nodes	Differential Press. (psi)	Time (sec)
51 - Upper Compartment	28.5	5.21
52 - Upper Compartment	27.5	5.23
53 - Upper Compartment	27.5	5.23
54 - Upper Compartment	27.4	5.25
55 - Upper Compartment	27.4	5.25

Across Steam Generator Vessel

Nodes	Differential Press. (psi)	Time (sec)
53 - 52	0.03	0.017
55 - 54	0.08	0.039

Across Steam Generator Separator Wall

Nodes	Differential Press. (psi)	Time (sec)
55-59	10.97	0.040

WBN

TABLE 6.2.1-27d
UNIT 2

PEAK DIFFERENTIAL PRESSURE - STEAM GENERATOR ENCLOSURE

Across Enclosure Walls

Nodes	Differential Press. (psi)	Time (sec)
51 - Upper Compartment	28.8	5.52
52 - Upper Compartment	27.9	5.54
53 - Upper Compartment	27.9	5.54
54 - Upper Compartment	27.7	5.56
55 - Upper Compartment	27.7	5.56

Across Steam Generator Vessel

Nodes	Differential Press. (psi)	Time (sec)
53 - 52	0.03	0.017
55 - 54	0.08	0.039

Across Steam Generator Separator Wall

Nodes	Differential Press. (psi)	Time (sec)
55-59	11.00	0.040

WBN

TABLE 6.2.1-28 (Sheet 1 of 2)

MASS AND ENERGY RELEASE RATES INTO PRESSURIZER ENCLOSURE

Time (sec)	Mass Flow (10^3 lbm/sec)	Energy Flow (10^6 Btu/sec)
0.0	0.0	0.0
0.00251	5.0473	3.0977
0.00502	5.2333	3.2013
0.01002	5.1051	3.1226
0.01251	5.0746	3.1029
0.01755	5.3833	3.2753
0.02505	5.5402	3.3601
0.03259	5.8746	3.5479
0.04002	5.9221	3.5716
0.05005	5.6865	3.4332
0.07250	5.7877	3.4868
0.09001	5.4917	3.3157
0.11253	5.9404	3.5710
0.13756	5.5454	3.3445
0.15755	5.6392	3.3979
0.17760	5.4721	3.3026
0.19254	5.5189	3.3291
0.21254	5.4725	3.3025
0.23508	5.5465	3.3446
0.27752	5.5345	3.3378
0.35027	5.3649	3.2411

TABLE 6.2.1-28 (Sheet 2 of 2)

MASS AND ENERGY RELEASE RATES INTO PRESSURIZER ENCLOSURE

Time (sec)	Mass Flow (10^3 lbm/sec)	Energy Flow (10^6 Btu/sec)
0.38001	5.2985	3.2031
0.41515	5.3825	3.2507
0.45006	5.2660	3.1842
0.57002	5.2492	3.1738
0.77015	5.1816	3.1336
1.00005	5.1562	3.1169
2.00015	5.0326	3.0400

WBN

TABLE 6.2.1-29

PRESSURIZER GEOMETRIC DATA

Node	Volume (ft^3)
51	2262
52	502
53	667
54	647

Flow Path	k	f	L_1(ft)	D_H(ft)	A(ft^2)	L_{EQ}(ft)	a/A
51-52	0.5	0.02	13.3	3.3	20.9	12.1	0.16
51-53	0.5	0.02	13.8	4.8	27.7	12.1	0.21
51-54	0.5	0.02	13.8	4.8	26.9	12.1	0.21
53-52	0.0	0.02	8.0	3.5	42.6	8.0	0.28
54-52	0.0	0.02	8.0	1.5	18.5	8.0	0.12
53-54	0.0	0.02	8.0	0.9	11.3	8.0	0.06
52-lower compartment	1.0	0.02	12.0	3.3	22.1	12.0	1.00
53-lower compartment	1.0	0.02	12.0	4.8	27.7	12.0	1.00
54-lower compartment	1.0	0.02	12.0	4.8	24.4	12.0	1.00

TABLE 6.2.1-29a

PEAK DIFFERENTIAL PRESSURE - PRESSURIZER ENCLOSURE

Across Enclosure Walls

Nodes	Differential Press. (psi)	Time (sec)
51 - Upper Compartment	11.4	0.06
52 - Upper Compartment	7.7	0.10
53 - Upper Compartment	7.7	0.10
54 - Upper Compartment	7.7	0.10

Across Pressurizer Vessel

Nodes	Differential Press. (psi)	Time (sec)
52 - 53	-0.04	0.038
52 - 54	-0.23	0.046
53 - 54	-0.20	0.050

TABLE 6.2.1-30 (Sheet 1 of 5)

MASS AND ENERGY RELEASE RATES 127 IN^2 COLD LEG

HISTORICAL INFORMATION

Time (sec)	Mass Flow (lbm/sec)	Energy Flow (Btu/sec)	Avg. Enthalpy (Btu/lbm)
0.00000	0	0	0.00
0.00251	1.1982845E+04	6.7296740E+06	561.61
0.00502	1.5308269E+04	8.5974676E+06	561.62
0.00751	1.7398501E+04	9.7720743E+06	561.66
0.01001	1.9131092E+04	1.0741761E+07	561.48
0.01253	1.9948352E+04	1.1193906E+07	561.14
0.01503	1.9716482E+04	1.1050978E+07	560.49
0.01753	2.1905036E+04	1.2288321E+07	560.98
0.02006	2.2170478E+04	1.2426731E+07	560.51
0.02255	2.1560830E+04	1.2069870E+07	559.81
0.02506	2.1315153E+04	1.1923450E+07	559.39
0.02751	2.1626356E+04	1.2094688E+07	559.26
0.03001	2.1729350E+04	1.2147779E+07	559.05
0.03254	2.2084361E+04	1.2345775E+07	559.03
0.03503	2.2542872E+04	1.2603165E+07	559.08
0.03757	2.2895385E+04	1.2800602E+07	559.09
0.04009	2.3203939E+04	1.2973383E+07	559.10
0.04255	2.3446963E+04	1.3108981E+07	559.09
0.04502	2.3464854E+04	1.3115753E+07	558.95
0.04752	2.3298089E+04	1.3017402E+07	558.73
0.05001	2.3145127E+04	1.2927663E+07	558.55
0.05266	2.3018004E+04	1.2853122E+07	558.39
0.05514	2.2950194E+04	1.2812973E+07	558.29
0.05757	2.2904460E+04	1.2785675E+07	558.22
0.06012	2.2779154E+04	1.2713027E+07	558.10
0.06257	2.2510846E+04	1.2559119E+07	557.91
0.06500	2.2164087E+04	1.2360966E+07	557.70
0.06763	2.1888594E+04	1.2203861E+07	557.54
0.07009	2.1850009E+04	1.2182079E+07	557.53
0.07259	2.2019590E+04	1.2278820E+07	557.63
0.07503	2.2242956E+04	1.2406073E+07	557.75
0.07759	2.2352310E+04	1.2468054E+07	557.80
0.08002	2.2278656E+04	1.2425609E+07	557.74
0.08253	2.2036897E+04	1.2287536E+07	557.59
0.08504	2.1670113E+04	1.2078517E+07	557.38
0.08752	2.1266578E+04	1.1848983E+07	557.16
0.09004	2.0857542E+04	1.1617001E+07	556.97
0.09260	2.0466616E+04	1.1395523E+07	556.79
0.09500	2.0201194E+04	1.1245397E+07	556.67
0.09751	2.0053059E+04	1.1161858E+07	556.62
0.10007	2.0025022E+04	1.1146521E+07	556.63

TABLE 6.2.1-30 (Sheet 2 of 5)

MASS AND ENERGY RELEASE RATES 127 IN^2 COLD LEG

HISTORICAL INFORMATION

Time (sec)	Mass Flow (lbm/sec)	Energy Flow (Btu/sec)	Avg. Enthalpy (Btu/lbm)
0.10515	2.0170943E+04	1.1230305E+07	556.76
0.11011	2.0365487E+04	1.1341279E+07	556.89
0.11505	2.0647554E+04	1.1501747E+07	557.05
0.12008	2.0944972E+04	1.1670752E+07	557.21
0.12502	2.0977664E+04	1.1688856E+07	557.20
0.13007	2.0780412E+04	1.1576250E+07	557.08
0.13509	2.0500682E+04	1.1417226E+07	557.92
0.14001	2.0096382E+04	1.1187990E+07	556.72
0.14508	1.9569603E+04	1.0889944E+07	556.47
0.15009	1.9235427E+04	1.0701397E+07	556.34
0.15504	1.9138491E+04	1.0647134E+07	556.32
0.16006	1.9034644E+04	1.0588880E+07	556.30
0.16505	1.8879080E+04	1.0501358E+07	556.24
0.17007	1.8748148E+04	1.0427857E+07	556.21
0.17514	1.8720580E+04	1.0412805E+07	556.22
0.18005	1.8785810E+04	1.0450146E+07	556.28
0.18504	1.8911550E+04	1.0521664E+07	556.36
0.19010	1.9101126E+04	1.0629209E+07	556.47
0.19507	1.9311878E+04	1.0748514E+07	556.58
0.20009	1.9465602E+04	1.0835436E+07	556.65
0.21252	1.9617023E+04	1.0920644E+07	556.69
0.22507	1.9458748E+04	1.0830336E+07	556.58
0.23759	1.9647389E+04	1.0937376E+07	556.68
0.25011	1.9804565E+04	1.1026138E+07	556.75
0.26253	1.9395307E+04	1.0793667E+07	556.51
0.27516	1.8760813E+04	1.0435112E+07	556.22
0.28761	1.8860759E+04	1.0492777E+07	556.33
0.30014	1.9381793E+04	1.0787950E+07	556.60
0.31261	1.9557340E+04	1.0886714E+07	556.66
0.32509	1.9428795E+04	1.0813221E+07	556.56
0.33757	1.9460687E+04	1.0831309E+07	556.57
0.35003	1.9510288E+04	1.0859152E+07	556.59
0.36251	1.9334731E+04	1.0759415E+07	556.48
0.37512	1.9237392E+04	1.0704384E+07	556.44
0.38764	1.9172556E+04	1.0667882E+07	556.41
0.40007	1.9255351E+04	1.0715044E+07	556.47
0.41263	1.9518505E+04	1.0864131E+07	556.61
0.42512	1.9566788E+04	1.0890843E+07	556.60
0.43769	1.9443279E+04	1.0820460E+07	556.51
0.45005	1.9309158E+04	1.0744438E+07	556.44
0.46260	1.9325193E+04	1.0753755E+07	556.46

TABLE 6.2.1-30 (Sheet 3 of 5)

MASS AND ENERGY RELEASE RATES 127 IN^2 COLD LEG

HISTORICAL INFORMATION

Time (sec)	Mass Flow (lbm/sec)	Energy Flow (Btu/sec)	Avg. Enthalpy (Btu/lbm)
0.47515	1.9427001E+04	1.0811564E+07	556.52
0.48751	1.9463982E+04	1.0832327E+07	556.53
0.50010	1.9412566E+04	1.0802979E+07	556.49
0.52505	1.9416927E+04	1.0805655E+07	556.51
0.55001	1.9520981E+04	1.0864335E+07	556.55
0.57500	1.9439249E+04	1.0817886E+07	556.50
0.60009	1.9432289E+04	1.0814194E+07	556.51
0.62502	1.9570908E+04	1.0892620E+07	556.57
0.65001	1.9484134E+04	1.0843384E+07	556.52
0.67502	1.9537413E+04	1.0873742E+07	556.56
0.70006	1.9557525E+04	1.0885106E+07	556.57
0.72503	1.9556471E+04	1.0884559E+07	556.57
0.75008	1.9566953E+04	1.0890551E+07	556.58
0.77503	1.9575425E+04	1.0895394E+07	556.59
0.80011	1.9613175E+04	1.0916838E+07	556.61
0.82503	1.9623035E+04	1.0922366E+07	556.61
0.85012	1.9607042E+04	1.0913377E+07	556.60
0.87505	1.9625149E+04	1.0923689E+07	556.62
0.90005	1.9642366E+04	1.0933451E+07	556.63
0.92504	1.9652418E+04	1.0939158E+07	556.63
0.95005	1.9665495E+04	1.0946566E+07	556.64
0.97509	1.9657157E+04	1.0941870E+07	556.64
1.00024	1.9674801E+04	1.0951903E+07	556.65
1.02501	1.9674211E+04	1.0951587E+07	556.65
1.05002	1.9685832E+04	1.0958208E+07	556.65
1.07501	1.9689581E+04	1.0960360E+07	556.66
1.10003	1.9688612E+04	1.0959861E+07	556.66
1.12501	1.9688440E+04	1.0959833E+07	556.66
1.15013	1.9691682E+04	1.0961746E+07	556.67
1.17512	1.9694412E+04	1.0963374E+07	556.67
1.20008	1.9690643E+04	1.0961334E+07	556.68
1.22506	1.9686074E+04	1.0958870E+07	556.68
1.25010	1.9682378E+04	1.0956913E+07	556.69
1.27506	1.9685597E+04	1.0958900E+07	556.70
1.30002	1.9688096E+04	1.0960455E+07	556.70
1.32505	1.9673388E+04	1.0952302E+07	556.71
1.35006	1.9668391E+04	1.0949690E+07	556.72
1.37504	1.9669445E+04	1.0950509E+07	556.73
1.40009	1.9673705E+04	1.0950139E+07	556.74
1.42508	1.9668652E+04	1.0950505E+07	556.75
1.45004	1.9667081E+04	1.0950053E+07	556.76

TABLE 6.2.1-30 (Sheet 4 of 5)

MASS AND ENERGY RELEASE RATES 127 IN^2 COLD LEG

HISTORICAL INFORMATION

Time (sec)	Mass Flow (lbm/sec)	Energy Flow (Btu/sec)	Avg. Enthalpy (Btu/lbm)
1.47501	1.9675943E+04	1.0955165E+07	556.78
1.50004	1.9668050E+04	1.0950970E+07	556.79
1.52500	1.9665596E+04	1.0949895E+07	556.80
1.55005	1.9671043E+04	1.0953307E+07	556.82
1.57502	1.9666568E+04	1.0951104E+07	556.84
1.60008	1.9662702E+04	1.0949279E+07	556.86
1.62509	1.9658419E+04	1.0947234E+07	556.87
1.65008	1.9652327E+04	1.0944186E+07	556.89
1.67508	1.9641445E+04	1.0938449E+07	556.91
1.70000	1.9631684E+04	1.0933366E+07	556.92
1.72523	1.9622211E+04	1.0928465E+07	556.94
1.75002	1.9611372E+04	1.0922805E+07	556.96
1.77506	1.9600265E+04	1.0917009E+07	556.98
1.80004	1.9586316E+04	1.0909616E+07	556.00
1.82507	1.9570844E+04	1.0901377E+07	557.02
1.85004	1.9558044E+04	1.0894662E+07	557.04
1.87501	1.9547428E+04	1.0889183E+07	557.06
1.90005	1.9533703E+04	1.0881955E+07	557.09
1.92505	1.9518588E+04	1.0873945E+07	557.11
1.95013	1.9504270E+04	1.0866400E+07	557.13
1.97508	1.9490671E+04	1.0854264E+07	557.15
2.00001	1.9475975E+04	1.0851512E+07	557.17
2.02504	1.9460138E+04	1.0843124E+07	557.20
2.05011	1.9443525E+04	1.0834315E+07	557.22
2.07503	1.9425610E+04	1.0824775E+07	557.24
2.10004	1.9406458E+04	1.0814547E+07	557.27
2.12507	1.9386749E+04	1.0804030E+07	557.29
2.15003	1.9366596E+04	1.0793269E+07	557.31
2.17504	1.9344857E+04	1.0781622E+07	557.34
2.20000	1.9321966E+04	1.0769339E+07	557.36
2.22510	1.9298174E+04	1.0756568E+07	557.39
2.25001	1.9274722E+04	1.0743996E+07	557.41
2.27507	1.9250836E+04	1.0731182E+07	557.44
2.30008	1.9225729E+04	1.0717684E+07	557.47
2.32510	1.9199767E+04	1.0703706E+07	557.49
2.35000	1.9189974E+04	1.0698897E+07	557.53
2.37503	1.9159580E+04	1.0682347E+07	557.55
2.40011	1.9117138E+04	1.0659079E+07	557.57
2.42510	1.9108543E+04	1.0654963E+07	557.60
2.45013	1.9096201E+04	1.0648650E+07	557.63
2.47510	1.9042948E+04	1.0633130E+07	557.65

TABLE 6.2.1-30 (Sheet 5 of 5)

MASS AND ENERGY RELEASE RATES 127 IN^2 COLD LEG

HISTORICAL INFORMATION

Time (sec)	Mass Flow (lbm/sec)	Energy Flow (Btu/sec)	Avg. Enthalpy (Btu/lbm)
2.50011	1.9042948E+04	1.0619918E+07	557.68
2.52510	1.9038224E+04	1.0617984E+07	557.72
2.55010	1.9011234E+04	1.0603416E+07	557.74
2.57508	1.8977430E+04	1.0585017E+07	557.77
2.60006	1.8963744E+04	1.0578049E+07	557.80
2.62512	1.8950653E+04	1.0571411E+07	557.84
2.65025	1.8927642E+04	1.0559161E+07	557.87
2.67504	1.8894421E+04	1.0541167E+07	557.90
2.70018	1.8869044E+04	1.0527674E+07	557.93
2.72514	1.8846108E+04	1.0515517E+07	557.97
2.75003	1.8827068E+04	1.0505586E+07	558.00
2.77504	1.8815133E+04	1.0499666E+07	558.04
2.80005	1.8795707E+04	1.0489482E+07	558.08
2.82511	1.8768598E+04	1.0474967E+07	558.11
2.85006	1.8741259E+04	1.0460340E+07	558.14
2.87505	1.8723005E+04	1.0450857E+07	558.18
2.90005	1.8704799E+04	1.0441382E+07	558.22
2.92505	1.8678392E+04	1.0427269E+07	558.25
2.95003	1.8650919E+04	1.0412569E+07	558.29
2.97508	1.8627102E+04	1.0399953E+07	558.32
3.00020	1.8605296E+04	1.0388476E+07	558.36

TABLE 6.2.1-31 (Sheet 1 of 2)

REACTOR CAVITY VOLUMES
HISTORICAL INFORMATION

COMPARTMENT NUMBER	COMPARTMENT LOCATION	VOLUME (ft^3)
1	Break Location	164.595
2	Lower Reactor Cavity	12,000.
3	Reactor Vessel Annulus	1.319
4	Reactor Vessel Annulus	1.938
5	Reactor Vessel Annulus	8.601
6	Reactor Vessel Annulus	8.601
7	Reactor Vessel Annulus	9.825
8	Reactor Vessel Annulus	17.202
9	Reactor Vessel Annulus	9.825
10	Reactor Vessel Annulus	17.202
11	Reactor Vessel Annulus	9.205
12	Reactor Vessel Annulus	17.202
13	Reactor Vessel Annulus	9.206
14	Reactor Vessel Annulus	17.202
15	Reactor Vessel Annulus	9.825
16	Reactor Vessel Annulus	17.202
17	Reactor Vessel Annulus	9.825
18	Reactor Vessel Annulus	17.202
19	Reactor Vessel Annulus	9.206
20	Reactor Vessel Annulus	17.202
21	Lower Containment	60,000.
22	Lower Containment	60,000.
23	Lower Containment	60,000.
24	Lower Containment	60,000.
25	Break Location	165.206
26	Inspection Annulus	165.819
27	Inspection Annulus	165.206
28	Inspection Annulus	164.595
29	Inspection Annulus	165.206
30	Inspection Annulus	165.819
31	Inspection Annulus	165.206
32	Upper Containment	651,000.
33	Reactor Vessel Annulus	1.404
34	Reactor Vessel Annulus	1.404
35	Reactor Vessel Annulus	1.938
36	Reactor Vessel Annulus	8.601
37	Reactor Vessel Annulus	8.601
38	Reactor Vessel Annulus	17.202
39	Reactor Vessel Annulus	17.202
40	Reactor Vessel Annulus	17.202
41	Reactor Vessel Annulus	17.202

TABLE 6.2.1-31 (Sheet 2 of 2)

REACTOR CAVITY VOLUMES
HISTORICAL INFORMATION

COMPARTMENT NUMBER	COMPARTMENT LOCATION	VOLUME (ft^3)
42	Reactor Vessel Annulus	17.202
43	Reactor Vessel Annulus	17.202
44	Reactor Vessel Annulus	17.202
45	Reactor Vessel Annulus	0.602
46	Reactor Vessel Annulus	0.602
47	Upper Reactor Cavity	15,500.
48	Ice Condenser	24,241.
49	Ice Condenser	28,760.
50	Ice Condenser	28,760.
51	Ice Condenser	28,760.
52	Ice Condenser	47,000.
53	Pipe Annulus	150.
54	Inspection Port	17.280
55	Inspection Port	17.280
56	Inspection Port	17.280
57	Inspection Port	17.280
58	Inspection Port	17.280
59	Inspection Port	17.280
60	Inspection Port	17.280
61	Inspection Port	17.280
62	Pipe Annulus	47.
63	Pipe Annulus	47.
64	Pipe Annulus	47.
65	Pipe Annulus	47.
66	Pipe Annulus	47.
67	Pipe Annulus	47.
68	Pipe Annulus	150.

TABLE 6.2.1-32 (Sheet 1 of 4)

FLOW PATH DATA (REACTOR CAVITY)
HISTORICAL INFORMATION

Between Compartments	k	f	Inertia Length (ft)	Hydraulic Diameter (ft)	Flow Area (ft^2)	Equiv. Length (ft)	Area Ratio a/A
1 to 3	0.4	0.02	1.6	0.3	2.7	1.2	0.28
2 to 22	2.9	0.02	28	5.8	36	19	0.0
3 to 34	1.0	0.02	0.7	0.4	1.2	0.7	1.0
4 to 35	0.0	0.02	3.6	0.4	0.7	3.6	1.0
5 to 36	0.0	0.02	3.3	0.4	2.6	3.3	1.0
6 to 37	0.0	0.02	3.3	0.4	2.6	3.3	1.0
7 to 9	1.0	0.02	4.9	0.4	1.0	4.1	1.0
8 to 10	0.0	0.02	6.6	0.4	2.6	6.6	1.0
9 to 11	1.04	0.02	4.6	0.4	1.0	3.7	0.46
10 to 12	0.0	0.02	6.6	0.4	2.6	6.6	1.0
11 to 13	1.04	0.02	4.8	0.4	1.0	4.0	0.46
12 to 14	0.0	0.02	6.6	0.4	2.6	6.6	1.0
13 to 15	1.04	0.02	4.6	0.4	1.0	3.7	0.46
14 to 16	0.0	0.02	6.6	0.4	2.6	6.6	1.0
15 to 17	1.04	0.02	4.9	0.4	1.0	4.1	0.5
16 to 18	0.0	0.02	6.6	0.4	2.6	6.6	1.0
17 to 19	1.0	0.02	4.6	0.4	1.0	3.7	0.46
18 to 20	0.0	0.02	6.6	0.4	2.6	6.6	1.0
19 to 4	1.0	0.02	5.4	0.4	0.5	5.4	1.0
20 to 6	0.0	0.02	5.0	0.4	2.6	5.0	1.0
21 to 22	2.0	0.02	38	40	1560	38	0.43
22 to 23	3.0	0.02	38	40	1560	38	0.47
23 to 24	2.0	0.02	38	40	1560	38	0.43
24 to 21	3.0	0.02	32	8.0	100	27	0.09
25 to 7	2.8	0.02	3.0	0.2	1.1	1.7	0.12
26 to 9	2.8	0.02	3.0	0.2	1.1	1.7	0.12
27 to 11	2.8	0.02	3.1	0.2	1.3	1.7	0.13
28 to 13	2.8	0.02	3.1	0.2	1.3	1.7	0.13
29 to 15	2.8	0.02	3.0	0.2	1.1	1.7	0.12
30 to 17	2.8	0.02	3.0	0.2	1.1	1.7	0.12
31 to 19	2.8	0.02	3.1	0.2	1.3	1.7	0.13
33 to 3	1.0	0.02	0.7	0.4	1.2	0.7	0.99
34 to 7	1.0	0.02	3.6	0.4	1.2	3.3	0.66
35 to 7	1.0	0.02	5.4	0.4	0.5	5.4	1.0
36 to 38	0.0	0.02	5.0	0.4	2.2	5.0	1.0
37 to 8	0.0	0.02	5.0	0.4	2.6	5.0	1.0
38 to 39	0.0	0.02	6.6	0.4	2.6	6.6	1.0
39 to 40	0.0	0.02	6.6	0.4	2.6	6.6	1.0
40 to 41	0.0	0.02	6.6	0.4	2.6	6.6	1.0
41 to 42	0.0	0.02	6.6	0.4	2.6	6.6	1.0
42 to 43	0.0	0.02	6.6	0.4	2.6	6.6	1.0
43 to 44	0.0	0.02	6.6	0.4	2.6	6.6	1.0

TABLE 6.2.1-32 (Sheet 2 of 4)

FLOW PATH DATA (REACTOR CAVITY)

HISTORICAL INFORMATION

Between Compartments	k	f	Inertia Length (ft)	Hydraulic Diameter (ft)	Flow Area (ft^2)	Equiv. Length (ft)	Area Ratio a/A
44 to 5	0.0	0.02	5.0	0.4	2.6	5.0	1.0
45 to 3	1.0	0.02	0.8	0.4	0.5	0.8	0.78
46 to 3	1.0	0.02	0.8	0.4	0.5	0.8	0.78
53 to 1	0.4	0.02	7.6	0.83	5.5	6.8	0.11
54 to 1	0.5	0.02	2.6	2.5	4.9	1.9	0.25
55 to 25	0.5	0.02	2.6	2.5	4.9	1.9	0.25
56 to 26	0.5	0.02	2.6	2.5	4.9	2.0	0.26
57 to 27	0.5	0.02	2.6	2.5	4.9	1.9	0.25
58 to 28	0.5	0.02	2.6	2.5	4.9	1.9	0.25
59 to 29	0.5	0.02	2.6	2.5	4.9	1.9	0.25
60 to 30	0.5	0.02	2.6	2.5	4.9	2.0	0.26
61 to 31	0.5	0.02	2.6	2.5	4.9	1.9	0.25
62 to 25	0.4	0.02	3.0	0.83	5.5	2.2	0.11
63 to 26	0.4	0.02	3.0	0.83	5.5	2.2	0.11
64 to 27	0.4	0.02	3.0	0.83	5.5	2.2	0.11
65 to 28	0.4	0.02	3.0	0.83	5.5	2.2	0.11
66 to 29	0.4	0.02	3.0	0.83	5.5	2.2	0.11
67 to 30	0.4	0.02	3.0	0.83	5.5	2.2	0.11
68 to 31	0.4	0.02	7.6	0.83	5.5	6.8	0.11
1 to 19	0.4	0.02	3.1	0.2	1.3	1.7	0.13
2 to 6	3.7	0.02	6.5	0.4	0.7	6.4	0.0
4 to 45	0.6	0.02	2.4	0.4	0.7	2.4	0.95
6 to 5	0.0	0.02	13.	0.4	0.7	13.	1.0
7 to 38	1.0	0.02	6.0	0.4	0.3	4.7	0.23
8 to 2	3.7	0.02	6.6	0.4	1.3	6.4	0.0
9 to 39	9.2	0.02	6.7	0.4	0.3	4.9	0.23
10 to 2	3.7	0.02	6.6	0.4	1.3	6.4	0.0
11 to 40	1.0	0.02	5.5	0.4	0.2	4.5	0.17
12 to 2	3.7	0.02	6.6	0.4	1.3	6.4	0.0
13 to 41	9.2	0.02	6.0	0.4	0.2	4.6	0.17
14 to 2	3.7	0.02	6.6	0.4	1.3	6.4	0.0
15 to 42	1.0	0.02	6.0	0.4	0.3	4.7	0.23
16 to 2	3.7	0.02	6.6	0.4	1.3	6.4	0.0
17 to 43	9.2	0.02	6.7	0.4	0.3	4.9	0.23
18 to 2	3.7	0.02	6.6	0.4	1.3	6.4	0.0
19 to 44	1.0	0.02	5.5	0.4	0.2	4.5	0.17
20 to 2	3.7	0.02	6.6	0.4	1.3	6.4	0.0
21 to 48	.7837	0.0	10.36	1.0	265.87	0.0	0.096
22 to 48	.7837	0.0	10.36	1.0	5	0.0	0.096
23 to 48	.7837	0.0	10.36	1.0	265.87	0.0	0.096

TABLE 6.2.1-32 (Sheet 3 of 4)

FLOW PATH DATA (REACTOR CAVITY)

HISTORICAL INFORMATION

24 to 48	.7837	0.0	10.36	1.0	5	0.0	0.096
25 to 3	0.4	0.02	1.6	0.3	265.87	1.2	0.28
26 to 7	0.4	0.02	3.0	0.2	5	1.7	0.12
27 to 9	0.4	0.02	3.0	0.2	265.87	1.7	0.12
28 to 11	0.4	0.02	3.1	0.2	5	1.7	0.13
29 to 13	0.4	0.02	3.1	0.2	2.7	1.7	0.13
30 to 15	0.4	0.02	3.0	0.2	1.1	1.7	0.12
31 to 17	0.4	0.02	3.0	0.2	1.1	1.7	0.12
33 to 46	2.2	0.02	3.3	0.4	1.3	3.3	0.25
34 to 46	2.2	0.02	3.3	0.4	1.3	3.3	0.25
35 to 47	1.1	0.02	3.0	0.4	1.1	1.5	0.0
37 to 36	0.0	0.02	13.	0.4	1.1	13.	1.0
38 to 8	0.0	0.02	13.	0.4	0.4	13.	1.0
39 to 10	0.0	0.02	13.	0.4	0.4	13.	1.0
					0.7		
					0.7		
					1.3		
					1.3		
40 to 12	0.0	0.02	13.	0.4	1.3	13.	1.0
41 to 14	0.0	0.02	13.	0.4	1.3	13.	1.0
42 to 16	0.0	0.02	13.	0.4	1.3	13.	1.0
43 to 18	0.0	0.02	13.	0.4	1.3	13.	1.0
44 to 20	0.0	0.02	13.	0.4	1.3	13.	1.0
45 to 33	0.0	0.02	3.3	0.4	0.4	3.3	0.25
48 to 49	0.0	0.1055	8.733	0.855	989.01	8.0	0.230
49 to 50	0.0	0.0592	12.278	0.855	982.47	16.0	0.239
50 to 51	0.0	0.0592	12.278	0.855	982.47	16.0	0.359
51 to 52	0.87979	0.1249	8.8558	0.855	982.47	8.0	0.359
52 to 32	1.43	0.0	2.8	1.0	2003.1		0.269
53 to 25	0.4	0.02	7.6	0.83	5.5	6.8	0.11
54 to 47	1.0	0.02	1.9	2.5	4.9	1.8	0.0
55 to 47	1.0	0.02	1.9	2.5	4.9	1.8	0.0
56 to 47	1.0	0.02	1.9	2.5	4.9	1.8	0.0
57 to 47	1.0	0.02	1.9	2.5	4.9	1.8	0.0
58 to 47	1.0	0.02	1.9	2.5	4.9	1.8	0.0
59 to 47	1.0	0.02	1.9	2.5	4.9	1.8	0.0
60 to 47	1.0	0.02	1.9	2.5	4.9	1.8	0.0
61 to 47	1.0	0.02	1.9	2.5	4.9	1.8	0.0
62 to 26	0.4	0.02	3.0	0.83	5.5	2.2	0.11
63 to 27	0.4	0.02	3.0	0.83	5.5	2.2	0.11
64 to 28	0.4	0.02	3.0	0.83	5.5	2.2	0.11
65 to 29	0.4	0.02	3.0	0.83	5.5	2.2	0.11
66 to 30	0.4	0.02	3.0	0.83	5.5	2.2	0.11
67 to 31	0.4	0.02	3.0	0.83	5.5	2.2	0.11
68 to 1	0.4	0.02	7.6	0.83	5.5	8.8	0.11

TABLE 6.2.1-32 (Sheet 4 of 4)

FLOW PATH DATA (REACTOR CAVITY)

HISTORICAL INFORMATION

1 to 25	1.0	0.02	5.4	1.5	9.6	2.6	0.47
2 to 37	3.7	0.02	6.5	0.4	0.7	6.4	0.0
4 to 47	1.1	0.02	3.0	0.4	0.7	1.5	1.0
5 to 46	9.2	0.02	7.8	2.0	0.5	18.	1.0
7 to 47	1.1	0.02	3.0	0.4	1.4	2.9	0.0
9 to 47	1.1	0.02	3.0	0.4	1.4	2.9	0.0
11 to 47	1.1	0.02	3.0	0.4	1.4	2.9	0.0
13 to 47	1.1	0.02	3.0	0.4	1.4	2.9	0.0
15 to 47	1.1	0.02	3.0	0.4	1.4	2.9	0.0
17 to 47	1.1	0.02	3.0	0.4	1.4	2.9	0.0
19 to 47	1.1	0.02	3.0	0.4	1.4	2.9	0.0
21 to 47	3.8	0.02	5.8	5.1	26.	4.0	0.04
22 to 47	3.8	0.02	9.1	12.	74.	4.2	0.10
23 to 47	3.8	0.02	8.3	11.	62.	4.0	0.08
24 to 47	3.9	0.02	6.3	5.5	32.	4.0	0.04
25 to 26	1.0	0.02	5.3	1.4	9.1	2.3	0.44
26 to 27	1.0	0.02	5.3	1.4	9.1	2.6	0.44
27 to 28	1.0	0.02	5.4	1.5	9.6	2.6	0.47
28 to 29	1.0	0.02	5.4	1.5	9.6	2.6	0.47
29 to 30	1.0	0.02	5.3	1.4	9.1	2.6	0.44
30 to 31	1.0	0.02	5.3	1.4	9.1	2.6	0.44
31 to 1	1.0	0.02	5.4	1.5	9.6	2.6	0.47
33 to 19	1.0	0.02	3.6	0.4	1.2	3.3	0.61
35 to 46	0.6	0.02	2.4	0.4	0.7	2.4	0.95
36 to 46	9.2	0.02	7.8	2.0	0.5	18.	0.95
45 to 34	0.0	0.02	3.3	0.4	0.4	3.3	0.25
53 to 21	1.0	0.02	7.2	17.	11.0	8.8	0.0
62 to 21	1.0	0.02	2.5	1.5	11.0	2.1	0.0
63 to 22	1.0	0.02	2.5	1.5	11.0	2.1	0.0
64 to 22	1.0	0.02	2.5	1.7	11.0	2.1	0.0
65 to 23	1.0	0.02	2.5	1.7	11.0	2.1	0.0
66 to 23	1.0	0.02	2.5	1.5	11.0	2.1	0.0
67 to 24	1.0	0.02	2.5	1.5	11.0	2.1	0.0
68 to 24	1.0	0.02	7.2	1.7	11.0	6.8	0.0

TABLE 6.2.1-33 (Sheet 1 of 2)

CONTAINMENT DATA (ECCS ANALYSIS)

I. Conservatively High Estimate of Containment Net Free Volume

Containment Area	Volume (ft^3)
Upper Compartment	651,000
Lower Compartment	271,400
Ice Condenser	169,400
Dead-Ended Compartments (includes all accumulator rooms, both fan compartments, instrument room pipe tunnel)	129,900

II. Initial Conditions

A.	Containment Pressure	15.0 psia
B.	Lowest Operational Containment Temperature for the Upper, Lower, and Dead-Ended Compartments	85°F 100°F
C.	Highest Refueling Water Storage Tank Temperature	100°F
D.	Lowest Temperature Outside Containment	5°F
E.	Highest Initial Spray Temperature	100°F
F.	Lowest Annulus Temperature	40°F

III. Structural Heat Sinks**

A. For Each Surface

1. Description of Surface
2. Conservatively High Estimate of Area Exposed to Containment Atmosphere — See Tables 6.2.1-34 through 6.2.1-36
3. Location in Containment by Compartment

TABLE 6.2.1-33 (Sheet 2 of 2)

CONTAINMENT DATA (ECCS ANALYSIS) (cont'd)

B. For Each Separate Layer of Each Surface		
	1. Material	
	2. Conservatively Large Estimate of Layer Thickness	See Tables 6.2.1-34 through 6.2.1-36
	3. Conservatively High Value of Material Conductivity	See Tables 6.2.1-34 through 6.2.1-36
	4. Conservatively High Value of Volumetric Heat Capacity	See Tables 6.2.1-34 through 6.2.1-36

IV. Spray System	
A. Runout Flow for a Spray Pump (Containment Spray)***	7700 gpm
B. Number of Spray Pumps Operating with No Diesel Failure	2/Unit
C. Number of Spray Pumps Operating with One Diesel Failure	1/Unit
D. Assumed Post Accident Initiation of Spray System	25 sec
V. Deck Fan	
A. Fastest Post Accident Initiation of Deck Fans	10 min
B. Conservatively High Flow Rate Per Fan	42,000 cfm
VI. Conservatively Low Hydrogen Skimmer System Flow Rate	100 cfm/each

** Structural heat sinks should also account for any surfaces neglected in containment integrity analysis.

*** Runout flow is for a break immediately downstream of the pump. In that event, the spray water will not enter the containment.

TABLE 6.2.1-34

MAJOR CHARACTERISTICS OF STRUCTURAL HEAT SINKS
INSIDE SEQUOYAH NUCLEAR PLANT CONTAINMENT - UPPER COMPARTMENT

Structure	Heat Transfer Area (ft^2)	Thickness and Material (as noted)	Thermal Conductivity (Btu/ft-hr-°F)	Volume Heat Capacity (Btu/ft^3-°F)
Operating Deck	4,452	1.1 ft concrete	0.84	30.24
	7,749	6.3 mils coating 1.1 ft concrete	0.087 0.84	29.8 30.24
	672	1.6 ft concrete	0.84	30.24
	11,445	6.3 mils coating 1.6 ft concrete	0.087 0.84	 30.24
	4,032	0.26 in. stainless steel 1.6 ft concrete	9.87 0.84	59.22 30.24
	798	15.7 mils coating 1.6 ft concrete	0.087 0.84	29.8 30.24
Containment Shell	22,890	7.8 mils coating 0.46 in. carbon steel	0.21 27.3	29.8 30.24
	18,375	7.8 mils coating 0.58 in. carbon steel	0.21 27.3	29.8 59.22
	2,100	7.8 mils coating 1.51 in. carbon steel	0.21 27.3	29.8 59.22
Miscellaneous Steel	4,095	7.8 mils coating 0.26 in. carbon steel	0.21 27.3	29.8 59.22
	3,559	7.8 mils coating	0.21 27.3	29.8 59.22
	3,539	7.8 mils coating 0.72 in. carbon steel	0.21 27.3	29.8 59.22
	273	7.8 mils coating 1.57 in. carbon steel	0.21 27.3	29.8 59.2

TABLE 6.2.1-35 (Sheet 1 of 2)

MAJOR CHARACTERISTICS OF STRUCTURAL HEAT SINKS
INSIDE SEQUOYAH NUCLEAR PLANT CONTAINMENT - LOWER COMPARTMENT

Structure	Heat Transfer Area (ft^2)	Thickness and Material (as noted)	Thermal Conductivity (Btu/ft-hr-°F)	Volume Heat Capacity (Btu/ft^3-°F)
Operating Deck	7,507	1.1 ft concrete	0.84	30.24
	2,971	1.6 mils coating 1.1 ft concrete	0.087 0.84	29.8 30.24
	2,131	1.6 ft concrete	0.84	30.24
	789	6.3 mils coating 1.84 ft concrete	0.087 0.84	29.8 30.24
	2,646	2.1 ft concrete	0.84	30.24
	210	6.3 mils coating 2.1 ft concrete	0.087 0.84	29.8 30.24
Crane Wall	14,752	1.6 ft concrete	0.84	30.24
	3,570	6.3 mils coating 1.6 ft concrete	0.087 0.84	29.8 30.24
Containment Floor	567	1.6 ft concrete	0.84	30.24
	7,612	6.3 mils coating 1.6 ft concrete	0.087 0.84	29.8 30.24
Interior Concrete	3,780	1.1 ft concrete	0.84	30.24
	567	1.1 ft concrete	0.84	30.24
	2,992	2.1 ft concrete	0.84	30.24
	2,384	0.26 in. stainless steel 2.1 ft concrete	9.8 0.84	59.2 30.24
	2,373	2.1 ft concrete	0.84	30.24
	1,480	6.3 mils coating 2.1 ft concrete	0.087 0.84	29.8 30.24

TABLE 6.2.1-35 (Sheet 2 of 2)

MAJOR CHARACTERISTICS OF STRUCTURAL HEAT SINKS
INSIDE SEQUOYAH NUCLEAR PLANT CONTAINMENT - LOWER COMPARTMENT

Structure	Heat Transfer Area (ft^2)	Thickness and Material (as noted)	Thermal Conductivity (Btu/ft-hr-°F)	Volume Heat Capacity (Btu/ft^3-°F)
Miscellaneous Steel	12,915	7.8 mils coating	0.22	14.7
		0.53 in. carbon steel	27.3	59.2
	7,560	7.8 mils coating	0.22	14.7
		0.78 in. carbon steel	27.3	59.2
	5,250	7.8 mils coating	0.22	14.7
		1.1 carbon steel	27.3	59.2
	2,625	7.8 mils coating	0.22	14.7
		1.45 in. carbon steel	27.3	59.2
	1,575	7.8 mils coating	0.22	14.7
		1.7 in. carbon steel	27.3	59.2

TABLE 6.2.1-36

MAJOR CHARACTERISTICS OF STRUCTURAL HEAT SINKS
INSIDE SEQUOYAH NUCLEAR PLANT CONTAINMENT - DEAD-ENDED COMPARTMENT

Structure	Heat Transfer Area (ft^2)	Thickness and Material (as noted)	Thermal Conductivity (Btu/ft-hr-°F)	Volume Heat Capacity (Btu/ft^3-°F)
Containment Shell	3,045	7.8 mils coating	0.22	14.7
		0.78 in. carbon steel	27.3	59.2
	4,305	7.8 mils coating	0.22	14.7
		1.1 in. carbon steel	27.3	59.2
	4,305	7.8 mils coating	0.22	14.7
		1.25 in. carbon steel	27.3	59.2
	3,780	7.8 mils coating	0.22	14.7
		1.37 in. carbon steel	27.3	59.2
	4,305	7.8 mils coating	0.22	14.7
		1.51 in. carbon steel	27.3	59.2
Crane Wall	7,255	1.6 ft concrete	0.84	30.24
	3,801	6.3 mils coating	0.087	14.7
		1.58 ft concrete	0.84	30.24
Containment Floor	4,809	6.3 mils coating	0.087	14.7
		2.1 ft concrete	0.84	30.24
Interior Concrete	9,870	1.1 ft concrete	0.84	30.24
	3,948	6.3 mils coating	0.087	14.7
		1.1 ft concrete	0.84	30.24
	5,376	1.58 ft concrete	0.84	30.24

Table 6.2.1-37

MAXIMUM REVERSE PRESSURE DIFFERENTIAL PRESSURE ANALYSIS
BASE CASE

Westinghouse ECCS structural heat transfer model

Sprays at runout flow

Offsite power available spray start time

Minimum containment temperature

Dead-ended volume is swept

Max. reverse differential pressure = 0.65 psi

Case	Variable	Change in Max. dP (psi)
1.	Ice condenser flow through the drains acts as 50% thermal efficient spray	+0.2
2.	Same as Case 1, except 100% thermal efficiency	+0.4
3.	Maximum containment temperature	-0.2
4.	Heat transfer coefficient to sump equals 5 times H_{max}	<0.1
5.	Same as Case 2, except drain flow rate times 1.5	+0.6
6.	Combination of Cases 2 and 4	+0.4
7.	1 bay of ice condenser doors remains open	-0.65
8.	Same as Case 6 except Equation (3) written as $H = H_{stag} + [H_{max} - H_{stag}]\ e^{-.025\ [t-tp]}$	+0.55
9.	Same as Case 6 except 5 times upper to lower resistance	+2.0
10.	RWST temperature = 105°F	+0.2

Table 6.2.1-38 (Sheet 1 of 3)

ICE CONDENSER STEAM EXIT FLOW VS. TIME VS. SUMP TEMPERATURE

Time (sec)	Sump Temp. (°F)	Ice Condenser Steam Exit Flow (lb/sec)
13.1	190.3	-1.74
13.8	190.6	-1.63
14.4	190.7	-1.76
15.0	190.9	-1.54
15.4	191.1	-1.37
15.9	191.2	-1.23
16.3	191.3	-.13
16.6	191.4	-.09
17.0	191.5	-.09
17.4	191.6	-.08
17.8	191.7	-.08
18.2	191.8	-.07
18.6	191.9	-.07
19.0	192.0	-.07
19.3	192.1	-.07
19.7	192.2	-.06
20.0	192.3	-1.04
20.3	192.4	-.93
20.9	192.5	-1.17
21.5	192.7	-1.43
21.8	192.8	-2.24
22.4	192.9	-2.95
23.0	193.1	-2.85
23.6	193.2	-2.64
23.9	193.3	-2.53
24.5	193.4	-2.34
25.1	193.8	-2.17
25.4	194.0	-2.05
25.7	194.1	-1.94

Table 6.2.1-38 (Sheet 2 of 3)

ICE CONDENSER STEAM EXIT FLOW VS. TIME VS. SUMP TEMPERATURE

Time (sec)	Sump Temp. (°F)	Ice Condenser Steam Exit Flow (lb/sec)
26.0	194.2	-1.85
26.6	194.6	-1.69
27.2	194.8	-1.58
27.5	194.9	-1.53
28.0	195.2	-1.45
29.5	195.6	-1.40
30.1	195.8	-1.42
30.7	196.0	-1.44
31.3	196.2	-1.45
31.9	196.3	-1.45
32.5	196.4	-1.43
33.1	196.5	-1.40
33.7	196.6	-1.36
34.3	196.8	-1.31
34.9	196.9	-1.26
35.5	196.9	-1.20
36.0	197.0	-1.115
36.9	197.2	-0.96
37.9	197.3	-0.80
38.9	197.4	-0.63
40.1	197.4	-0.44
41.3	197.5	-0.29
42.2	197.5	-0.20
44.0	197.4	-.09
44.9	197.3	-0.4
45.4	197.3	.12
46.7	197.2	.19
47.6	197.0	.20
48.9	196.9	.19

Table 6.2.1-38 (Sheet 3 of 3)

ICE CONDENSER STEAM EXIT FLOW VS. TIME VS. SUMP TEMPERATURE

Time (sec)	Sump Temp. (°F)	Ice Condenser Steam Exit Flow (lb/sec)
49.8	196.7	.17
51.2	196.5	.12
52.3	196.4	.07
53.6	196.1	.01
54.4	196.0	-.01
55.2	195.9	-.03
56.2	195.7	-.05
57.1	195.5	-.07
58.0	195.4	-.10
59.0	195.2	-.17
59.9	195.0	-.14
60.9	194.9	-.15
61.6	194.7	-.17
62.8	194.5	-.18
63.7	194.3	-.20
64.7	194.2	-.22
65.6	194.0	-.24
66.6	193.8	-.31
67.5	193.6	-.41
68.4	193.5	-.60
69.4	193.3	.20
70.3	193.2	.63
71.3	193.0	.84
72.2	192.9	1.05
73.2	192.7	1.25
74.1	192.6	1.39
75.1	192.5	1.54
76.0	192.4	1.66
77.0	192.3	1.78

TABLE 6.2.1-39 (Sheet 1 of 3)
UNIT 1

MASS AND ENERGY RELEASE RATES FOR SPECIFIED STEAM LINE BREAKS

I. Run 1 - 1.4 ft^2 Break, 100.6% Power, MFIV Failure

Time (sec)	Mass Flow Rate (lbm/sec)	Enthalpy (Btu/sec)
.00	.00	.00
.20	10371.88	1187.12
3.00	9033.73	1192.53
3.20	8961.	1193.
3.40	9671.	1193.
9.40	8390.27	1197.30
9.60	2308.78	1198.59
19.00	1704.32	1203.37
22.60	1528.31	1204.12
26.20	1396.01	1204.40
30.00	1287.64	1204.48
38.00	1128.43	1204.30
46.20	1030.01	1203.97
54.40	971.66	1203.67
62.40	935.95	1203.45
78.60	896.37	1203.16
95.00	877.33	1203.00
209.60	864.25	1202.89
213.40	834.65	1202.60
225.00	675.30	1200.46
240.40	447.40	1194.59
248.20	354.03	1190.46
252.00	319.65	1188.62
255.80	292.07	1187.02
263.60	255.53	1184.53
271.20	237.43	1183.11
279.00	228.69	1182.37
302.00	222.25	1181.81
600.40	223.41	1181.83
601.20	208.63	1180.09
603.80	142.45	1172.94
605.60	115.41	1168.40
607.40	93.00	1164.32
611.20	57.71	1155.28
612.40	45.77	1152.75
614.00	27.10	1150.57
614.20	23.95	1150.42
614.80	21.27	1150.35
618.00	.00	.00
700.00	.00	.00

TABLE 6.2.1-39 (Sheet 2 of 3)
UNIT 1

MASS AND ENERGY RELEASE RATES FOR SPECIFIED STEAM LINE BREAKS (cont'd)

II. Run 2 - 0.6 ft^2 Split, 30% Power, AFW Runout

Time (sec)	Mass Flow Rate (lbm/sec)	Energy Release Rate (Btu/sec)
.00	.00	.00
0.20	1377.37	1635205.38
14.40	1297.93	1544515.50
19.20	1148.52	1372031.75
39.80	799.93	961796.44
74.20	591.91	712912.75
108.60	514.54	619721.06
600.20	506.66	610231.19
642.40	558.3	672400.
650.20	558.3	672400.
697.00	551.12	663808.25
701.80	518.67	624704.75
706.40	439.88	529580.69
711.20	283.70	340478.06
715.80	176.42	210478.03
718.20	137.63	163588.05
721.80	127.18	150999.95
725.00	128.03	152020.34
728.60	126.86	150609.86
732.00	125.64	149143.52
735.40	124.38	147622.28
739.00	122.78	145698.16
742.40	121.14	143721.11
745.80	118.72	140802.95
749.00	116.43	138053.45
752.20	113.71	134779.92
755.40	110.29	130657.75
766.40	90.70	107101.64
784.40	35.84	41662.02
791.20	18.15	20908.96
792.40	13.65	15709.18
793.20	9.16	10535.82
793.40	.00	.00
800.00	.00	.00

TABLE 6.2.1-39 (Sheet 3 of 3)
UNIT 1

MASS AND ENERGY RELEASE RATES FOR SPECIFIED STEAM LINE BREAKS (cont'd)

III. Run 3 - 0.35 ft^2 Split, 30% Power, AFW Runout

Time (sec)	Mass Flow Rate (lbm/sec)	Energy Release Rate (Btu/sec)
.00	.00	.00
.20	807.03	958075.38
20.80	768.03	913652.69
49.80	535.07	642170.50
99.40	406.46	489273.72
198.60	334.69	403113.91
600.20	342.0	411900.
650.40	380.0	457600.
683.20	381.8	459700.
1013.40	358.95	432301.00
1019.20	328.69	395899.63
1031.00	183.54	220495.66
1036.80	129.39	154851.70
1042.60	91.15	108555.84
1050.20	81.96	97461.79
1055.60	77.52	92110.46
1056.60	81.11	96429.80
1062.80	78.98	93867.66
1068.60	78.04	92727.79
1074.40	77.67	92278.16
1080.20	75.92	90173.65
1085.80	74.58	88556.09
1091.20	72.66	86245.25
1096.40	70.91	84130.01
1101.20	69.47	82400.01
1106.20	67.25	79727.20
1110.80	64.32	76203.61
1115.20	61.83	73196.51
1119.80	58.39	69060.92
1169.40	7.73	8893.20
1169.80	7.09	8159.84
1170.20	6.36	7316.84
1170.80	4.74	5456.56
1171.00	.00	.00
1200.00	.00	.00

TABLE 6.2.1-39 (Sheet 1 of 5)
UNIT 2

MASS AND ENERGY RELEASE RATES FOR SPECIFIED STEAM LINE BREAKS

Full-Power DER with an MSIV Single Failure Assumed

Time (sec)	Mass Flow Rate (lbm/sec)	Enthalpy (Btu/sec)
	Steam Piping Blowdown	
0.00	0.00	0.00
0.0001	10409.30	1191.30
3.61	10409.30	1191.30
3.6101	0.00	0.00
	Faulted Steam Generator Blowdown	
0.00	0.00	0.00
0.20	11481.30	1190.20
1.00	10907.97	1191.97
3.00	9895.68	1195.24
5.20	9256.51	1197.28
9.20	8316.87	1199.72
9.40	2139.00	1200.20
13.40	1914.41	1202.04
16.60	1711.18	1203.33
19.80	1540.69	1204.08
23.00	1406.91	1204.39
27.00	1278.93	1204.47
32.00	1164.50	1204.37
37.00	1083.59	1204.17
42.00	1028.99	1203.96
47.20	990.74	1203.78
57.40	946.40	1203.52
67.40	923.28	1203.36
87.60	899.42	1203.18
134.00	885.44	1203.07
174.20	883.29	1203.05
177.40	851.48	1202.76
207.00	421.03	1193.55
210.40	378.48	1191.65

TABLE 6.2.1-39 (Sheet 2 of 5)
UNIT 2

MASS AND ENERGY RELEASE RATES FOR SPECIFIED STEAM LINE BREAKS

Time (sec)	Mass Flow Rate (lbm/sec)	Enthalpy (Btu/sec)
213.60	343.75	1189.93
216.80	314.06	1188.30
220.20	287.73	1186.74
223.40	268.03	1185.43
226.80	251.79	1184.24
230.20	239.71	1183.28
233.40	230.95	1182.55
240.00	219.45	1181.55
246.60	213.37	1181.00
253.20	210.18	1180.70
600.40	208.75	1180.49
606.60	102.37	1166.10
612.00	51.14	1153.78
617.00	21.08	115035
617.20	0.00	0.00
800.00	0.00	0.00

30%-Power 0.60 ft^2 Split Break with an AFW Runout Protection Single Failure Assumed

Time (sec)	Mass Flow Rate (lbm/sec)	Energy Flow Rate (Btu/sec)
0.00	0.00	0.00
0.20	1372.60	1630221.00
5.80	1293.47	1539423.00
15.60	1280.15	1254195.00
20.00	1127.00	1346983.00
24.00	1026.90	1229945.00
28.00	949.95	1139445.00
32.00	887.53	1065723.00
36.00	836.01	1004670.00
40.00	793.05	953609.31
48.00	726.08	873769.31
56.00	676.46	814408.00
64.00	638.43	768816.50

TABLE 6.2.1-39 (Sheet 3 of 5)
UNIT 2

MASS AND ENERGY RELEASE RATES FOR SPECIFIED STEAM LINE BREAKS

Time (sec)	Mass Flow Rate (lbm/sec)	Energy Flow Rate (Btu/sec)
72.00	608.55	732911.88
88.00	566.18	681943.62
104.00	541.48	652195.81
120.00	530.83	639368.12
449.40	537.84	647812.69
456.80	519.09	625206.38
464.00	483.78	582613.00
478.80	382.17	459791.31
486.00	343.19	412601.81
493.40	316.99	380855.09
500.80	301.31	361859.19
508.00	292.44	351098.69
522.80	284.40	341359.00
600.60	283.64	340435.41
602.80	264.49	317193.59
603.80	247.57	296662.81
605.00	219.86	263089.19
607.00	181.54	216675.30
611.20	163.08	194360.09
669.00	155.91	185688.00
830.00	99.66	117867.70
843.20	83.80	98821.17
862.20	56.24	65881.32
889.60	20.15	23228.71
894.80	8.88	10210.10
895.00	0.00	0.00
900.00	0.00	0.00

30%-Power 0.35 ft^2 Split Break with an AFW Runout Protection Single Failure Assumed

Time (sec)	Mass Flow Rate (lbm/sec)	Energy Flow Rate (Btu/sec)
0.00	0.00	0.00
0.20	801.77	952220.69

TABLE 6.2.1-39 (Sheet 4 of 5)
UNIT 2

MASS AND ENERGY RELEASE RATES FOR SPECIFIED STEAM LINE BREAKS

Time (sec)	Mass Flow Rate (lbm/sec)	Energy Flow Rate (Btu/sec)
8.40	764.37	909340.50
17.60	778.91	926066.81
20.80	772.19	918356.62
25.20	710.33	847131.88
30.80	653.99	781720.81
36.40	611.00	731486.38
44.00	565.37	677894.88
49.60	538.75	646514.81
60.80	498.11	598437.50
71.80	468.71	563523.50
83.00	445.59	535991.81
105.40	412.10	496009.19
127.60	389.13	468513.69
150.00	372.27	448295.31
172.40	361.20	435001.81
194.60	357.84	430964.59
362.80	364.89	439435.00
600.40	366.99	441951.19
635.20	396.98	477914.50
762.20	393.63	473911.19
768.80	379.23	456649.09
771.00	369.60	445095.50
773.40	355.05	427611.00
775.80	335.40	403969.41
778.00	313.01	377011.50
780.40	284.67	342837.69
785.00	222.43	267595.81
787.40	195.15	234563.00
789.80	171.04	205339.91
792.00	150.99	181033.50
794.40	131.83	157819.70
796.80	115.47	137993.80
800.20	95.08	113308.50

TABLE 6.2.1-39 (Sheet 5 of 5)
UNIT 2

MASS AND ENERGY RELEASE RATES FOR SPECIFIED STEAM LINE BREAKS

Time (sec)	Mass Flow Rate (lbm/sec)	Energy Flow Rate (Btu/sec)
1193.20	52.56	62053.44
1250.80	22.96	26729.50
1284.80	4.80	5516.81
1285.00	0.00	0.00
1300.00	0.00	0.00

WBN

TABLE 6.2.1-40

STEAM LINE BREAK CASES FOR CORE INTEGRITY

Case	Type of Break	Boric Acid Concentration (ppm)
1	Hypothetical with offsite power, downstream of the flow restrictor	0
2	Hypothetical without offsite power, downstream of the flow restrictor	0
3	Credible - Uniform	0
4	Credible - Nonuniform	0

WBN

TABLE 6.2.1-41

LINE BREAK[1] DESCRIPTIONS FOR MASS AND ENERGY RELEASES

100.6% Power - AFW Pump Runout Protection Failure

100.6% Power - Main Steam Line Isolation Valve (MSIV) Failure

100.6% Power - Feedwater Isolation Valve (FWIV) Failure (Unit 1)

100.6% Power - Feedwater Regulation Valve (FRV) Failure (Unit 2)

0% Power - AFW Pump Runout Protection Failure

0% Power - Main Steam Line Isolation Valve (MSIV) Failure

Notes:

(1) For 1.4 ft^2 break

WBN

TABLE 6.2.1-42

SMALL BREAK DESCRIPTIONS FOR MASS AND ENERGY

Break Size (ft^2)	Description
0.677	30% Power - AFW Pump Runout Protection Failure (Unit 1)
0.944	30% Power 0 AFW Pump Runout Protection Failure (Unit 2)
0.60	30% Power - AFW Pump Runout Protection Failure
0.35	30% Power - AFW Pump Runout Protection Failure
0.843	100.6% Power - AFW Pump Runout Protection Failure (Unit 1)
0.86	100.6% Power - AFW Pump Runout Protection Failure (Unit 2)

WBN

TABLE 6.2.1-43

LARGE BREAK ANALYSIS - ASSOCIATED TIMES

UNIT 1

Case	Maximum Lower Compartment Temperature (°F)	Time, T_{max} (sec)
1.4 ft^2, 100.6% Power - AFW Pump Runout Protection Failure	324.2	49.5
1.4 ft^2, 100.6% Power - FWIV Failure	324.5	56.9
1.4 ft^2, 100.6% Power - MSIV Failure	324.4	55.9
1.4 ft^2, 0% Power - AFW Pump Runout Protection Failure	322.7	40.4
1.4 ft^2, 0% Power - MSIV Failure	323.2	39.8

UNIT 2

Case	Maximum Lower Compartment Temperature (°F)	Time, T_{max} (sec)
1.4 ft^2, 100.6% Power - AFW Pump Runout Protection Failure	324.1	66.2
1.4 ft^2, 100.6% Power - FRV Failure	324.3	49.6
1.4 ft^2, 100.6% Power - MSIV Failure	324.4	55.5
1.4 ft^2, 0% Power - AFW Pump Runout Protection Failure	322.7	37.3
1.4 ft^2, 0% Power - MSIV Failure	323.2	39.5

TABLE 6.2.1-44

SMALL BREAK ANALYSIS - SMALL SPLIT - ASSOCIATED TIMES

UNIT 1

Case[1] (ft^2)	Maximum Lower Compartment Temperature (°F)	Time, T_{max} (sec)
0.843	325.5	101.5
0.677	325.8	605.1
0.60	325.9	606.1
0.35	325.1	608.5

[1]All with AFW pump runout protection failure and 30% power, except that 0.843 ft^2 break is at 100.6% power.

UNIT 2

Case[1] (ft^2)	Maximum Lower Compartment Temperature (°F)	Time, T_{max} (sec)
0.86	325.6	101.5
0.944	325.6	75.5
0.6	325.9	449.4
0.35	325.0	609.4

[1]All with AFW pump runout protection failure and 30% power, except that 0.86 ft^2 break is at 100.6% power.

TABLE 6.2.1-45

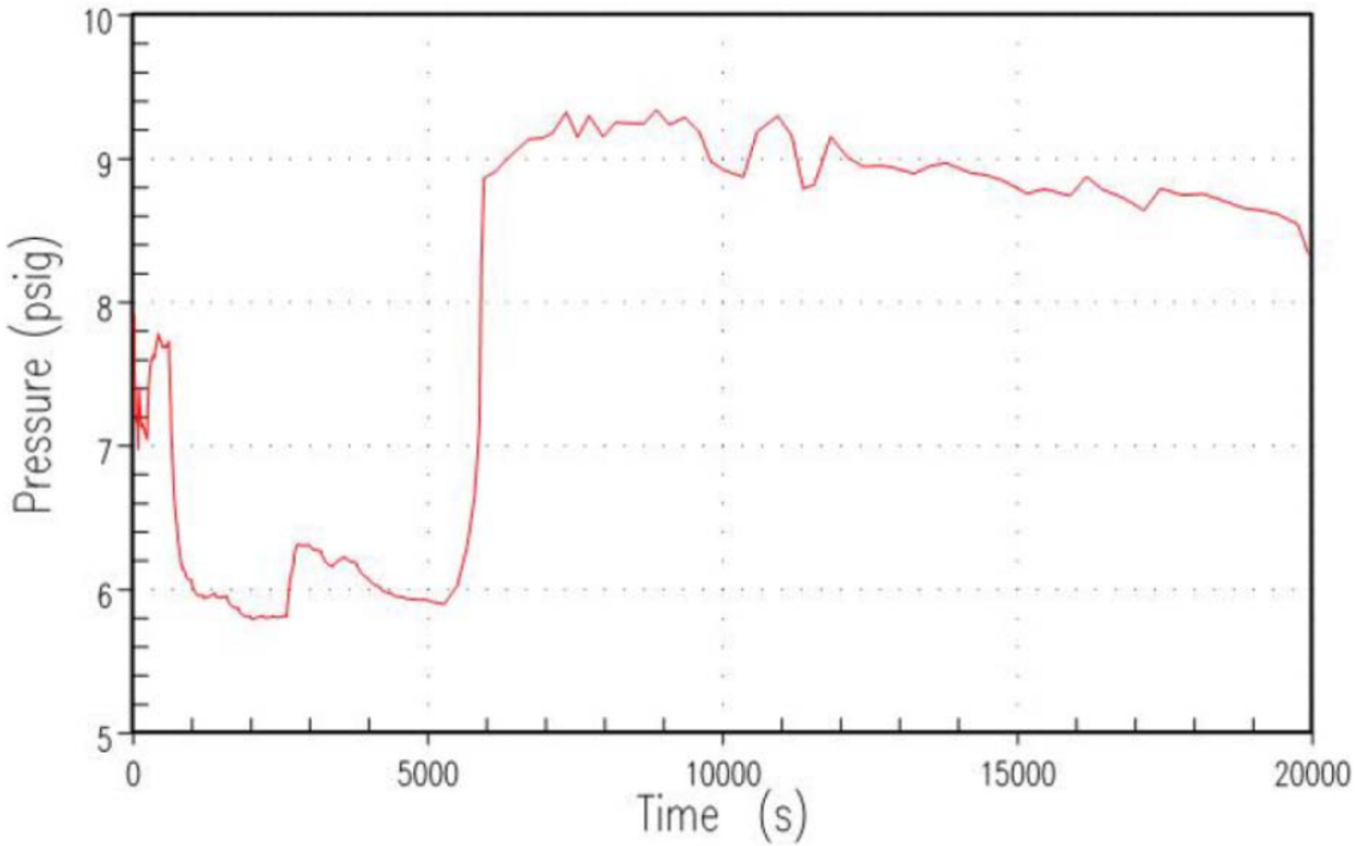
SAFETY INJECTION FLOW MINIMUM SAFETY INJECTION

Injection Mode

RCS Pressure (psia)	Total Flow (gpm)
28.2	4587.0

Recirculation Mode
(Without Residual Heat Removal (RHR) Spray)

RCS Pressure (psia)	Total Flow (gpm)
0	3200.3

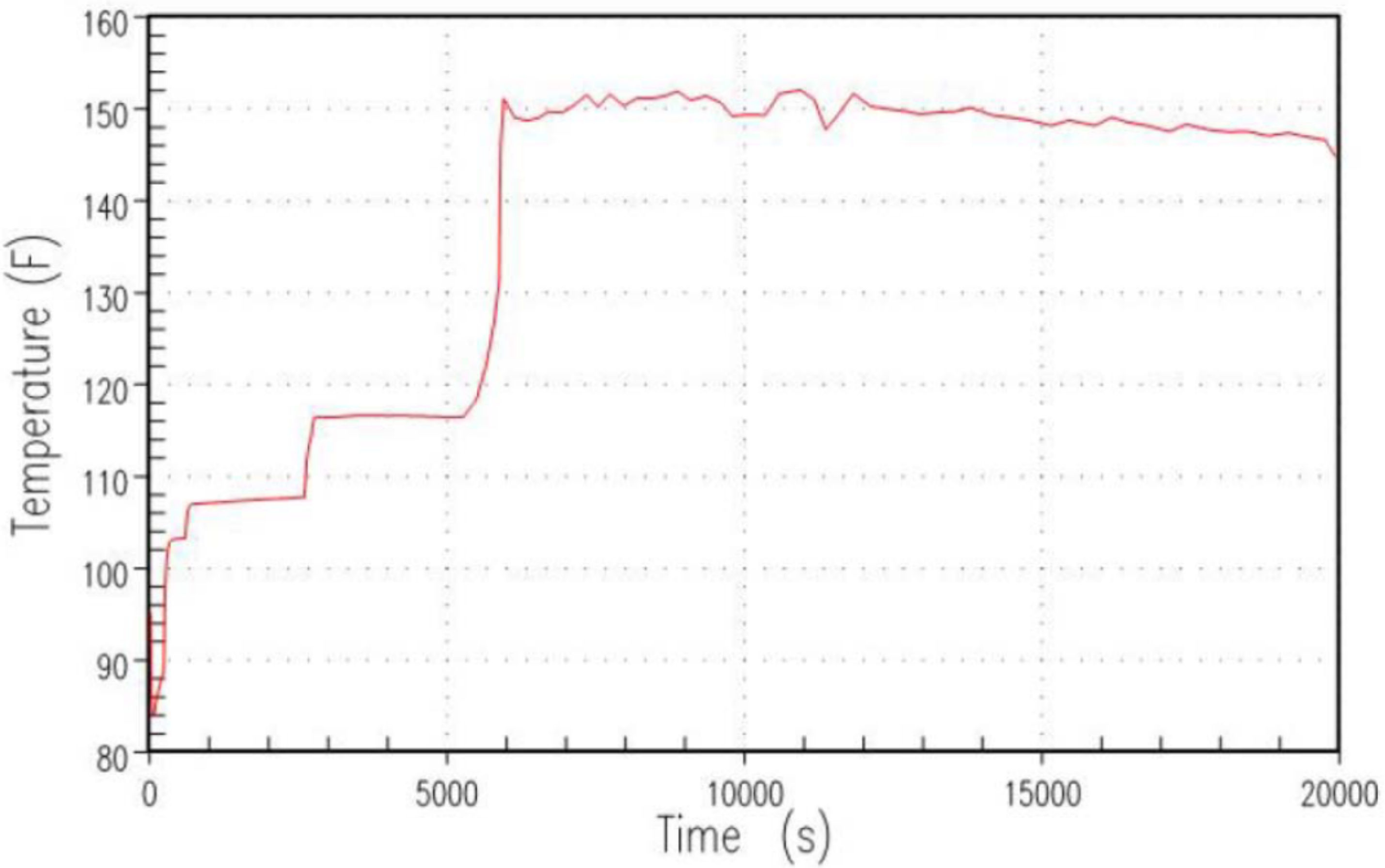


AMENDMENT 2

WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

Containment Pressure
Versus Time

FIGURE 6.2.1-1

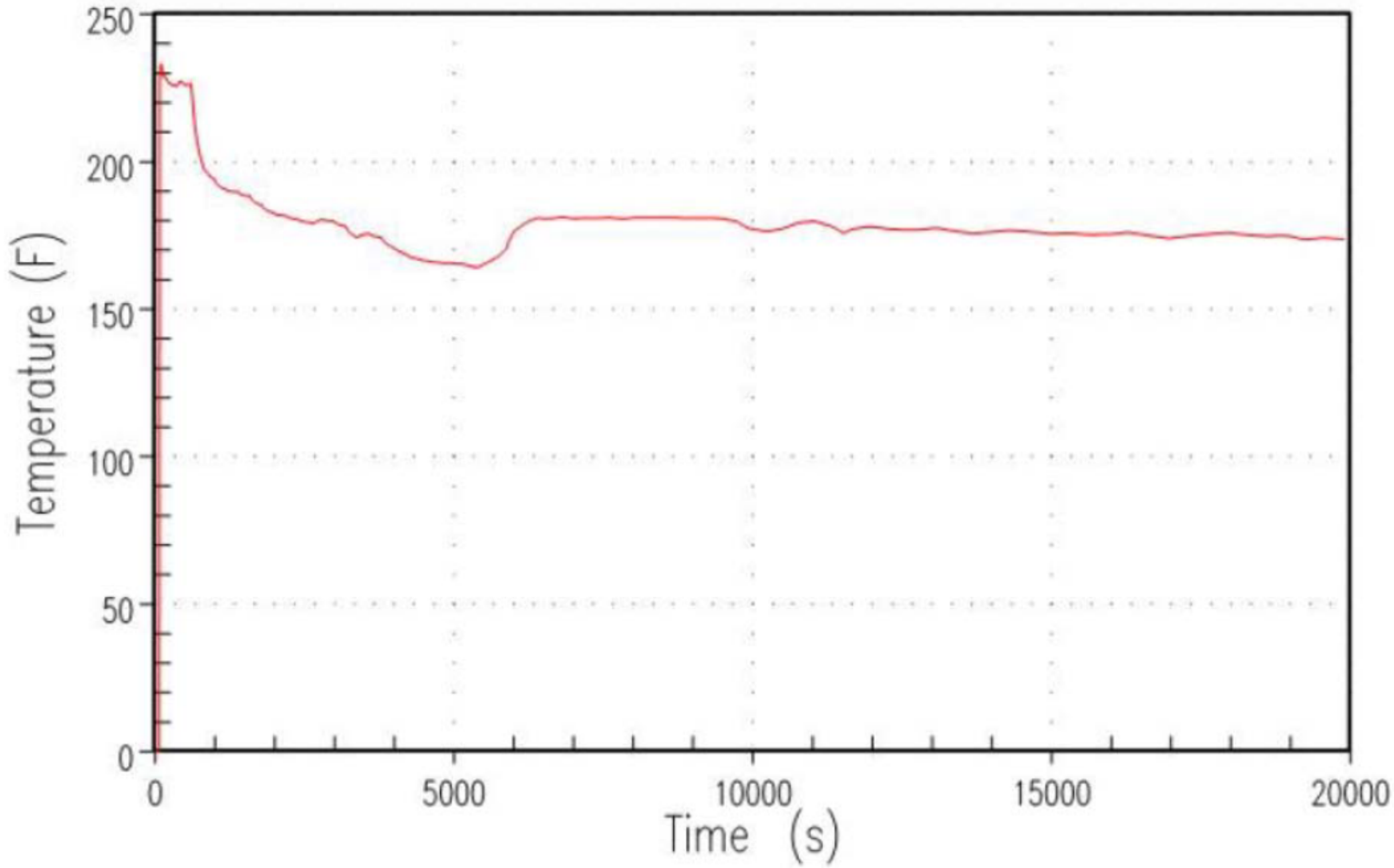


AMENDMENT 2

WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

Upper Compartment Temperature
Versus Time

FIGURE 6.2.1-2a

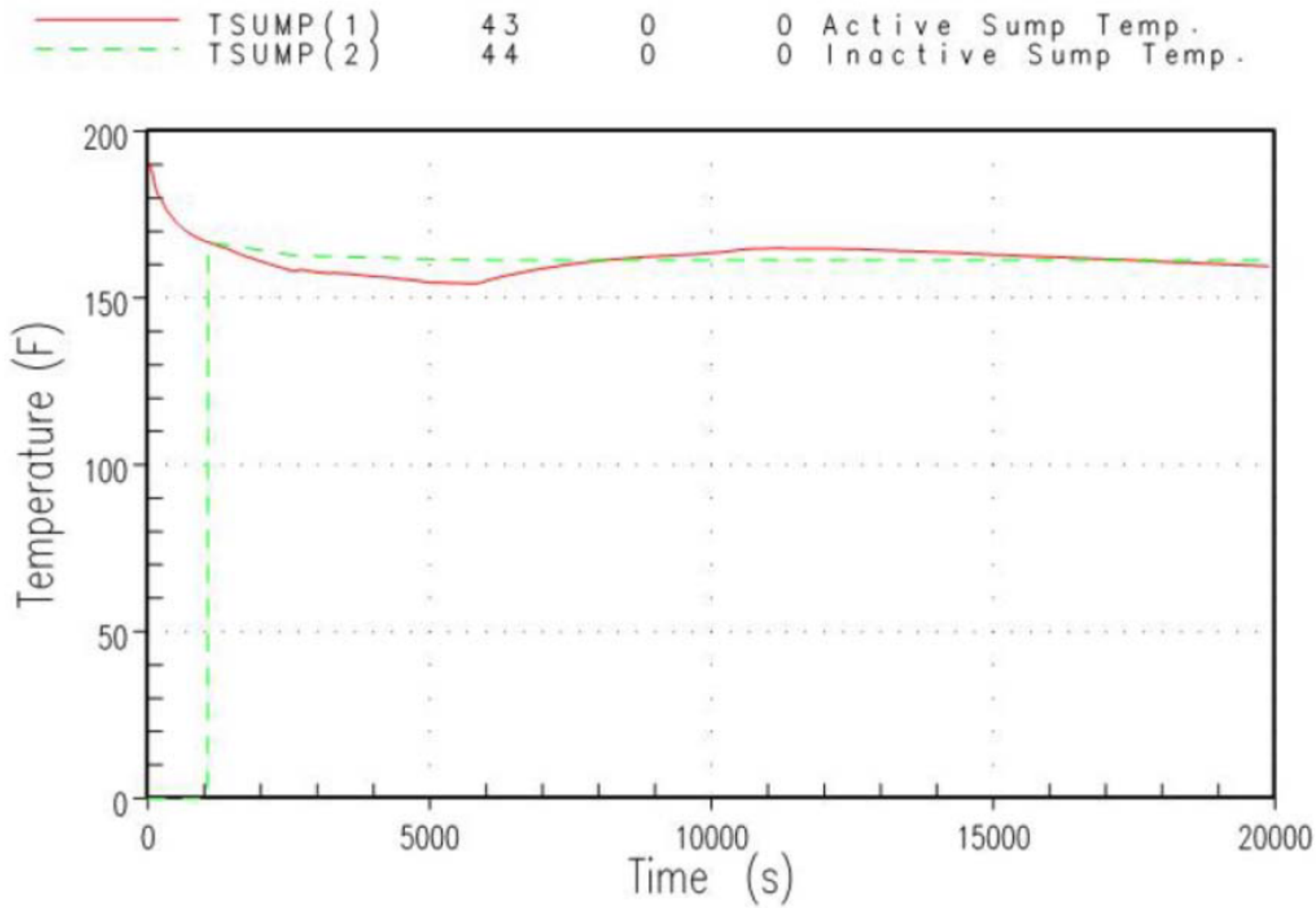


AMENDMENT 2

WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

Lower Compartment Temperature
Versus Time

FIGURE 6.2.1-2b

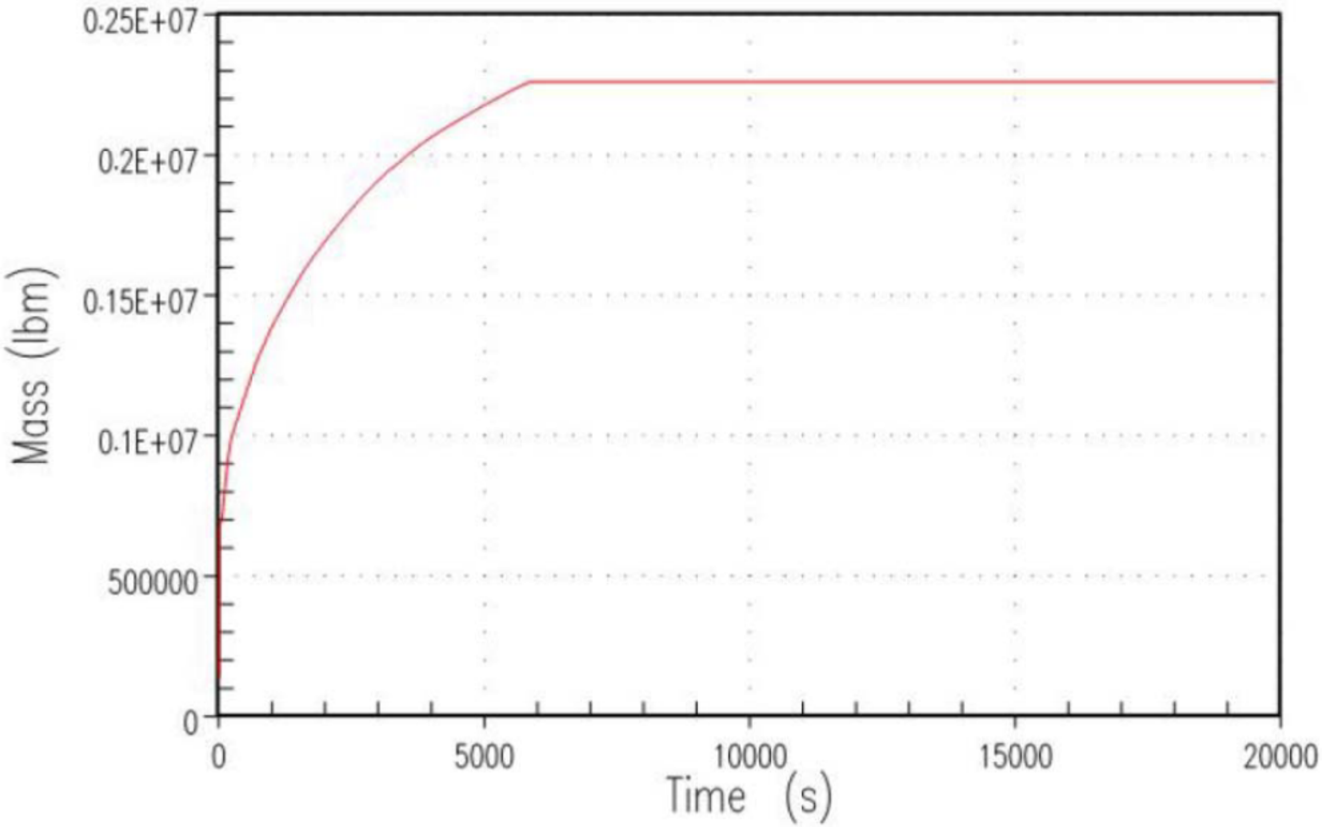


AMENDMENT 2

WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

Lower Compartment Temperature
Versus Time

FIGURE 6.2.1-3

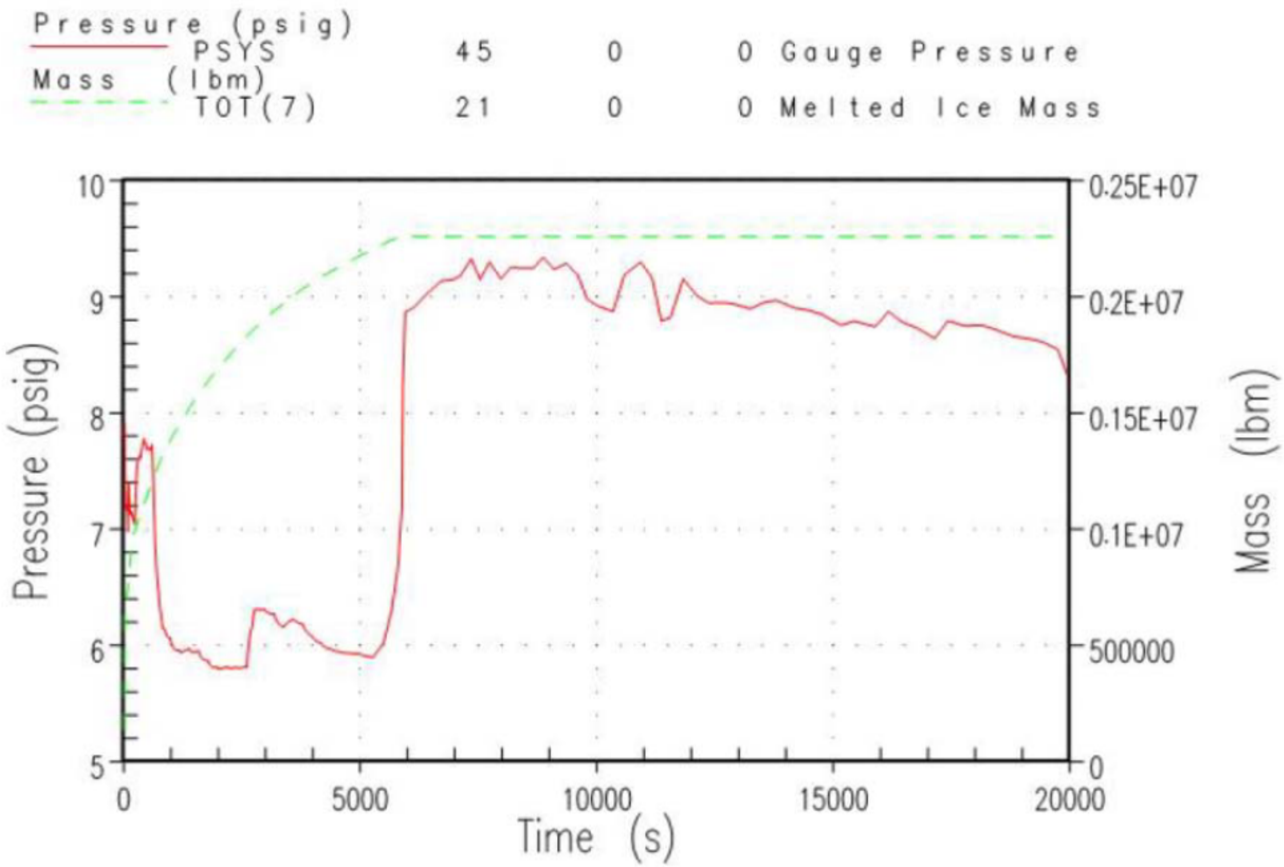


AMENDMENT 2

WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

Melted Ice Mass
Transient

FIGURE 6.2.1-4

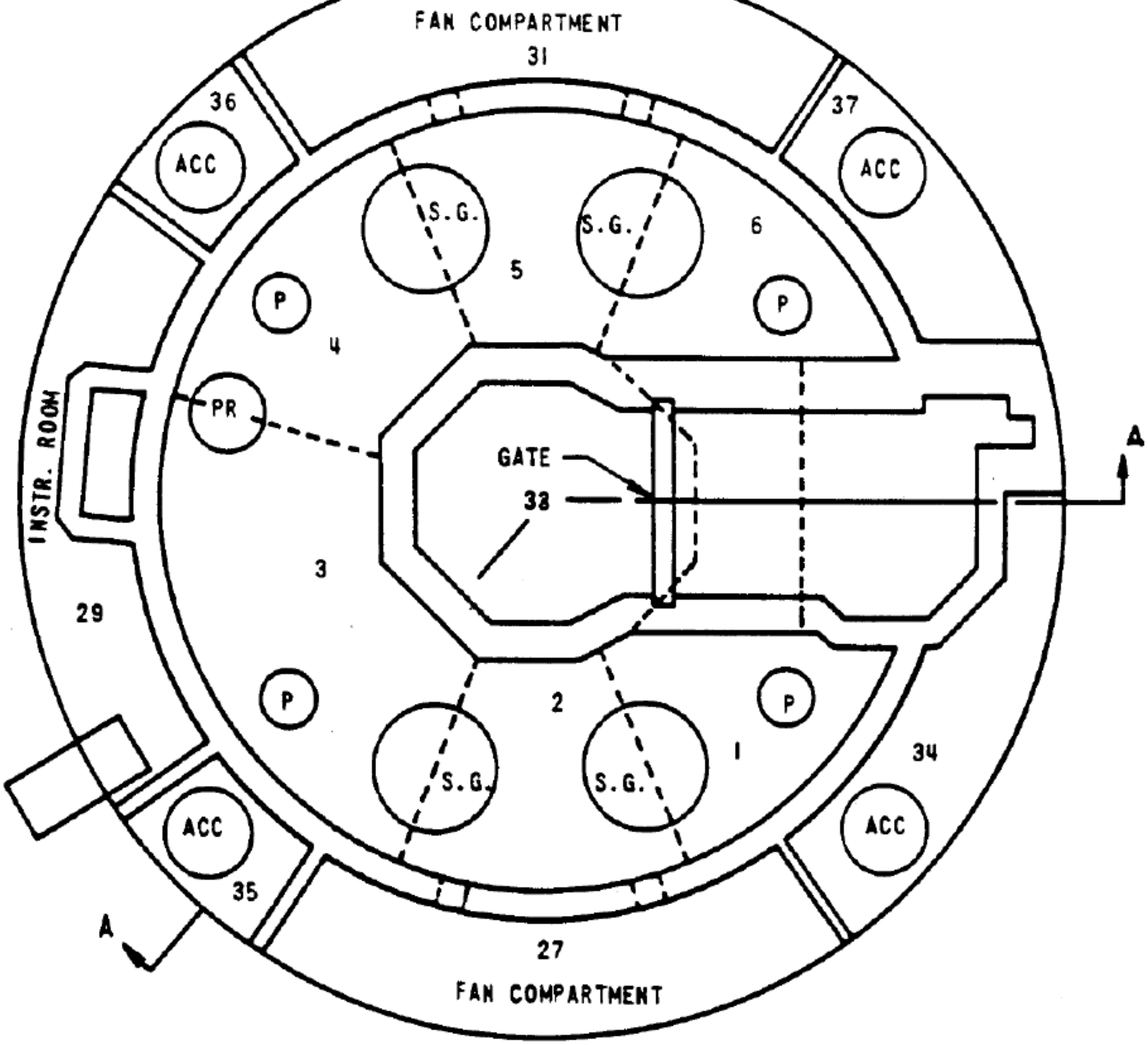


AMENDMENT 2

WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

Comparison of containment
Pressure Versus Ice Melt

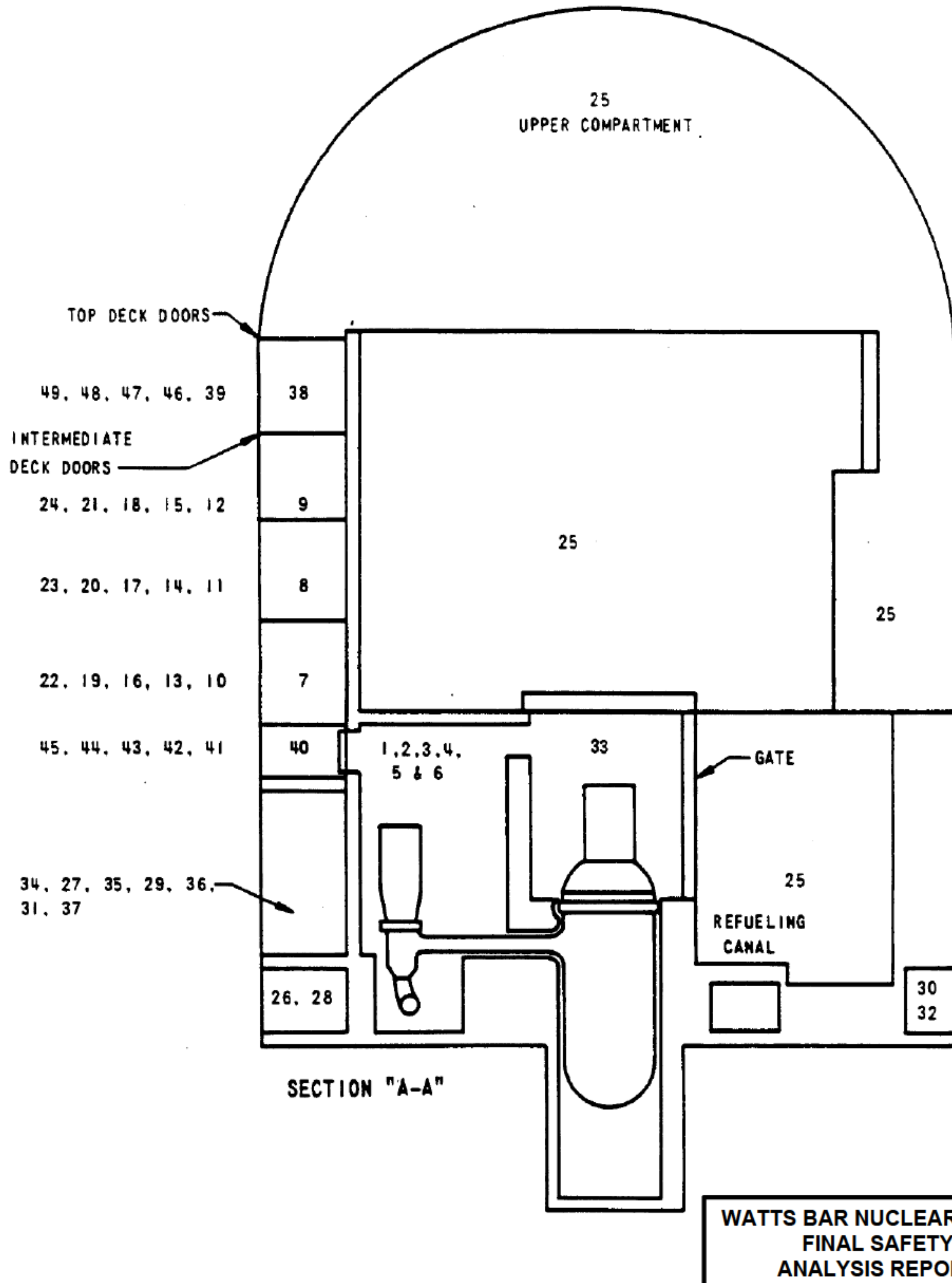
FIGURE 6.2.1-4a



WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

Plan at Equipment Rooms
Elevation

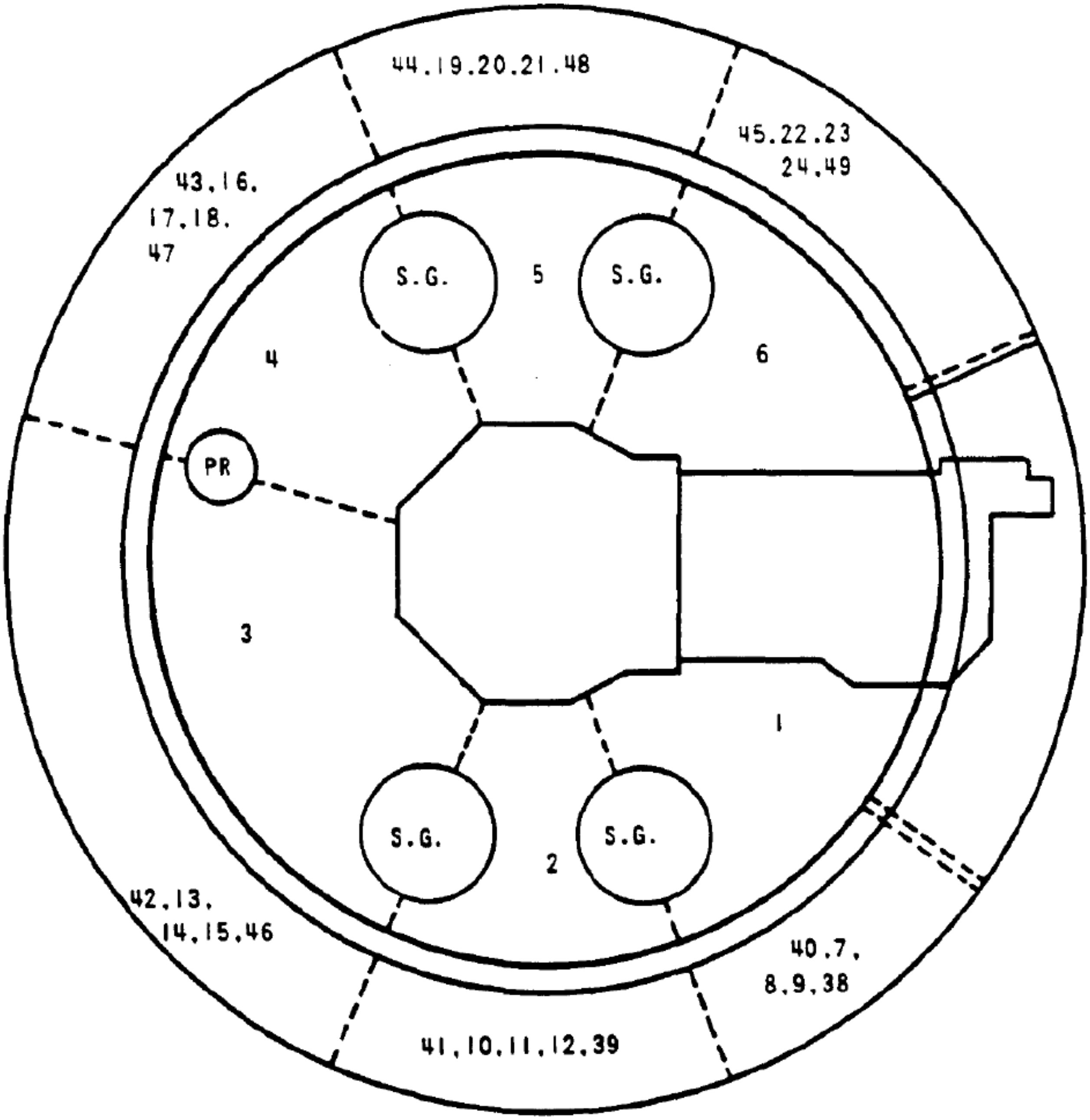
Figure 6.2.1-5



WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

Containment
Section View

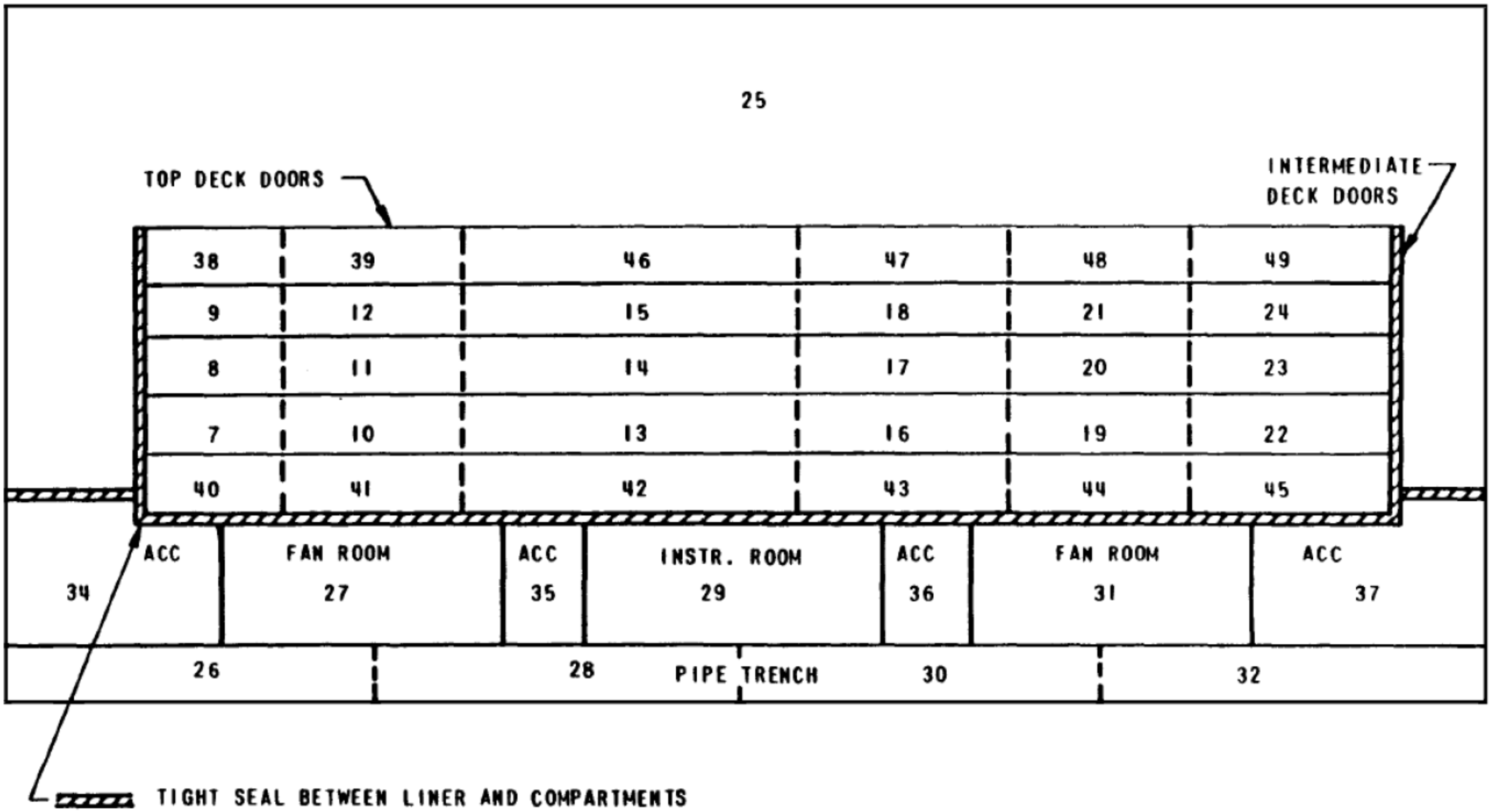
Figure 6.2.1-6



WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

Plan View at Ice Condenser Elevation – Ice Condenser Compartments

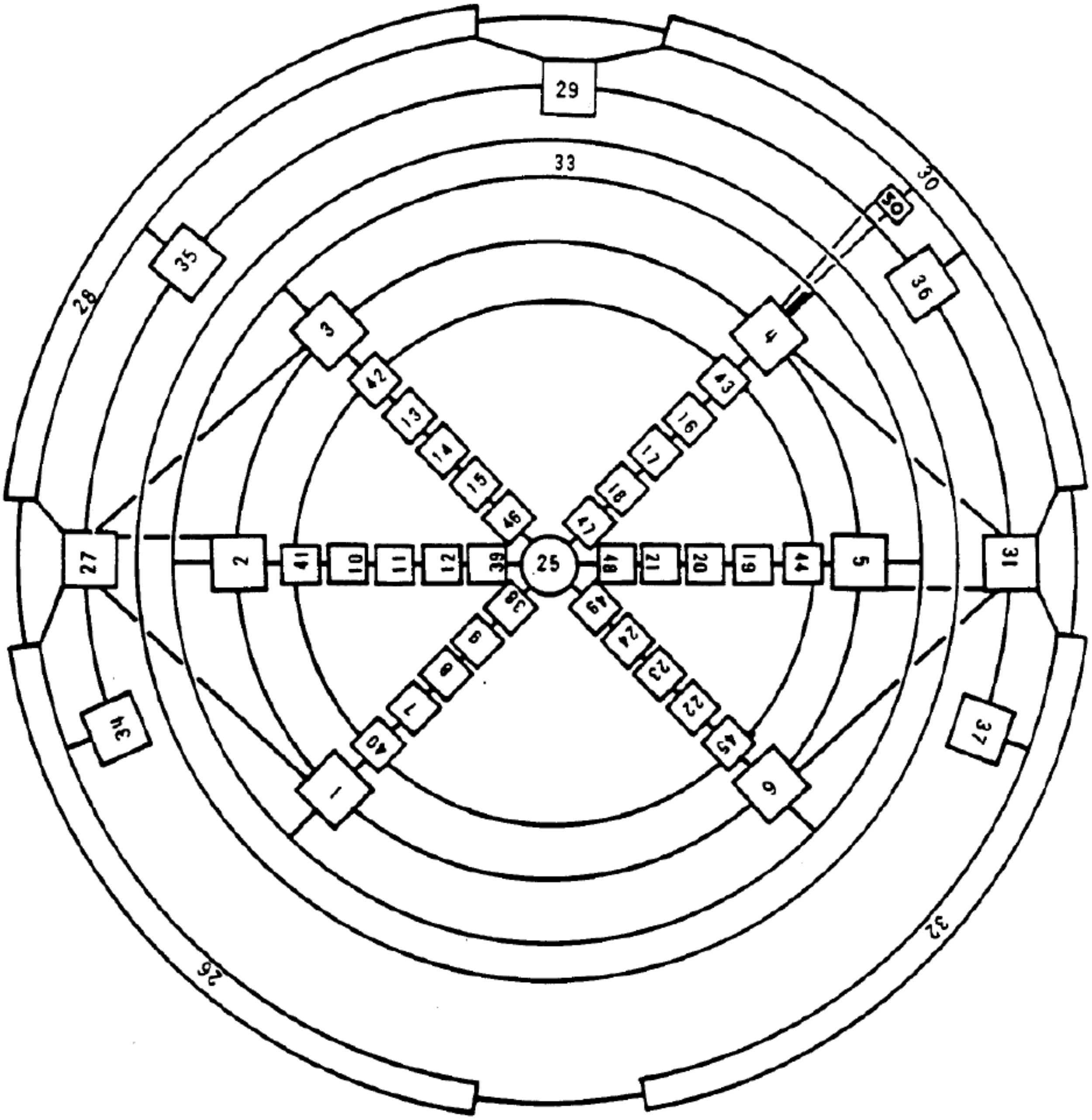
Figure 6.2.1-7



WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

Layout of
Containment Shell

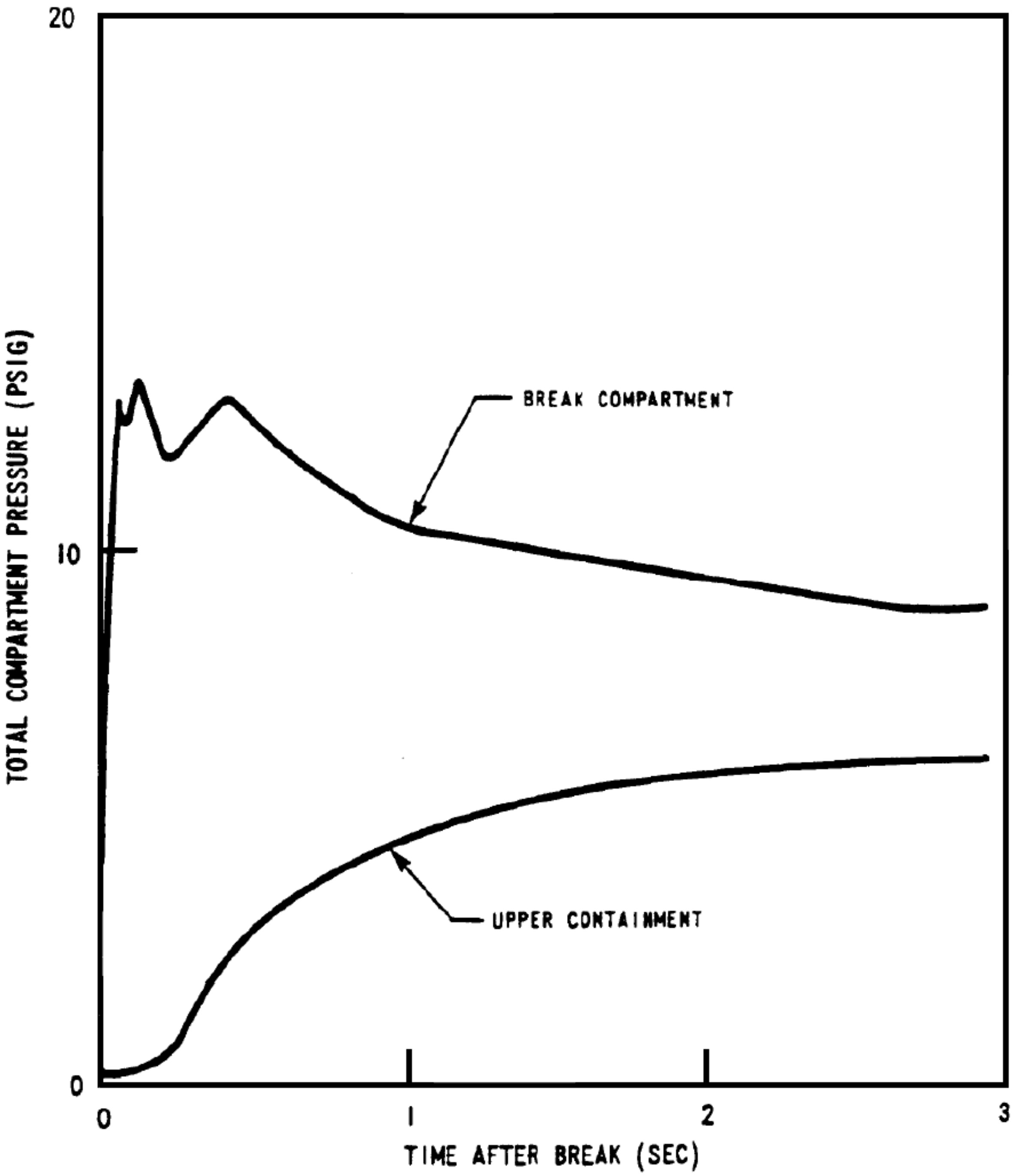
FIGURE 6.2.1-8



WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

TMD Code
Network

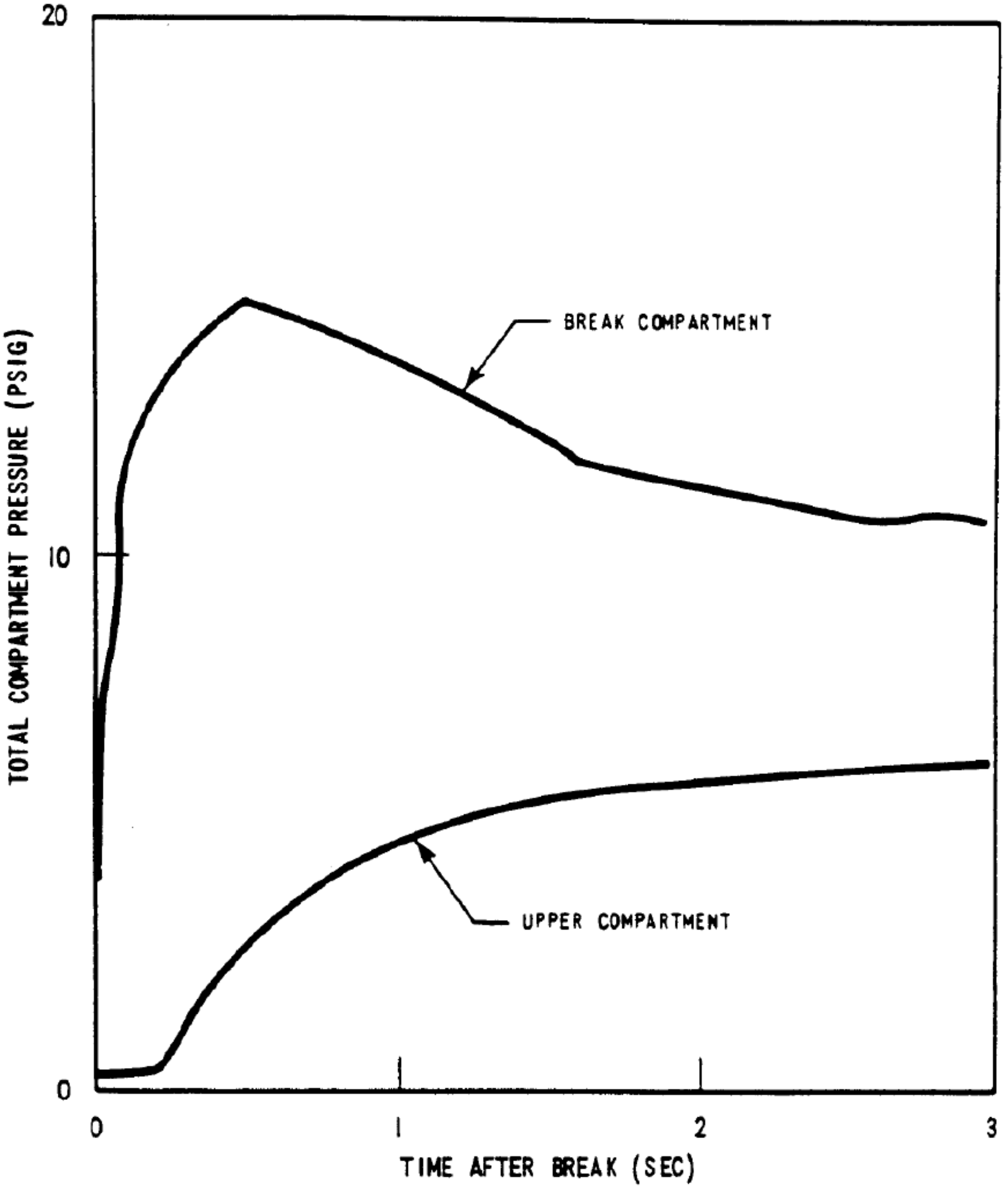
Figure 6.2.1-9



WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

Upper and Lower Compartment Pressure Transient for Worst Case Break Compartment (Element 1) Having a DEHL Break

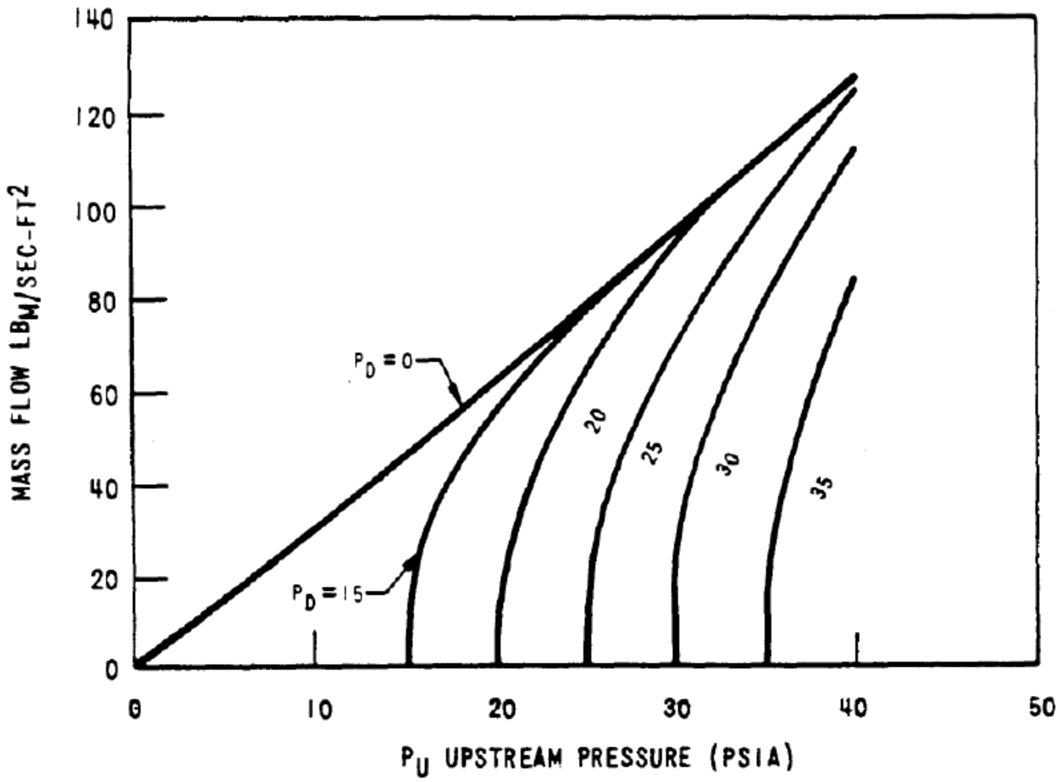
Figure 6.2.1-10

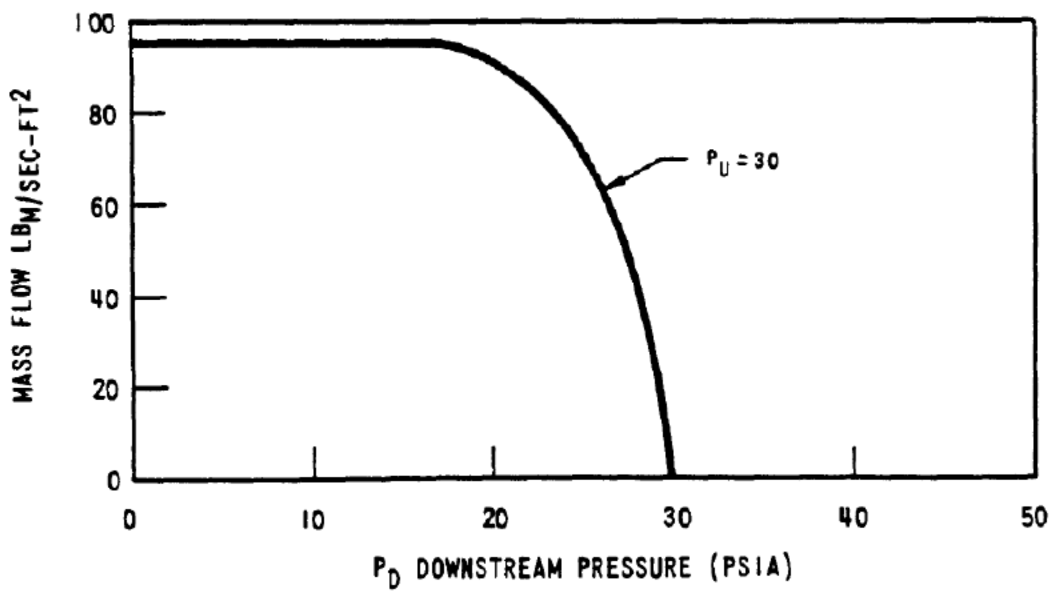


WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

Upper and Lower Compartment Pressure Transient for Worst Case Break Compartment (Element 1) Having a DECL Break

Figure 6.2.1-11

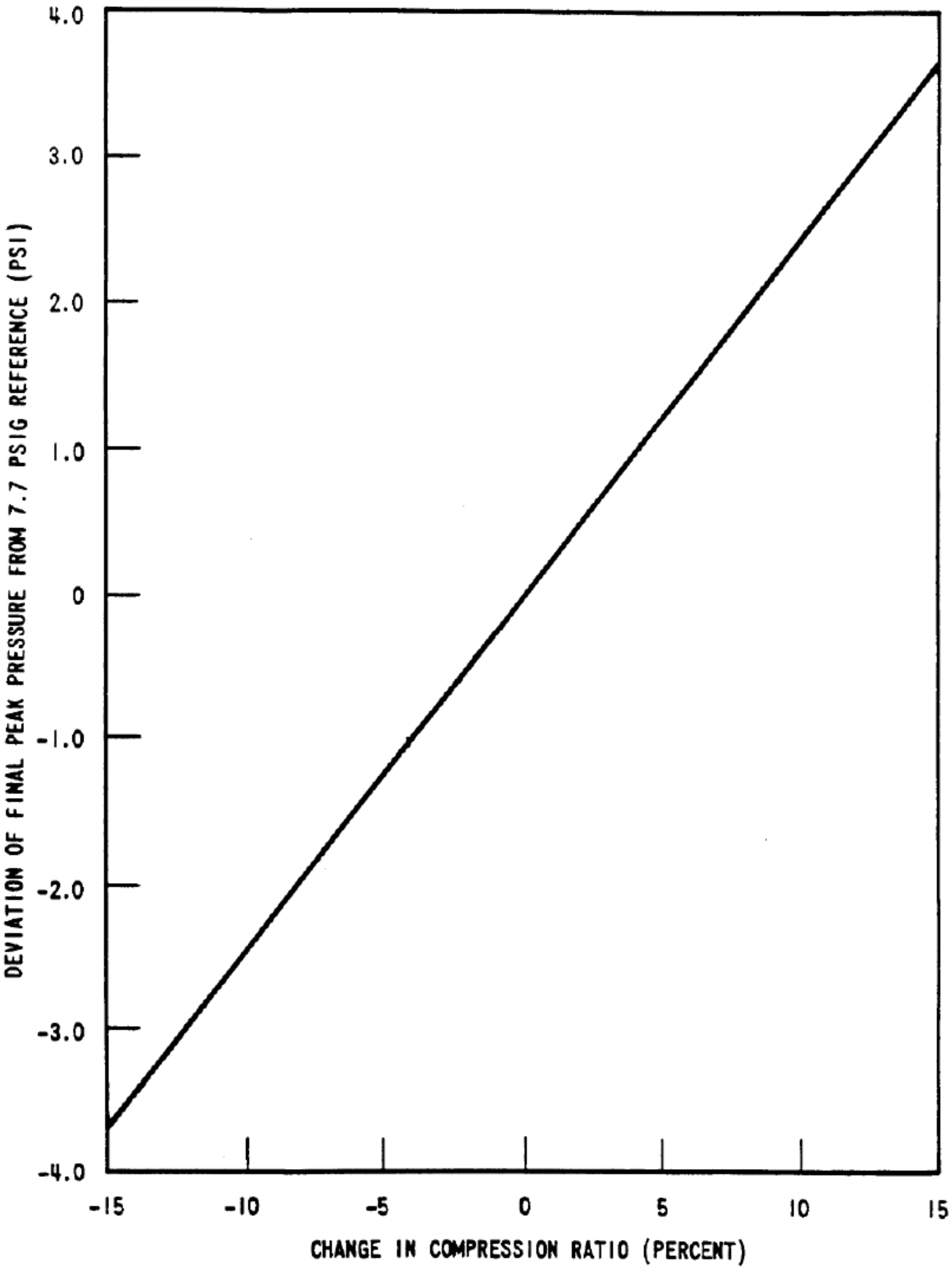




WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

Illustration of
Choked Flow
Characteristics

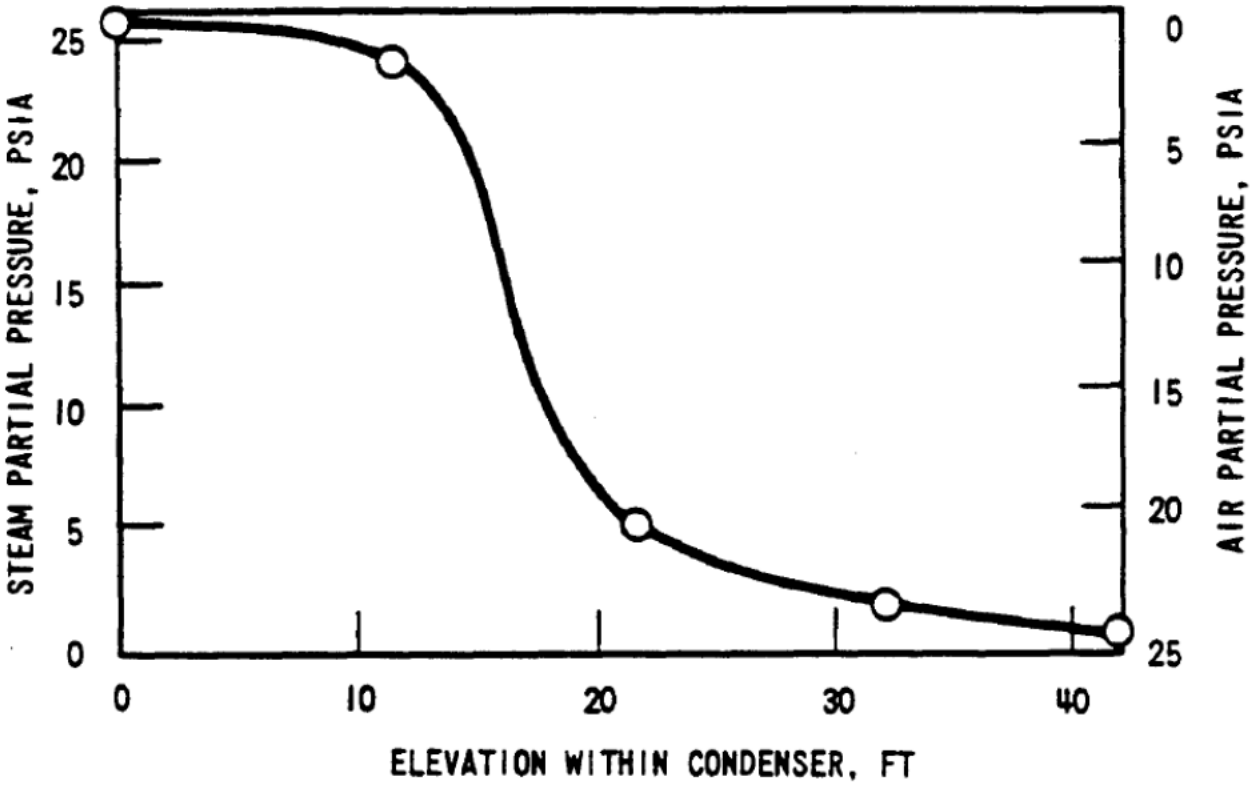
Figure 6.2.1-12



WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

Sensitivity of
Peak Pressure to Air
Compression Ratio

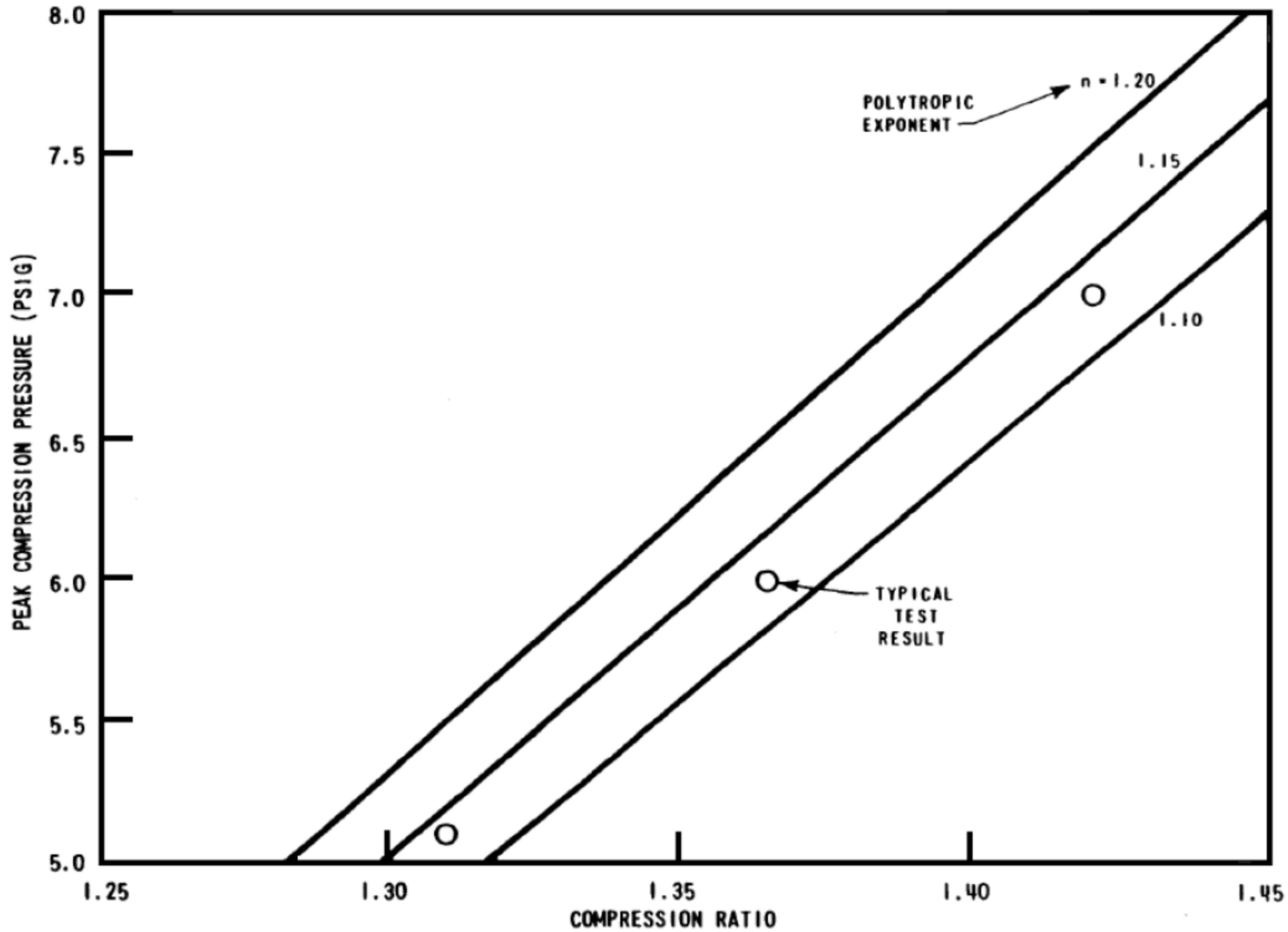
Figure 6.2.1-13



WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

Steam Concentration
in a Vertical
Distribution Channel

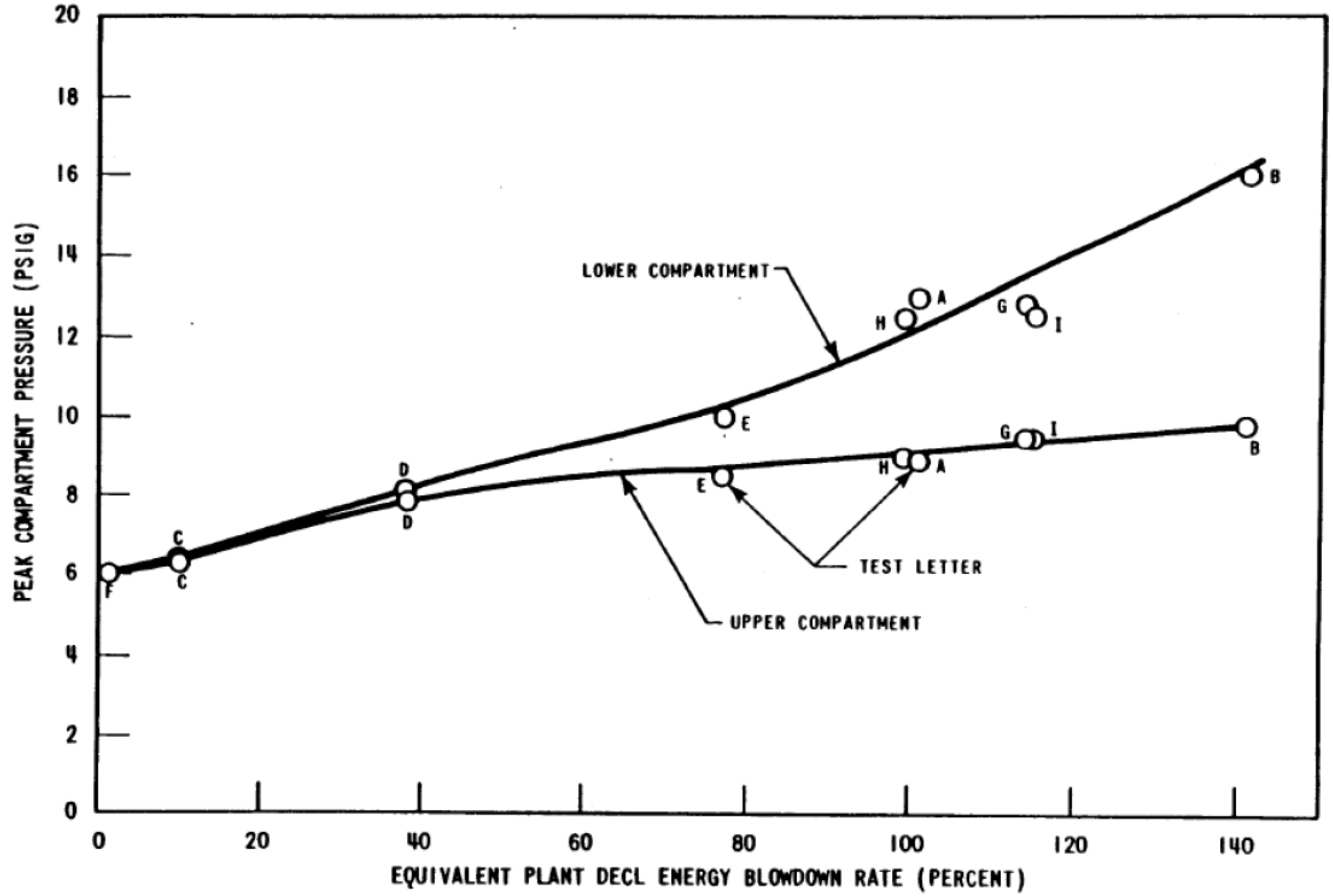
Figure 6.2.1-14



WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

Peak Compression Pressure
Versus Compression Ratio

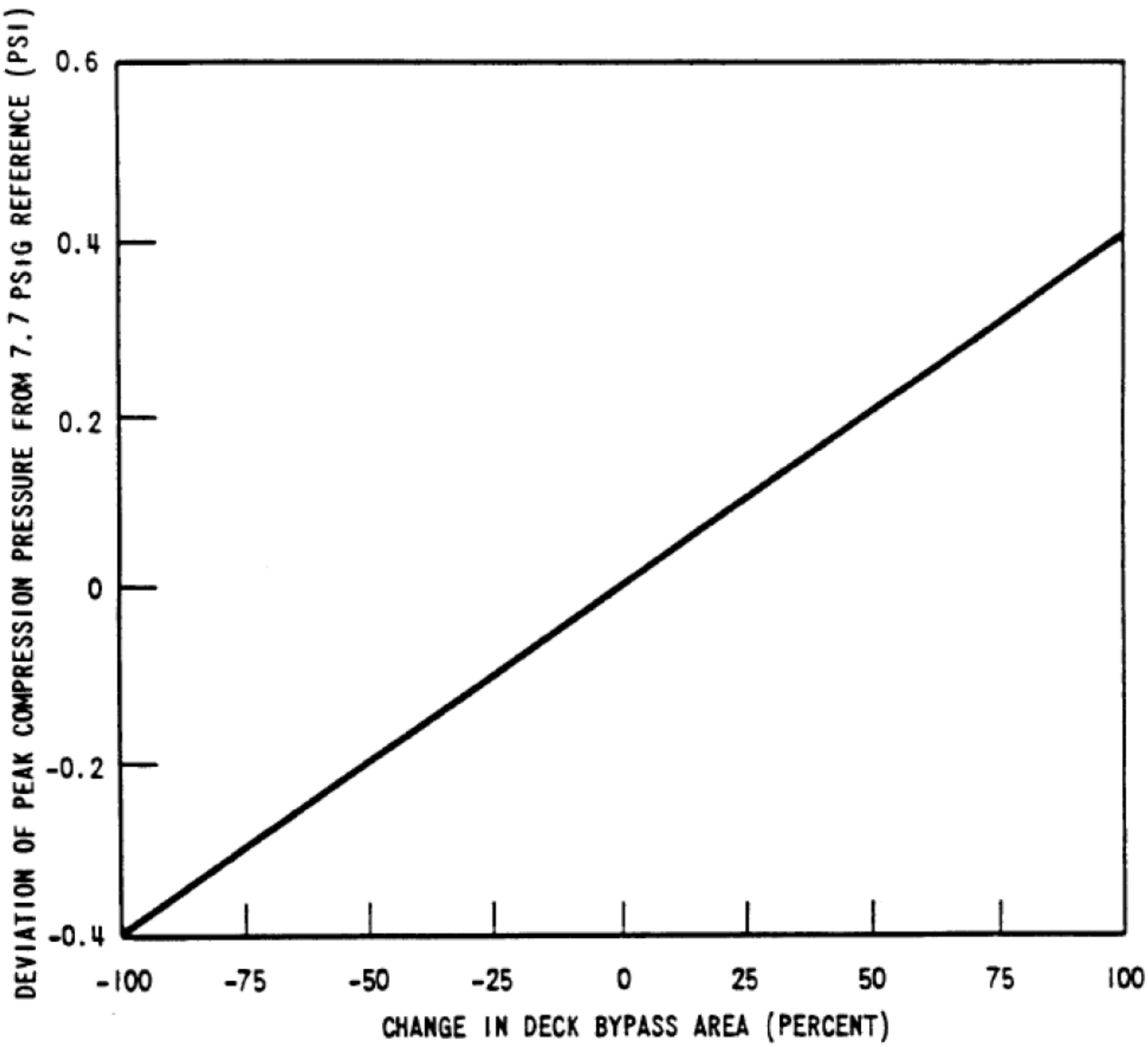
FIGURE 6.2.1-15



WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

Peak Compartment Pressure
Versus Blowdown Rate

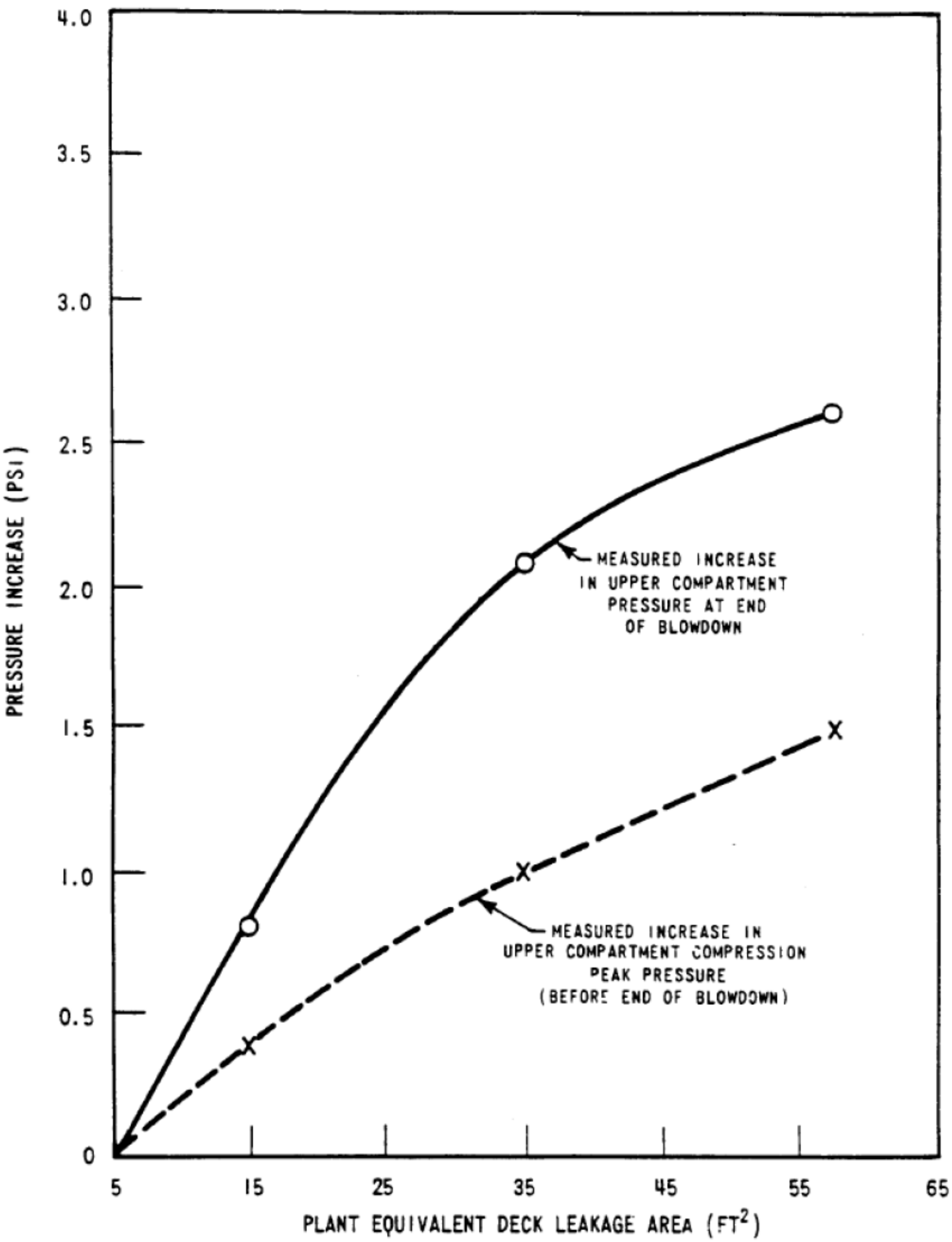
FIGURE 6.2.1-16



WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

Sensitivity of Peak
Compression Pressure
to Deck Bypass

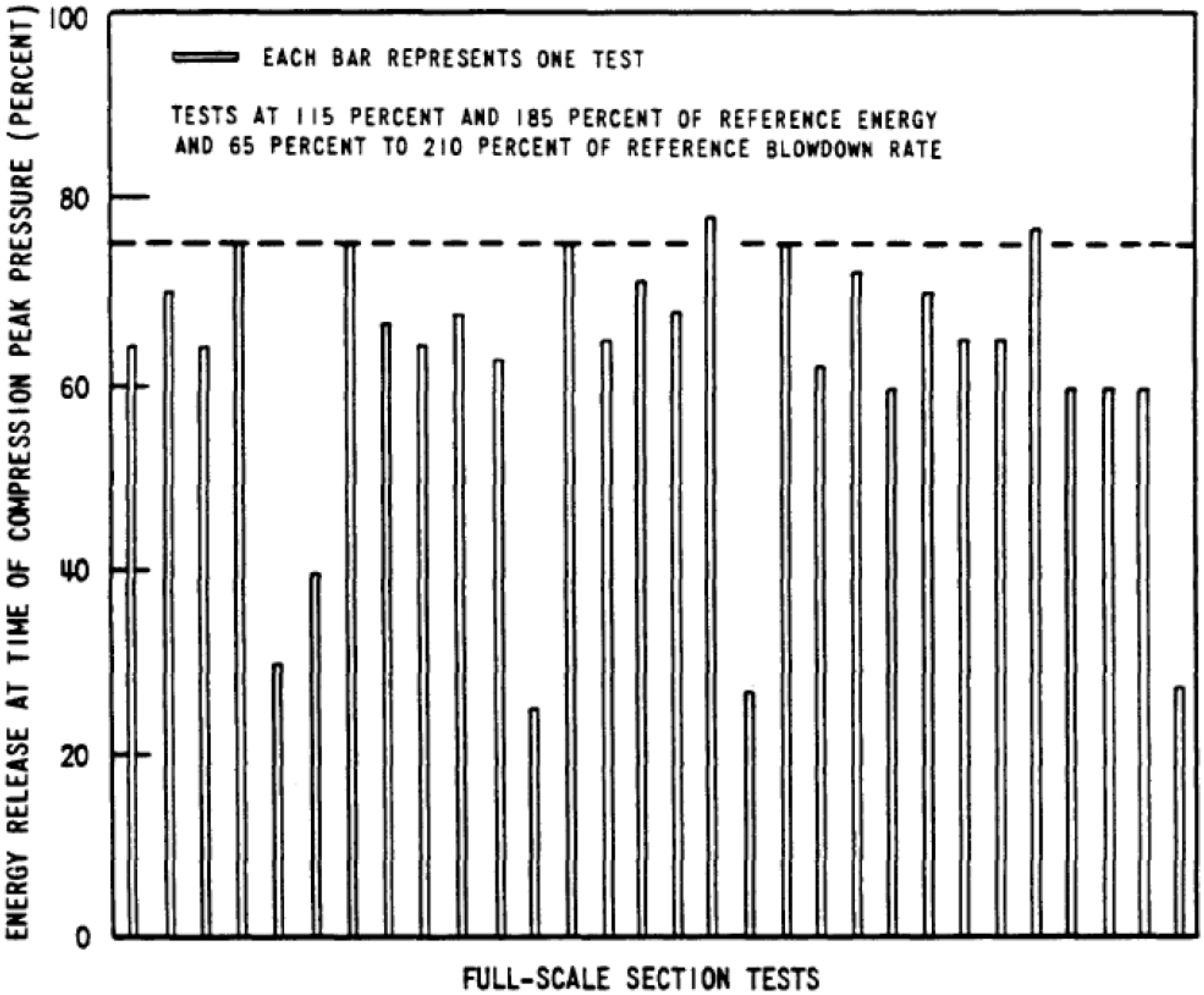
Figure 6.2.1-17



WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

Pressure Increase Versus
Deck Area from
Deck Leakage Tests

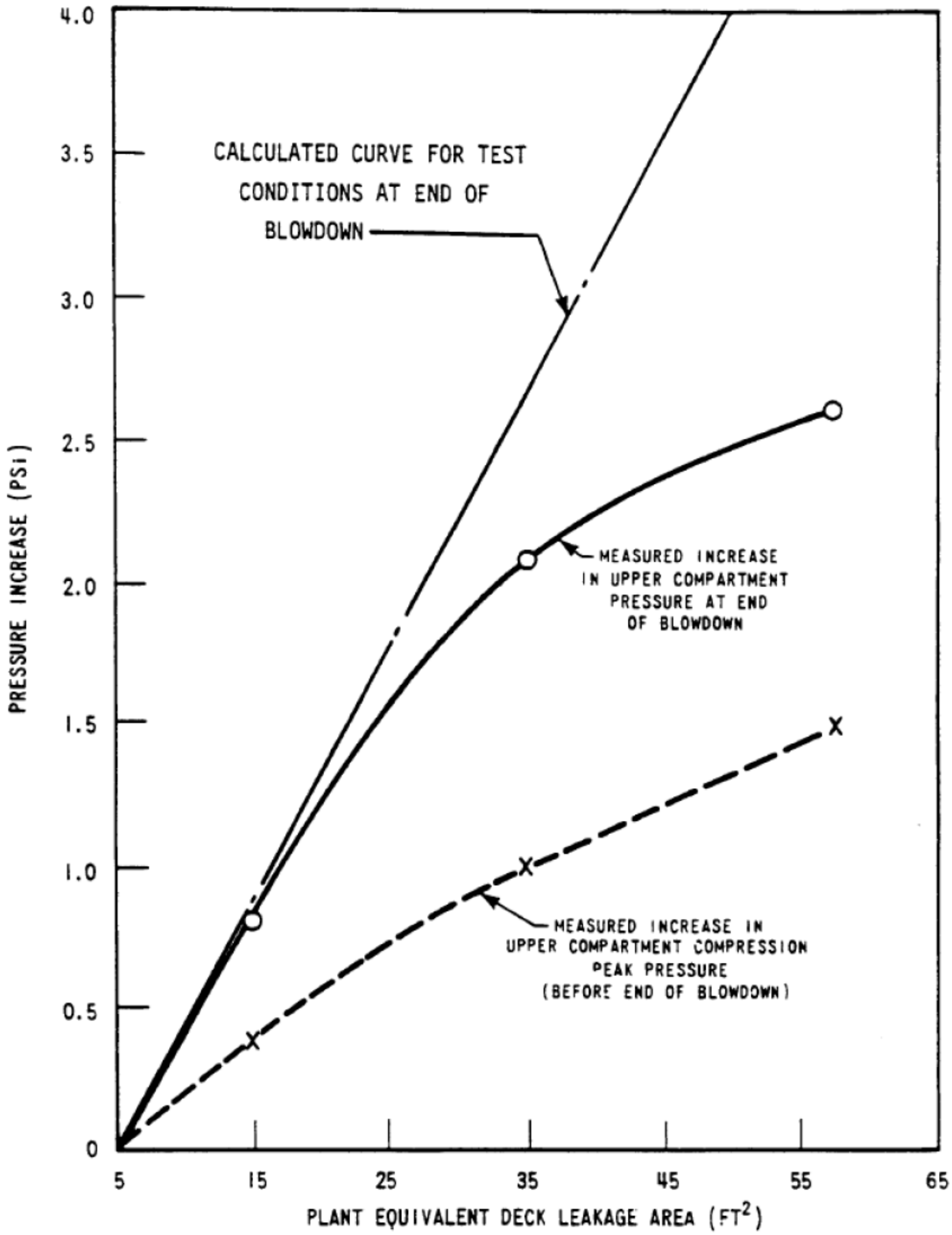
Figure 6.2.1-18



WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

Energy Release at time of Compression Peak Pressure from Full-Scale Section Tests with 1-Foot Diameter Baskets

Figure 6.2.1-19



WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

Pressure Increase Versus
Deck Area from
Deck Leakage Tests

Figure 6.2.1-20

FIGURES 6.2.1-21 THRU 6.2.1-26

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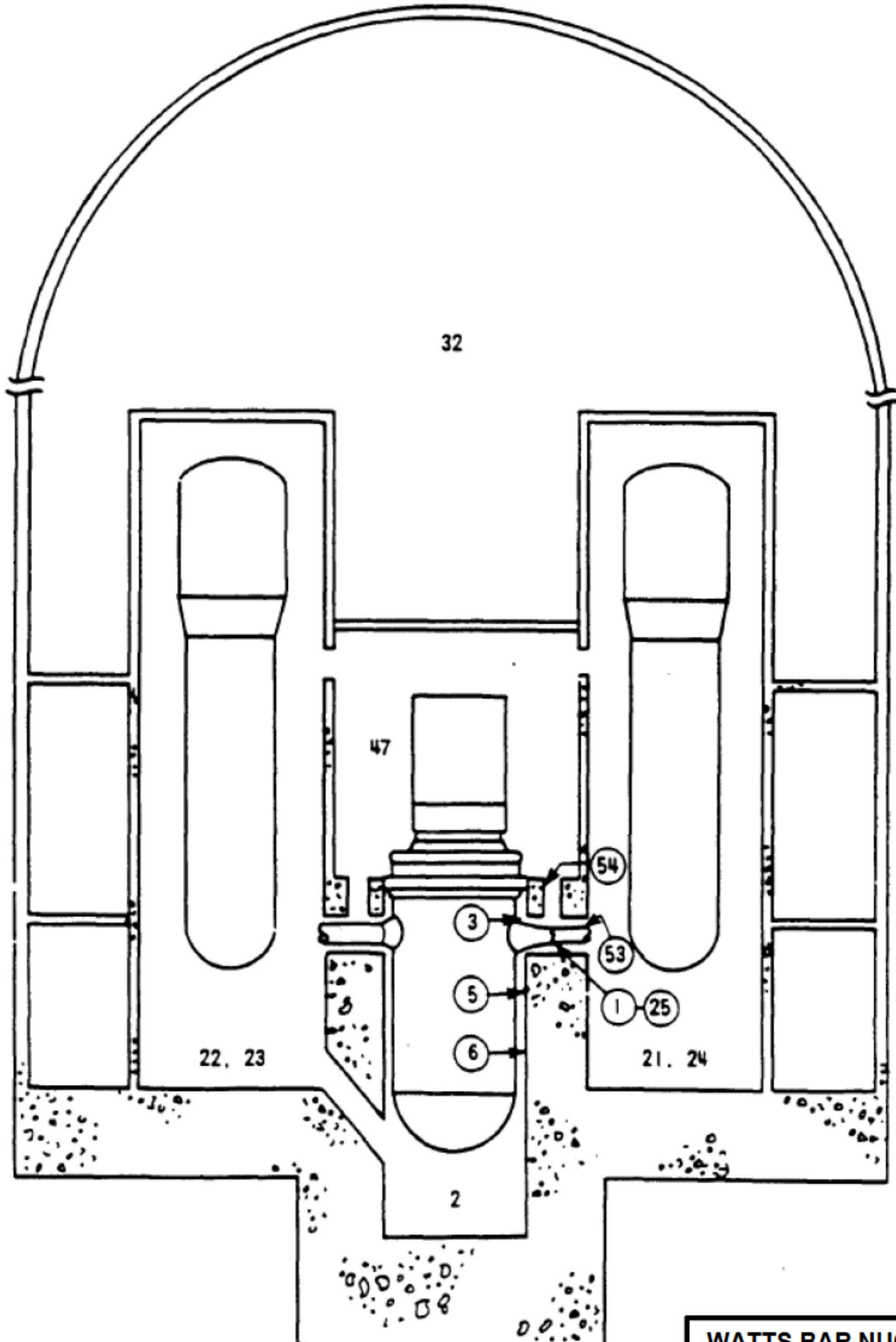
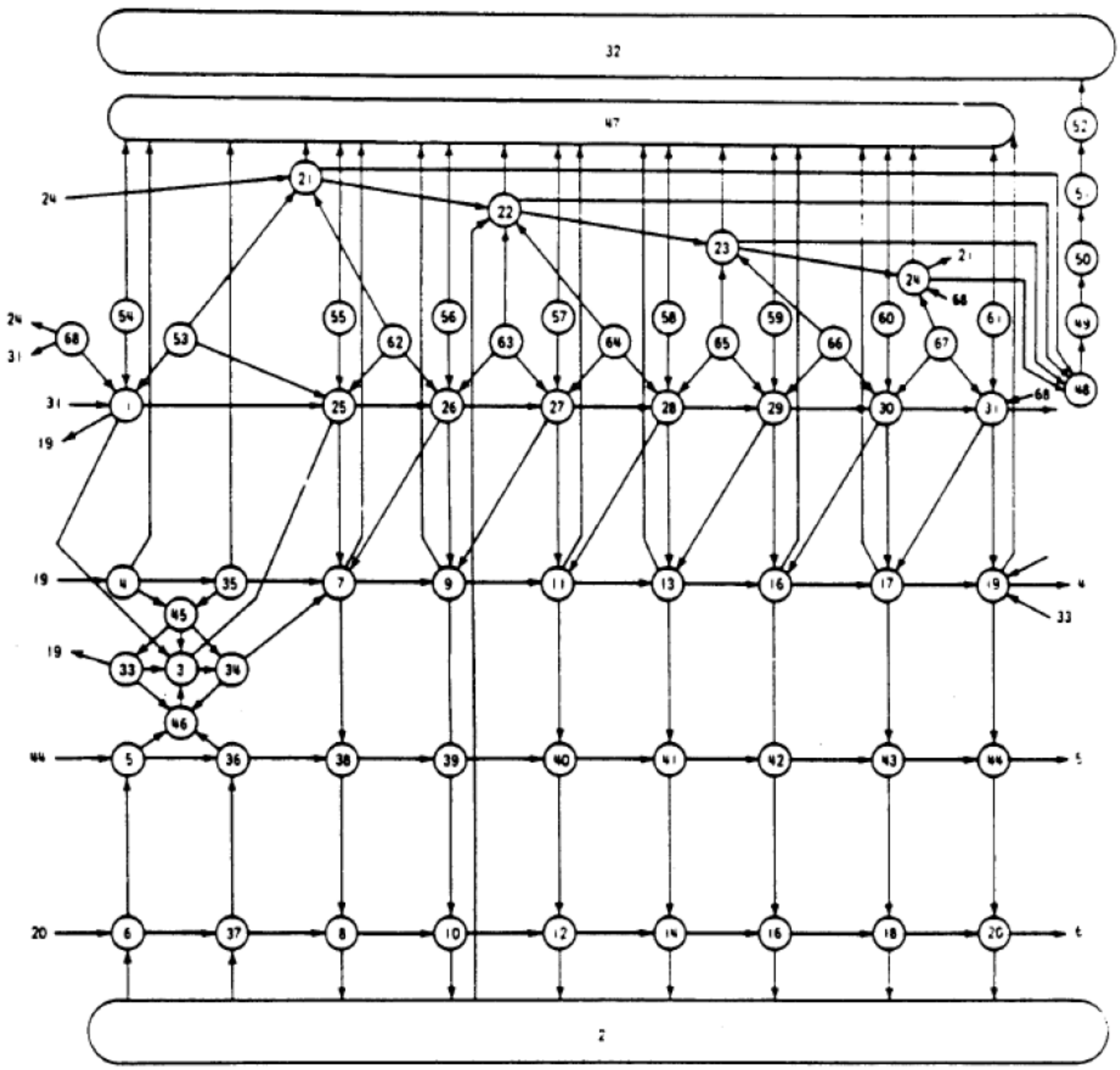


WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

Containment Model
Schematic

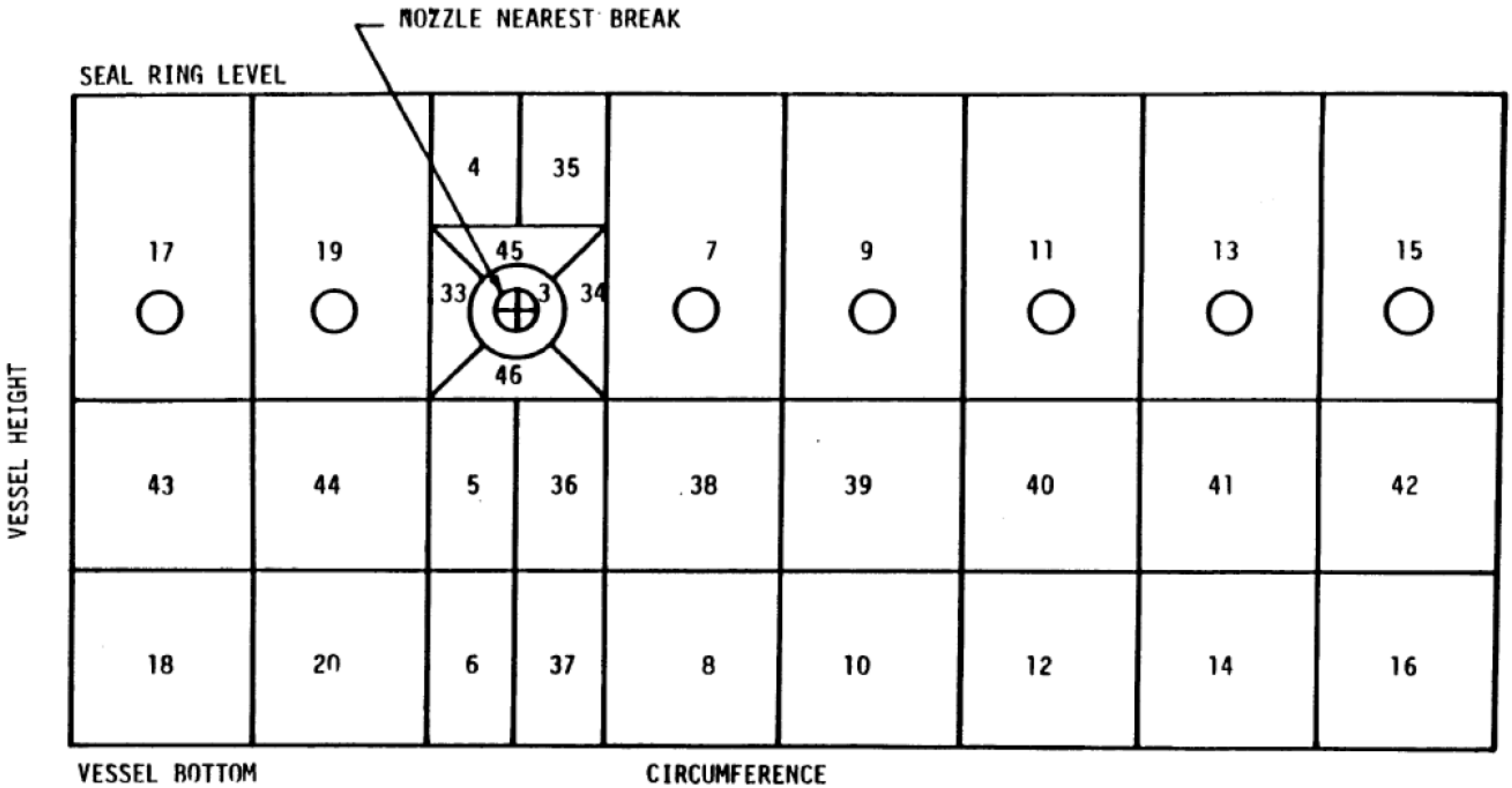
Figure 6.2.1-27



WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

Reactor Cavity TMD Network

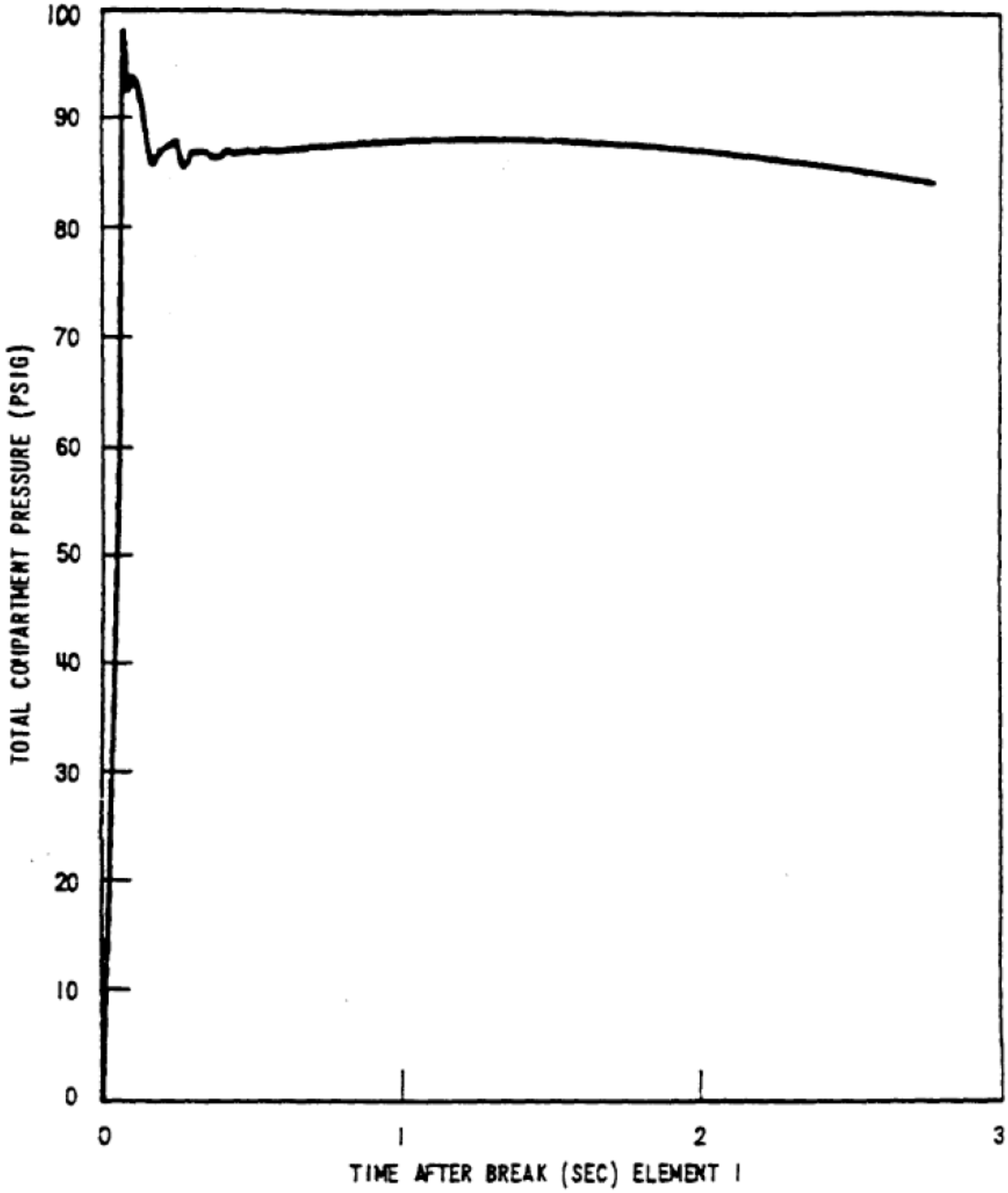
Figure 6.2.1-28



WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

Reactor Vessel Annulus

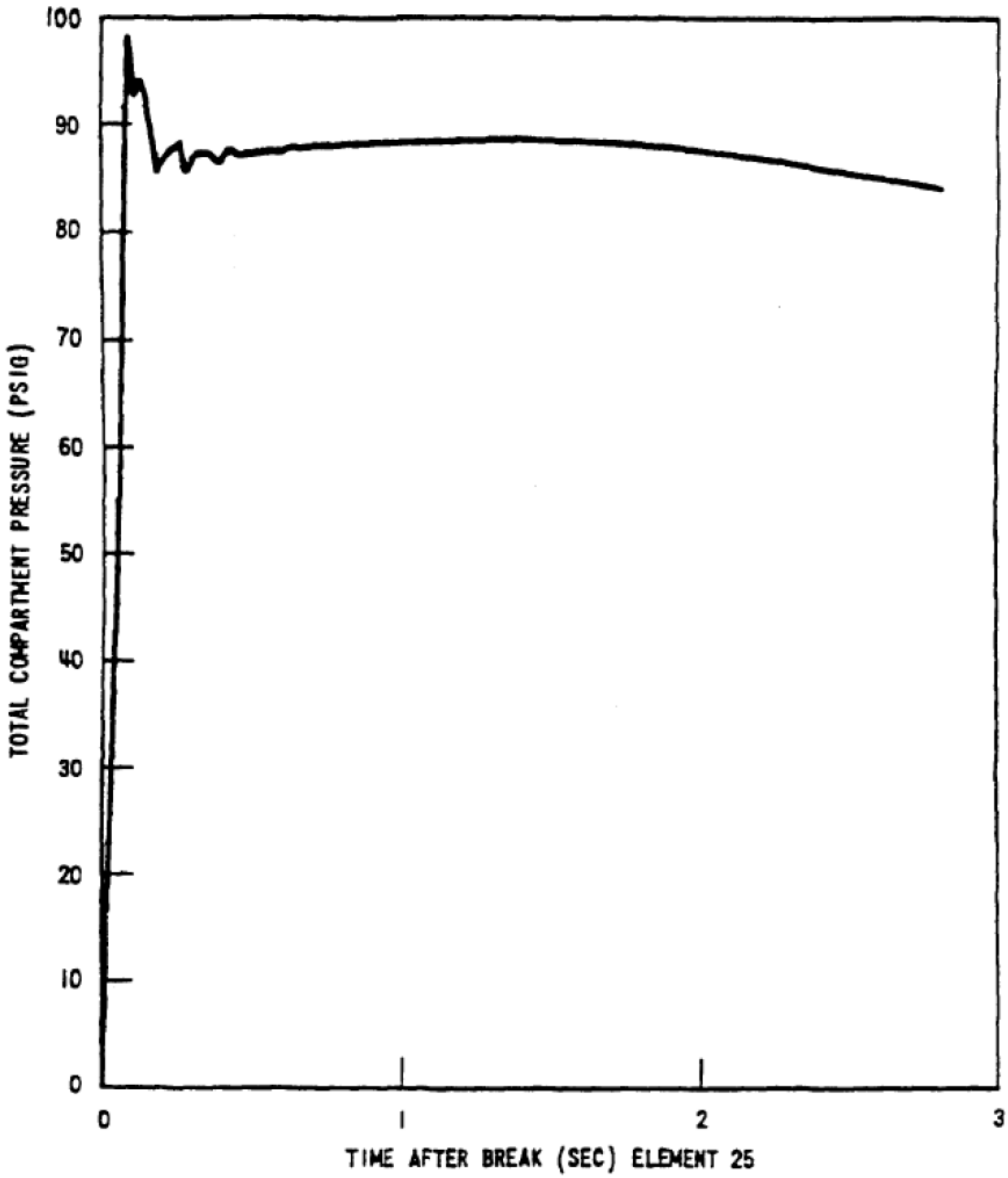
FIGURE 6.2.1-29



WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

127 Square Inch
Cold Leg Break
(Reactor Cavity Analysis)

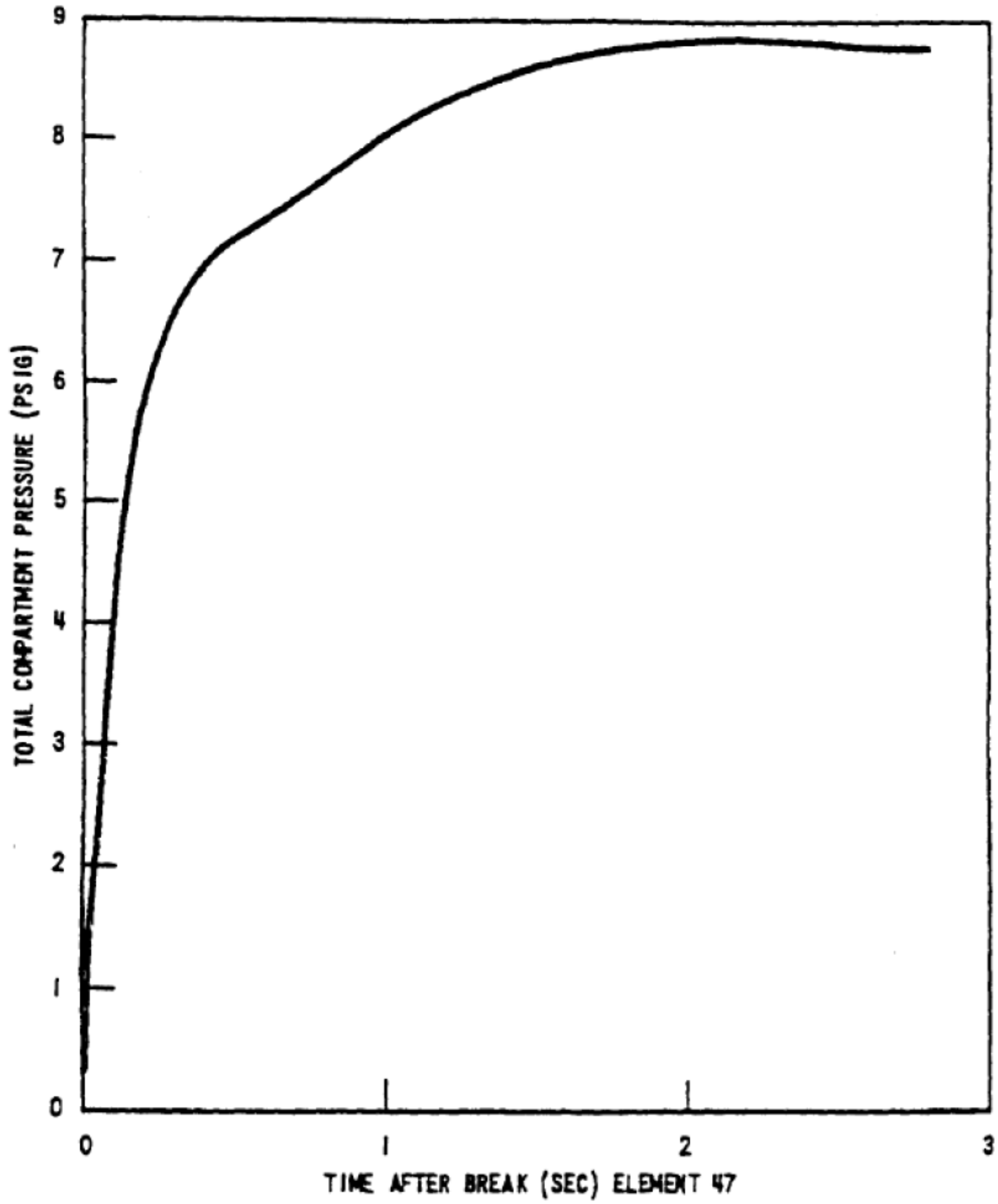
Figure 6.2.1-30



WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

127 Square Inch
Cold Leg Break
(Reactor Cavity Analysis)

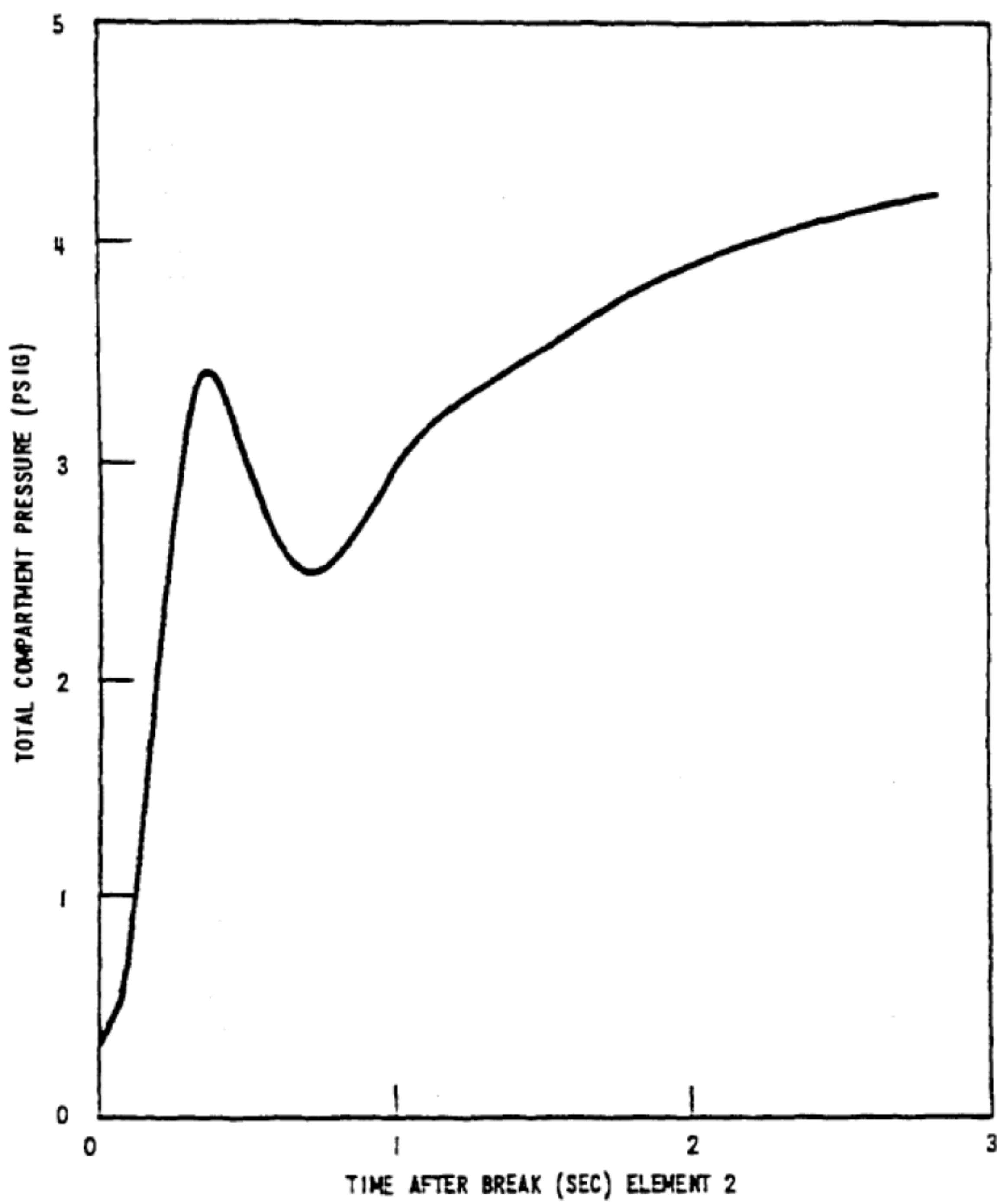
Figure 6.2.1-31



WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

127 Square Inch
Cold Leg Break
(Reactor Cavity Analysis)

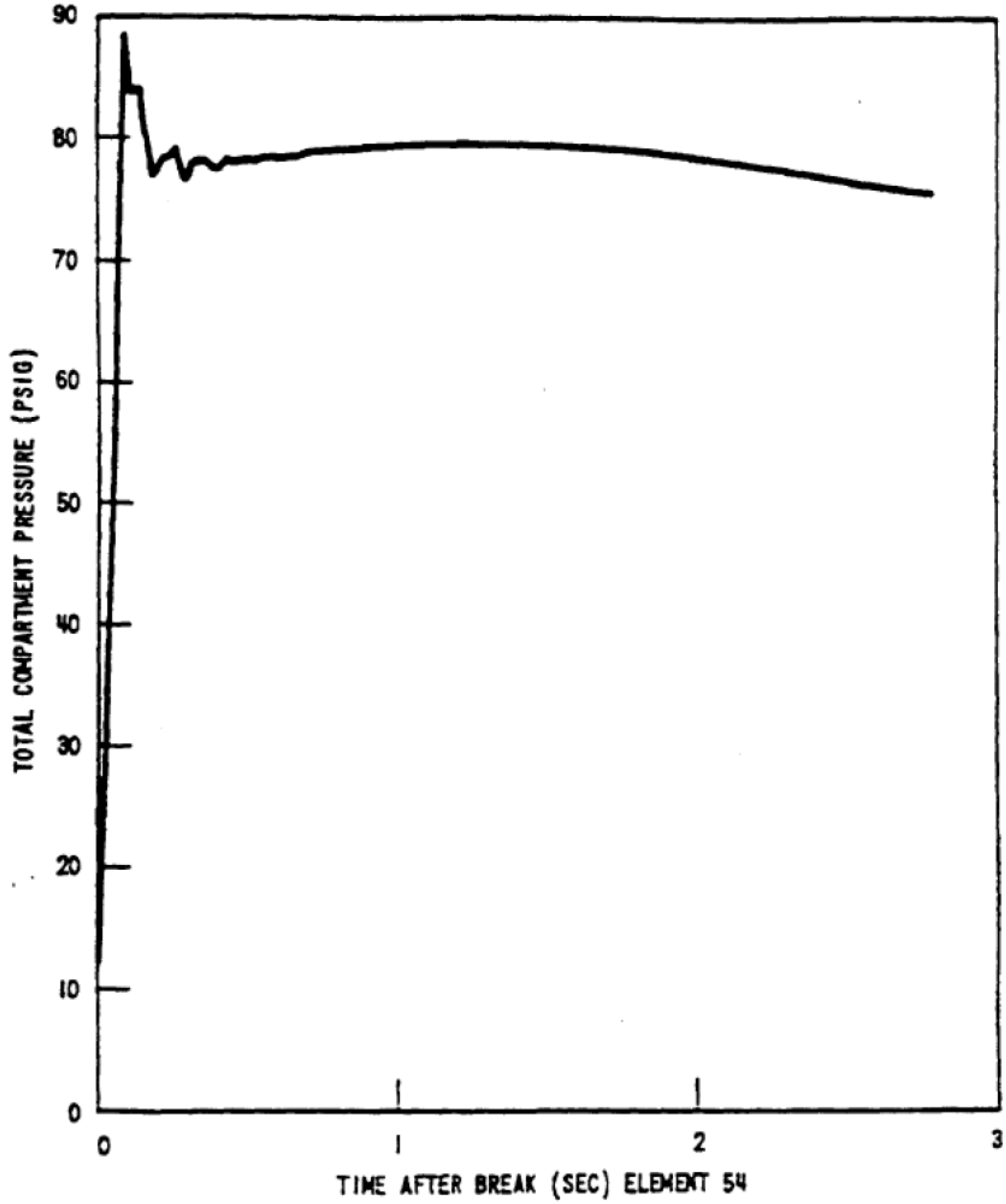
Figure 6.2.1-32



WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

127 Square Inch
Cold Leg Break
(Reactor Cavity Analysis)

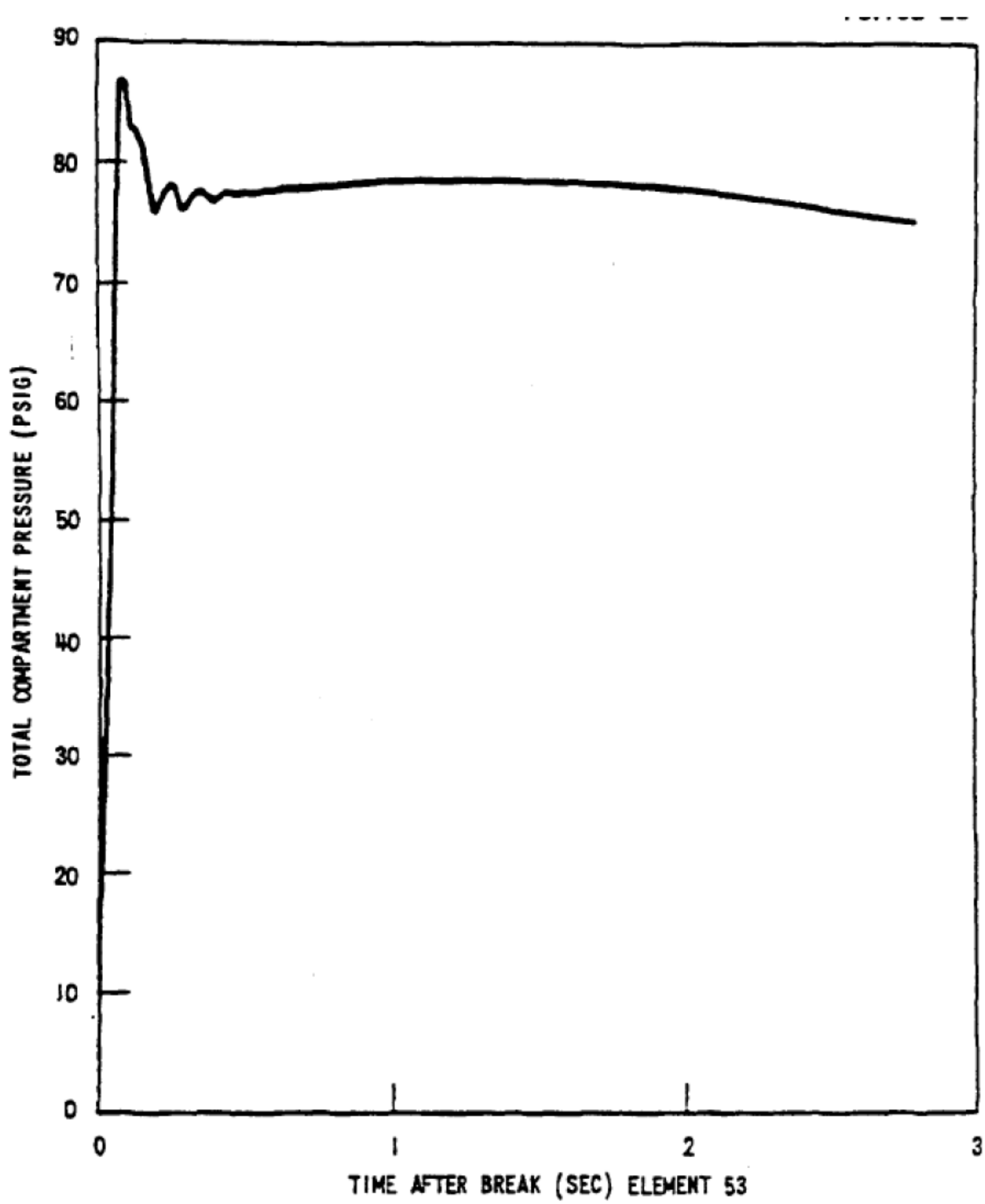
Figure 6.2.1-33



WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

127 Square Inch
Cold Leg Break
(Reactor Cavity Analysis)

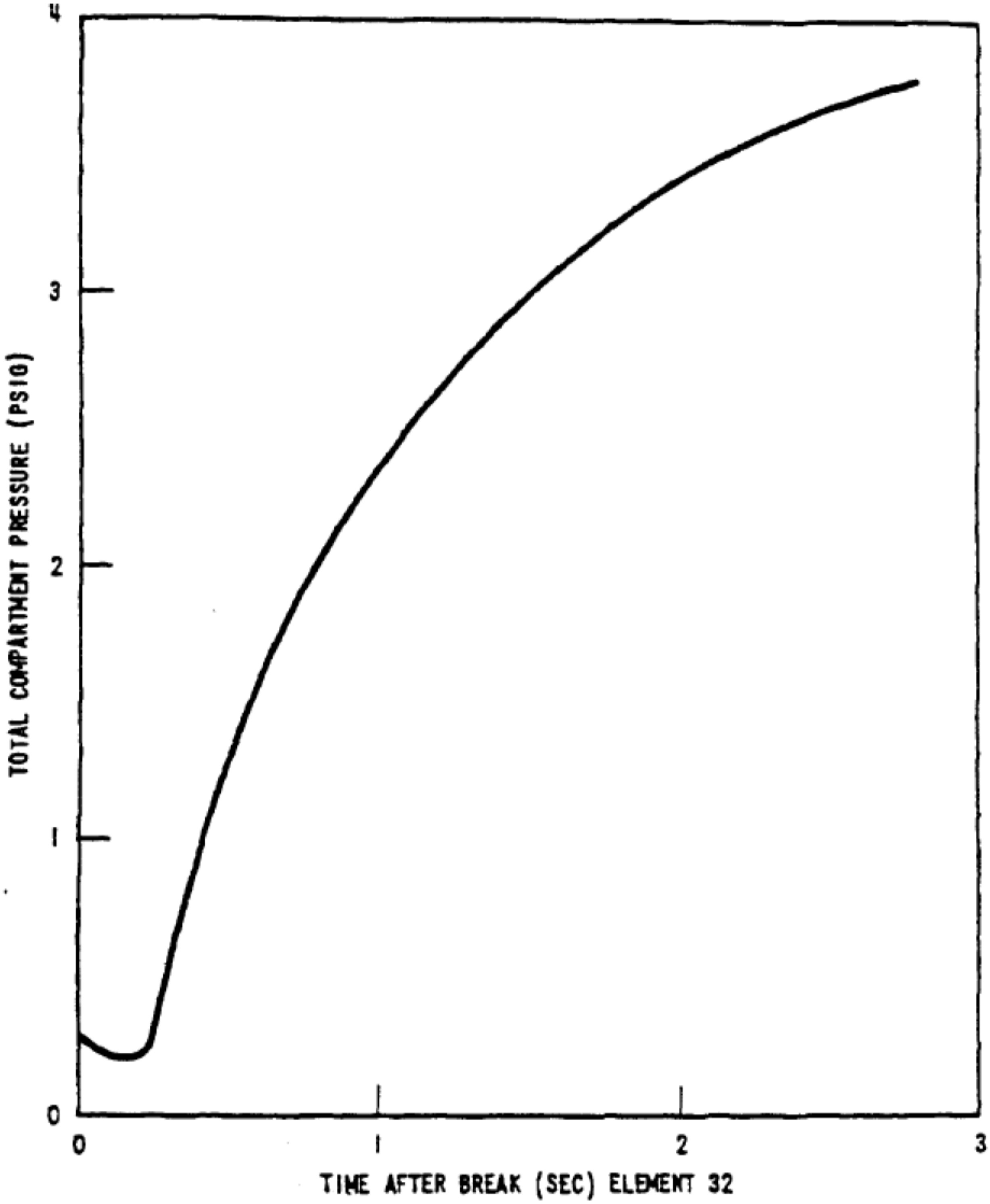
Figure 6.2.1-34



WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

127 Square Inch
Cold Leg Break
(Reactor Cavity Analysis)

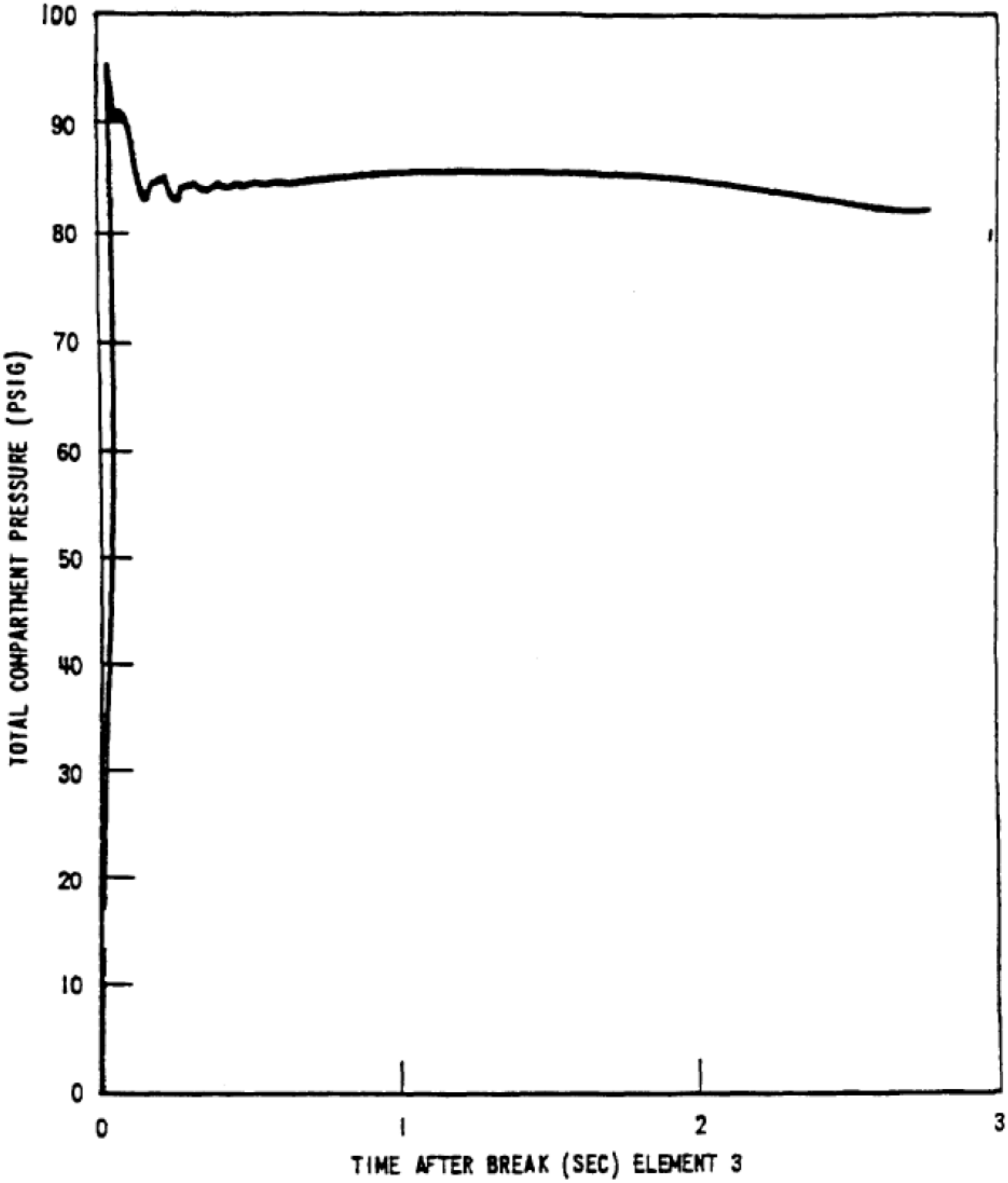
Figure 6.2.1-35



WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

127 Square Inch
Cold Leg Break
(Reactor Cavity Analysis)

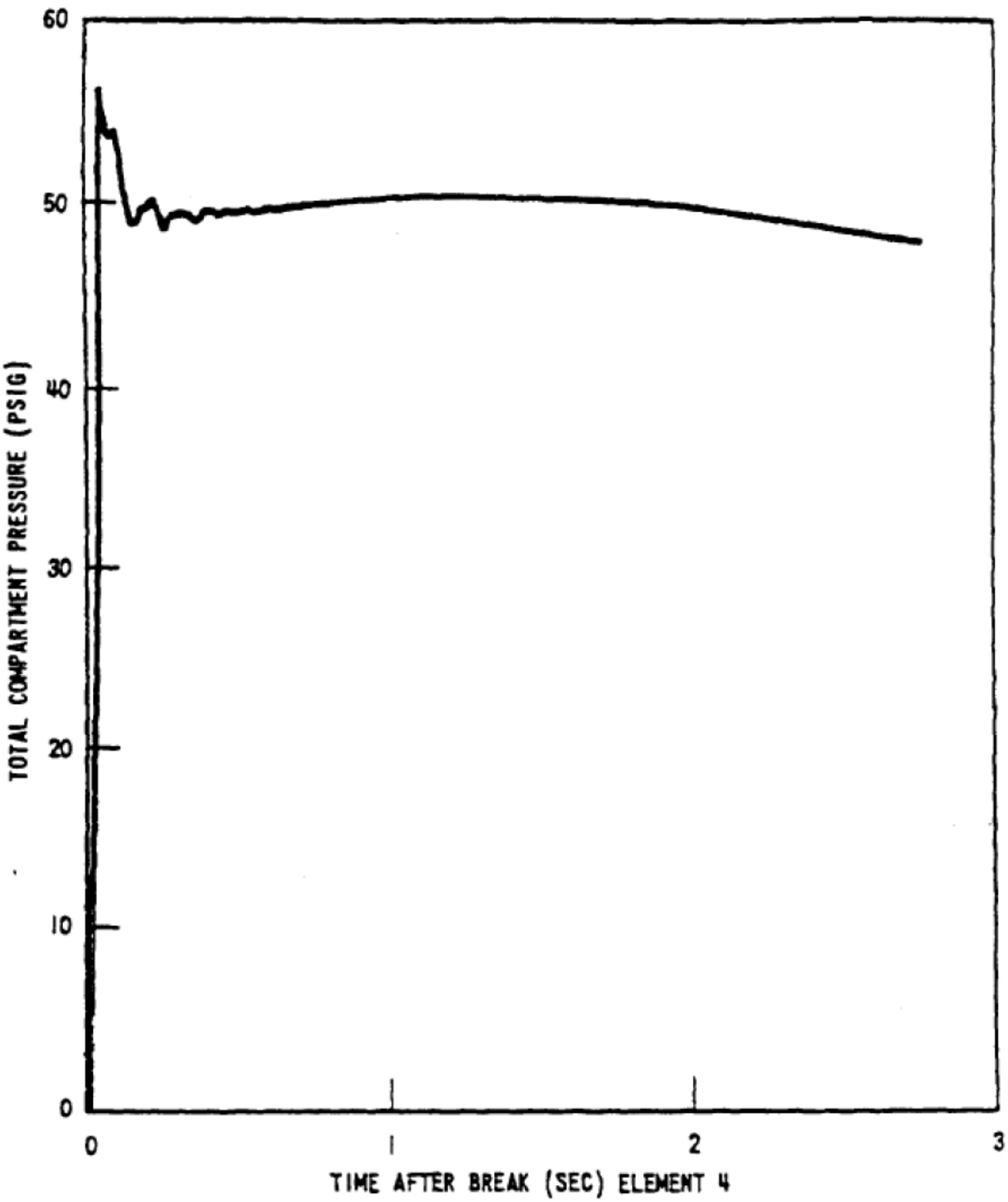
Figure 6.2.1-36



WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

127 Square Inch
Cold Leg Break
(Reactor Cavity Analysis)

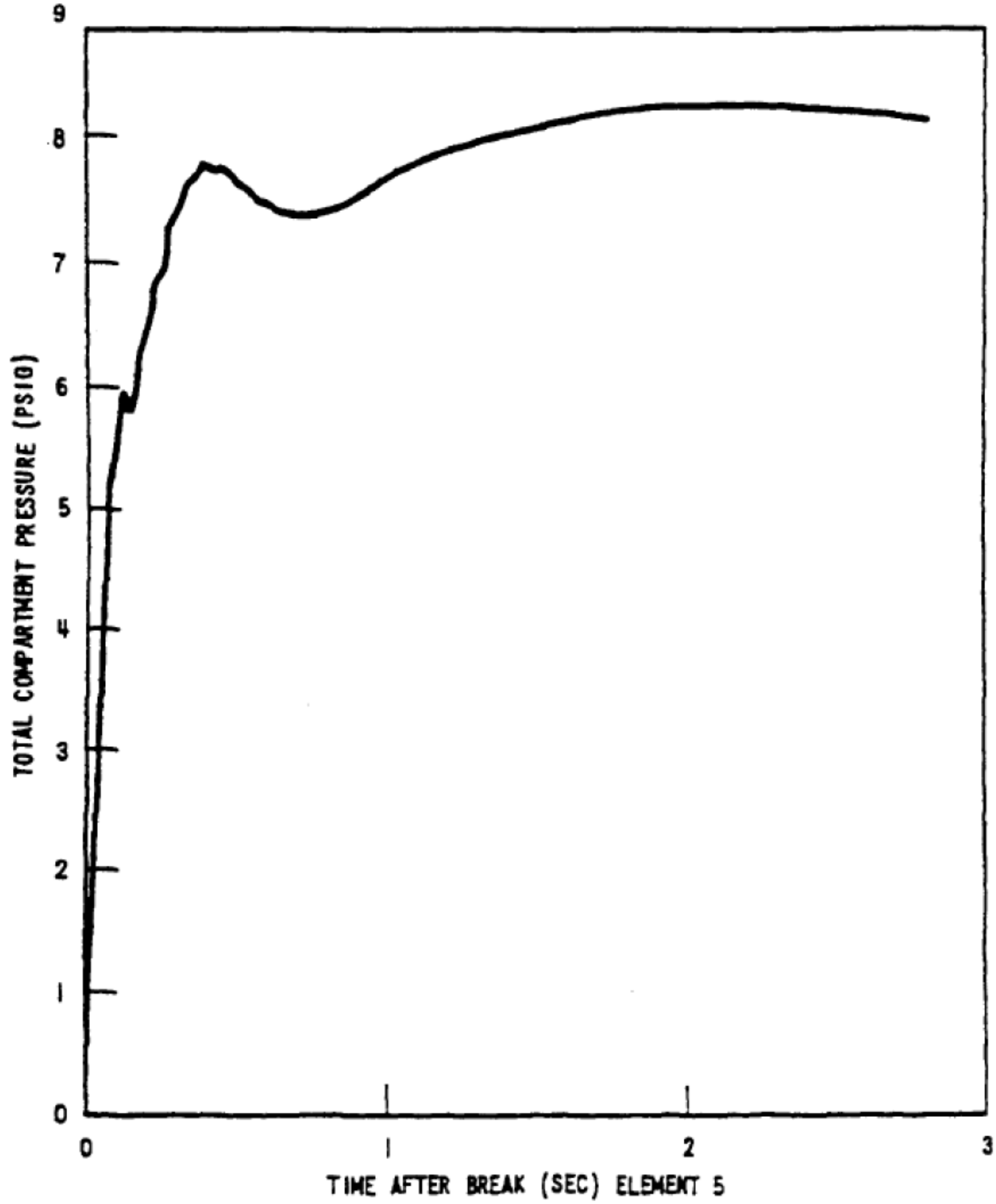
Figure 6.2.1-37



WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

127 Square Inch
Cold Leg Break
(Reactor Cavity Analysis)

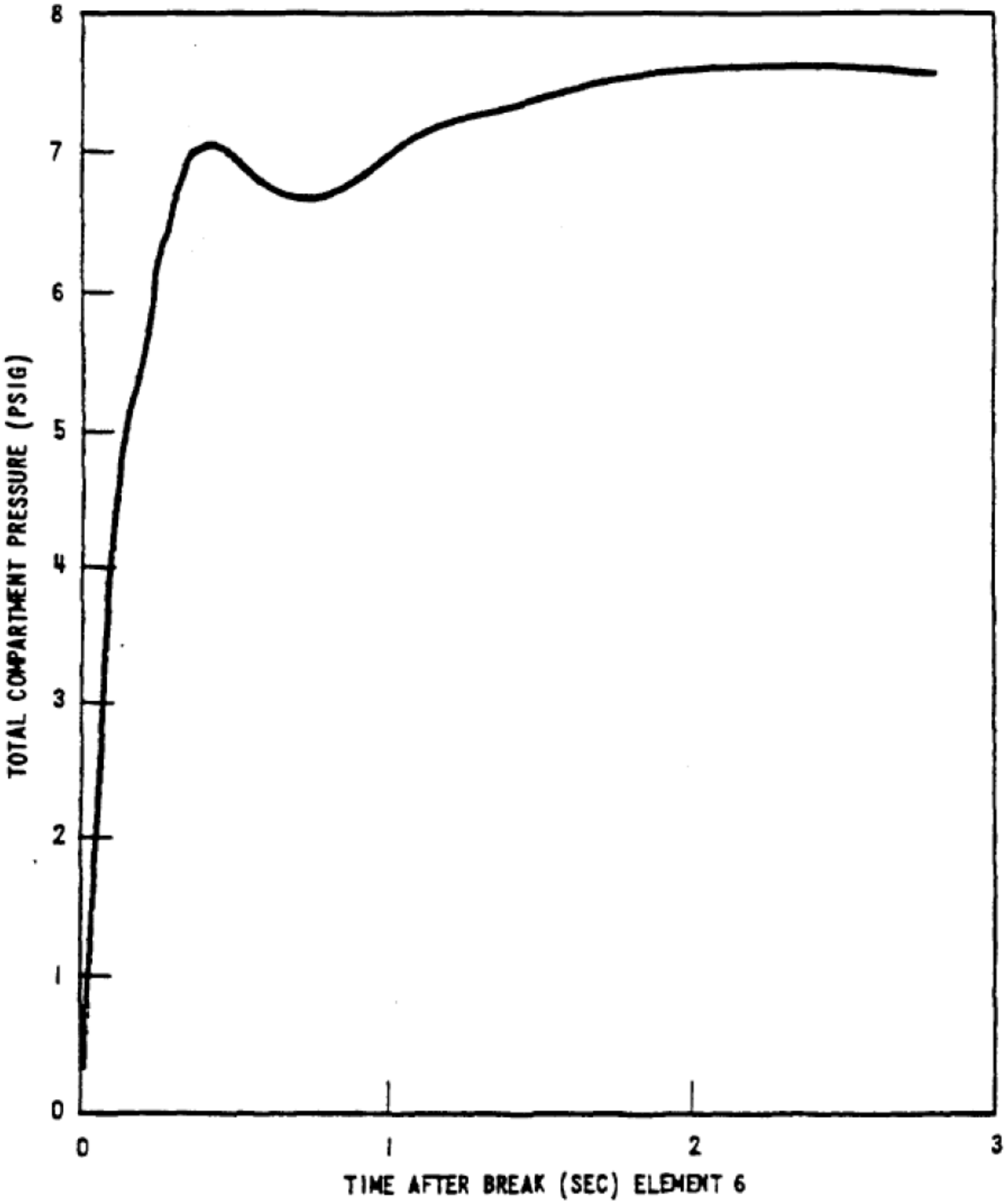
Figure 6.2.1-38



WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

127 Square Inch
Cold Leg Break
(Reactor Cavity Analysis)

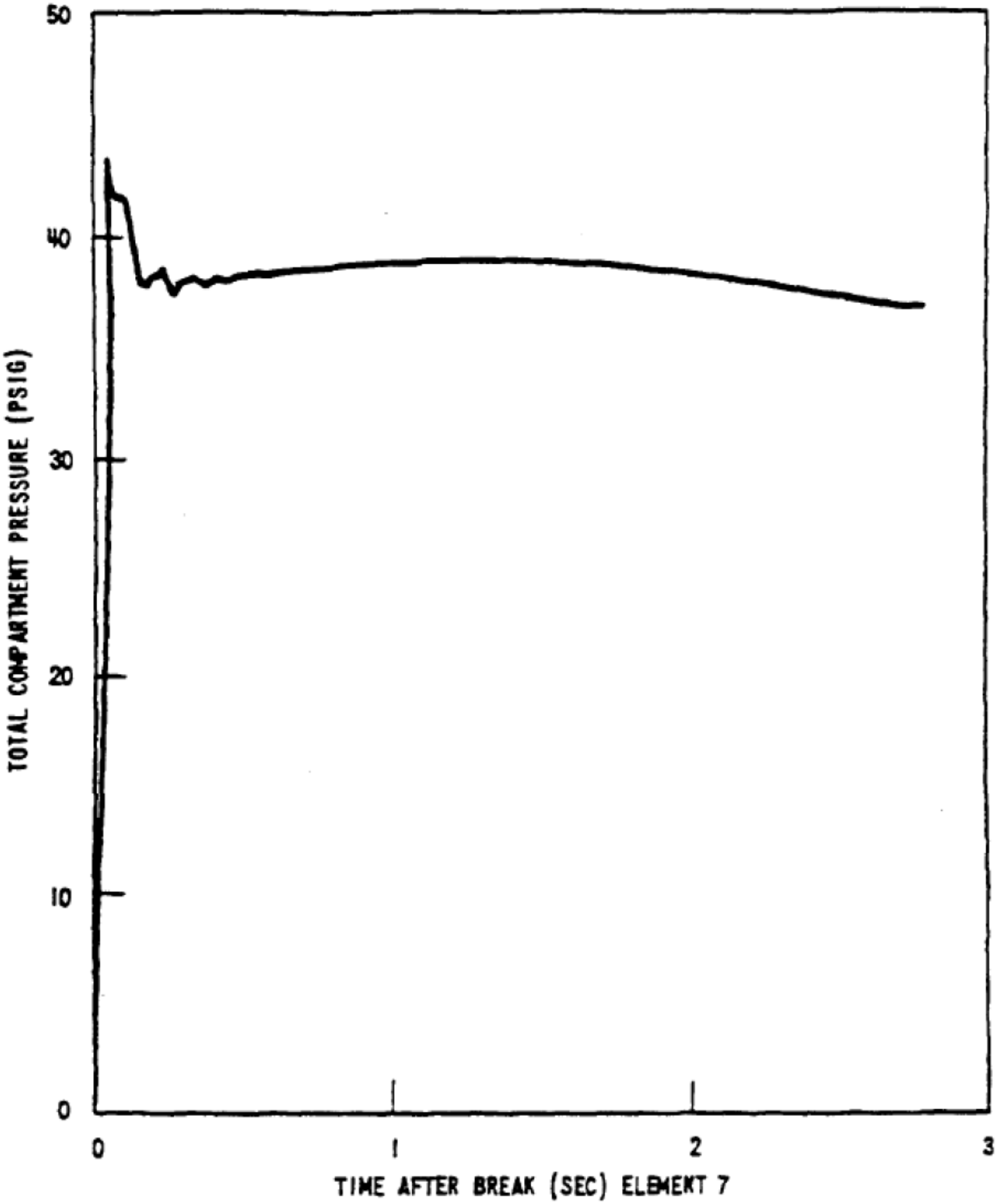
Figure 6.2.1-39



WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

127 Square Inch
Cold Leg Break
(Reactor Cavity Analysis)

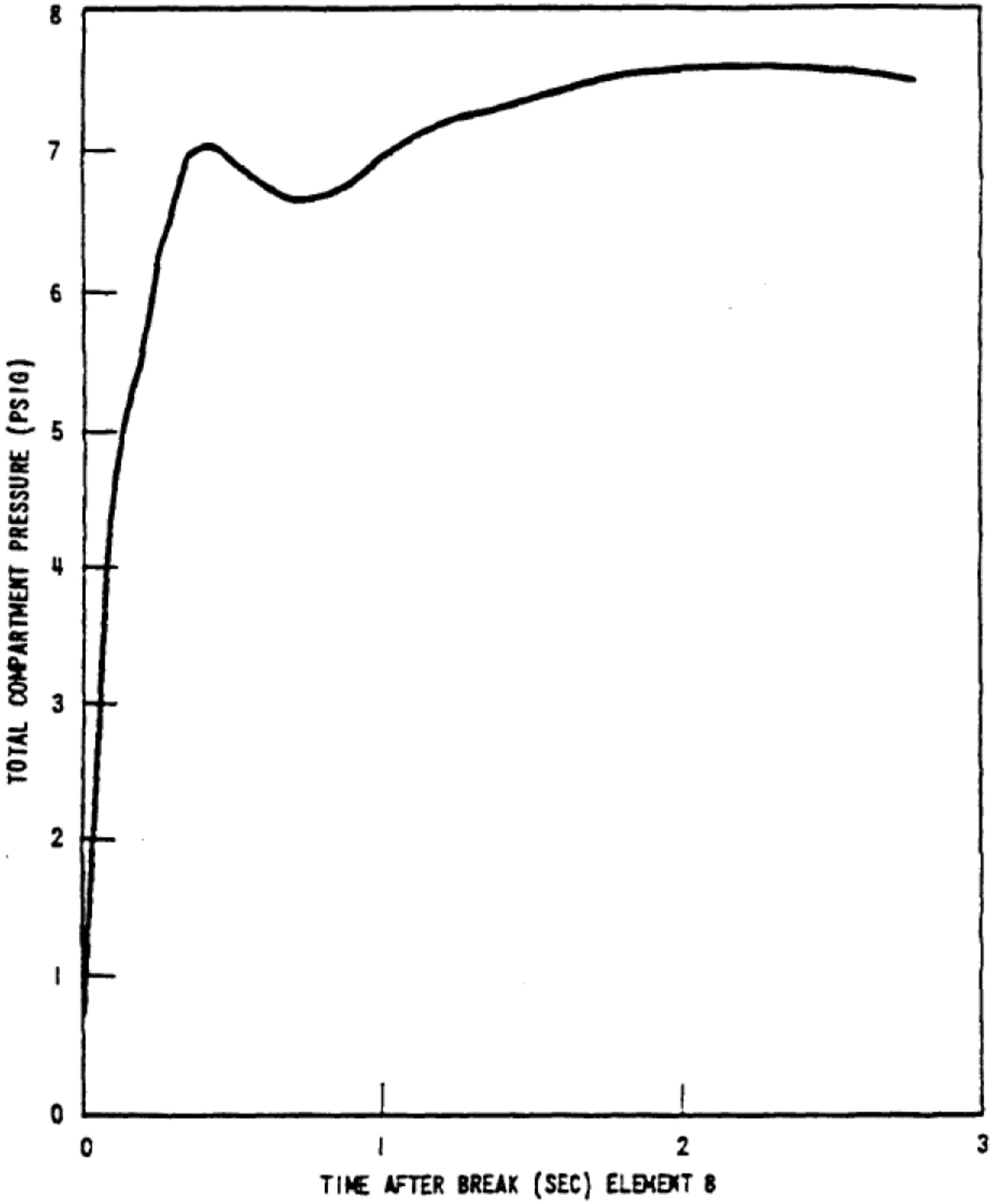
Figure 6.2.1-40



WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

127 Square Inch
Cold Leg Break
(Reactor Cavity Analysis)

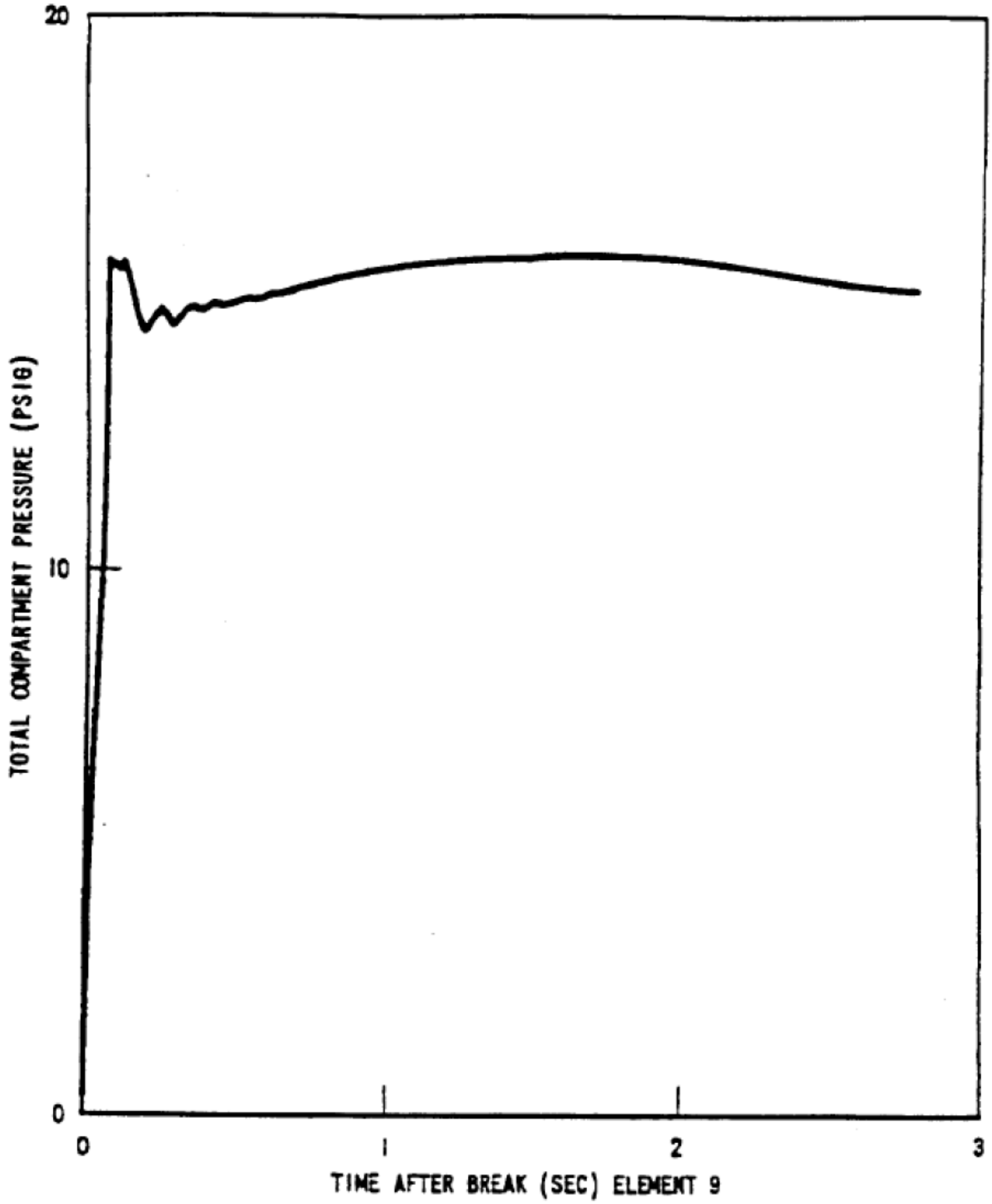
Figure 6.2.1-41



WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

127 Square Inch
Cold Leg Break
(Reactor Cavity Analysis)

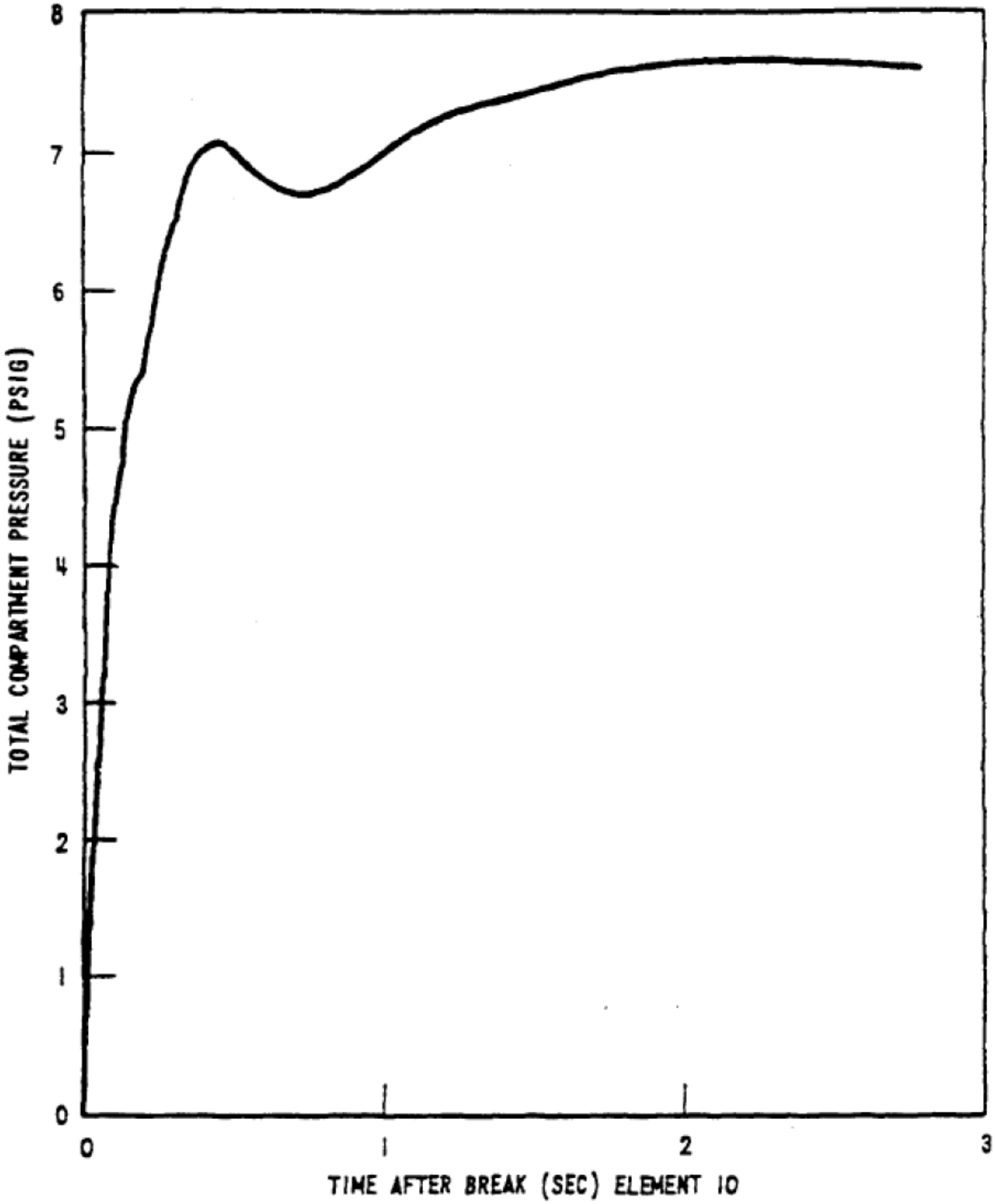
Figure 6.2.1-42



WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

127 Square Inch
Cold Leg Break
(Reactor Cavity Analysis)

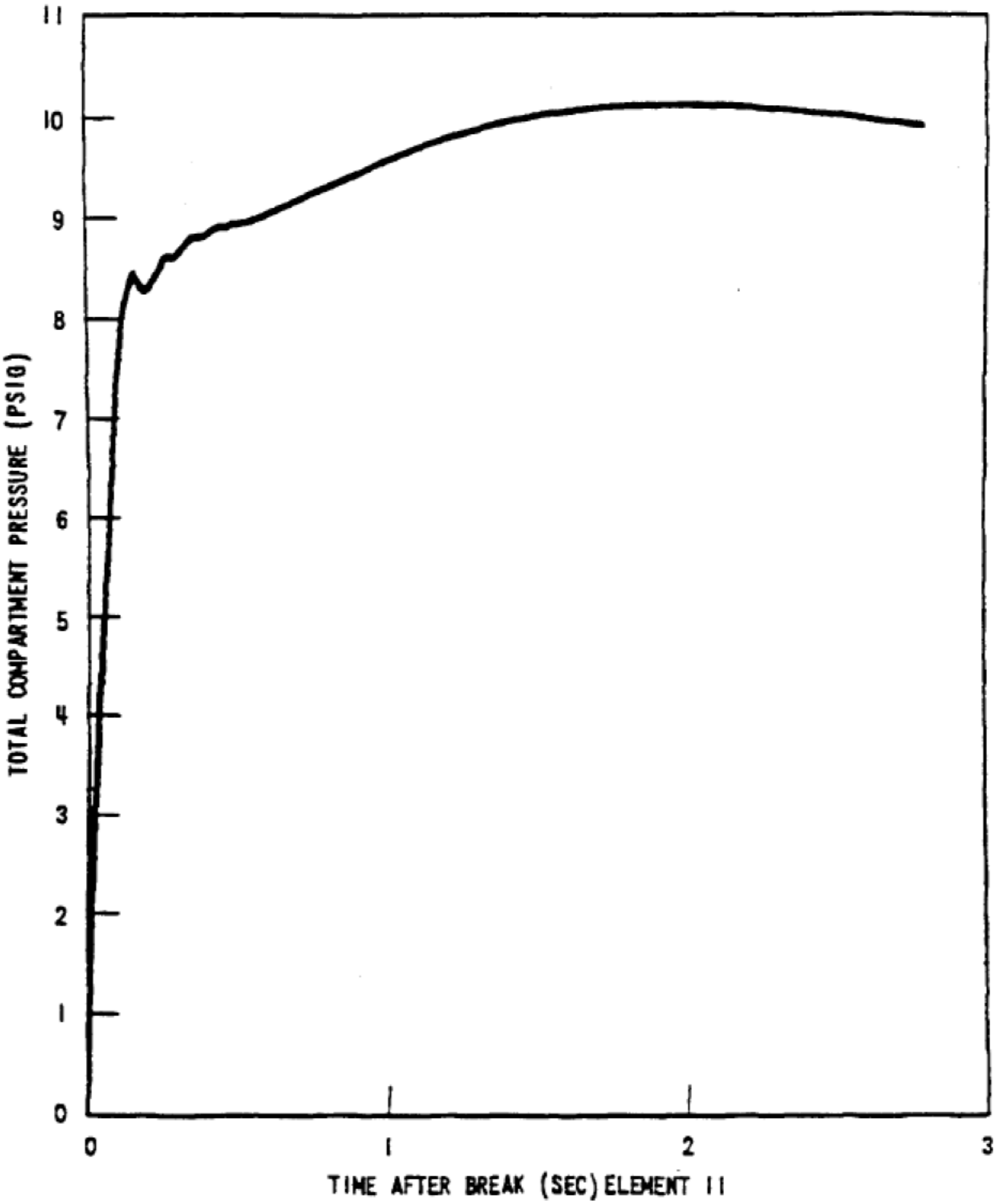
Figure 6.2.1-43



WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

127 Square Inch
Cold Leg Break
(Reactor Cavity Analysis)

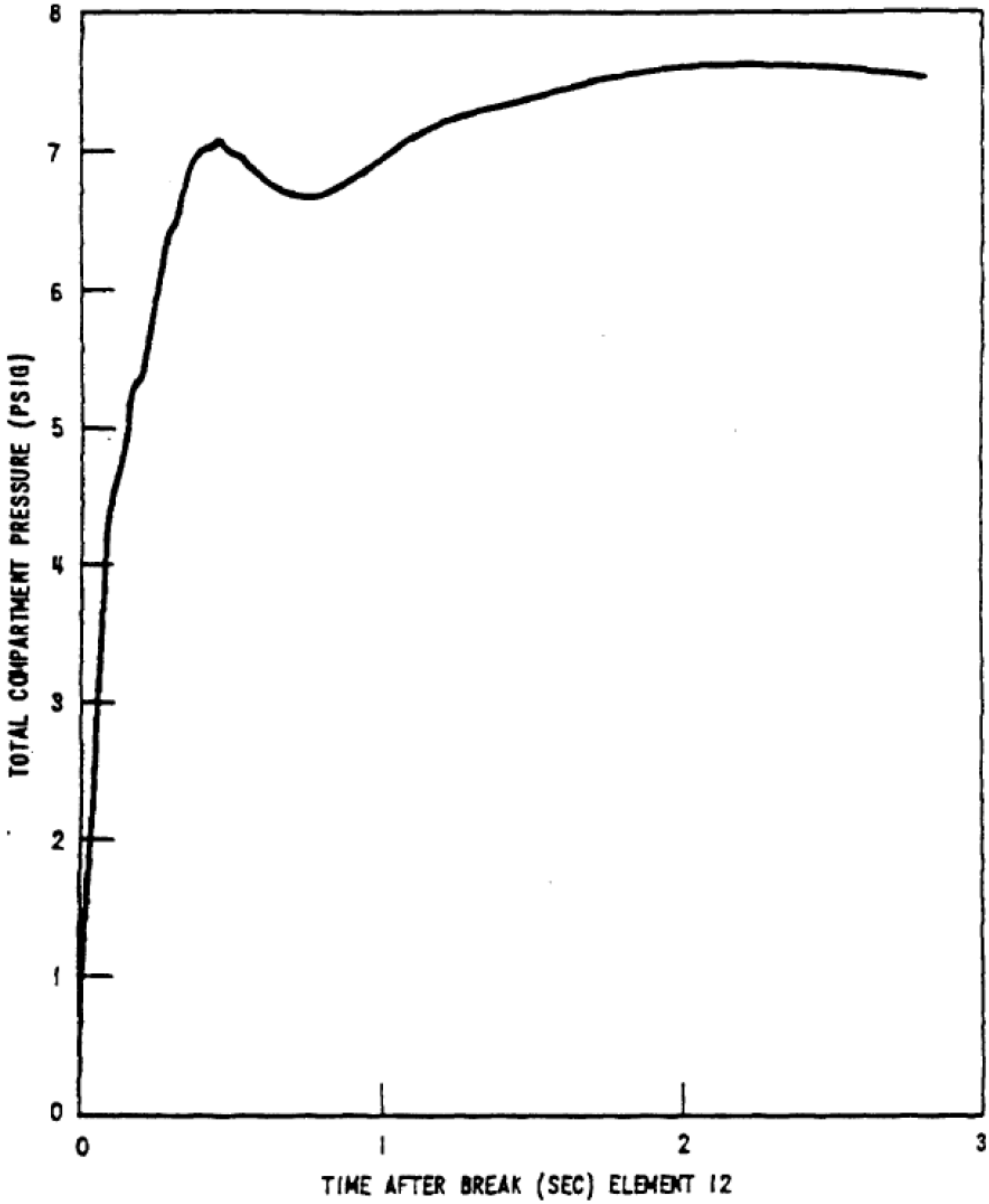
Figure 6.2.1-44



WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

127 Square Inch
Cold Leg Break
(Reactor Cavity Analysis)

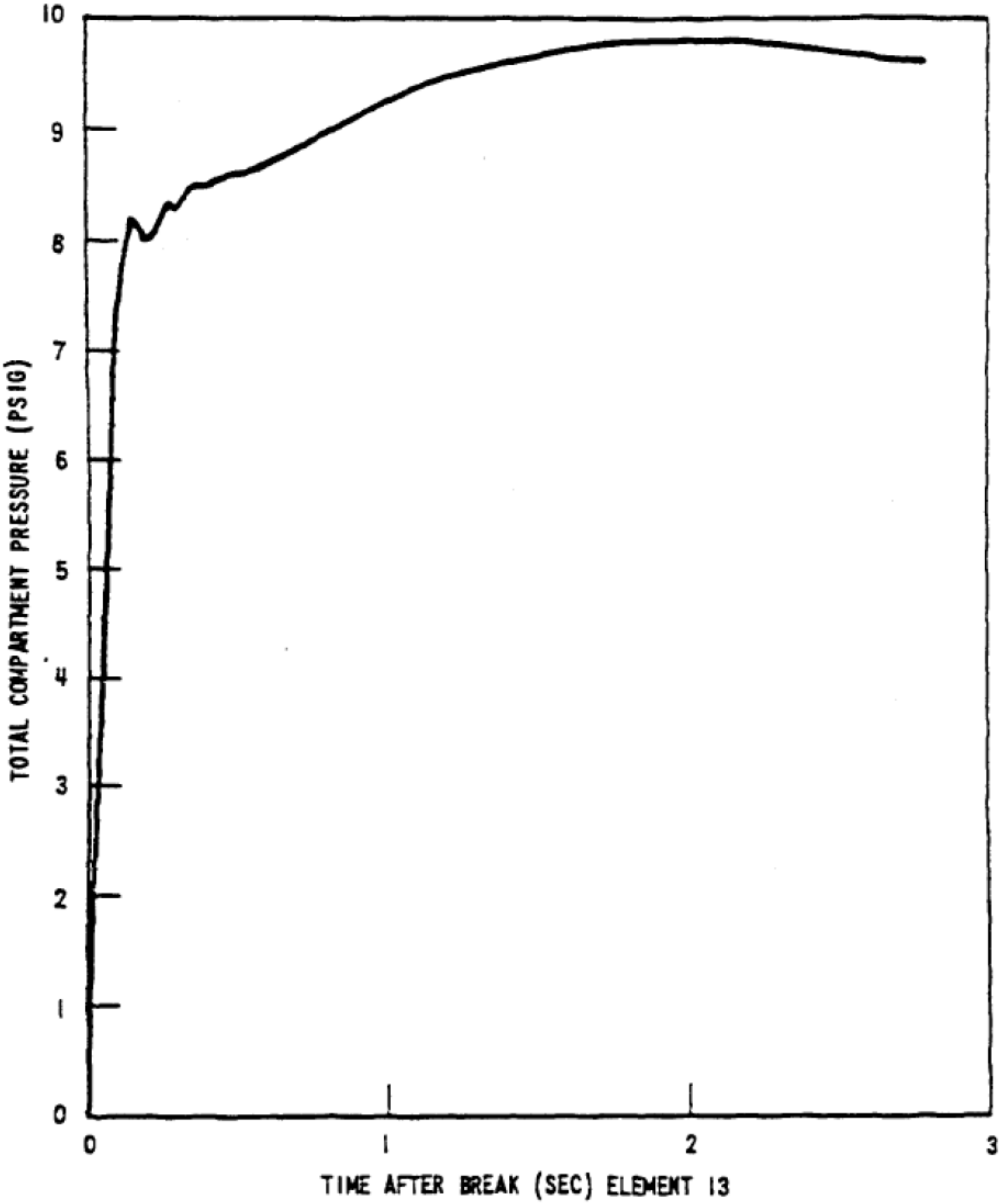
Figure 6.2.1-45



WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

127 Square Inch
Cold Leg Break
(Reactor Cavity Analysis)

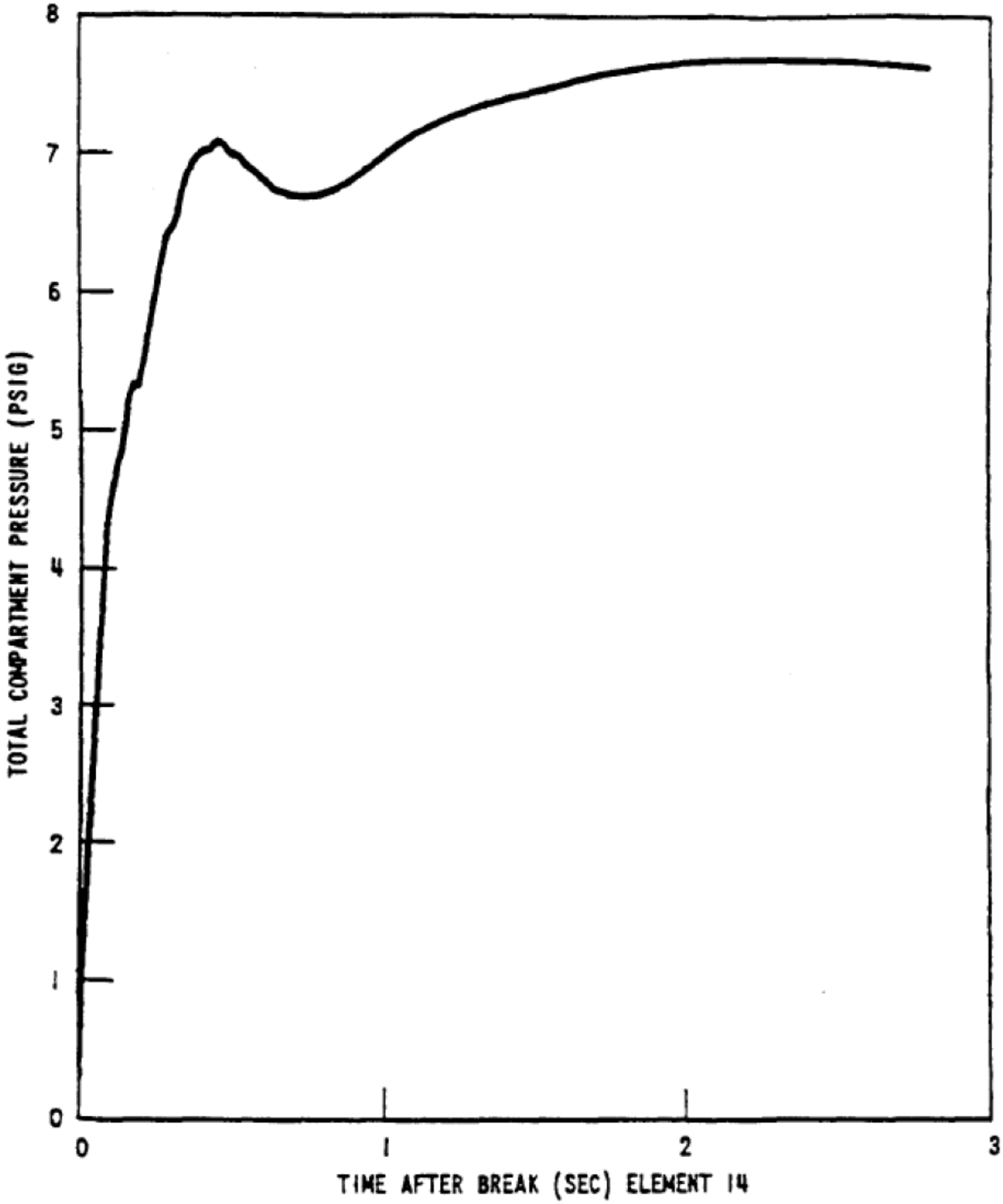
Figure 6.2.1-46



WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

127 Square Inch
Cold Leg Break
(Reactor Cavity Analysis)

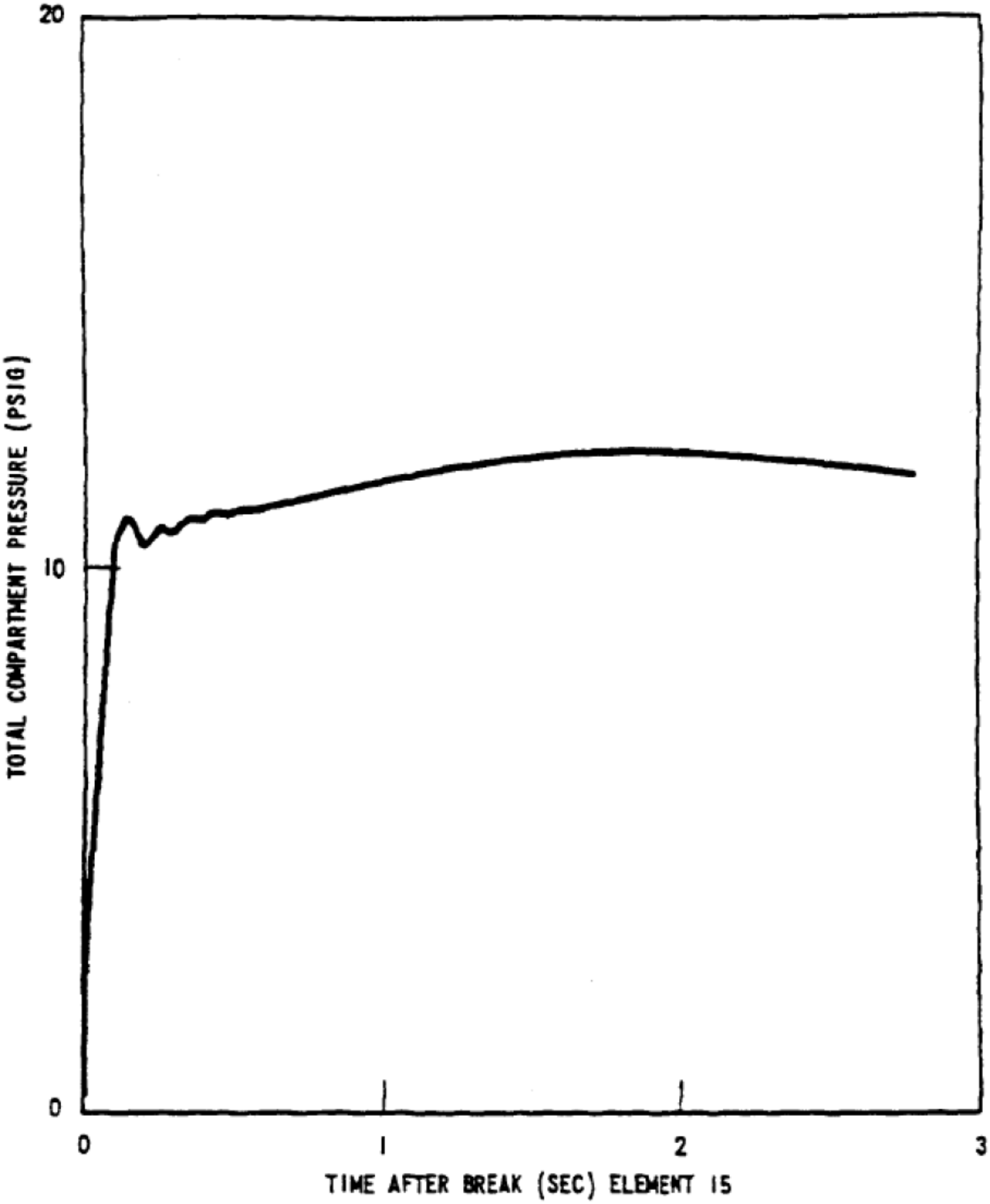
Figure 6.2.1-47



WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

127 Square Inch
Cold Leg Break
(Reactor Cavity Analysis)

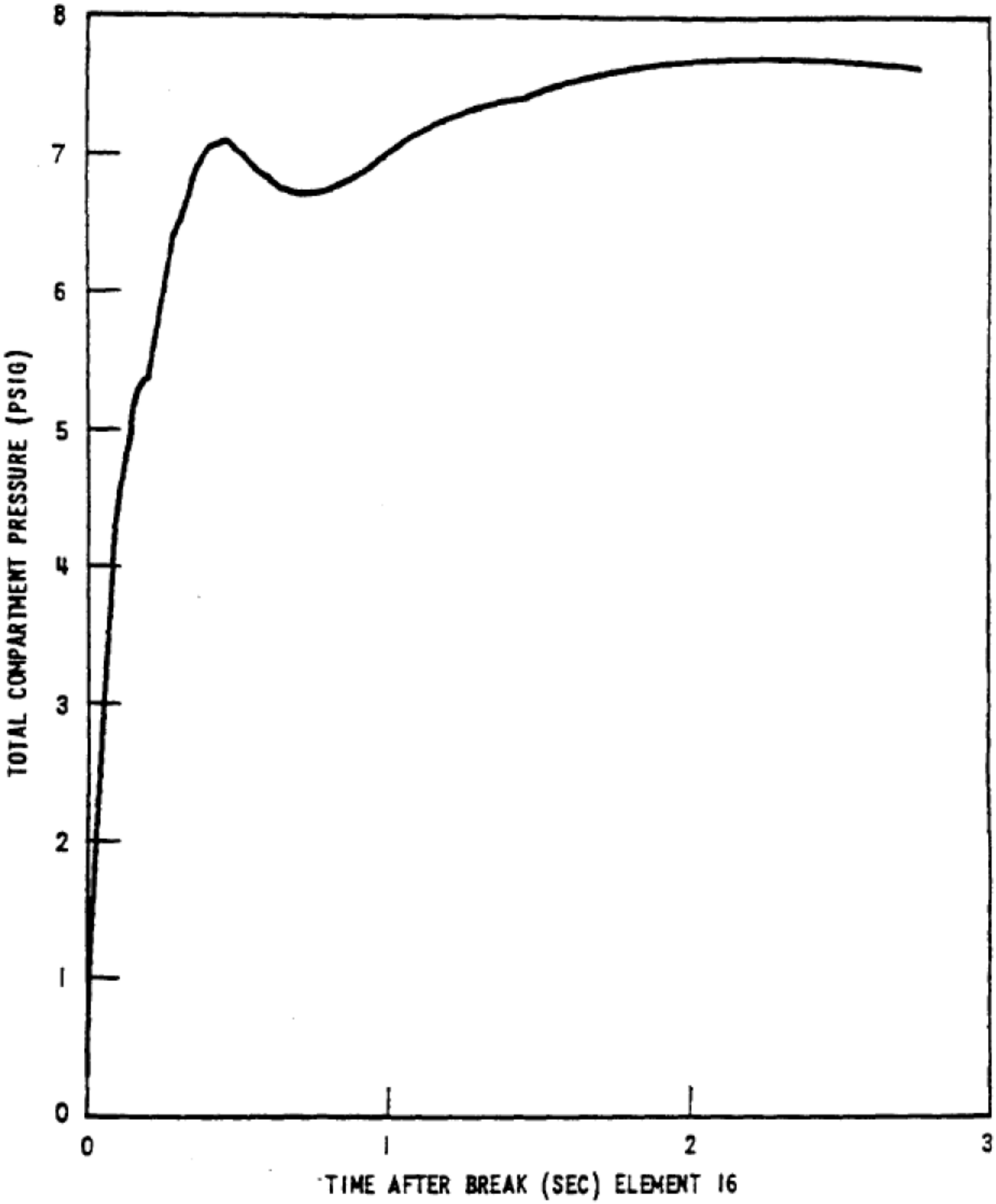
Figure 6.2.1-48



WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

127 Square Inch
Cold Leg Break
(Reactor Cavity Analysis)

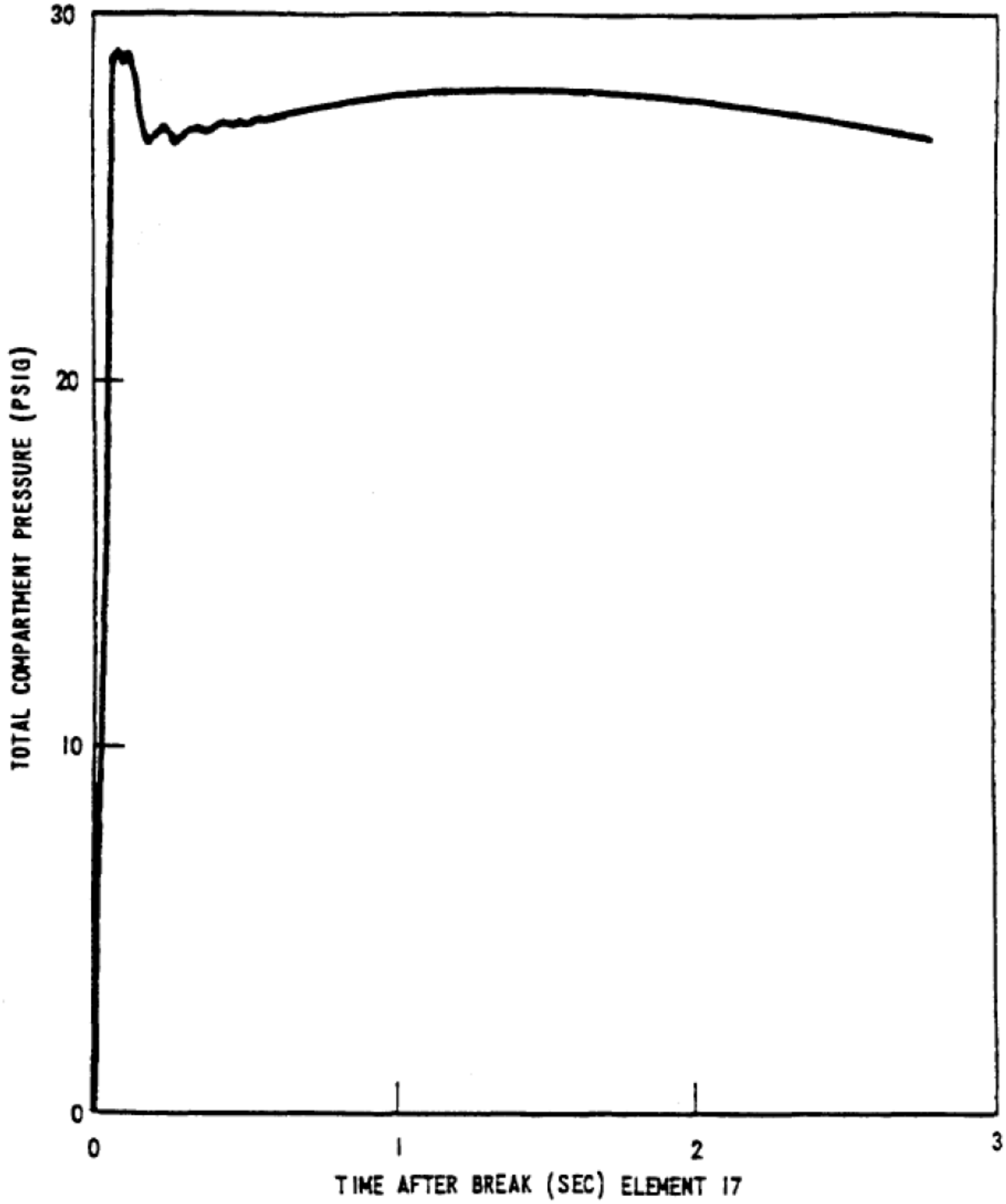
Figure 6.2.1-49



WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

127 Square Inch
Cold Leg Break
(Reactor Cavity Analysis)

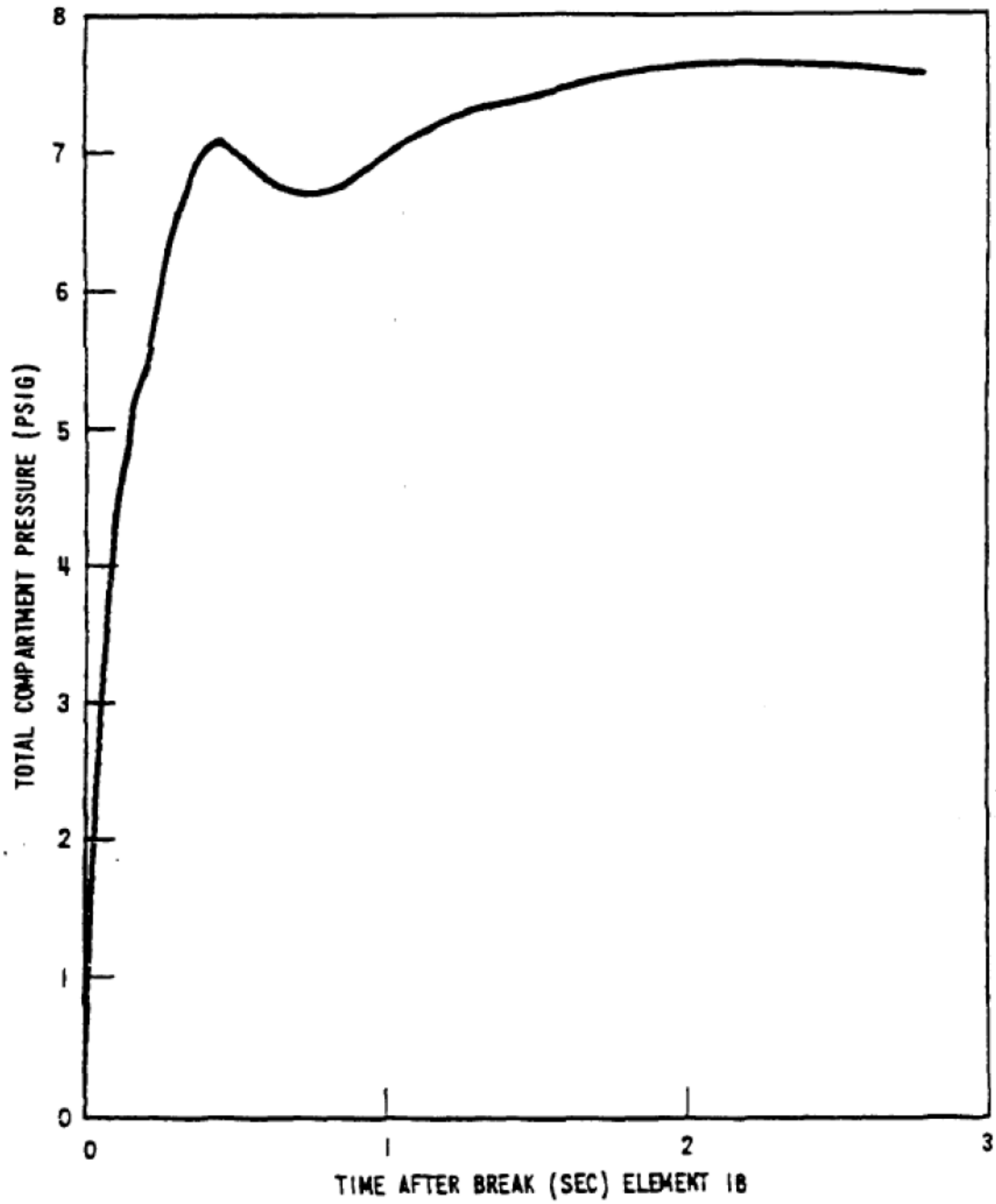
Figure 6.2.1-50



WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

127 Square Inch
Cold Leg Break
(Reactor Cavity Analysis)

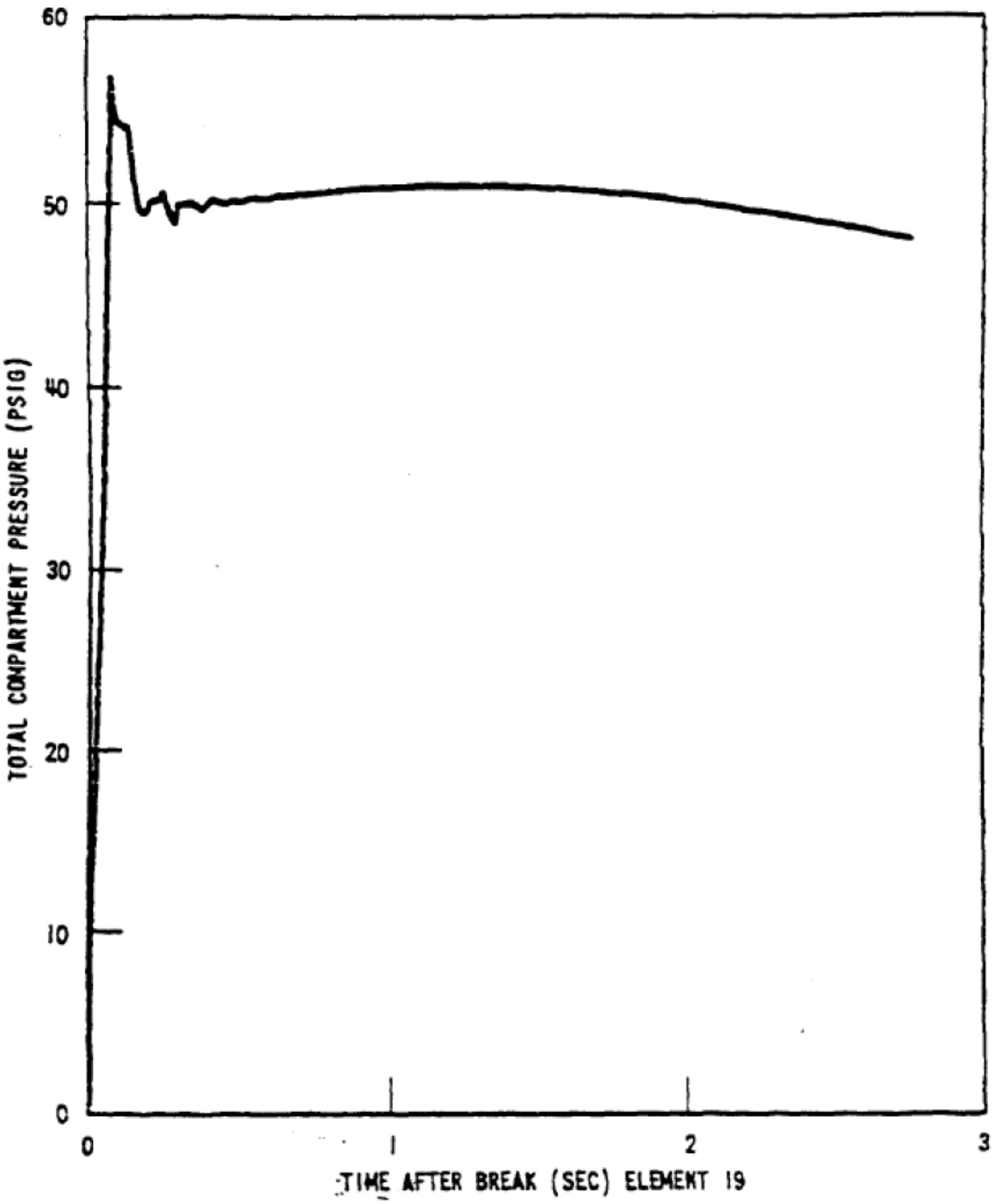
Figure 6.2.1-51



WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

127 Square Inch
Cold Leg Break
(Reactor Cavity Analysis)

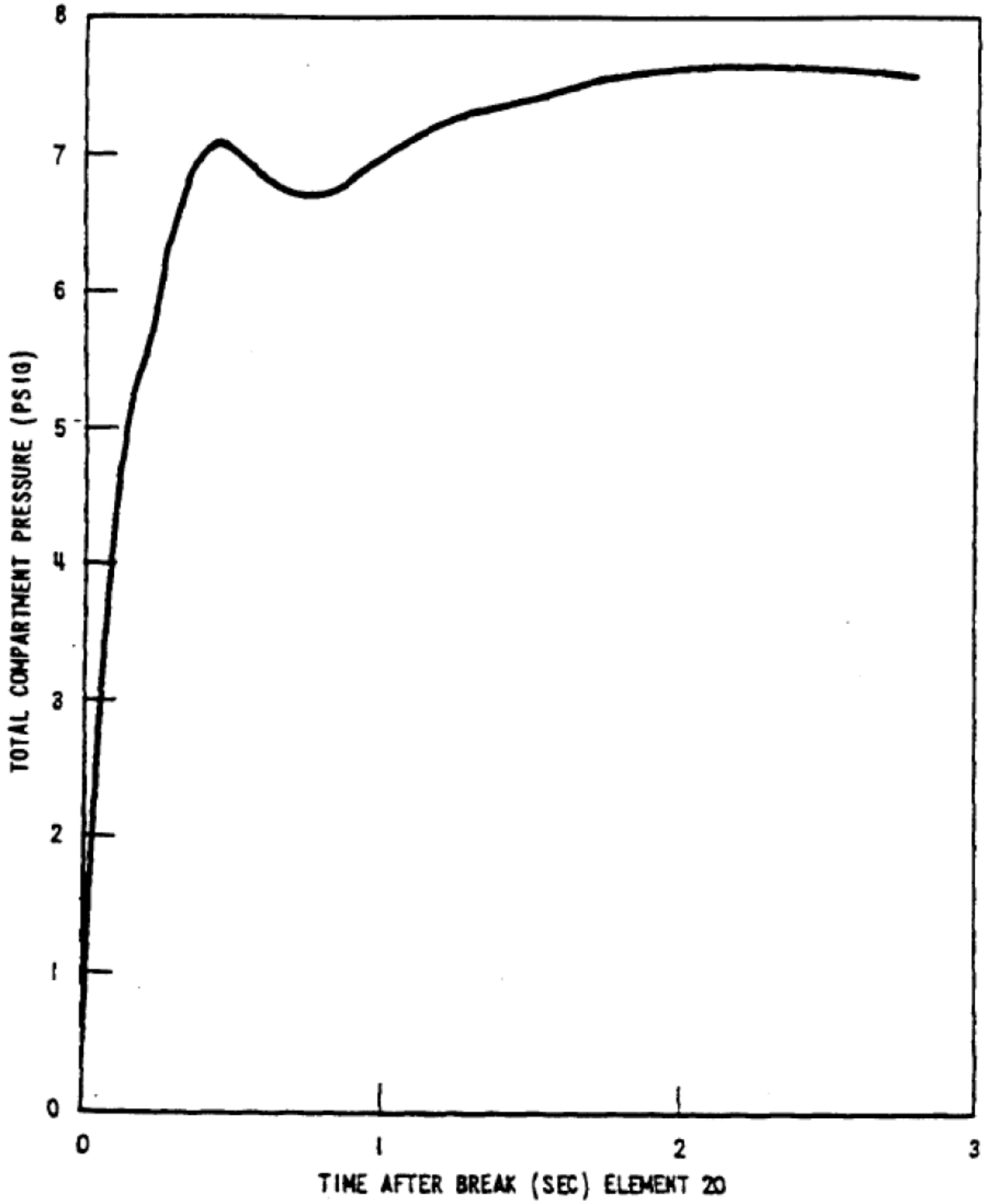
Figure 6.2.1-52



WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

127 Square Inch
Cold Leg Break
(Reactor Cavity Analysis)

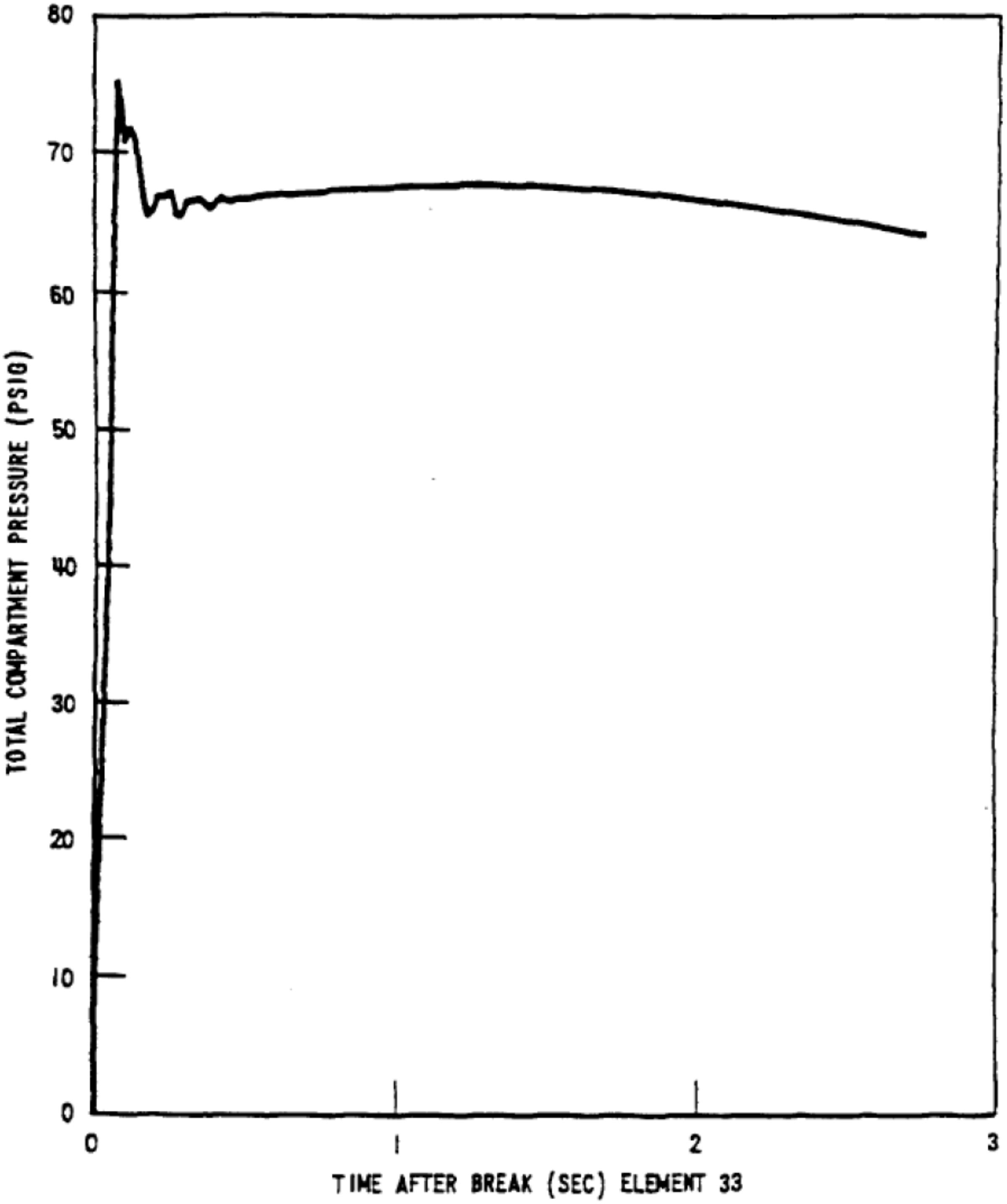
Figure 6.2.1-53



WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

127 Square Inch
Cold Leg Break
(Reactor Cavity Analysis)

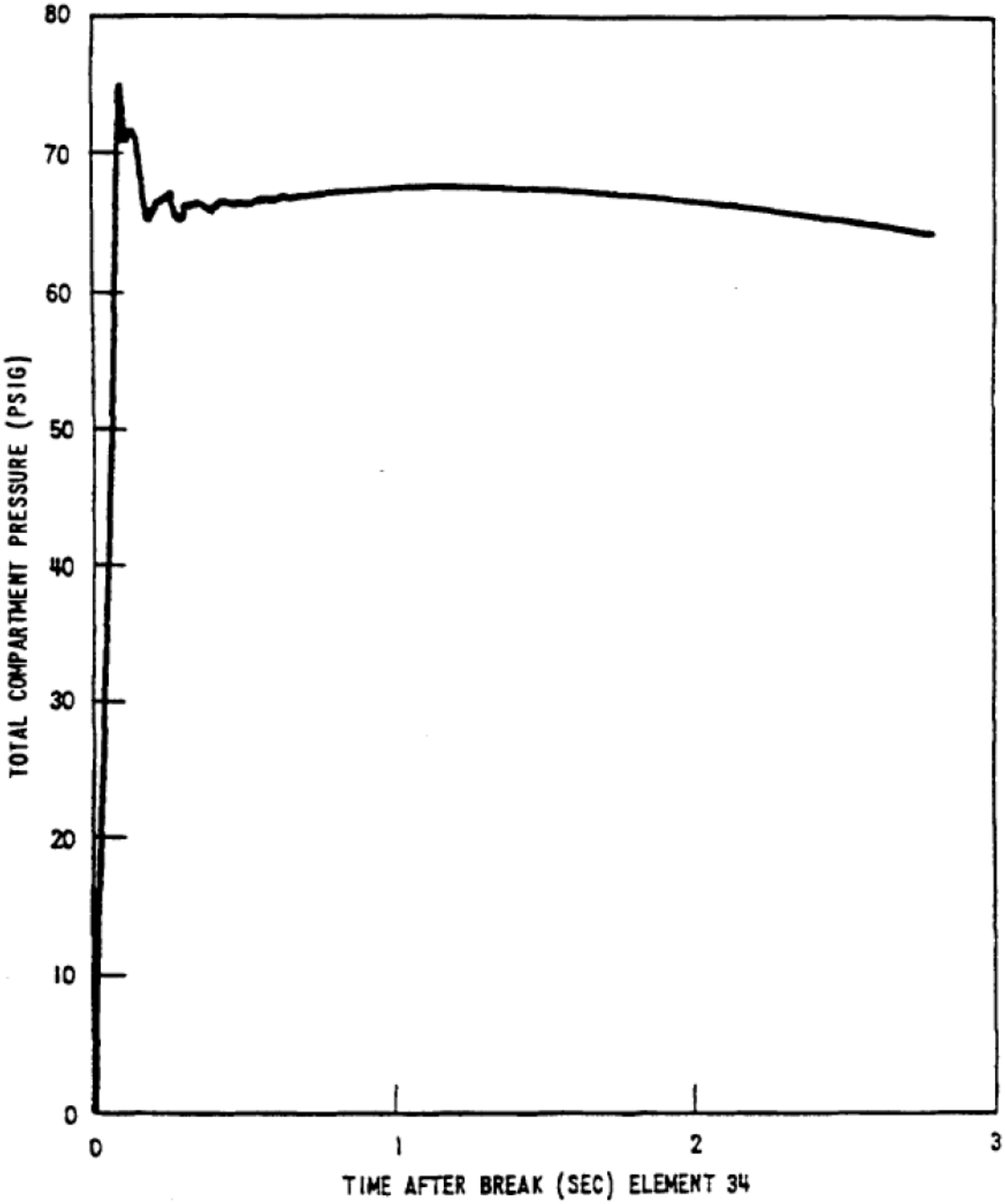
Figure 6.2.1-54



WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

127 Square Inch
Cold Leg Break
(Reactor Cavity Analysis)

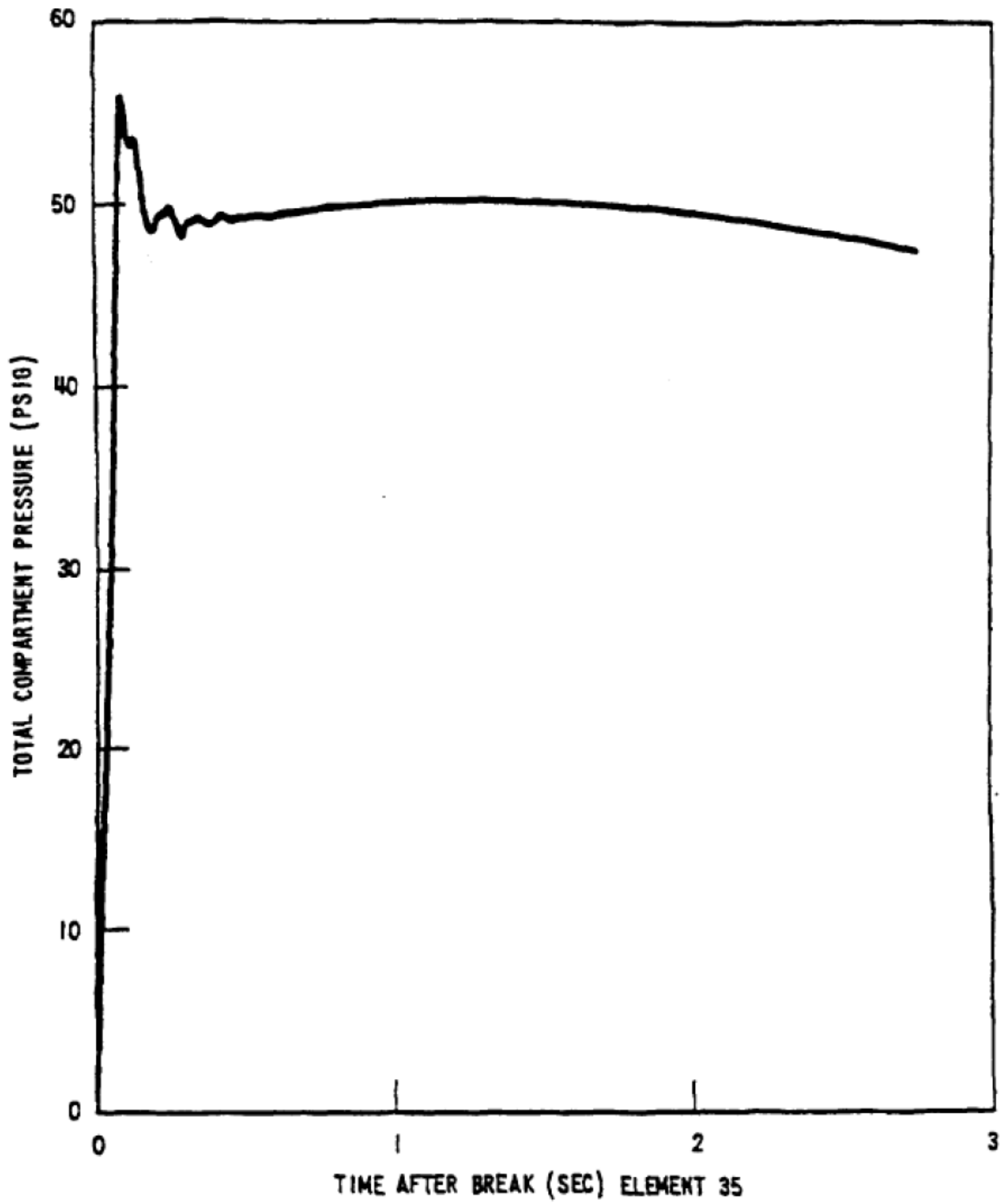
Figure 6.2.1-55



WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

127 Square Inch
Cold Leg Break
(Reactor Cavity Analysis)

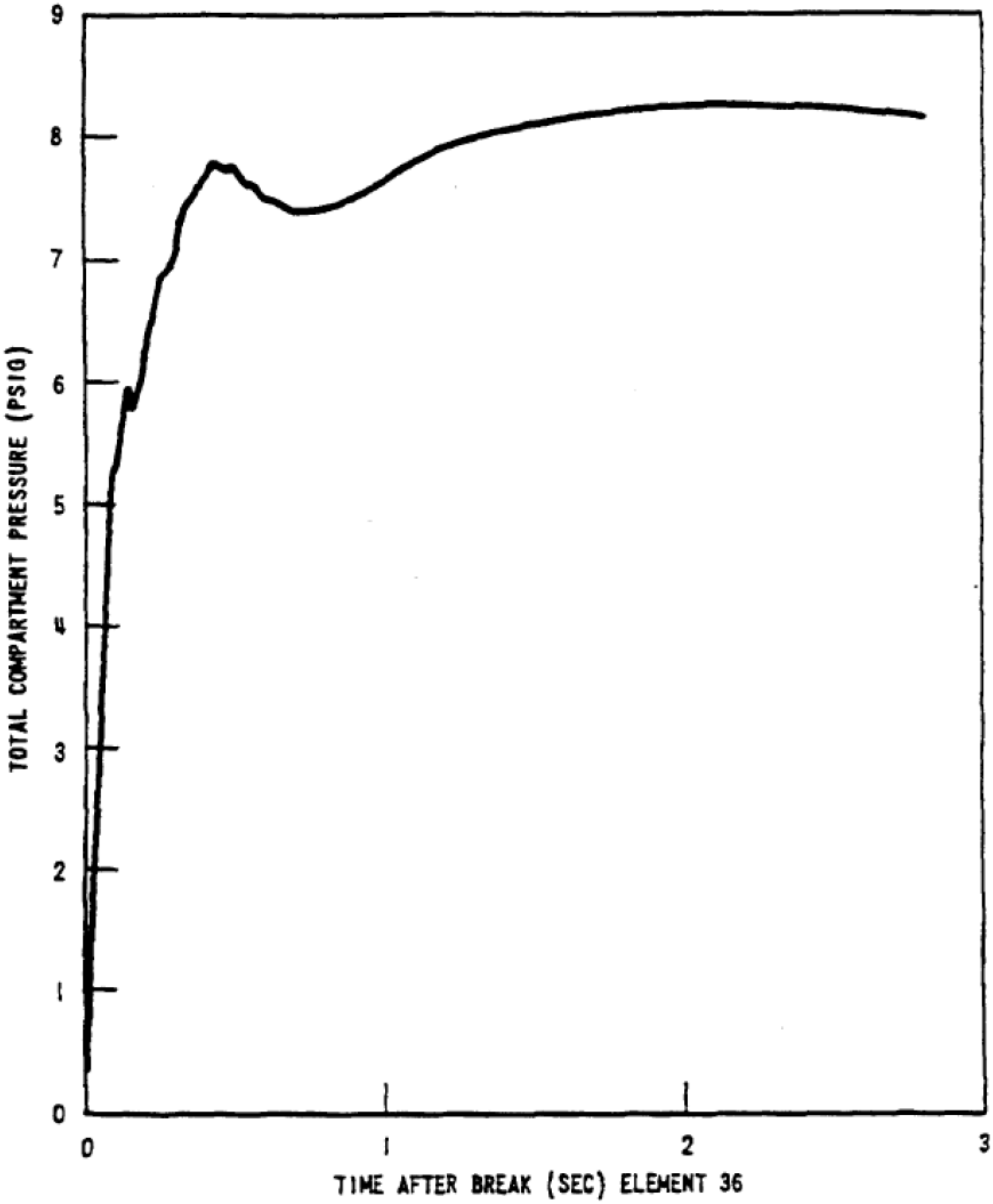
Figure 6.2.1-56



WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

127 Square Inch
Cold Leg Break
(Reactor Cavity Analysis)

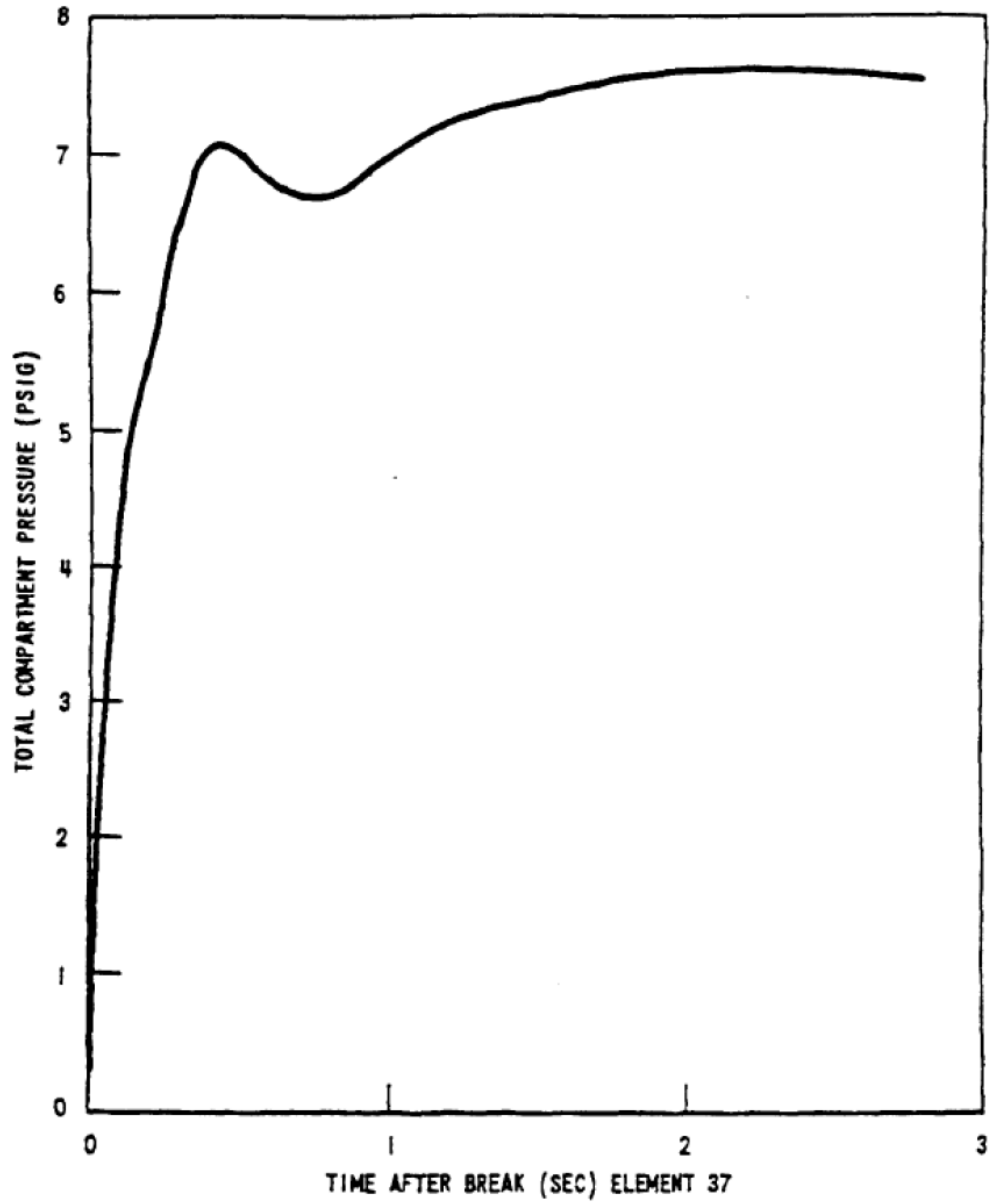
Figure 6.2.1-57



WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

127 Square Inch
Cold Leg Break
(Reactor Cavity Analysis)

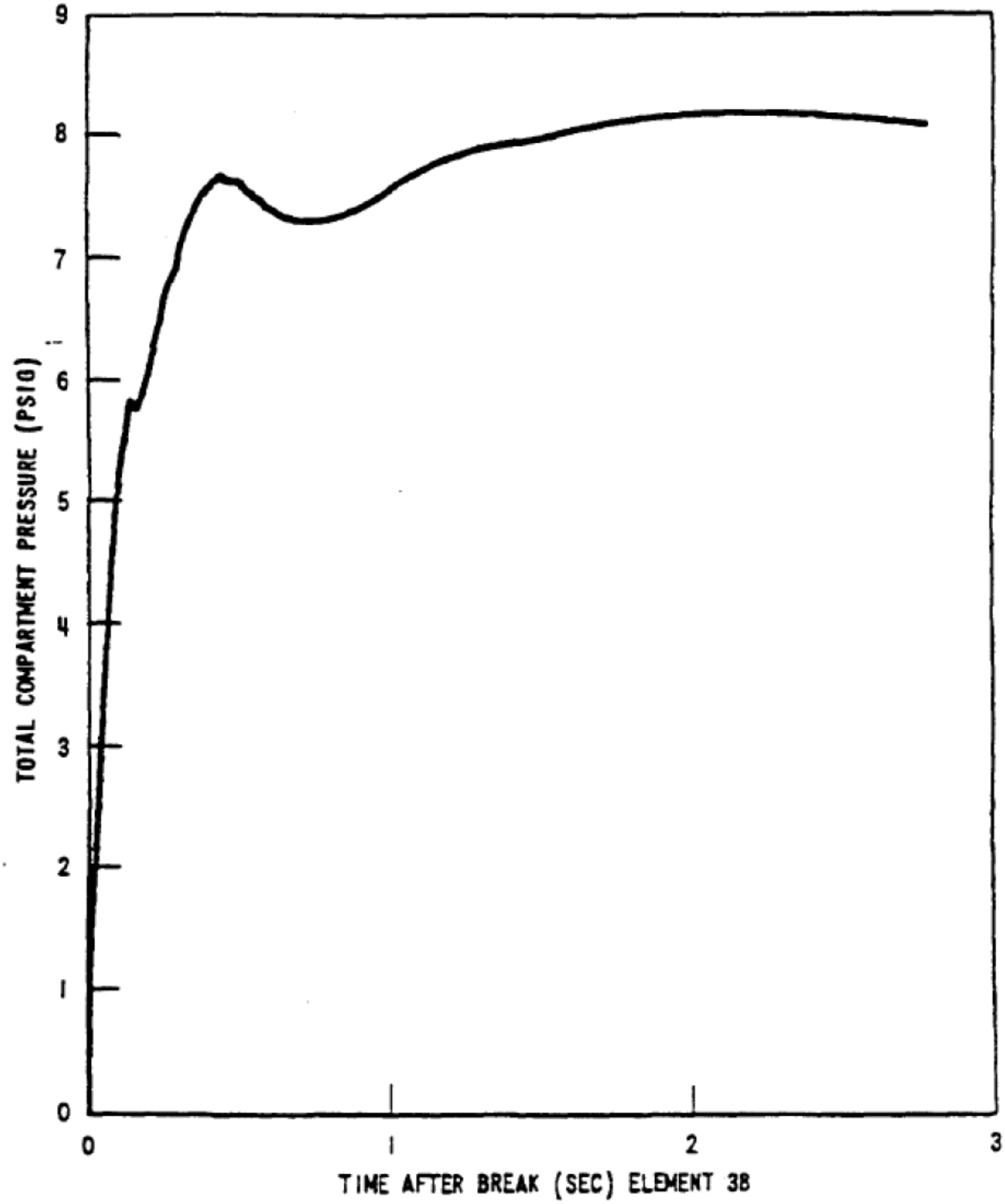
Figure 6.2.1-58



WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

127 Square Inch
Cold Leg Break
(Reactor Cavity Analysis)

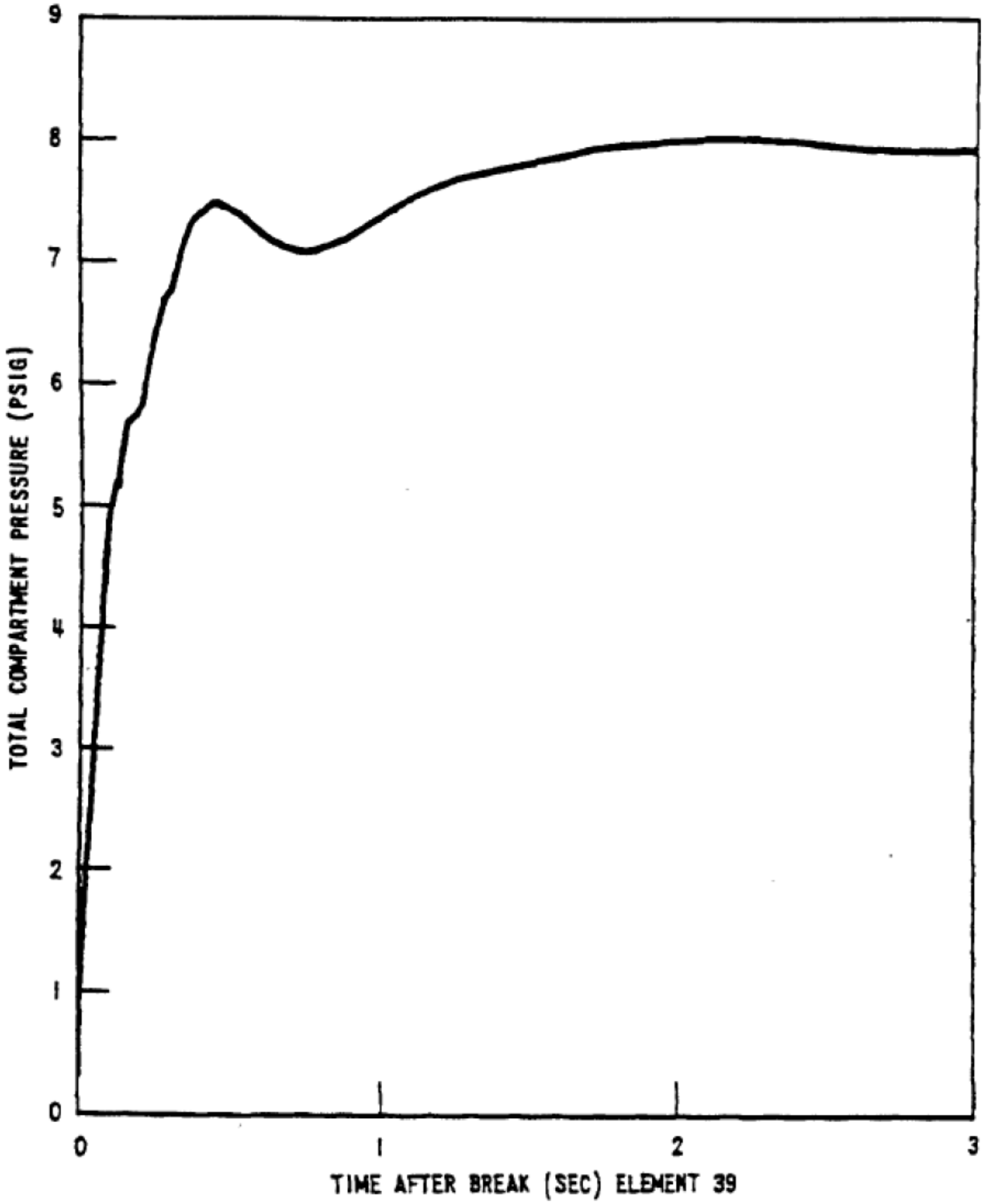
Figure 6.2.1-59



WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

127 Square Inch
Cold Leg Break
(Reactor Cavity Analysis)

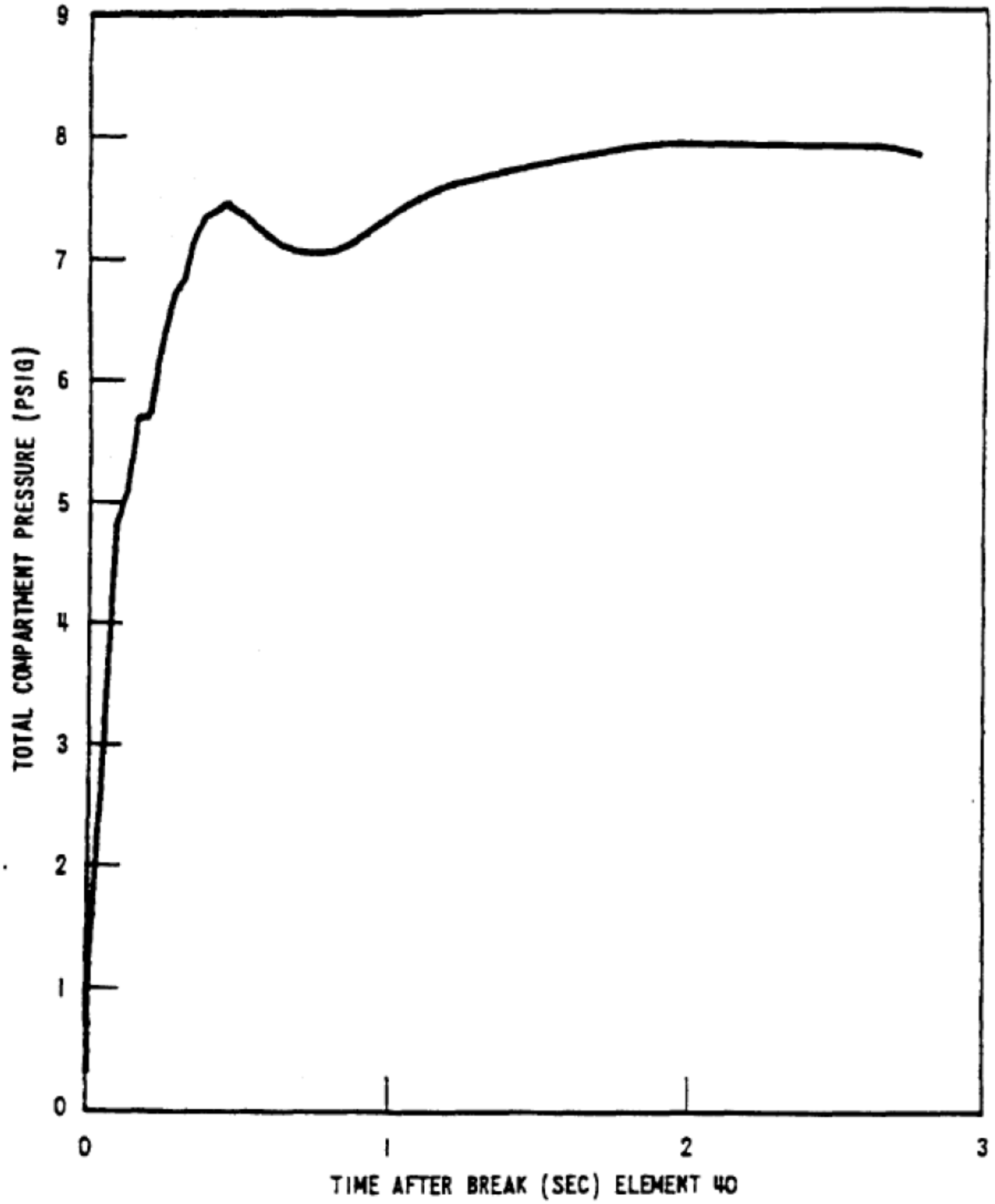
Figure 6.2.1-60



WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

127 Square Inch
Cold Leg Break
(Reactor Cavity Analysis)

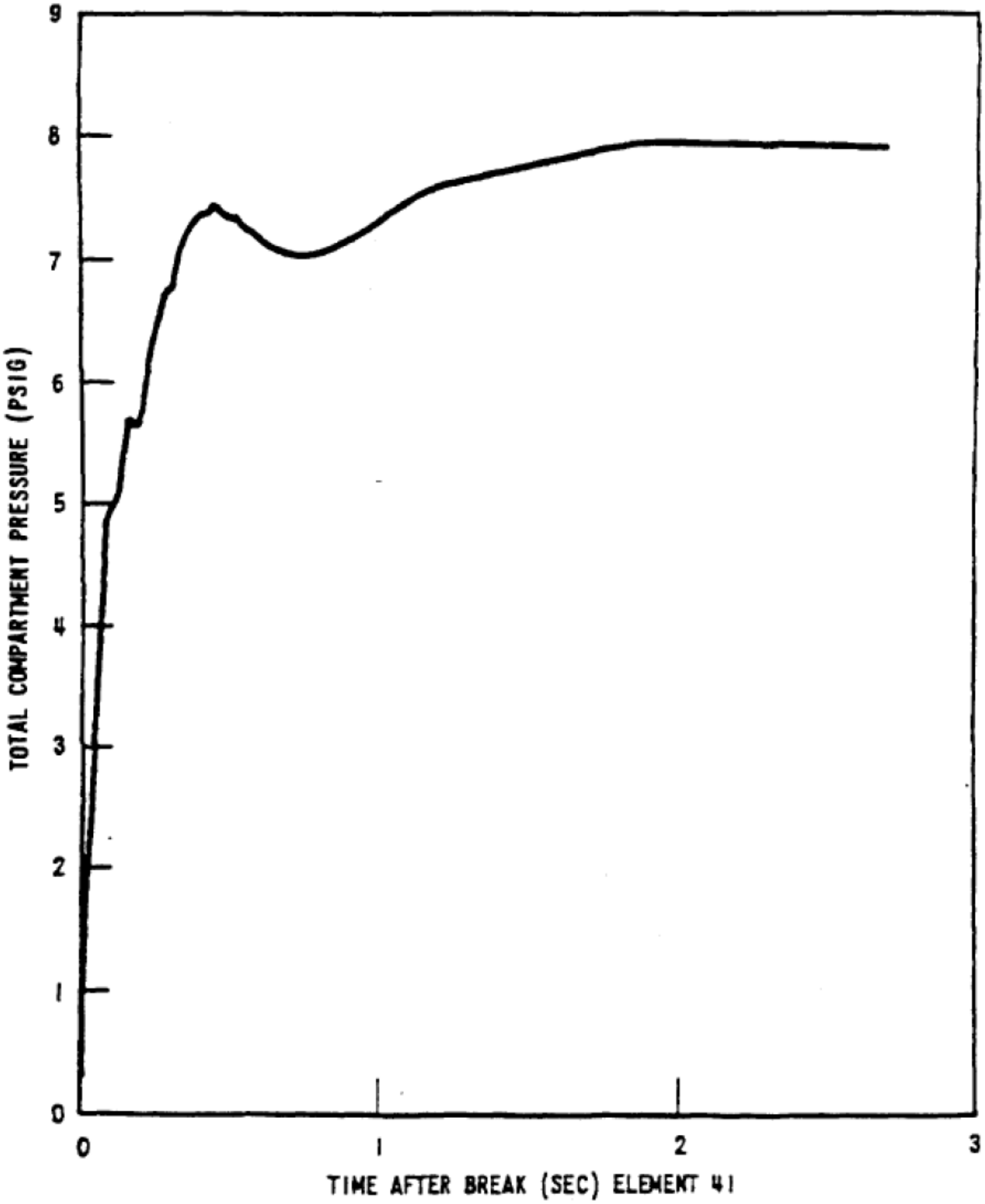
Figure 6.2.1-61



WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

127 Square Inch
Cold Leg Break
(Reactor Cavity Analysis)

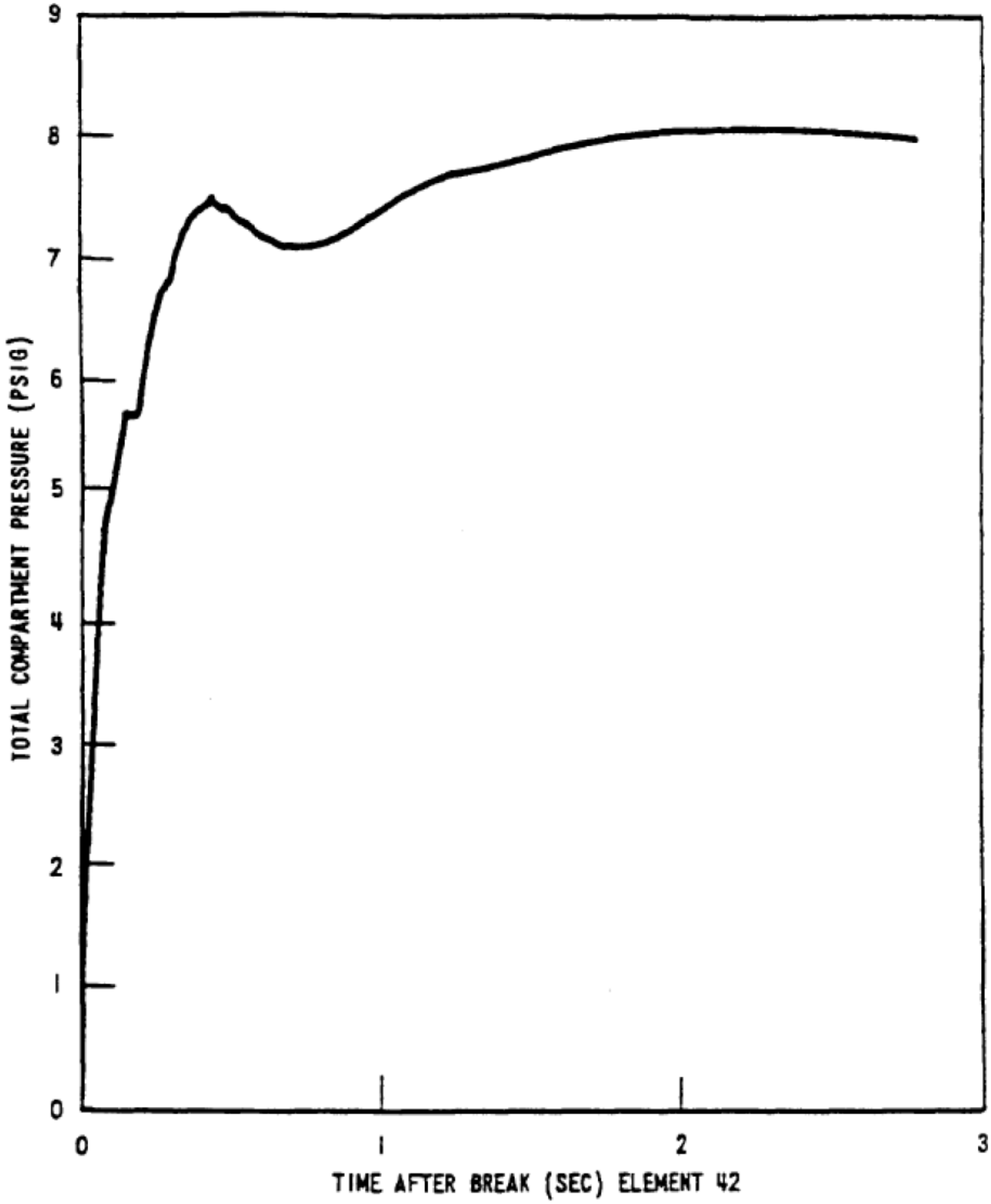
Figure 6.2.1-62



WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

127 Square Inch
Cold Leg Break
(Reactor Cavity Analysis)

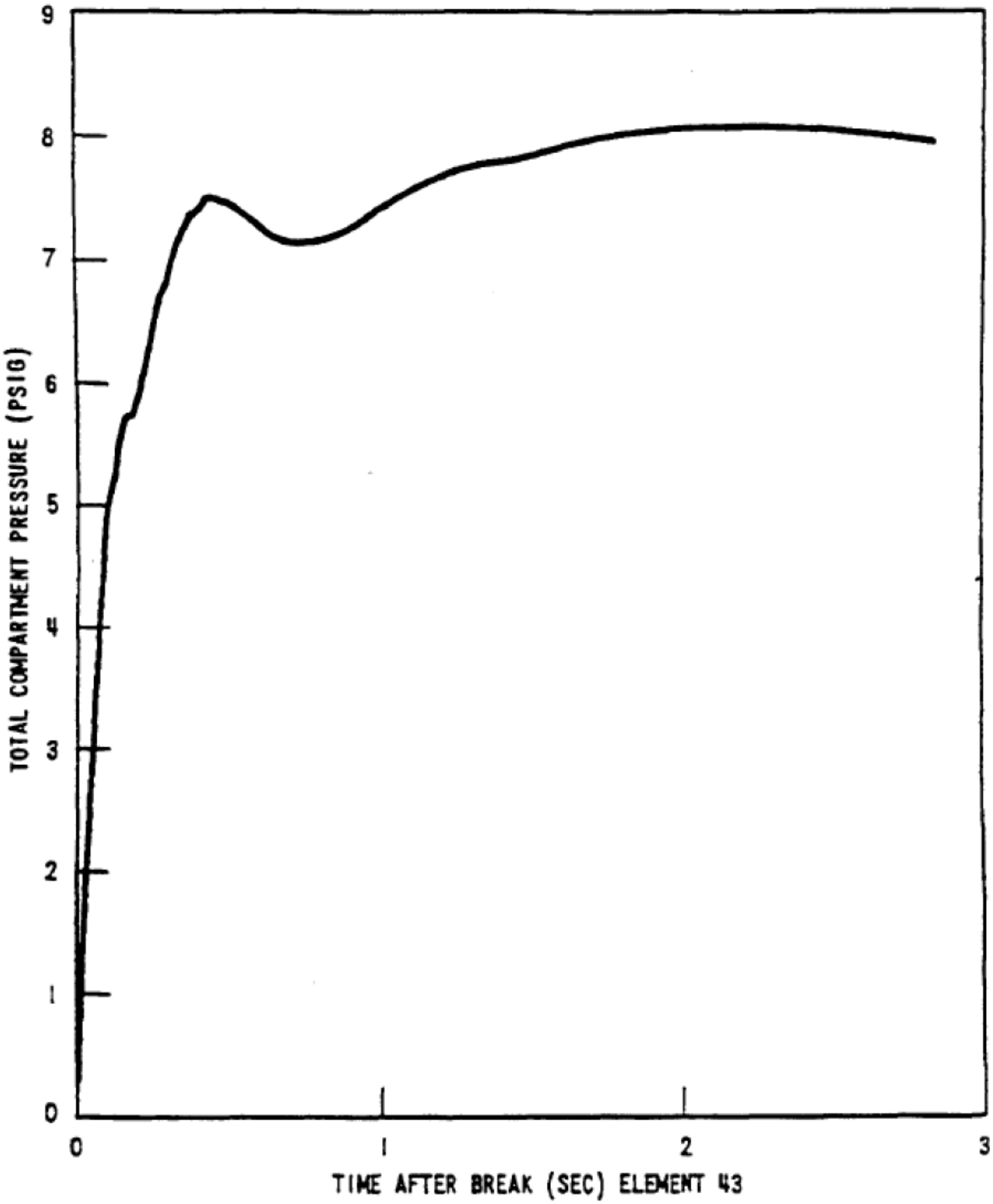
Figure 6.2.1-63



WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

127 Square Inch
Cold Leg Break
(Reactor Cavity Analysis)

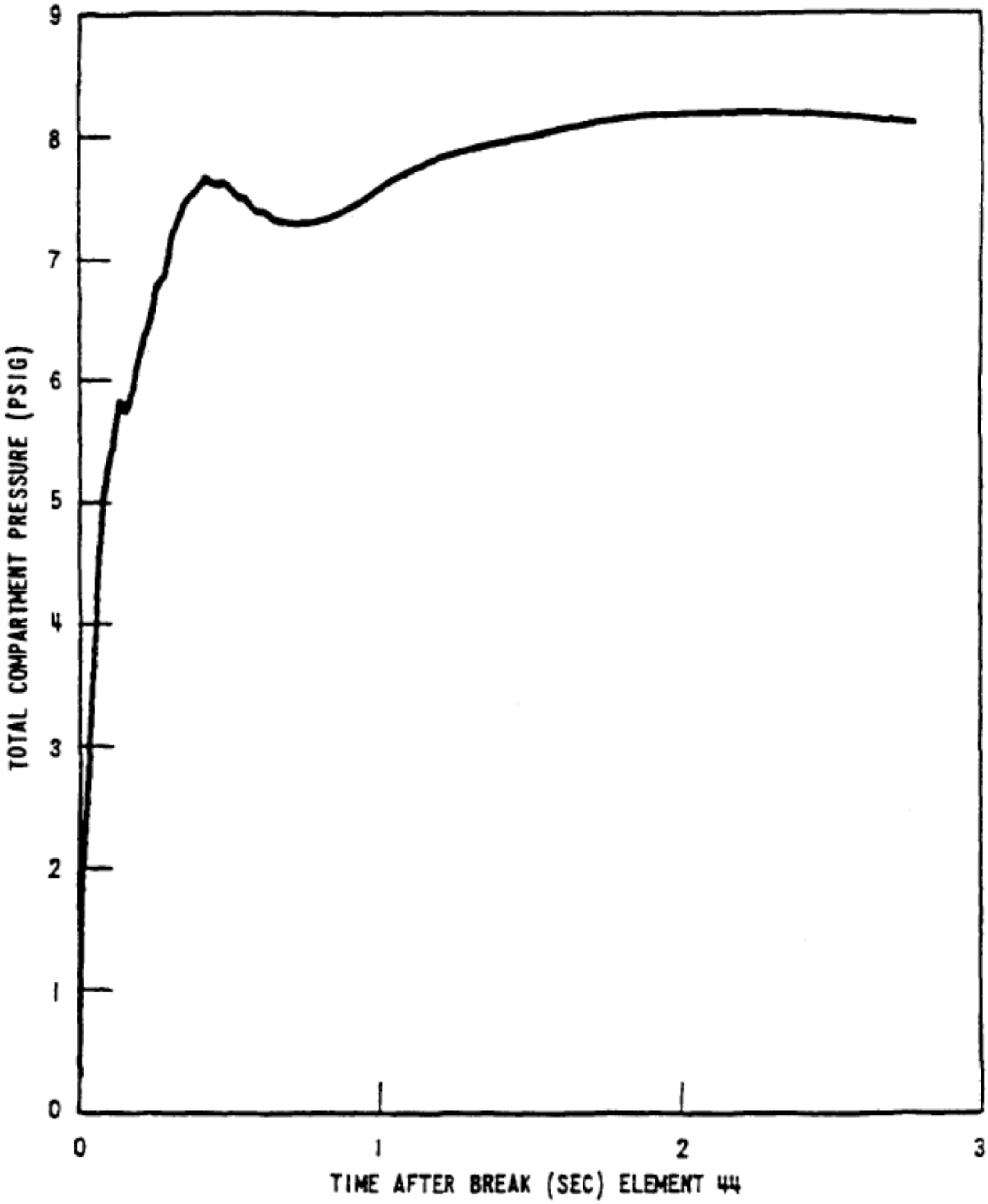
Figure 6.2.1-64



WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

127 Square Inch
Cold Leg Break
(Reactor Cavity Analysis)

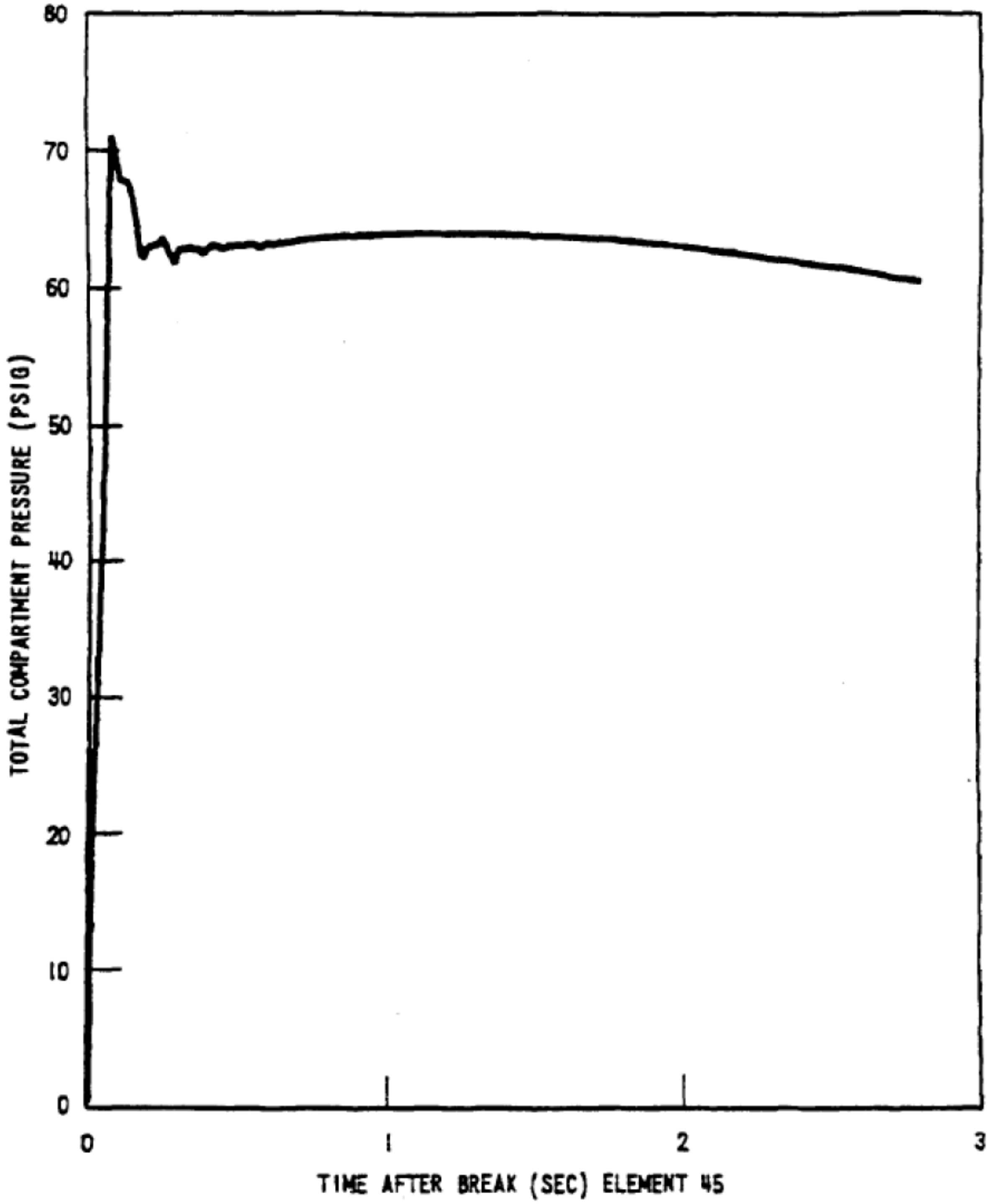
Figure 6.2.1-65



WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

127 Square Inch
Cold Leg Break
(Reactor Cavity Analysis)

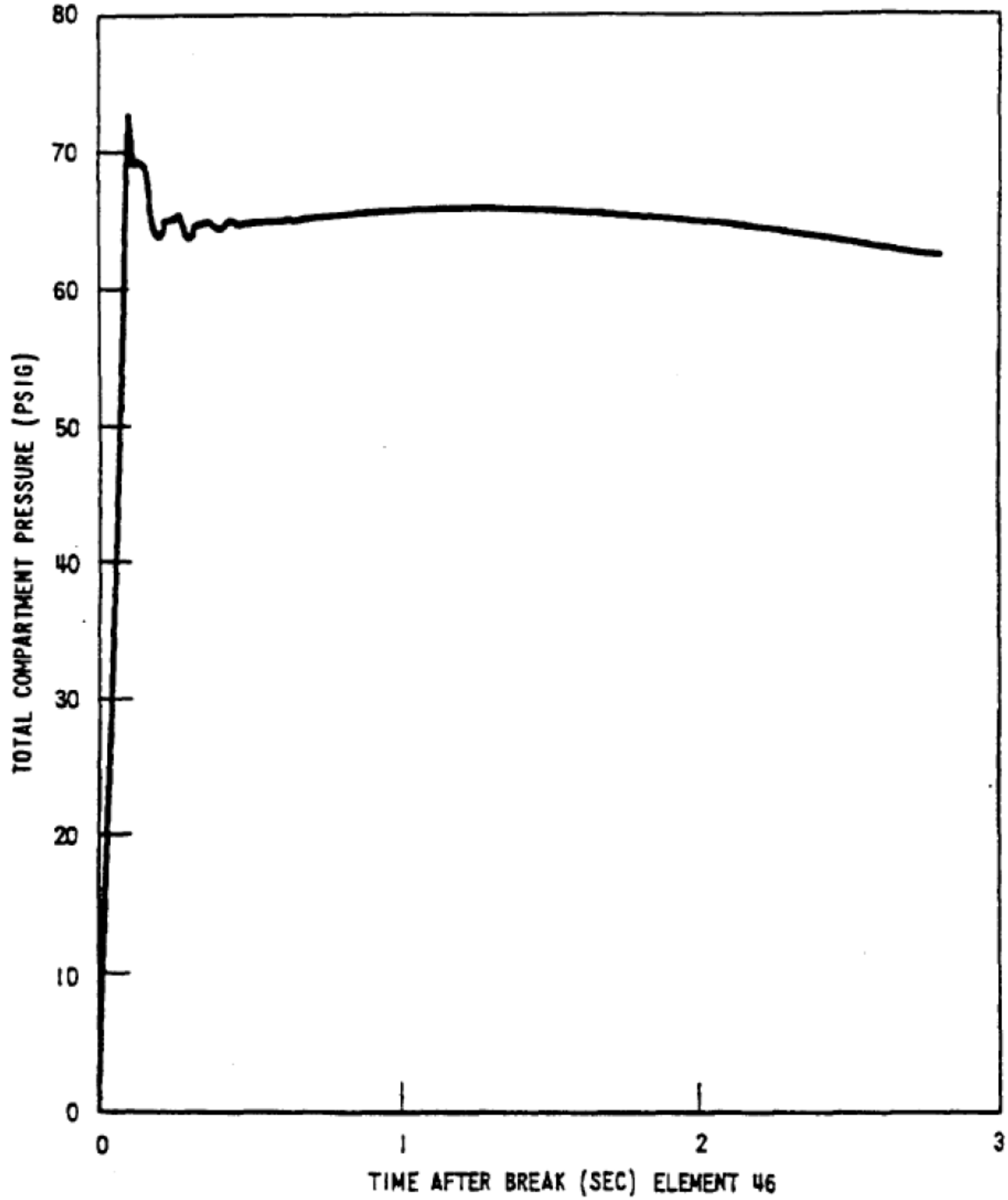
Figure 6.2.1-66



WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

127 Square Inch
Cold Leg Break
(Reactor Cavity Analysis)

Figure 6.2.1-67

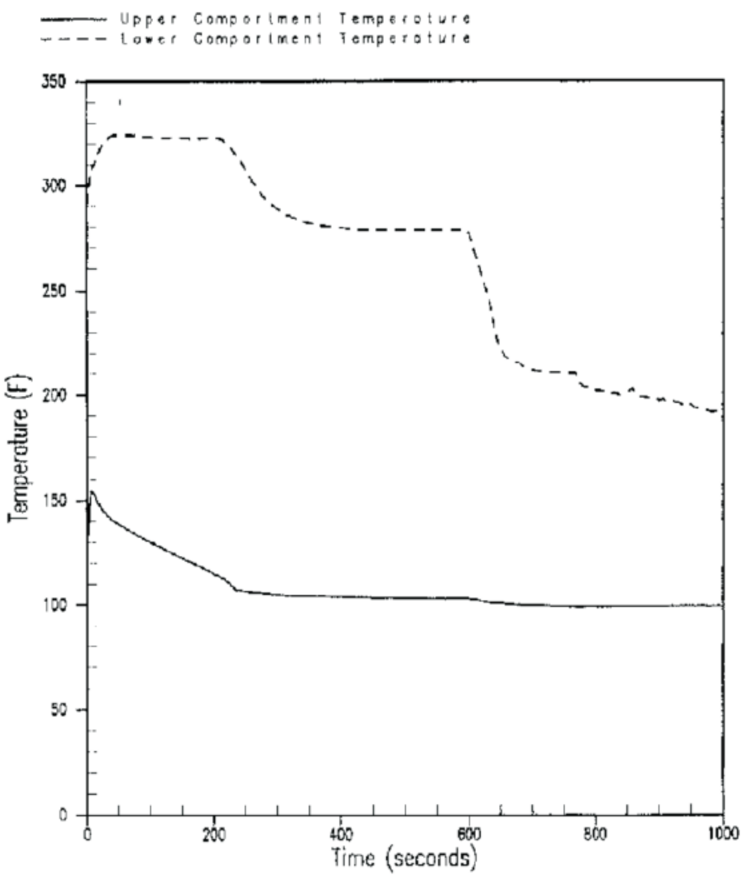


WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

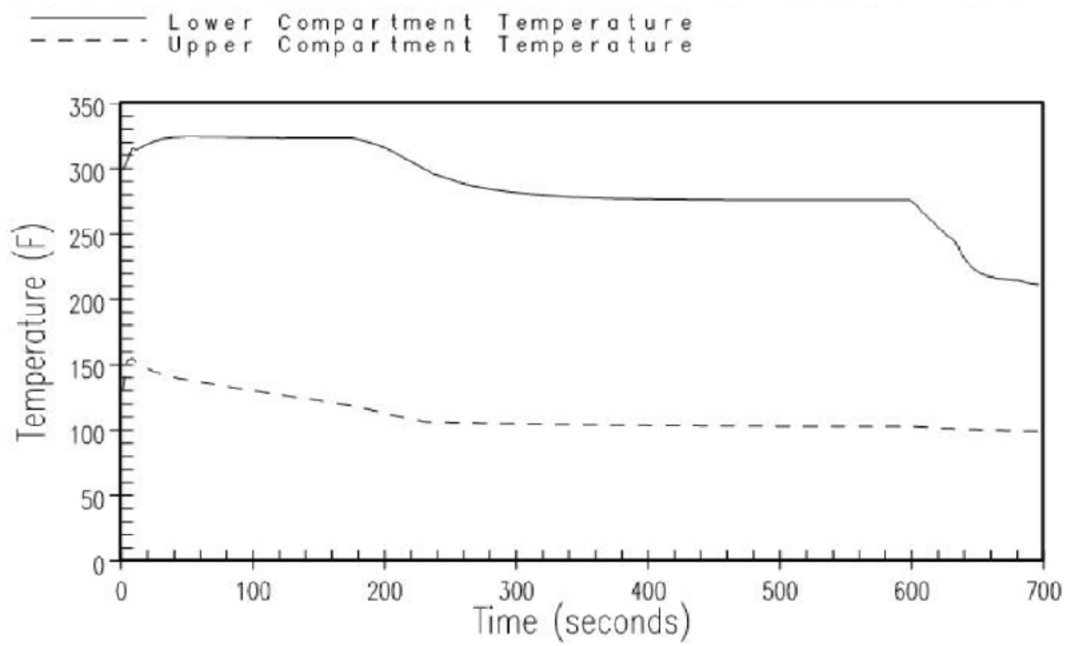
127 Square Inch
Cold Leg Break
(Reactor Cavity Analysis)

Figure 6.2.1-68

Unit 1 – 100% Power, 1.4FT^2 DER, MFIV Single Failure



Unit 2 – 100.6% Power, 1.4ft^2/Loop, FIV Failure

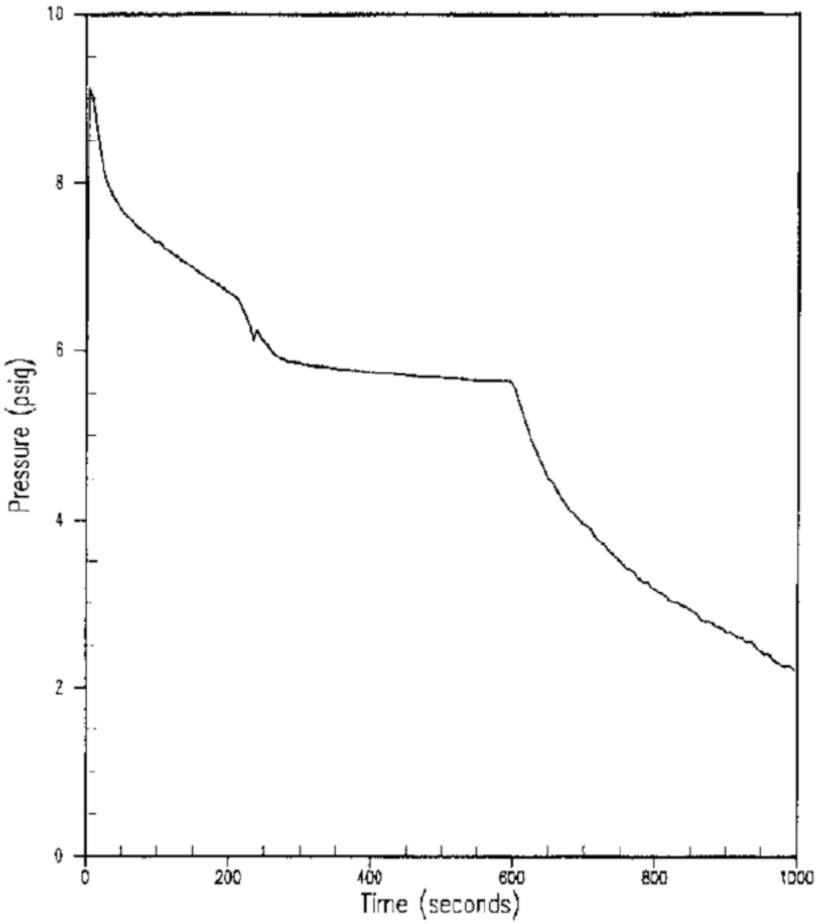


WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

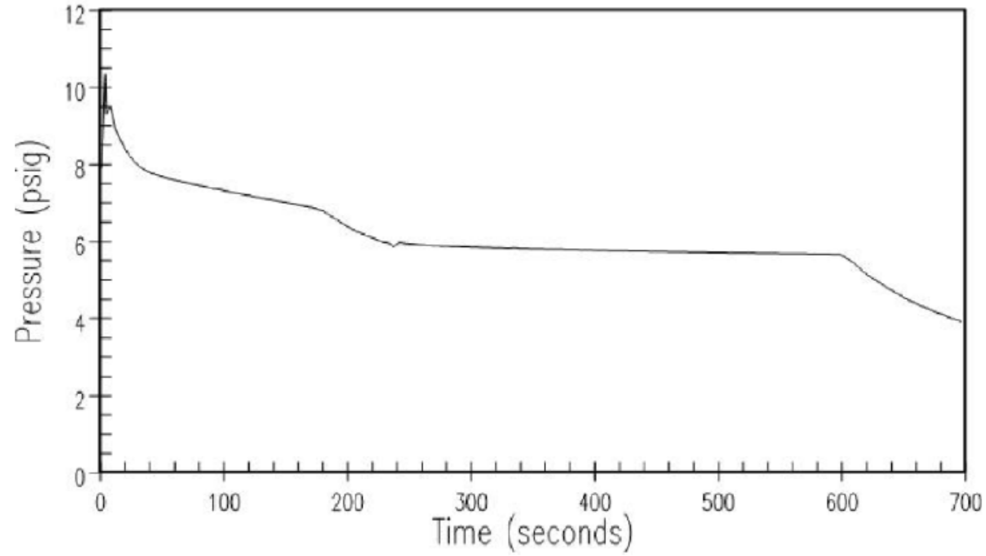
Compartment Temperature Response

FIGURE 6.2.1-69

Unit 1 – 100% Power, 1.4FT^2 DER, MFIV Single Failure



Unit 2 – Lower Compartment Pressure 100.6% Power, 1.4ft^2/Loop, FIV Failure

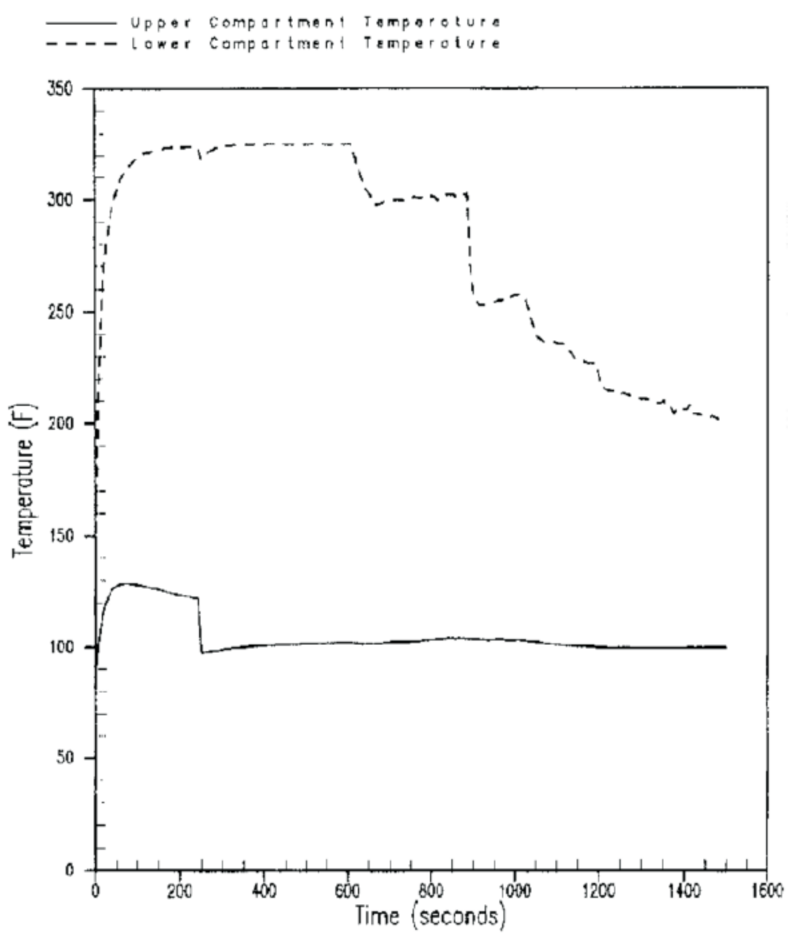


WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

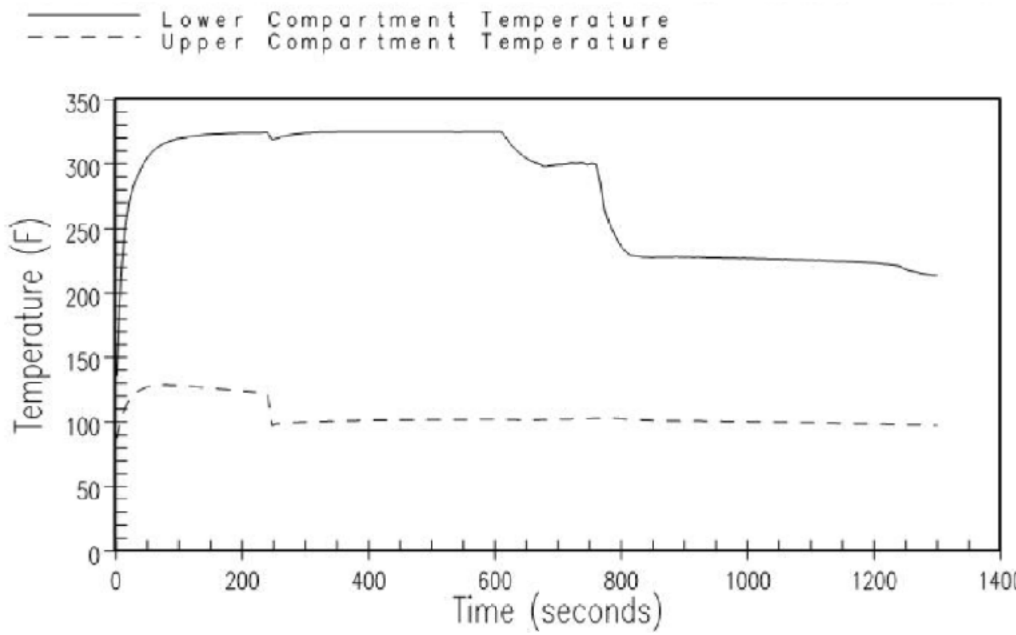
Compartment Pressure Response

FIGURE 6.2.1-70

Unit 1 – 30% Power, 0.350 Ff² Split Break, AFW Run Out Protection Single Failure



Unit 2 – 30% Power, 0.350 Ff² Split Break, AFW Run Out Protection Single Failure

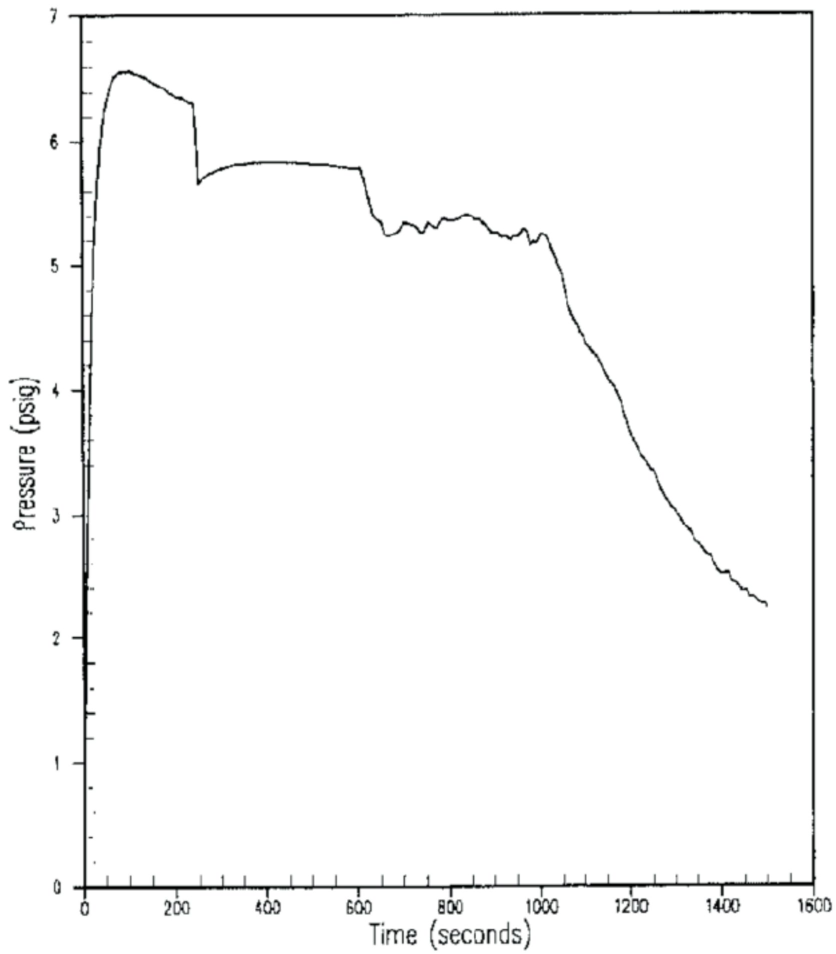


WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

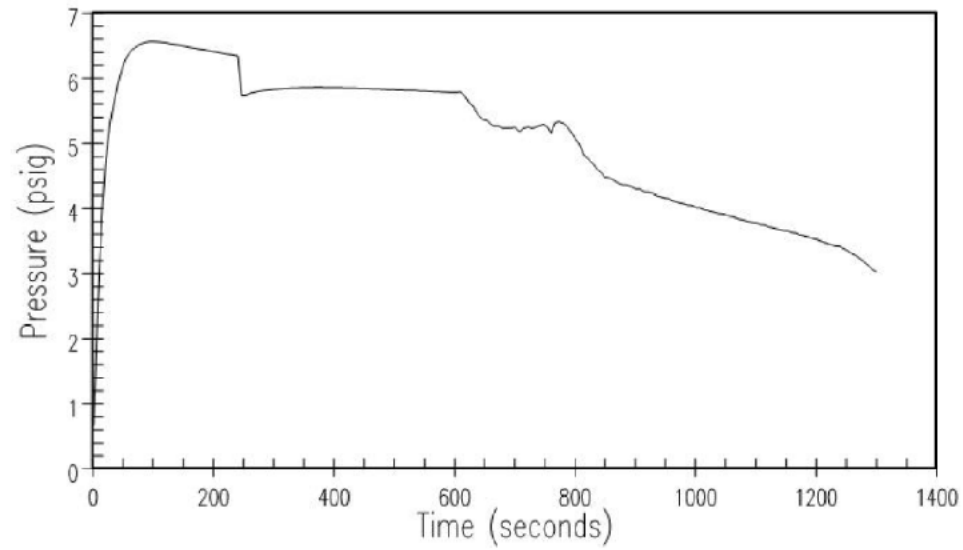
Compartment Temperature Response

FIGURE 6.2.1-71

Unit 1 – 30% Power, 0.350 Ff^2 Split Break, AFW Run Out Protection Single Failure



Unit 2 – 30% Power, 0.35 Ff^2 Split Break, AFW Run Out Protection Failure

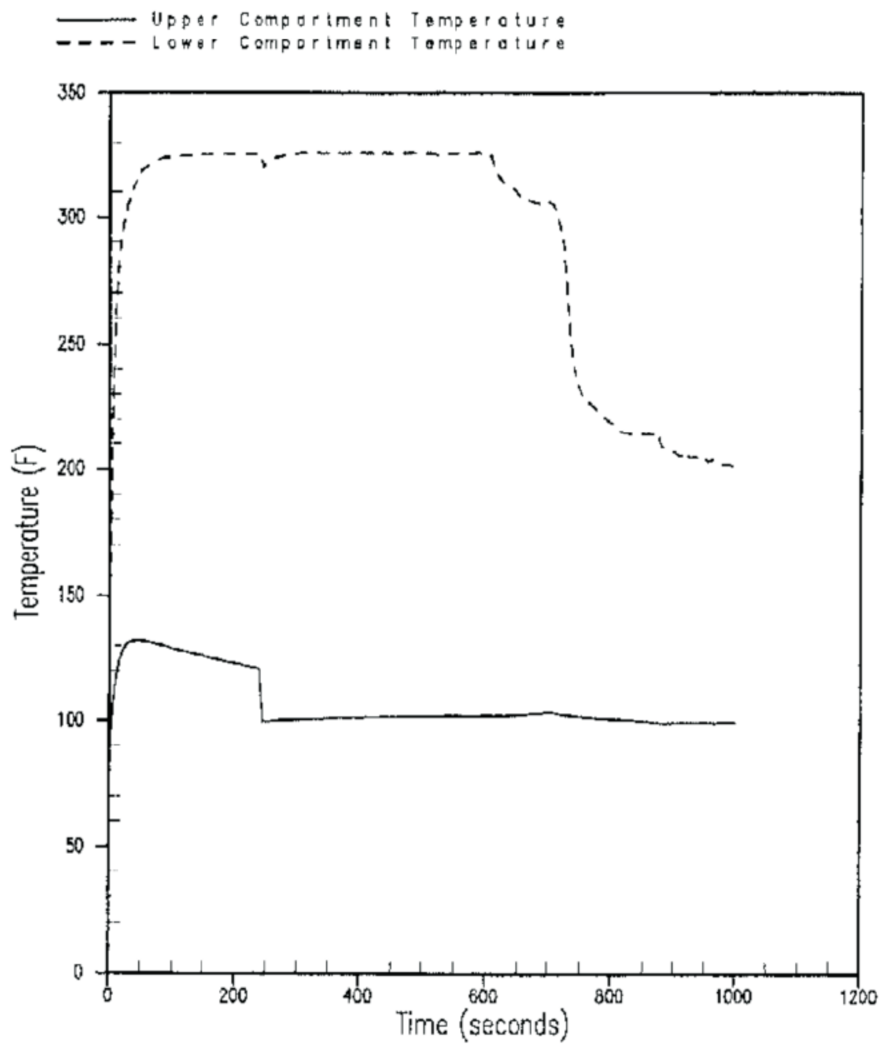


WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

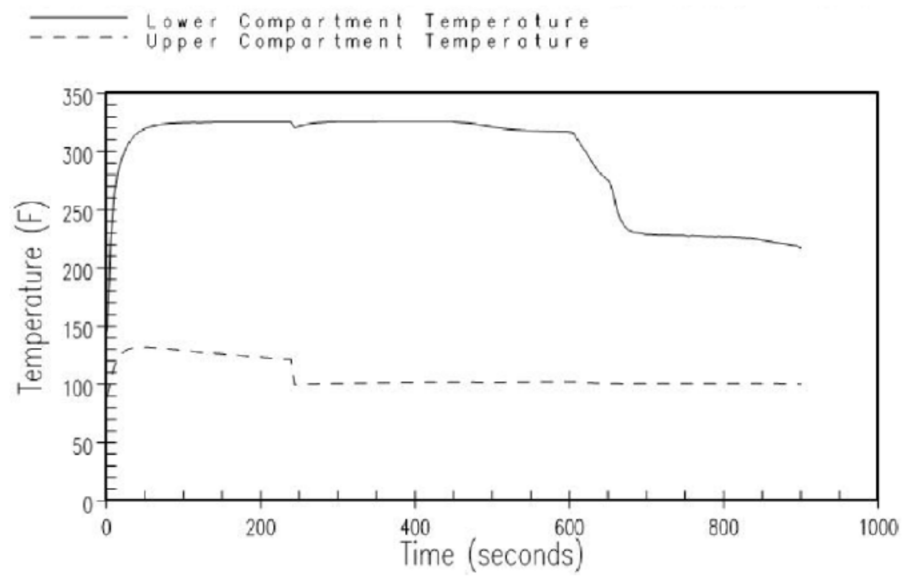
Compartment Pressure
Response

FIGURE 6.2.1-72

Unit 1 – 30% Power, 0.600 Ff² Split Break, AFW Run Out Protection Single Failure

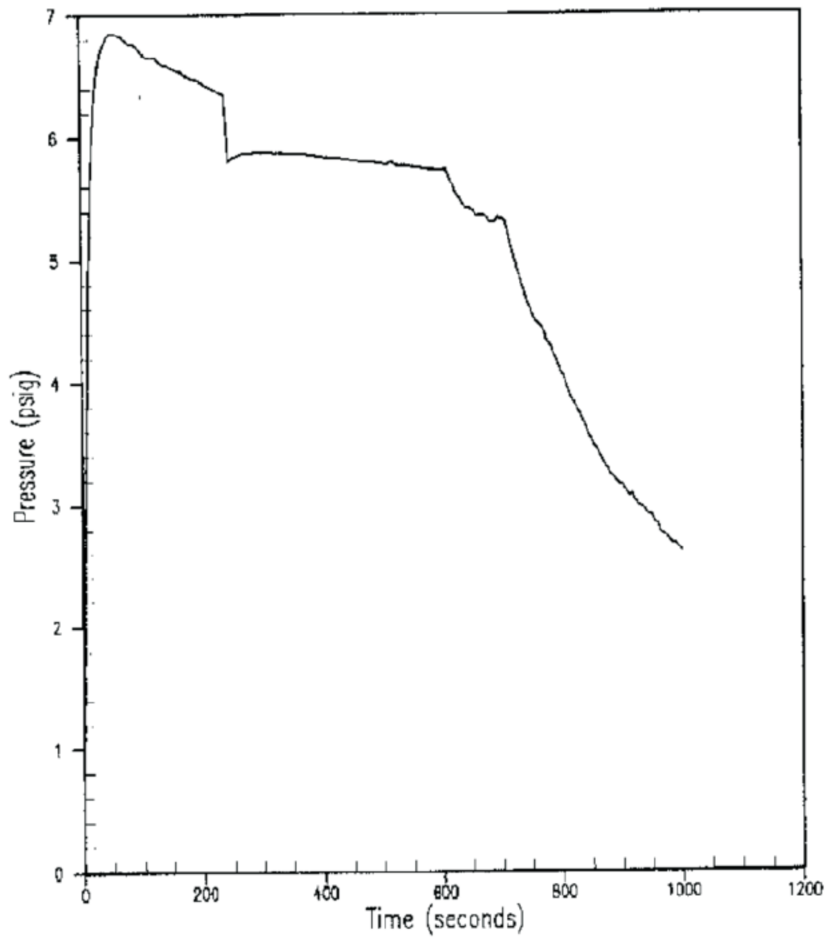


Unit 2 – 30% Power, 0.6 Ff² Split Break, AFW Run Out Protection Failure

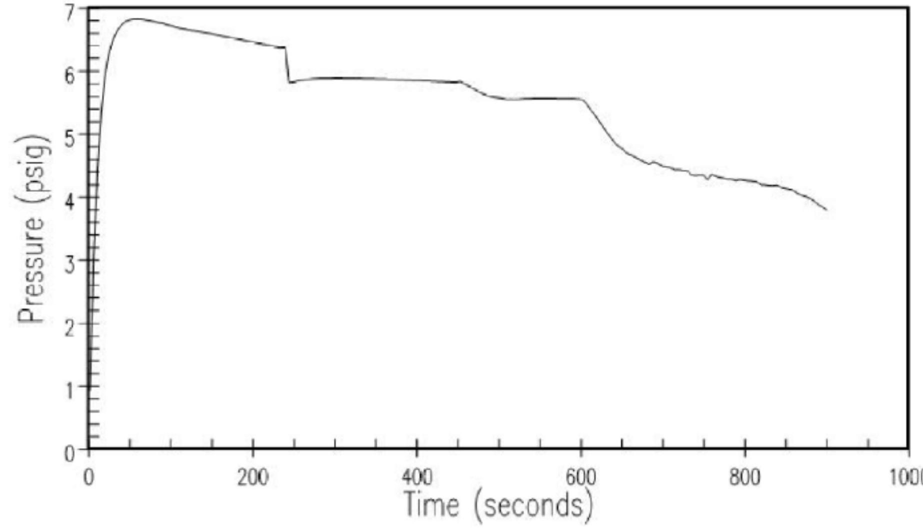


WATTS BAR NUCLEAR PLANT FINAL SAFETY ANALYSIS REPORT
Compartment Temperature Response FIGURE 6.2.1-73

Unit 1 – 30% Power, 0.600 Ff^2 Split Break, AFW Run Out Protection Single Failure



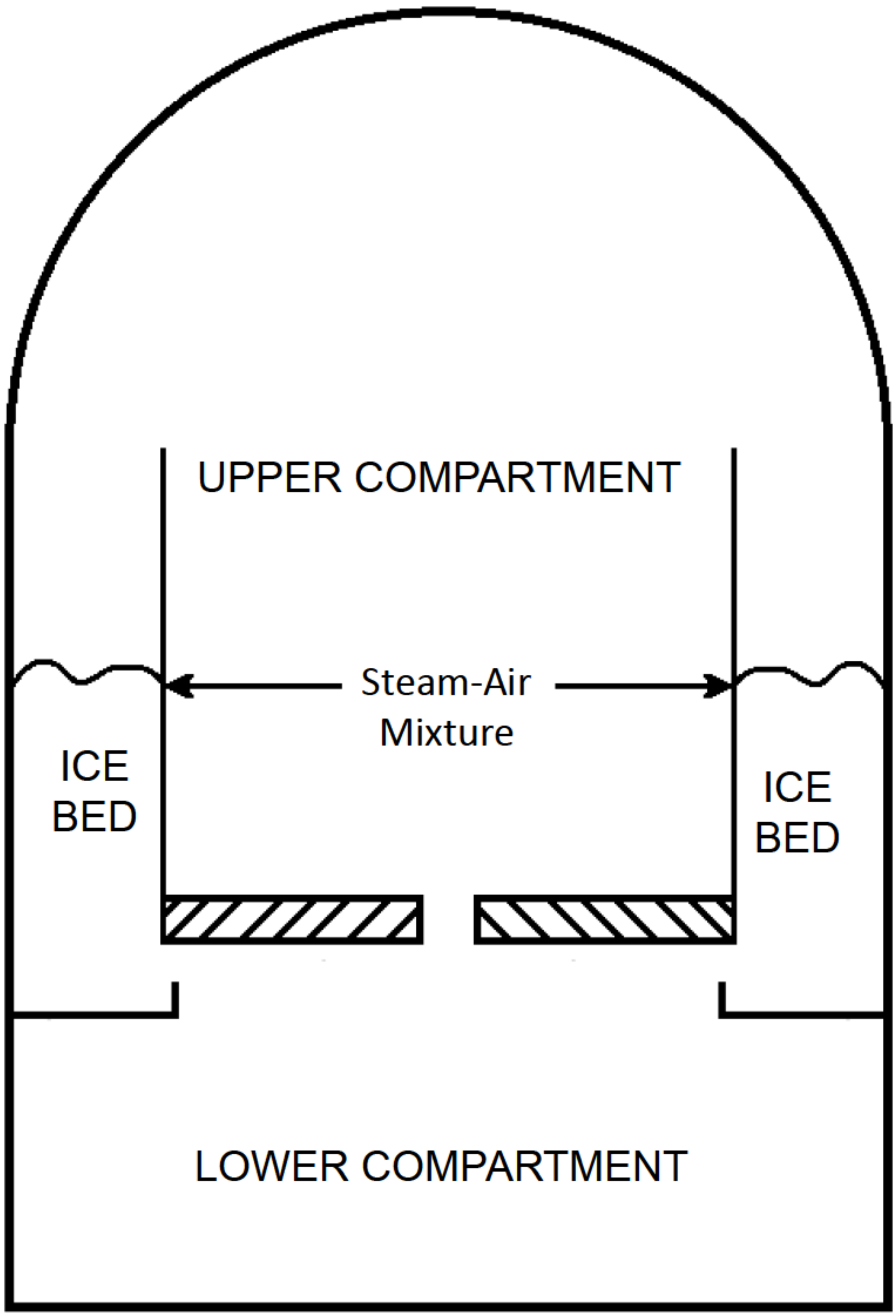
Unit 2 – 30% Power, 0.6 Ff^2 Split Break, AFW Run Out Protection Failure



WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

Compartment Pressure Response

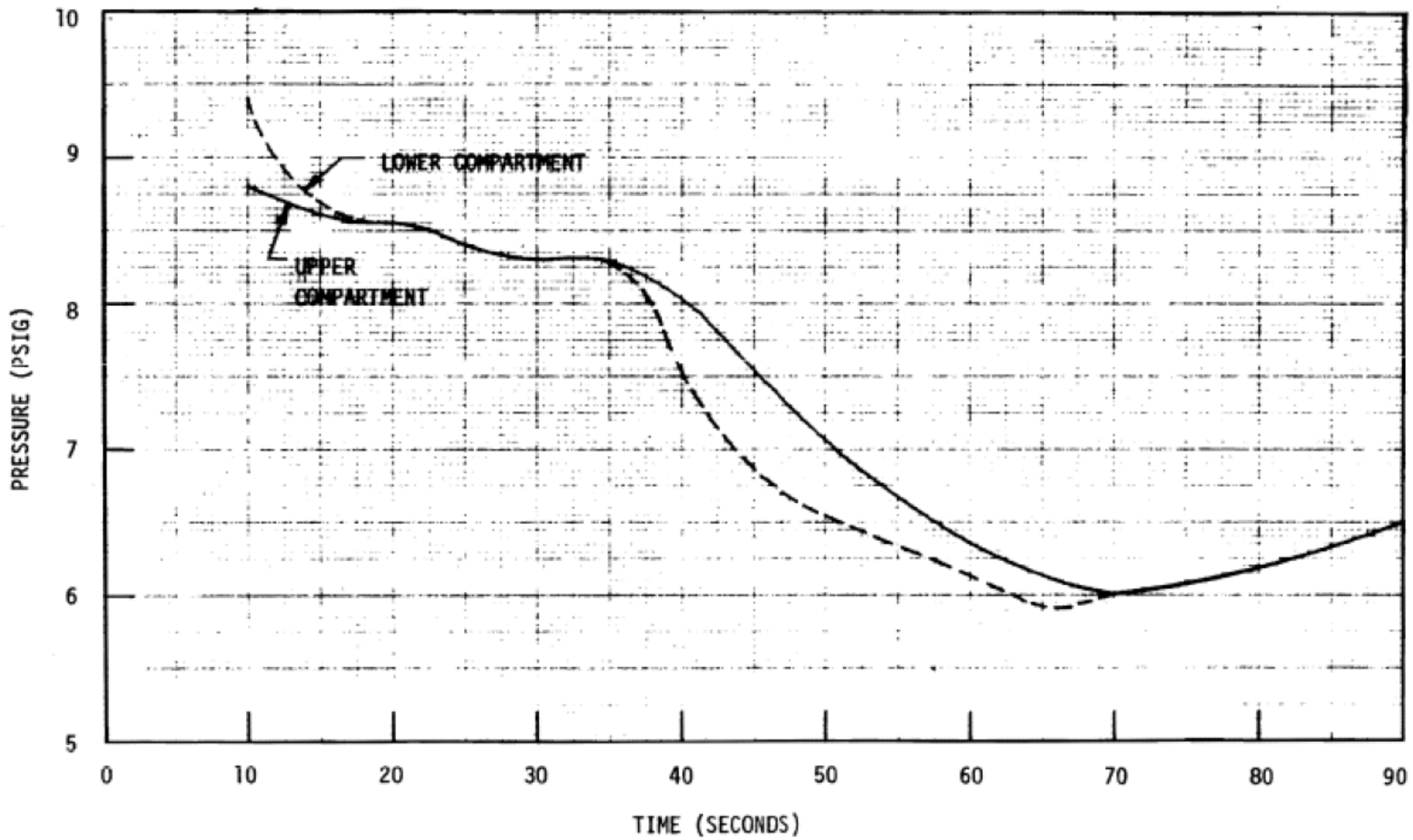
FIGURE 6.2.1-74



WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

Maximum Reverse
Pressure Differential
Model

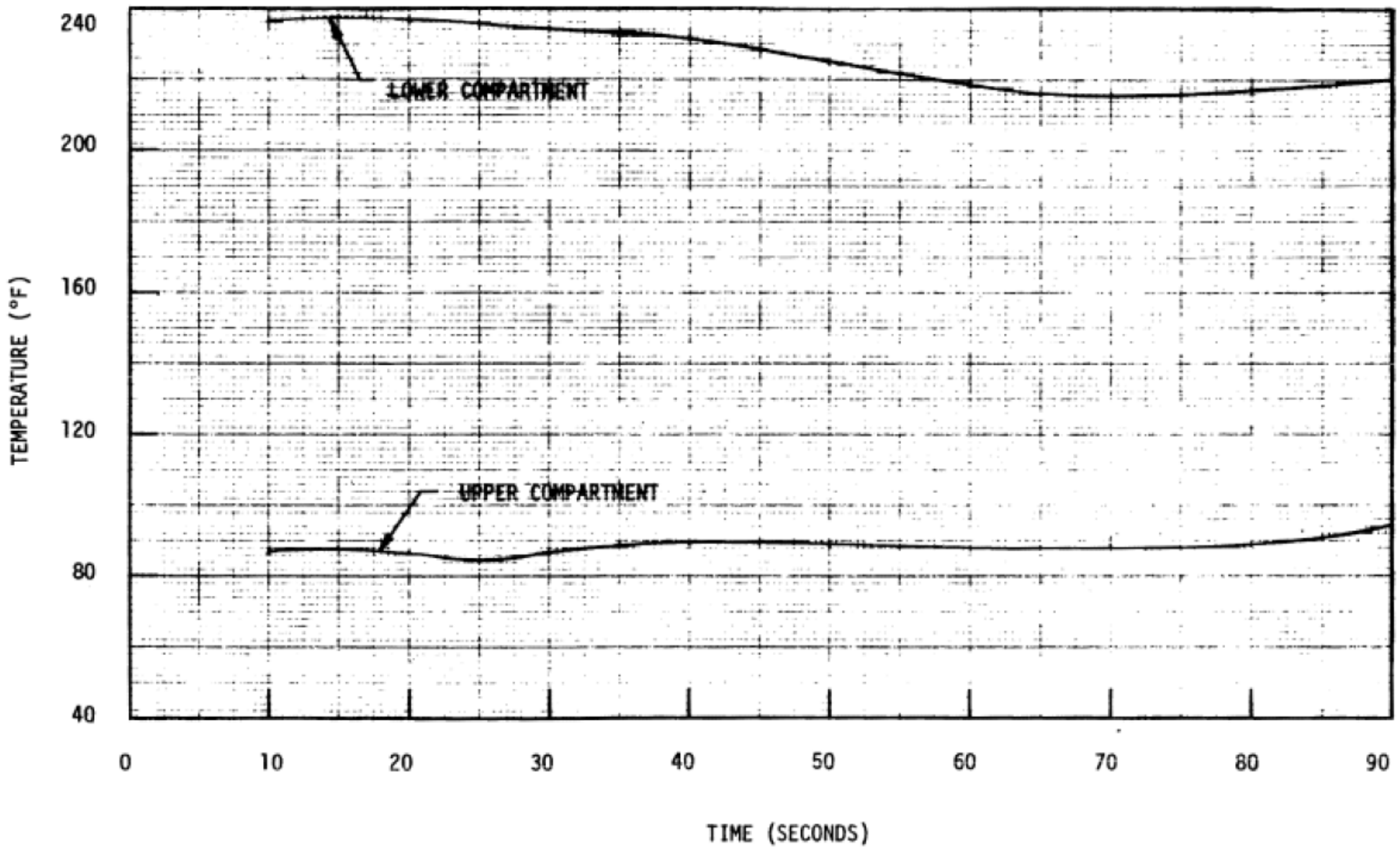
Figure 6.2.1-75



WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

Maximum Reverse Pressure
Differential Upper and Lower
Compartment Pressures

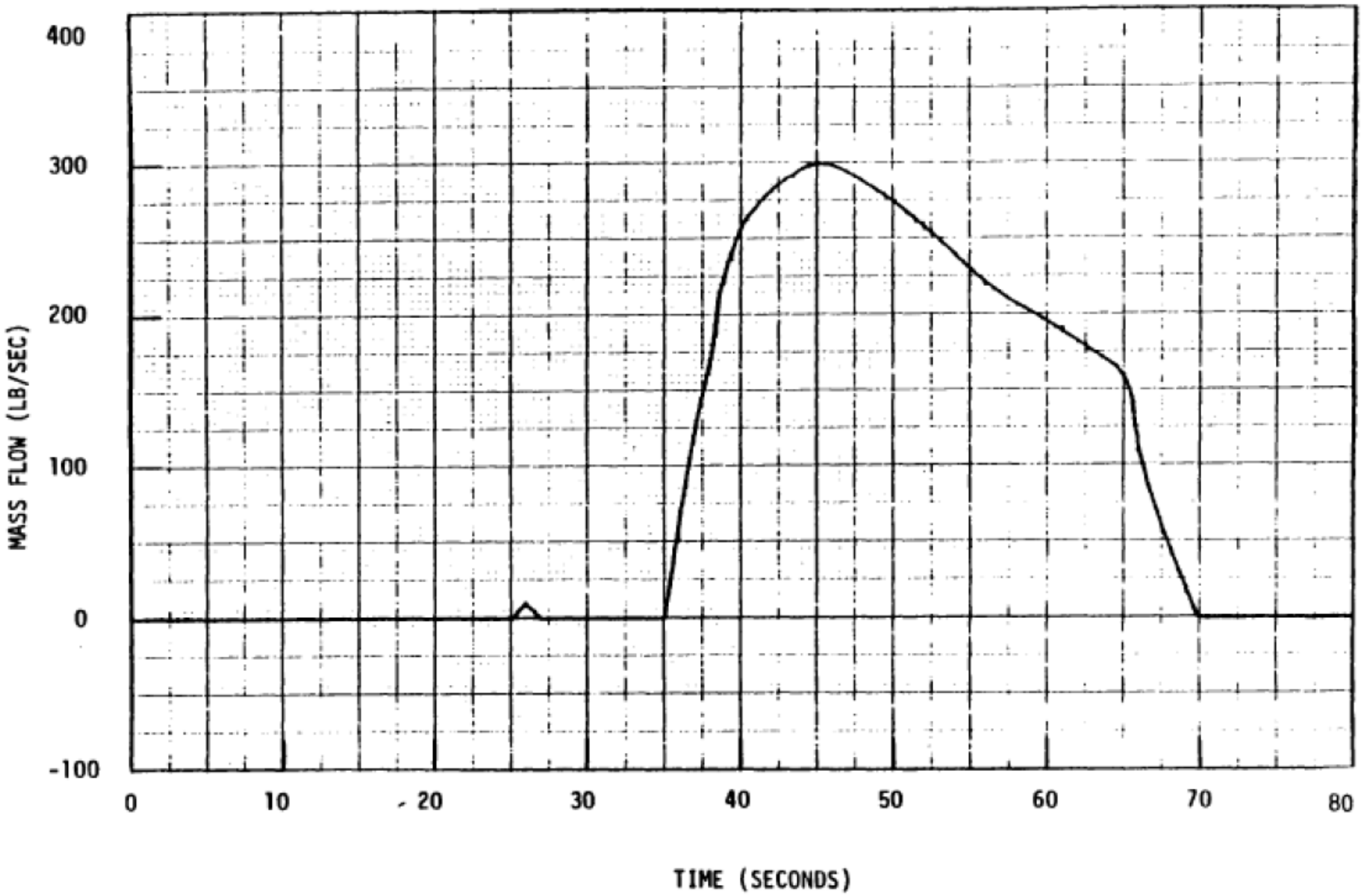
FIGURE 6.2.1-76



WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

Maximum Reverse Pressure Differential Upper and Lower Compartment Temperatures

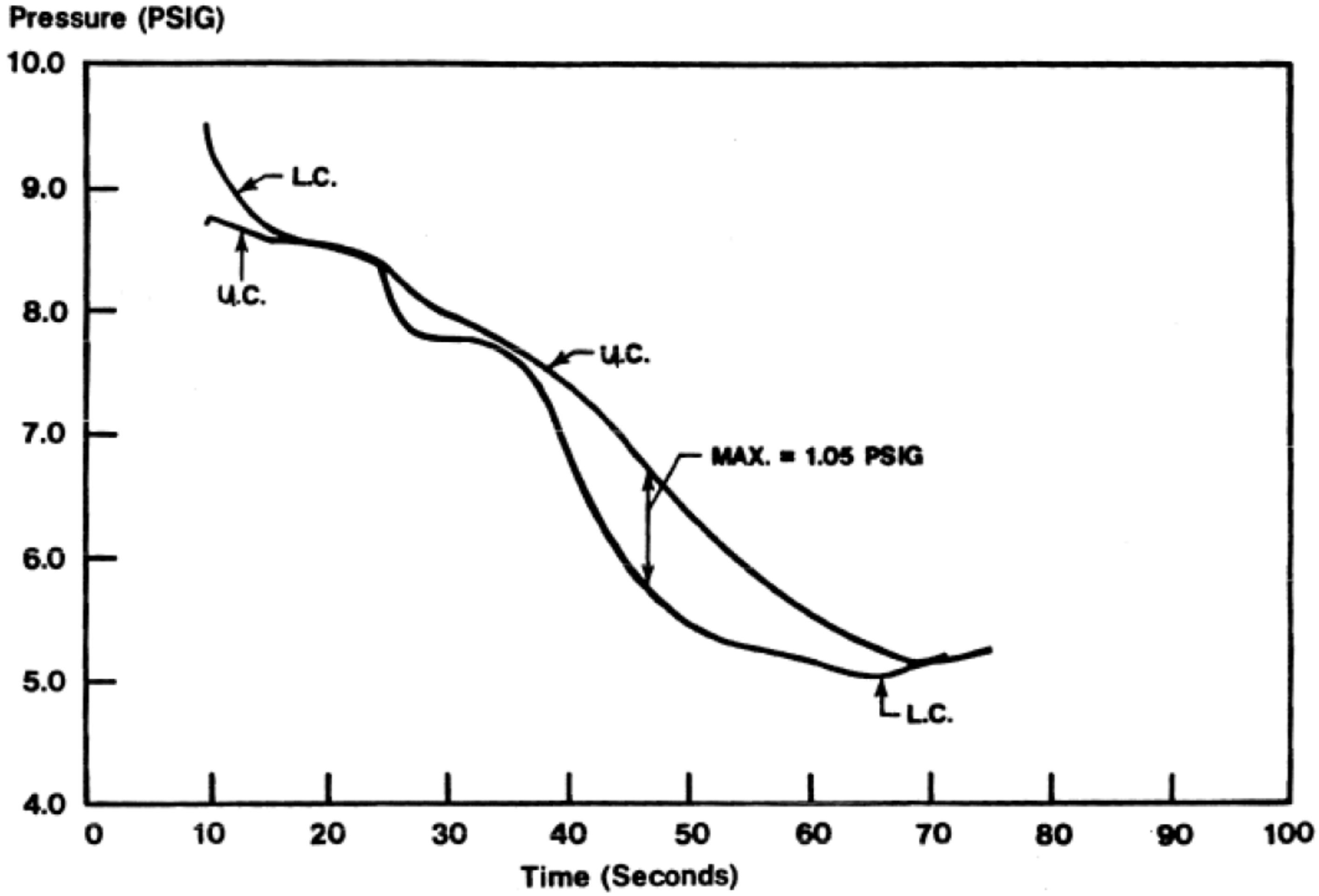
FIGURE 6.2.1-77



WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

Maximum Reverse Pressure
Differential Upper to Lower
Compartment Flowrates

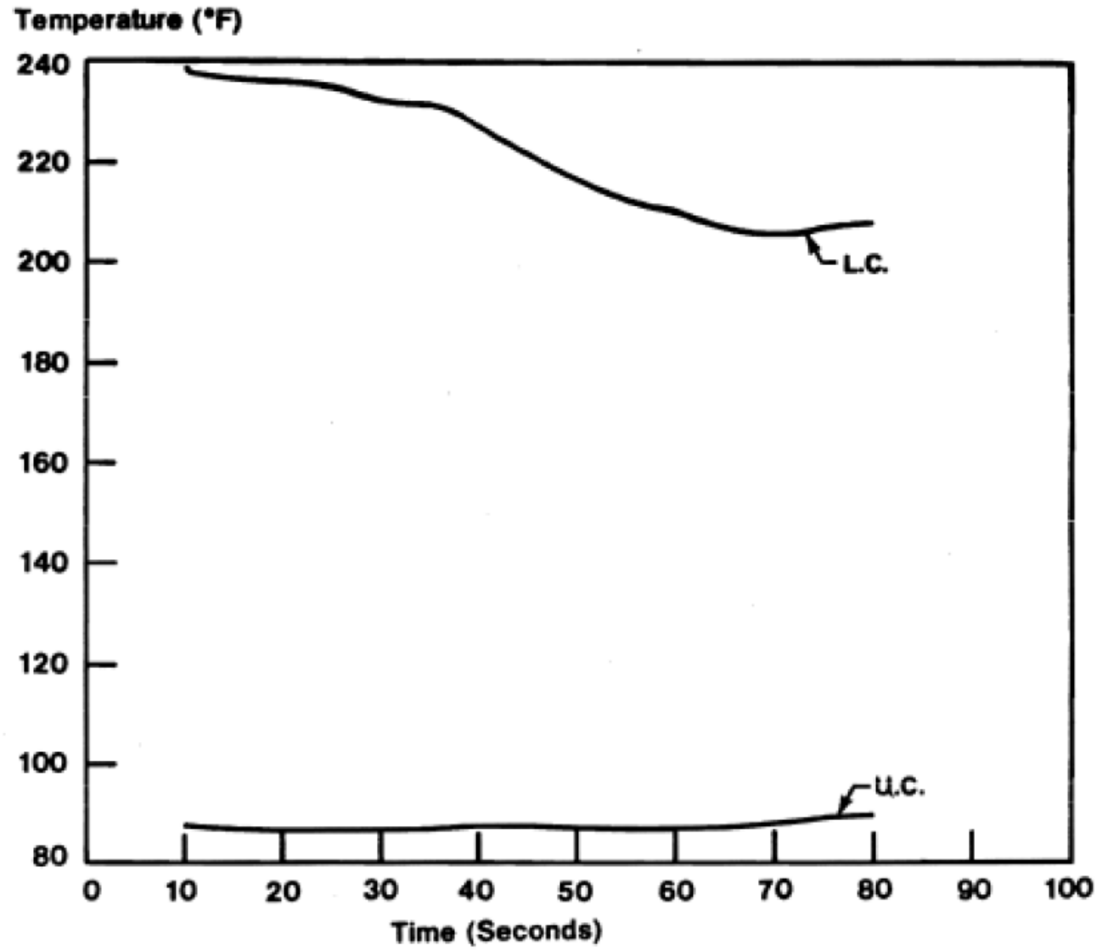
FIGURE 6.2.1-78



WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

Maximum Reverse Pressure Differential Case 6 Upper and Lower Compartment Pressures

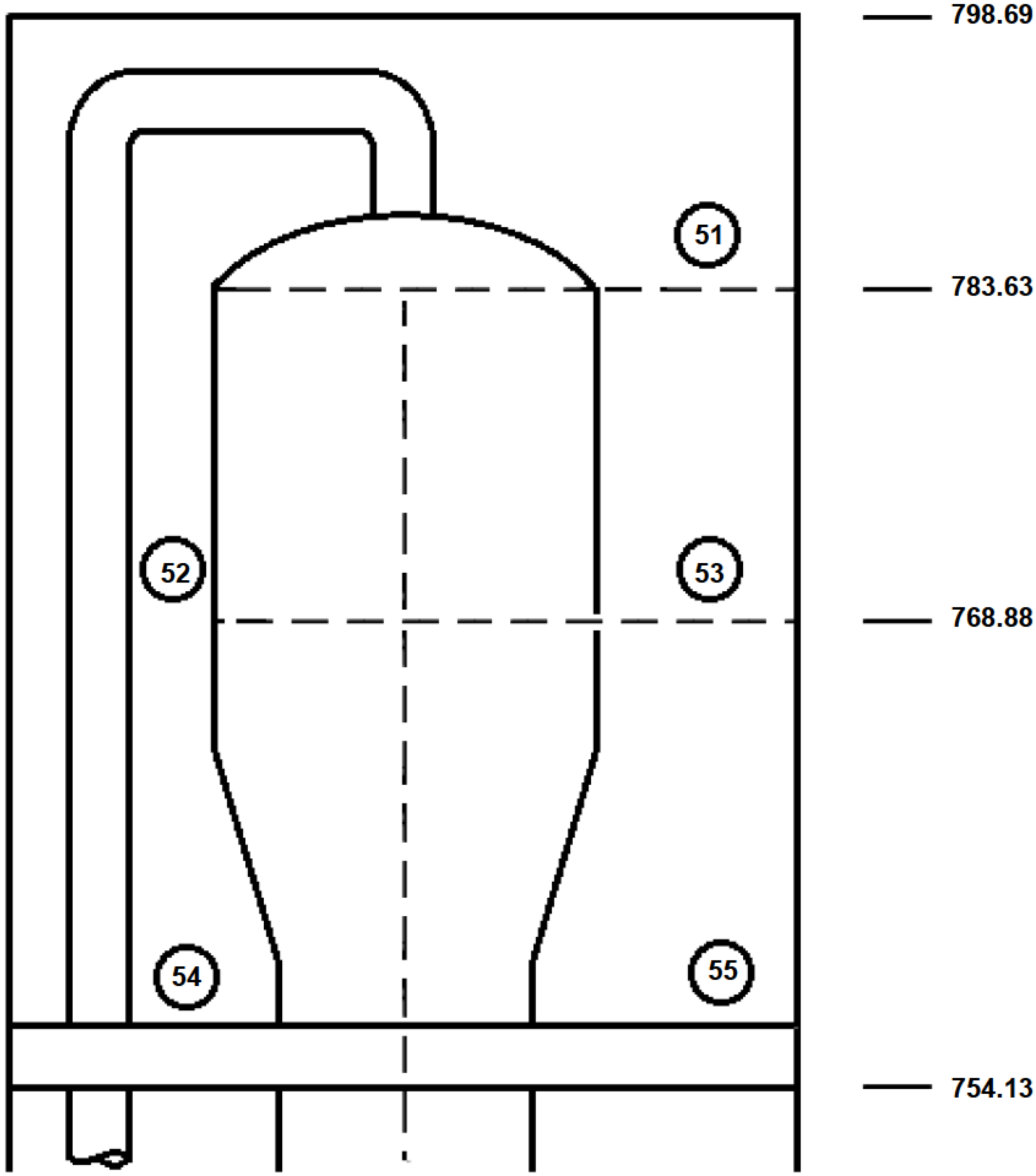
FIGURE 6.2.1-79



WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

Maximum Reverse Pressure Differential Case 6 Upper and Lower Compartment Temperatures

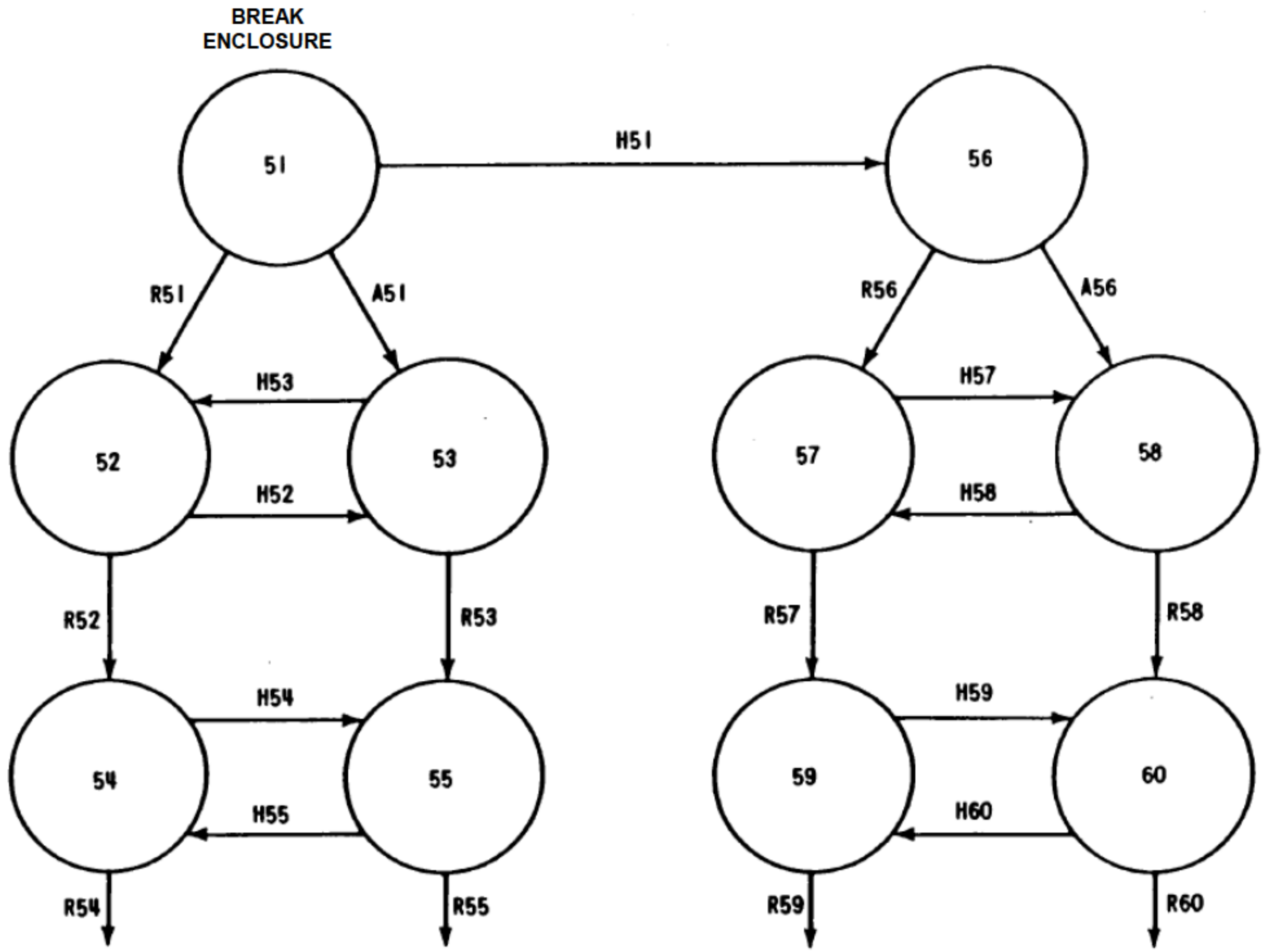
FIGURE 6.2.1-80



WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

Steam Generator
Enclosure Nodalization

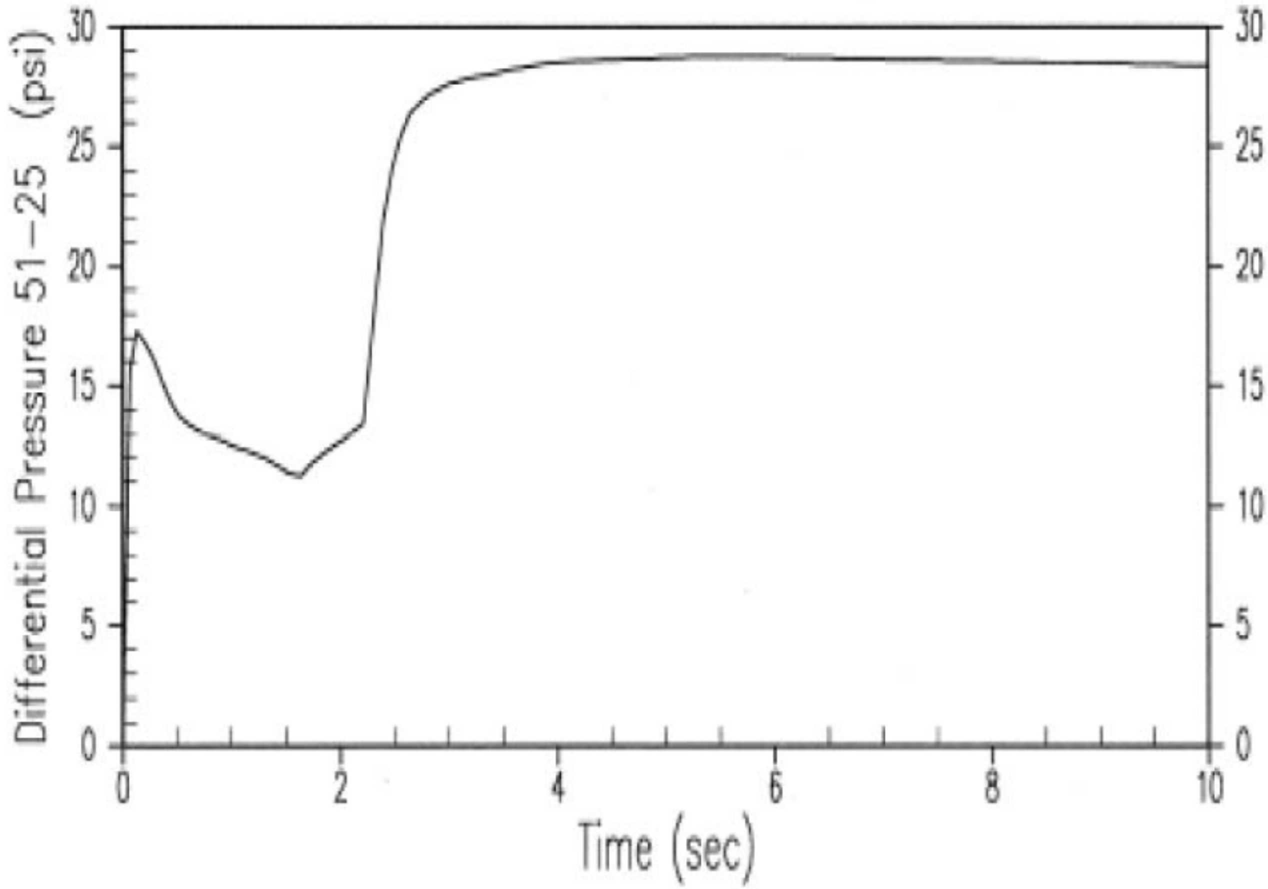
Figure 6.2.1-81



WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

Flow Paths for TMD Steam Generator Enclosure Short-Term Pressure Analysis

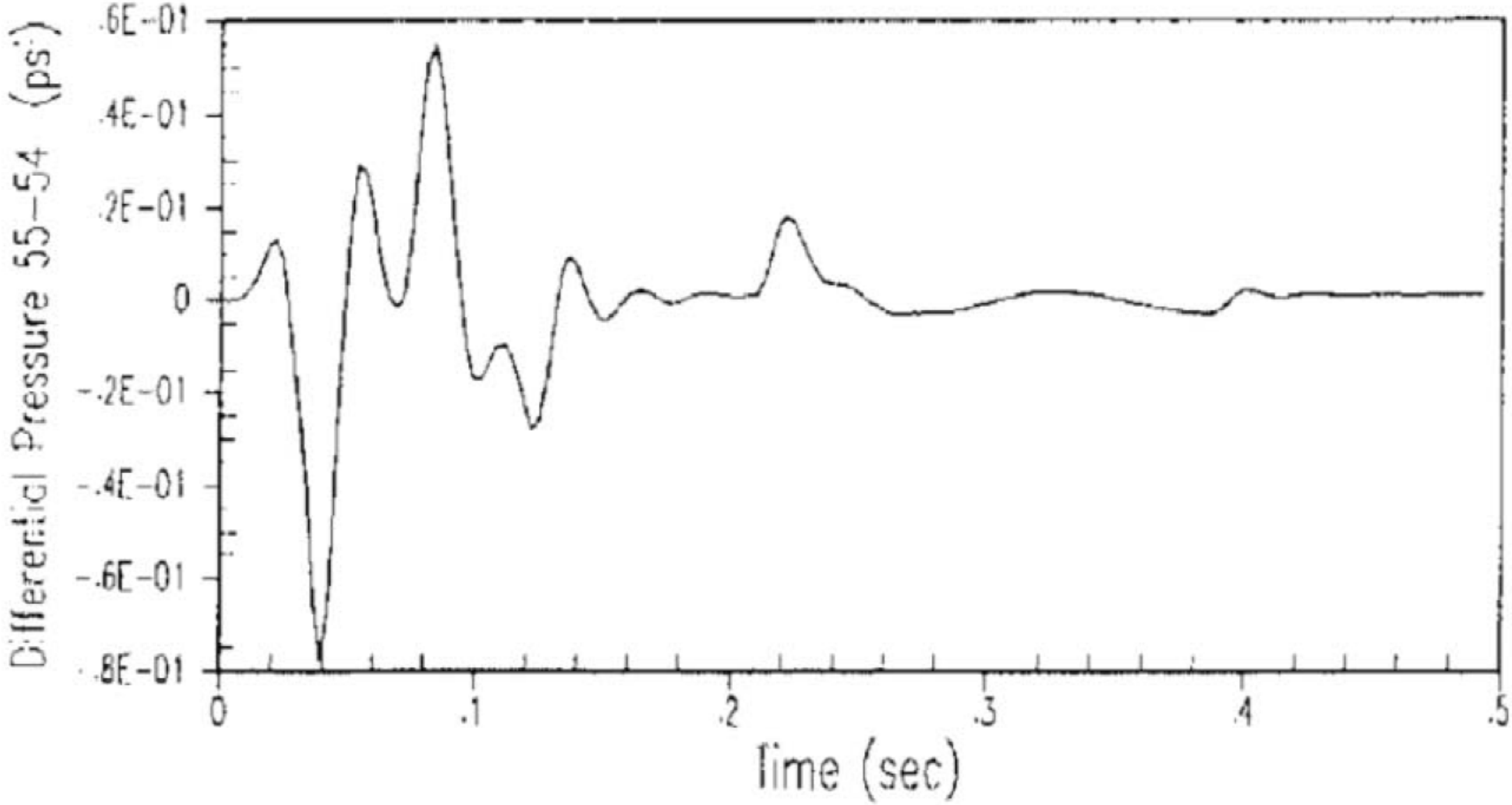
FIGURE 6.2.1-82



WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

Pressure Transient Between Break
Element and Upper Compartment
(Steam Generator Enclosure Analysis)

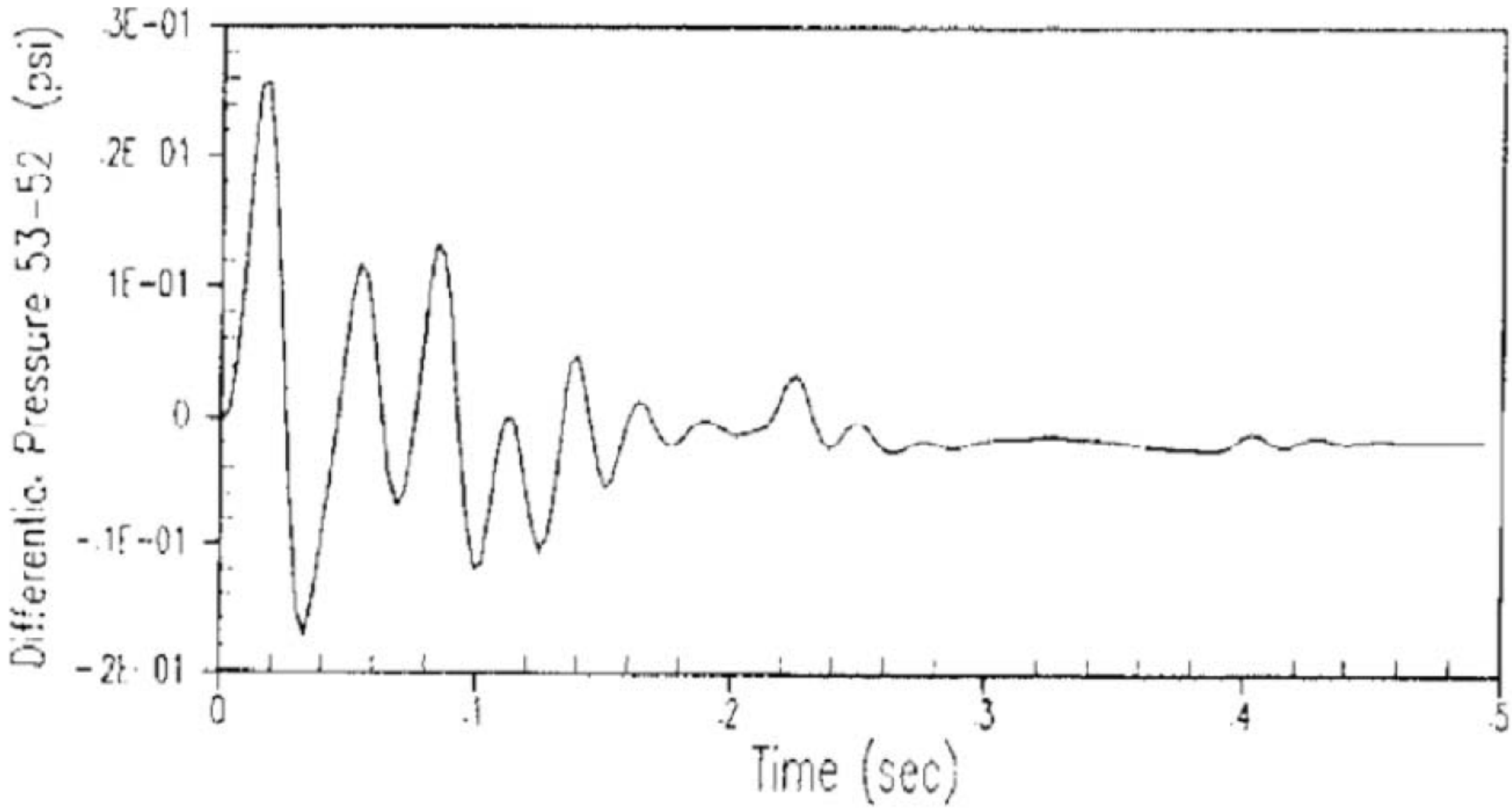
FIGURE 6.2.1-83



WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

Differential Pressure Transient
Across the Steam Generator Vessel
(Steam Generator Enclosure Analysis)

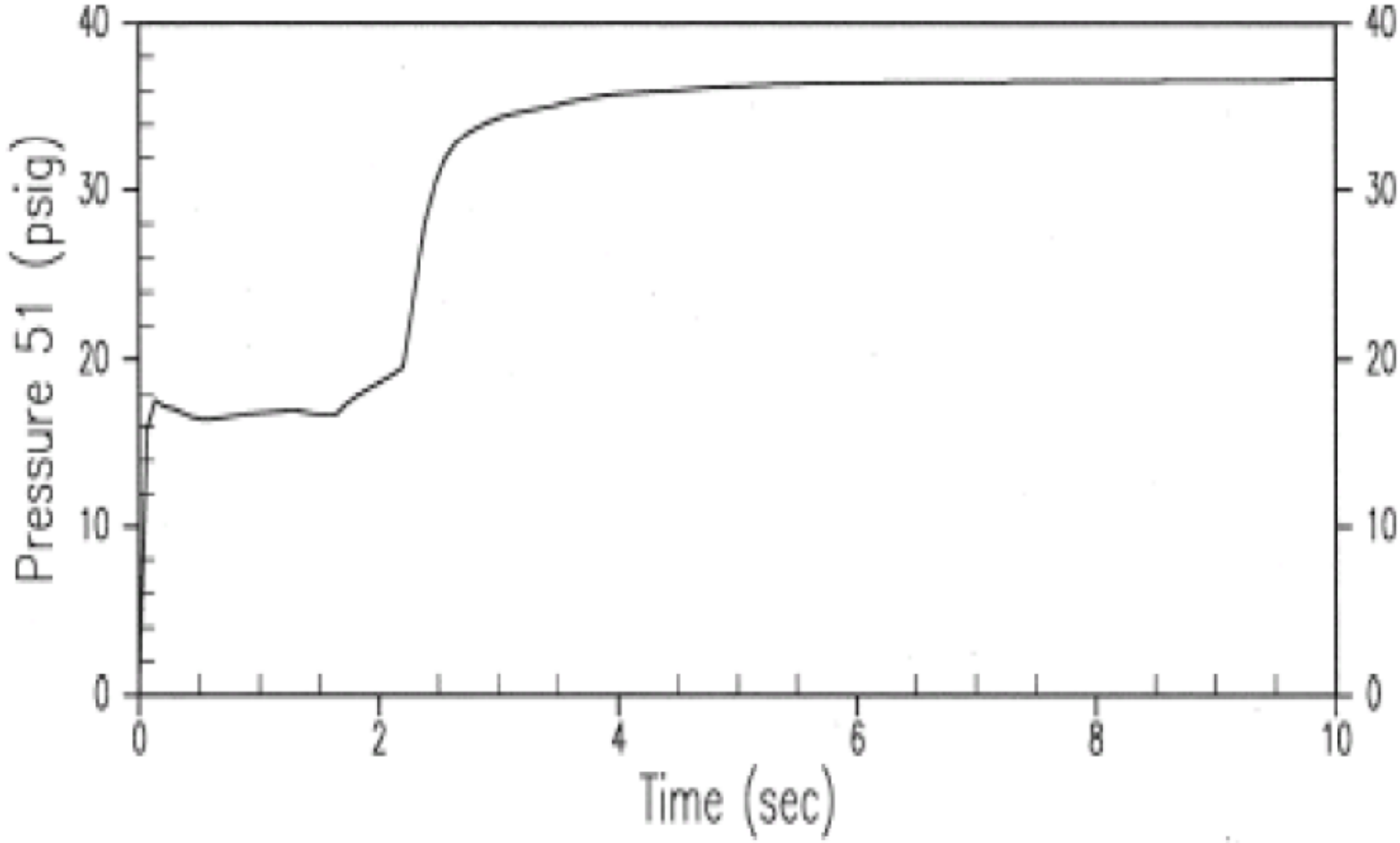
FIGURE 6.2.1-84



WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

Differential Pressure Transient
Across the Steam Generator Vessel
(Steam Generator Enclosure Analysis)

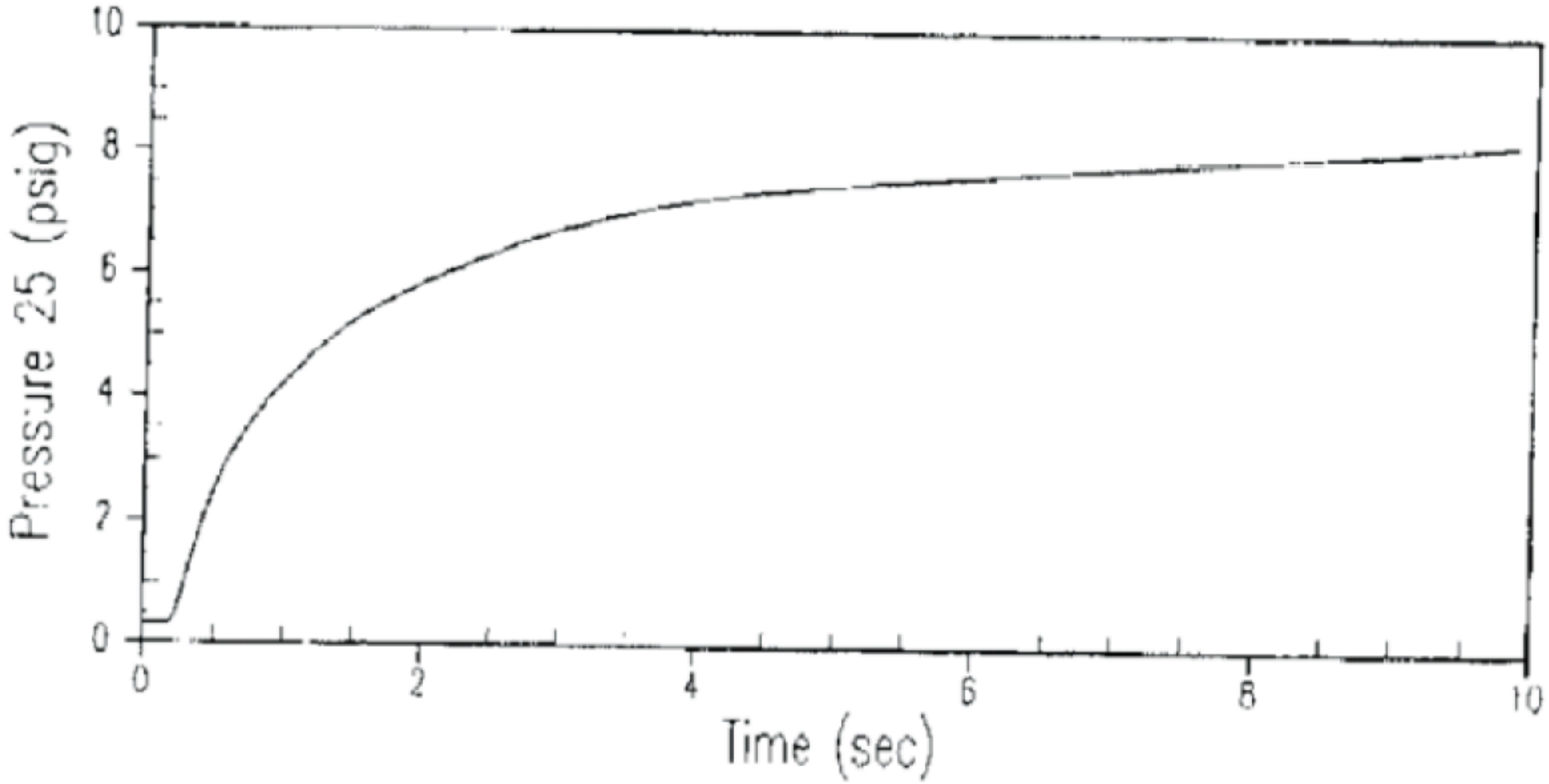
FIGURE 6.2.1-85



WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

Pressure Versus Time
for the Break Element
(Steam Generator Enclosure Analysis)

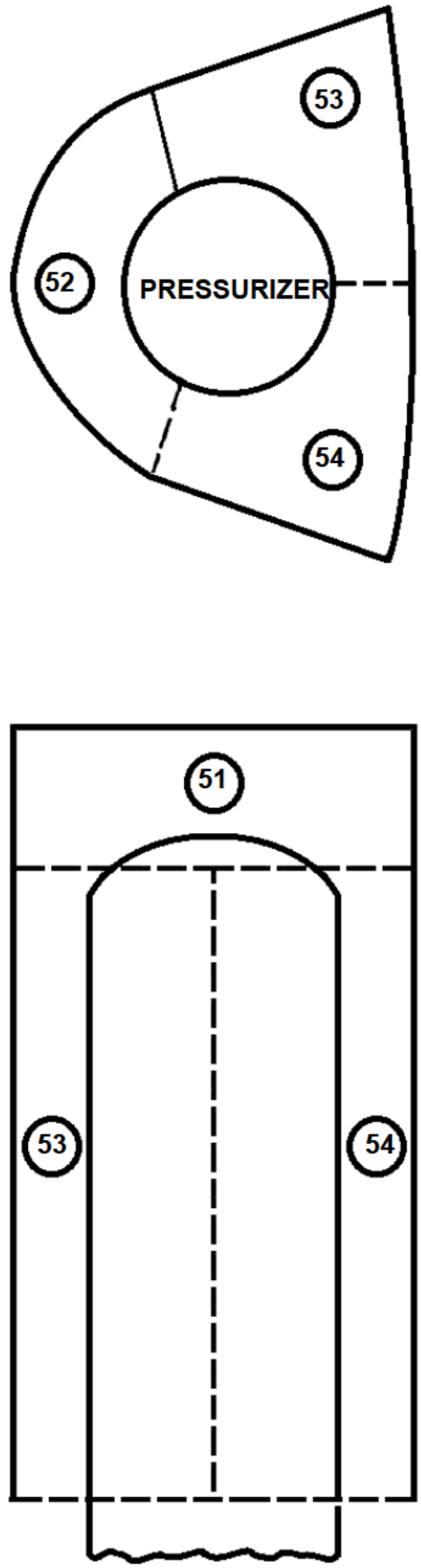
FIGURE 6.2.1-86



WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

Upper Compartment Pressure
Versus Time
(Steam Generator Enclosure Analysis)

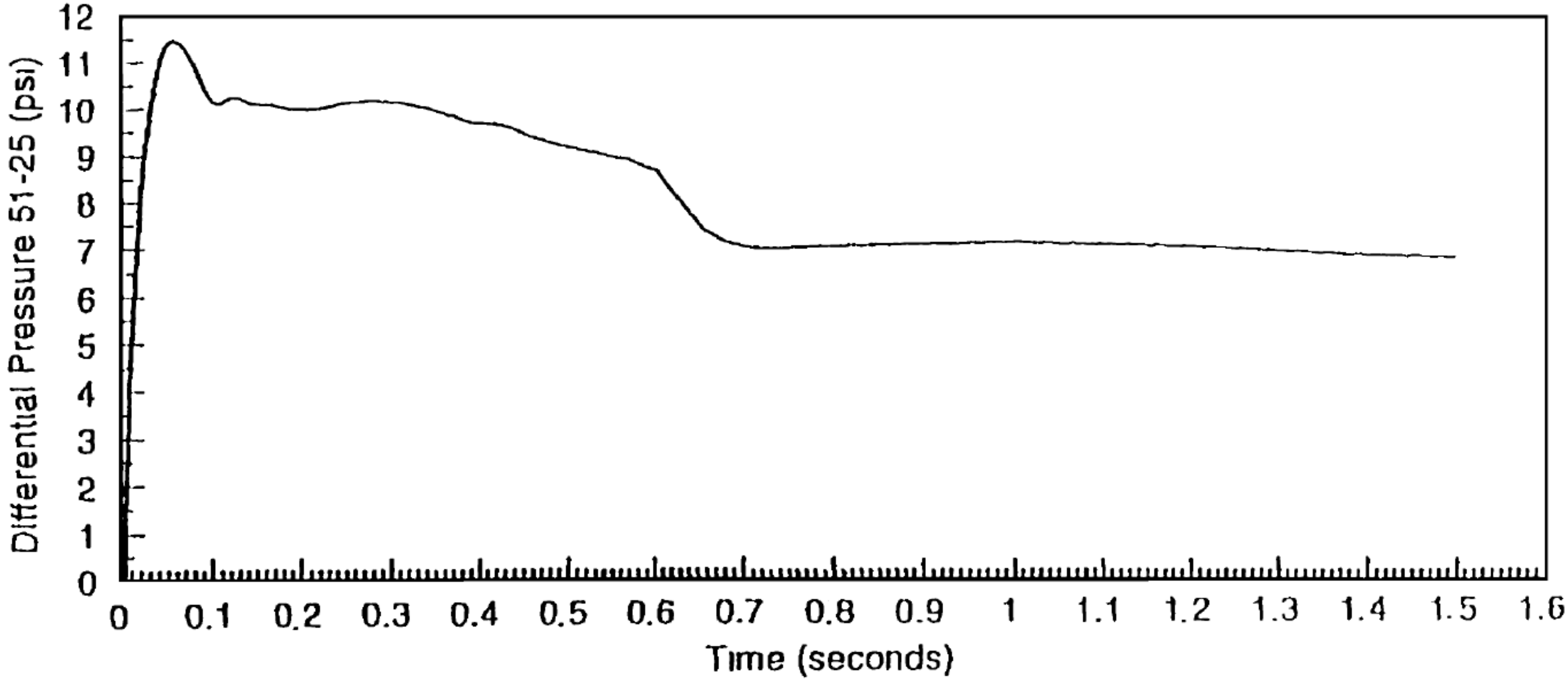
FIGURE 6.2.1-86a



WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

Pressure Enclosure
Nodalization Analysis

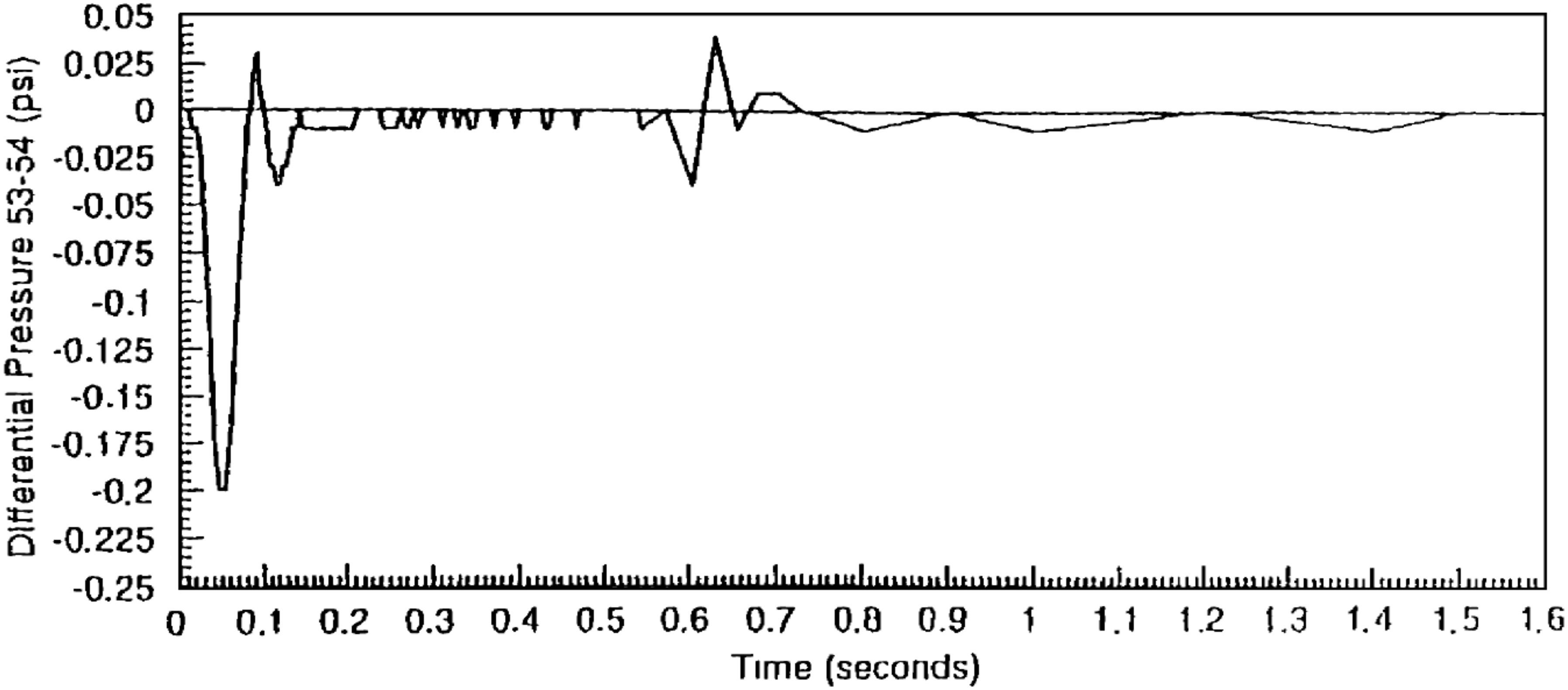
Figure 6.2.1-87



WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

Pressure Transient Between Break Element and Upper Compartment (Pressurizer Enclosure Analysis)

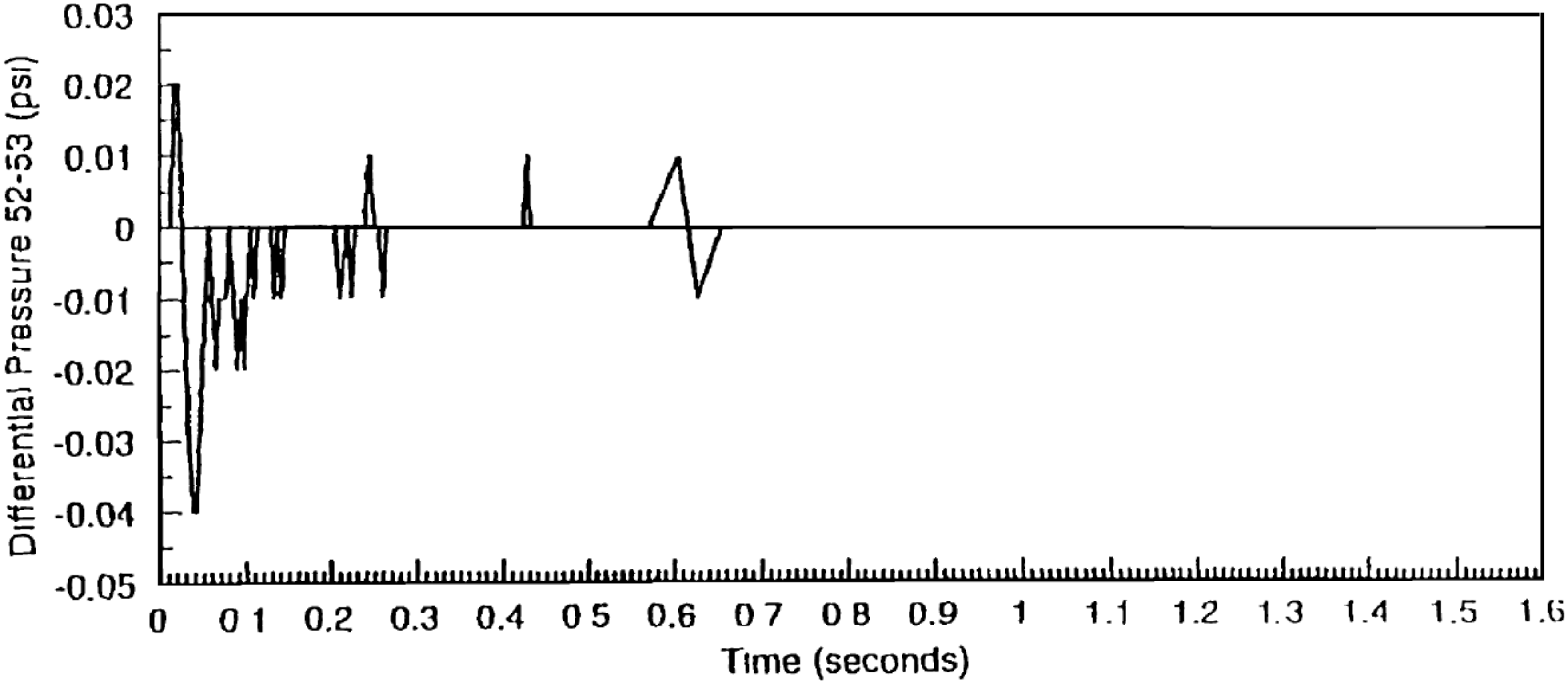
FIGURE 6.2.1-88



WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

Pressure Differential Across the
Pressurizer Vessel
(Pressurizer Enclosure Analysis)

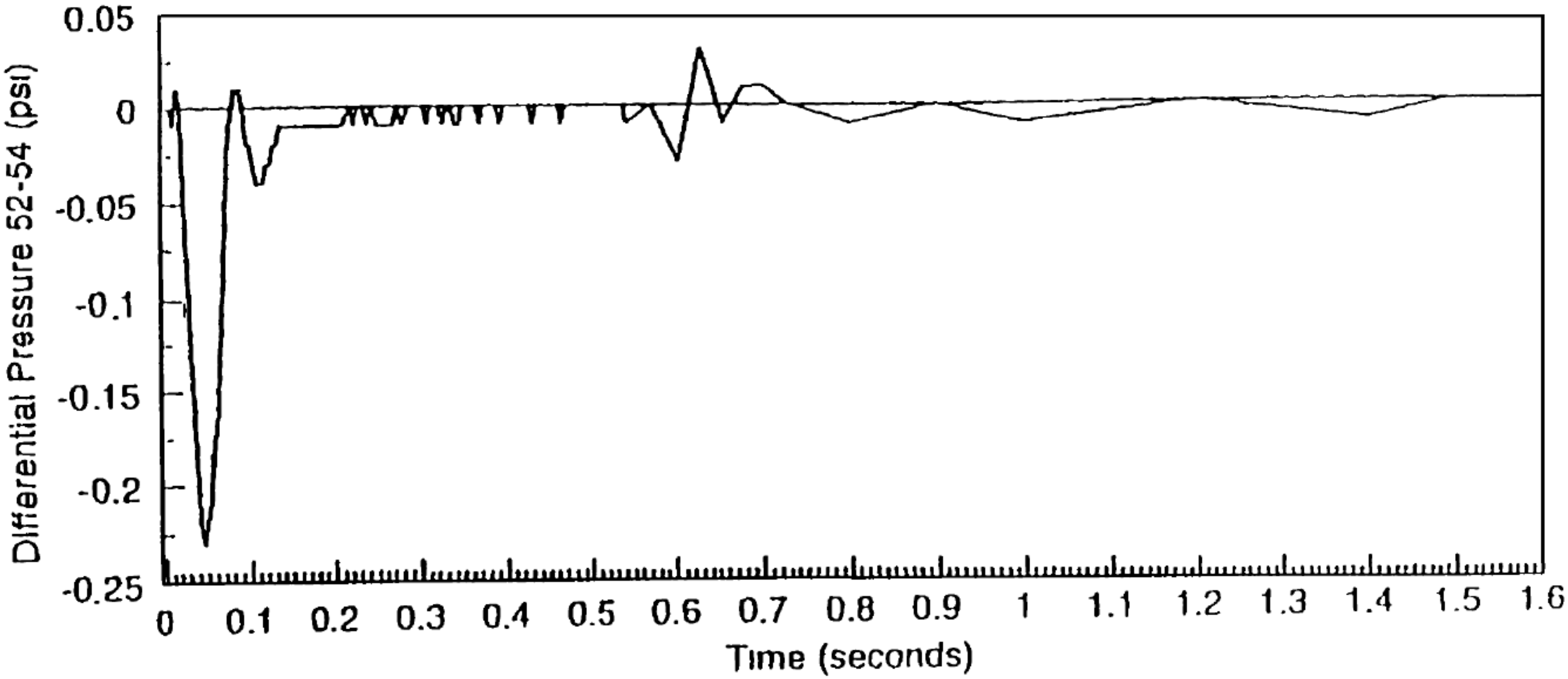
FIGURE 6.2.1-89



WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

Pressure Differential Across the Pressurizer Vessel (Pressurizer Enclosure Analysis)

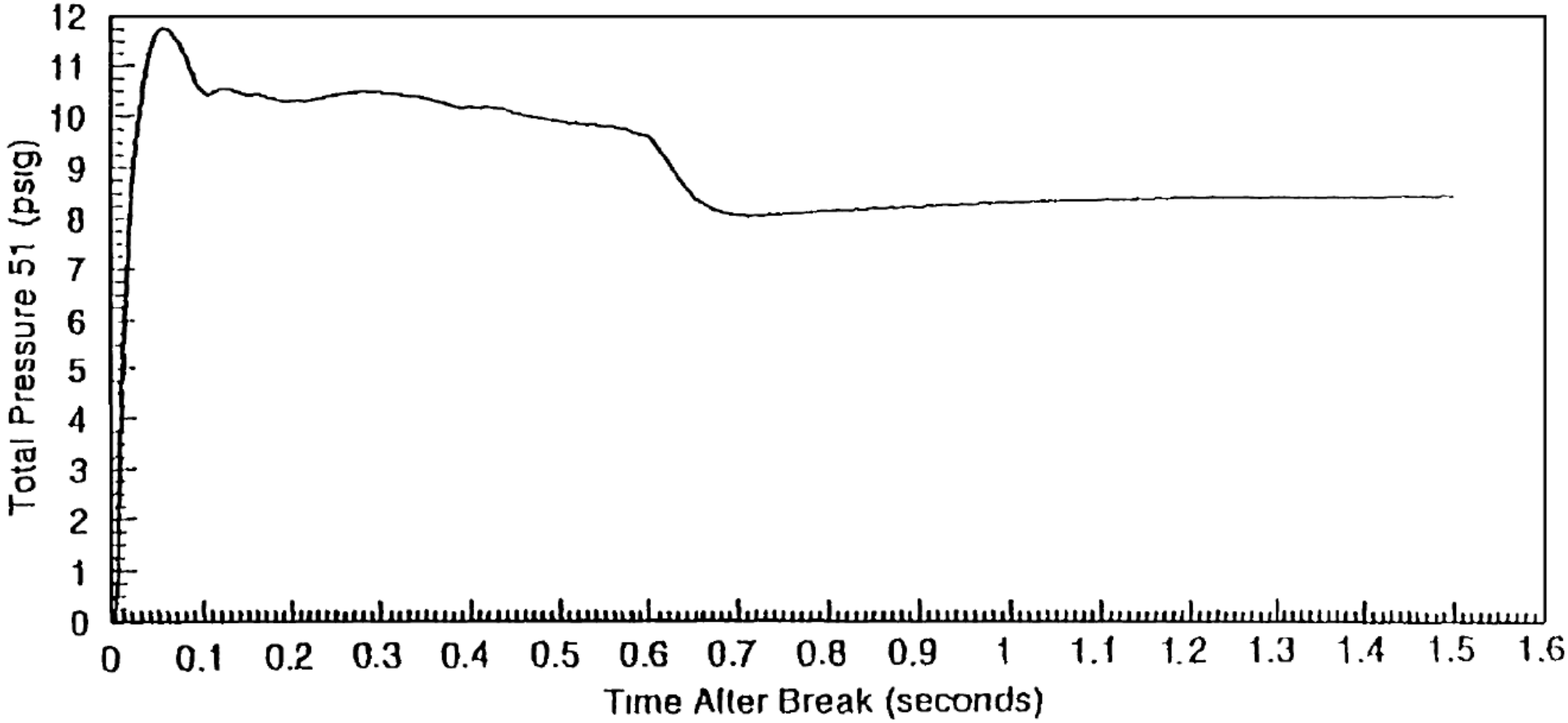
FIGURE 6.2.1-90



WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

Pressure Differential Across the Pressurizer Vessel (Pressurizer Enclosure Analysis)

FIGURE 6.2.1-91



WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

Pressure Versus Time
for the Break Element
(Pressurizer Enclosure Analysis)

FIGURE 6.2.1-92

6.2.2 CONTAINMENT HEAT REMOVAL SYSTEMS

Adequate containment heat removal capability for the ice condenser reactor containment is provided by the ice condenser (Section 6.7), the air return fan system (Section 6.8), and two separate containment heat removal spray systems whose components operate in the sequential modes described in Section 6.2.2.2. One of these heat removal spray systems is the containment spray system, and the second is the residual heat removal spray system, which is a portion of the residual heat removal system (Section 6.3).

Minimum engineered safety feature performance of the containment heat removal systems is achieved with the following:

1. Ice condenser (Section 6.7)

2. One train of the air return fan system

3. One train of the containment spray system

4. One train of the residual heat removal spray system (not required for steam or feed line break)

Each spray system consists of two trains of redundant equipment per reactor unit. There are four spray headers per unit. Two headers are supplied from separate trains of the containment spray system; the other two are supplied by separate trains of the RHR spray system. Each individual train consists of a pump, a heat exchanger, appropriate control valves, required piping, and a header with nozzles located in the upper compartment of the containment with flow directed to obtain full coverage of the containment upper volume during an emergency. The systems use borated water supplied from the refueling water storage tank and/or the recirculation sump.

6.2.2.1 Design Bases

The primary design basis for the containment heat removal spray systems is to spray cool water into the containment atmosphere when appropriate in the event of a loss-of-coolant accident or secondary side break and thereby ensure that the containment pressure cannot exceed the containment shell maximum internal pressure of 15.0 psig at 250°F, which corresponds to the code design internal pressure of 13.5 psig at 250°F (see Section 3.8.2). This protection is afforded for all pipe break sizes up to and including the hypothetical instantaneous circumferential rupture of the reactor coolant loop resulting in unobstructed flow from both pipe ends. After the ice has melted, the containment spray system and the residual heat removal spray system become the sole systems for removing energy directly from the containment. The containment heat removal systems are designed to provide a means of removing containment heat without loss of functional performance in the post-accident containment environment and operate without benefit of maintenance for the duration of time to restore and maintain containment conditions at atmospheric pressure. Although the water in the core after a loss-of-coolant accident is quickly subcooled by the emergency core cooling system (Section 6.3), the design of heat removal capability of each containment heat removal system is based on the conservative assumption that the core residual heat is released to the containment as steam which eventually melts all ice in the ice condenser.

The containment spray system provides two redundant heat removal trains. The system is designed such that both trains are automatically started by high-high containment pressure signal. The signal actuates, as required, all controls for positioning all valves to their operating position and starts the pumps. The operator can also manually actuate the entire system from the control room. Either of the two trains containing a pump, heat exchanger, and associated valving and spray headers is independently capable of delivering a minimum flowrate of 4,000 gpm.

The containment heat removal spray systems are designed to withstand the design basis earthquake and the operational basis earthquake without loss of function. They satisfy the TVA Class B Mechanical Requirements. The heat exchanger tube side (CSS) meets TVA Class B and the shell side (ERCW) meets TVA Class C mechanical requirements. The containment heat removal spray systems maintain their integrity and do not suffer loss of ability to perform their minimum required function due to normal operation, faults of moderate frequency, infrequent faults, and limiting faults.

Sufficient redundancy for all supporting systems necessary for minimum operational requirements of the containment heat removal spray systems is provided and complies with the single failure criteria for engineered safety features. Separate divisions on essential raw cooling water supply, power equipment heat exchangers, pumps, valves, and instrumentation are provided in order to have two completely separated trains.

The system is provided with overpressure protection from excessive pressures that could otherwise result from temperature changes, interconnection with other systems operating at higher pressures, or other means.

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

1. Shielding within guidelines of 10CFR20 and 10CFR100.

2. Collection of discharges from pressure relieving devices.

3. Remote means for isolating any sections under anticipated malfunction or failure conditions.

4. Means to detect and control radioactivity leakage into the environs to limits consistent with guidelines set forth in 10CFR20 and 10CFR100.

During accident conditions, cooling of the containment spaces is provided by the ice condenser system, containment heat removal spray systems, and the air return system. In addition, during non-LOCA accidents, the lower compartment cooler (LCC) fans are utilized to recirculate air throughout the lower containment spaces to prevent hot pockets from developing. The LCC units may be operated continuously throughout all accidents, except an MSLB, which do not initiate a containment Phase B isolation signal.. After a MSLB, all four coolers (2 Train A and 2 Train B) are started, although only 2 are required, within 1-1/2 hours to 4 hours after the MSLB to recirculate air throughout the lower containment spaces, to prevent hot spots from developing. During or after a LOCA, the LCC units, including their fans, are not required to be operable

6.2.2.2 System Design

The containment spray systems consist of two separate trains of equal capacity with each train independently capable of meeting system requirements. This system can be supplemented with two residual heat removal system pumps and two residual heat exchangers in parallel, with associated piping, valves and individual spray headers in the upper containment volume. Each train includes a pump, heat exchanger, ring header with nozzles, isolation valves and associated piping, and instrumentation and controls. Partial flow from an RHR system pump through its associated heat exchanger can be used to supplement each train. Independent electrical power supplies are provided for equipment in each containment spray train. In addition each train is provided with electrical power from separate emergency diesel generators in the event of a loss of offsite electrical power. During normal operation, all of the equipment is idle and the associated isolation valves are closed.

Upon system activation during a LOCA or other high energy line break, adequate containment cooling is provided by the containment spray systems whose components operate in sequential modes. These modes are: 1) spraying a portion of the contents of the refueling water storage tank into the containment atmosphere using the containment spray pumps; 2) after the refueling water storage tank has been drained, but while there is still ice remaining in the ice condenser, recirculation of water from the containment sump through the containment spray pumps, through the containment spray heat exchangers, and back to the containment (This spray is useful in reducing sump water temperatures.); 3) diversion of a portion of the recirculation flow from the residual heat removal system to additional spray headers. RHR spray operation is initiated manually by the operator only if the emergency core cooling system and containment spray system are both operating in the recirculation mode. If switchover to recirculation occurs prior to 1 hour after initiation of the LOCA, RHR spray operation can be commenced 1 hour after initiation of the LOCA. If switchover to recirculation occurs later than 1 hour after initiation of the LOCA, RHR spray operation can be commenced after completion of the switchover procedure.

The spray water from the containment and RHR spray systems returns from the upper compartment to the lower compartment through two 14 inch drains in the bottom of the refueling canal. The curbing around the personnel access door and the equipment access hatch on the operating deck directs spray water flow towards the refueling canal. The air-water mixture entering the air return fans will be rerouted inside the polar crane wall through the accumulator rooms utilizing curbing, the floor hatch cover and floor drainage system.

The flow diagram for this system is presented in Figure 6.2.2-1.

Component Description

Pumps

The containment spray system flow is provided by two centrifugal type pumps driven by electric motors. The motors, which can be powered either normally or from an emergency source are direct coupled and non-overloading to the end of the pump curve. The design head of the containment spray pump is sufficient to ensure rated capacity with a minimum level in the refueling water storage tank or the containment sump when pumping against a head equivalent to the sum of the maximum pressure of the containment post LOCA/HELB, the elevational head between the pump discharge and the uppermost spray nozzles, and the equipment and piping friction losses. See Table 6.2.2-1 for design parameters and Figure 6.2.2-2 for characteristic curves.

The residual heat removal pumps which also provide flow to the containment heat removal spray system are described in Section 5.5.7.2.1 and Table 5.5-8.

Each residual heat removal pump provides a design flow of 2000 gpm (1475 gpm analytical for Unit 2) for upper containment spray.

Each containment spray pump is powered by a horizontal squirrel cage induction motor. Pump motor parameters are presented in Table 6.2.2-1.

Net Positive Suction Head (NPSH)

The plant and piping layout of the containment spray system ensures that the pump NPSH requirements are met at maximum runout conditions with the containment spray pumps taking suction from either the refueling water storage tank or the containment sump. The NPSH available from the containment sump is calculated using the maximum credible sump water temperature (190°F) with no credit taken for containment overpressure or height of water above the RHR Sump Strainer Assembly. Available NPSH parameters are given in Table 6.3-12.

Heat Exchangers

The containment spray heat exchangers are the vertical counter flow U-tube type with tubes welded to the tube sheet. Borated water from either the refueling water storage tank or the containment sump circulates through the tube side. Design parameters are presented in Table 6.2.2-2.

Piping

All containment heat removal spray system piping in contact with borated water is austenitic stainless steel. All piping joints are welded except for the flanged connection at the pump or where disassembly of the joint may be required. On Unit 1, flanged connections are also used where the cut spray piping was reassembled during restoration of the steel containment vessel dome steel following installation of the replacement steam generators.

Spray Nozzles and Ring Headers

Each containment spray ring header provides 4000 gpm minimum and contains 263 hollow cone ramp bottom nozzles, each of which is capable of a design flow of 15.2 gpm with a 40 psi differential pressure. These nozzles have an approximately 3/8-inch diameter spray orifice and are not subject to clogging by particles less than 1/4 inch in maximum dimension. The nozzles produce a drop size spectrum with a median diameter of less than 700 microns in diameter at 40 psid. The spray solution is completely stable and soluble at all temperatures of interest in the containment and, therefore, does not precipitate or otherwise interfere with nozzle performance. Each nozzle header is independently oriented to maximize coverage of the containment volume inside the crane wall. This arrangement prohibits any flow into the ice condenser.
The residual heat removal spray ring headers contain 147 nozzles, Unit 1, or 146 nozzles, Unit 2, per header and deliver a design flow of 2000 gpm (1475 gpm analytical) per header. Each RHR spray header will perform its required design function with 142 unobstructed spray nozzles. They have the same design characteristics as the headers in the containment spray system.

Refueling Water Storage Tank

During the injection phase immediately following a LOCA or HELB, the containment spray is supplied from the refueling water storage tank.

Recirculation Sump

The recirculation sump is described in Section 6.3.2.2 under the discussion of the recirculation mode.

Material Compatibility

All parts of the containment spray system in contact with borated water are austenitic stainless steel or equivalent corrosion resistant material.

6.2.2.3 Design Evaluation

Performance of the containment heat removal system is evaluated through analyses of the design basis accident and various other cases described in Chapter 15 and Section 6.2.1. The analyses were performed using the LOTIC code and show that the containment heat removal systems are capable of keeping the containment pressure below the containment maximum internal pressure of 15 psig, which corresponds to the code design internal pressure of 13.5 psig at 250°F (see Section 3.8.2) even when it is assumed that the minimum engineered safety features are operating. Section 6.2.1 presents a description of the analytical methods and models which were used along with verification of pertinent items from Waltz Mill tests, and curves showing the calculated performance of important variables following the design-basis loss-of-coolant accident.

The design basis accident containment pressure analysis input assumptions and results are presented in Section 6.2.1.3.3.

The containment spray systems provide two full-capacity heat removal systems for the containment, each of which is sized as described in Section 6.2.2.1 to remove heat at a rate which precludes an increase of the containment maximum internal pressure above 15.0 psig, which corresponds to the code design internal pressure of 13.5 psig at 250°F (see Section 3.8.2).

All spray headers and spray nozzles are located inside the containment in the upper compartment and can withstand, without loss of function or maintenance, the post-accident containment environment. The remainder of the systems, with the exception of the refueling water storage tanks, which includes all active components, are located in the Auxiliary Building and, therefore, are not affected by wind, tornado, or snow and ice conditions.

The design is based on the spray water being raised to the saturation temperature of the containment in falling through the steam-air mixture within the building. The minimum fall path of the droplets is approximately 75 ft from the 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 nozzles. Figures 6.2.2-3 through 6.2.2-6 depict the containment spray coverage for the containment spray system.

Except for the refueling water storage tank water supplied by the safety injection system, the containment spray system initially operates independently of other engineered safety features. For extended operation in the recirculation mode, water is supplied to the containment RHR spray headers through the residual heat removal pumps and residual heat removal exchangers. One containment spray system train, supplemented by one RHR spray train, when required, provides adequate heat removal capability to limit containment pressure below design (see Section 6.2.1.3). RHR spray is required only after switchover to the recirculation mode and no earlier than 1 hour after initiation of the LOCA. At this time one RHR pump can provide sufficient RHR spray as well as adequate core flow via the high head (one centrifugal charging and one safety injection) pumps. (See Section 6.3.3 for the performance evaluation of the RHR pumps in their core cooling function.)

All active components of the system were analyzed to show that the failure of any single active component does not prevent fulfilling the design function. This analysis is summarized in Table 6.2.2-3. A single failure in the residual heat removal system will not prevent long-term use of the spray system. The analyses of the loss-of-coolant accident presented in Chapter 15 reflect the single failure analysis. Each of the spray trains provides complete backup for the other.

An analysis of the spray return drains located in the refueling canal has been made to show that they are adequately sized for a maximum RHR and containment spray flow and ensures an adequate water supply in the lower compartment to satisfy pump NPSH requirements.

The passive portions of the spray systems located within the containment are designed to withstand, without loss of functional performance, a post accident containment environment and to operate without benefit of maintenance.

The spray headers which are located in the upper containment volume are separated from the reactor and primary coolant loops by the operating deck and inner wall of the ice bed. These spray headers are therefore protected from missiles originating in the lower compartment.

This evaluation shows that the containment spray systems can withstand expected conditions during the 40-year life of the plant without loss of capability to perform the required safety functions. Specifically, the system achieved this by having been designed to meet applicable General Design Criteria (GDC) as follows:

1. The systems can withstand the effects of natural phenomena as required by GDC 2.

2. The systems are designed to accommodate the effects of and be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents including loss of coolant as required by GDC 4.

3. The systems are not shared with another nuclear power unit as required by GDC 5.

4. The systems are designed to be capable of being inspected and tested to ensure reliability throughout their life as required by GDC 39 and 40.

5. The systems are designed adequately to provide post-accident cooling inside the primary containment to reduce the containment pressure and temperature following any LOCA and maintain them at acceptable levels, as required by GDC 38.

6. The systems are designed to aid in the control and removal of fission products, hydrogen, oxygen, and other substances which may be released into the reactor containment following postulated accidents to assure that containment integrity is maintained, as required by GDC 41.

<u>Tritium Production Core Evaluation (Unit 1 Only)</u>

The effects of converting WBN to a Tritium Production Core (TPC) on the containment heat removal systems, have been evaluated. The evaluation includes the effect of adding a leading edge flow meter (LEFM) into the main feedwater piping and the effect of uprating the plant nominal power level by 1.4%.

Based on the conclusions in Section 6.2.1.3.12 that mass and energy releases for LOCA and secondary side pipe ruptures do not have an adverse impact on the primary containment functionality due to incorporation of a TPC, the containment heat removal system will not be impacted.

6.2.2.4 Testing and Inspections

Performance tests of the active components in the system are performed in the manufacturer's plant and followed by in-place preoperational testing.

Capability is provided to test initially and subsequently on a routine basis to the extent practical the operational startup sequence and performance capability of the containment spray system including the transfer to alternate power sources. Capability to test periodically the delivery capacity of the containment spray system at a position as close to the spray header as is practical and for obstruction of the spray nozzles is provided. As part of the preoperational test program, the containment spray nozzles are physically verified to pass an unobstructed flow of air. The air is introduced into the headers through an air test connection on each header.

Initially, the containment spray system is hydrostatically tested to the applicable code test pressure.

All periodic tests of individual components or the complete containment spray system are controlled to ensure that plant safety is not jeopardized and that undesirable transients do not occur.

The containment spray system is designed to comply with ASME Section XI, "Inservice Inspection of Nuclear Reactor Coolant System." Detailed test procedures are given in Chapter 14.

6.2.2.5 Instrumentation Requirements

The containment spray system is actuated either manually from the control room or external to the main control room or automatically by the coincidence of two sets out of four protection set loops monitoring the lower containment pressure. The high-high containment pressure signal starts the containment spray pumps and positions all valves to their operating configuration.

The operation of the containment spray system is verified by instrument readout in the control room. Pump motor breakers energize indicating lights on the control panel to show power is being supplied to the pump motors. Status lights on the main control panel indicate valve position and are energized independently of the valve actuation signal.

To protect the pumps from low flow conditions, a mini-flow recirculation line is provided to allow pump discharge to be circulated back into the pump suction line. This line is opened by a motor-operated valve when flow in the discharge line drops below that required for pump protection or if, upon starting, flow is not achieved in the spray header within a preset time interval. Elbow taps in each discharge line provide a delta-p measurement to monitor the flow rate and provide the flow signal for the control room flow indicators and to control the minimum flow recirculation valve.

Local instruments monitor the following parameters: containment spray pump suction and discharge pressure, heat exchanger inlet temperature, heat exchanger inlet and outlet pressure, and containment spray test line flow.

In the event of a main control room evacuation, the necessary control functions are transferable to outside of the main control room in order to assure that the system can be aligned and locked to prevent inadvertent operation and to manually initiate system operation if necessary. The control transfer is provided for the spray pumps, containment spray isolation valves, and containment sump isolation valves.

The system is designed as Seismic Category I. The instrumentation and associated interconnected wiring and cables are physically and otherwise separated so that a single event cannot cause malfunction of the entire system.

6.2.2.6 Materials

All parts of the containment spray system in contact with borated water are austenitic stainless steel or equivalent corrosion-resistant material. None of these materials produce radiolytic or pyrolytic decomposition products that can interfere with this or other engineered safety features.

REFERENCES

None

TABLE 6.2.2-1

CONTAINMENT SPRAY PUMP/MOTOR DESIGN PARAMETERS

Pump	
Quantity Per Unit	2
Design Pressure, psig	300
Design Temperature, °F	250
Design Flow Rate, gpm	4000
Design Head (1), ft	435

Motor	
Horsepower, hp	700
Service Factor	1.15
Voltage, V	6600
Phase	3
Cycles, Hz	60

Note:

(1) UNIT 2 ONLY - The minimum acceptable developed head at 4,000 gpm is 391.5 ft (includes 5% pump wear allowance and is based on LO-LO level in the RWST), thereby assuring that the safety limit for accident analysis (372.9 ft at 4000 gpm) is met.

Preoperational testing results indicate that initial CS pump performance is somewhat below the certified pump curves, but well in excess of the minimum required CS pump performance as evaluated in LTR-SEE-I-15-19.

WBN

TABLE 6.2.2-2

CONTAINMENT SPRAY HEAT EXCHANGER DESIGN PARAMETERS

Quantity Per Unit	2
Type	Counter Flow
Percent Tubes Plugged	10%
Heat Transfer Per Unit, Btu/hr	12.85×10^7
Shell-Side Flow, gpm	5,200
Tube-Side Flow, gpm	4000
Tube-Side Inlet Temperature, °F	190
Shell-Side Inlet Temperature, °F	85
Tube-Side Outlet Temperature, °F	124.7
Shell-Side Outlet Temperature, °F	129.5
Design Pressure (Shell/Tube), psig	150/300
Design Temperature (Shell/Tube), °F	200/250
Heat Exchanger UA, Btu/hr - °F	2.44×10^6

TABLE 6.2.2-3 (Sheet 1 of 5)

FAILURE MODES AND EFFECTS ANALYSIS

ITEM NO. COMPONENT	FUNCTION	FAILURE MODE	POTENTIAL CAUSE	METHOD OF DETECTION	EFFECTS ON SYSTEM	EFFECT ON PLANT	REMARKS
INJECTION MODE							
1. Containment Spray Pump A-A	To pump borated water to spray nozzles in the containment	Fails to start	Mechanical failure or electrical failure due to loss of power	Indication via switches HS-72-27A & -27C	CS System Train "A" is lost	None. Two 100% capacity trains are provided.	Only one train is required to mitigate accident consequences.
2. Containment Spray Pump B-B	To pump borated water to spray nozzles in the containment	Fails to start	Mechanical failure or electrical failure due to loss of power	Indication via switches HS-72-10A & -10C	CS System Train "B" is lost	None. Two 100% capacity trains are provided.	Only one train is required to mitigate accident consequences.
3. Valve FCV 72-21	Provide water flow path to CS PMP B-B	Fails closed	Mechanical failure blockage or spurious electrical signal	Valve position indication via switch HS-72-21A	CS System Train "B" is inoperable	None. CS System Train "A" is used to supply water to nozzles	Only one train is required to mitigate accident consequences.
4. Valve FCV 72-22	Provide water flow path to CS PMP A-A	Fails Closed	Mechanical failure blockage or spurious electrical signal	Valve position indication via switch HS-72-22A	CS System Train "A" is inoperable	None. CS System train "B" is used to supply water to nozzles	Only one train is required to mitigate accident consequences.
5. Valve FCV 72-02	Water flow path to containment spray nozzles	Fails to open	Mechanical failure or loss of power to valve motor	Valve position indication via switches HS-72-2A/HS-72-2B*	CS System Train "B" is lost	None. Two 100% capacity trains are provided.	Only one train is required to mitigate accident consequences.
		Recloses after opening	Mechanical failure or spurious electrical signal	Valve position indication via switches HS-72-2A/HS-72-2B	CS System Train "B" is lost	None. Two 100% capacity trains are provided.	Only one train is required to mitigate accident consequences.
6. Valve FCV 72-039	Water flow path to containment spray nozzles	Fails to open	Mechanical failure or loss of power to valve motor	Valve position indication via switches HS-72-39A/HS-72-39B	CS System Train "A" is lost	None. Two 100% capacity trains are provided.	Only one train is required to mitigate accident consequences.

TABLE 6.2.2-3 (Sheet 2 of 5)

FAILURE MODES AND EFFECTS ANALYSIS

ITEM NO. COMPONENT	FUNCTION	FAILURE MODE	POTENTIAL CAUSE	METHOD OF DETECTION	EFFECTS ON SYSTEM	EFFECT ON PLANT	REMARKS
		Recloses after opening	Mechanical failure or spurious electrical signal	Valve position indication via switches HS-72-39A/HS-72-39B	CS System Train "A" is lost	None. Two 100% capacity trains are provided.	Only one train is required to mitigate accident consequences.
7. Emergency power to Train A	Provide power to PMP A motor & all MOV's in Train A	Fails	Diesel generator shutdown Bd. 1A-A failure	Voltmeters in Control Room	CS System Train "A" is lost	None. Two 100% capacity trains are provided.	Only one train is required to mitigate accident consequences.
8. Emergency Power to Train B	Provide power to PMP B motor & all MOV's in Train B	Fails	Diesel generator shutdown Bd. 1B-B failure	Voltmeters in Control Room	CS System Train "B" is lost	None. Two 100% capacity trains are provided.	Only one train is required to mitigate accident consequences.
9. FCV-72-34	Isolate pump A-A discharge from suction	Fails open	Mechanical failure or spurious electrical signal	Valve position indication via switch HS-72-34A	Water flowing into spray nozzles will be diminished.	Train "A" capability reduced. No impact on plant; operator can put 100% train B in operation.	Only one train is required to mitigate accident consequences.
10. FCV-72-13	Isolate pump B-B discharge from suction	Fails open	Mechanical failure or spurious electrical signal	Valve position indication via switch HS-72-13A	Water flowing into spray nozzles will be reduced.	Train "B" capability reduced. No impact on plant; operator can put 100% train A in operation.	Only one train is required to mitigate accident consequences.
11. Check VA. 72-506	Flow path to CS Pump A-A	Valve stuck closed	Mechanical failure	Pump discharge flow indicator FI-72-34	Train A is inoperable	None. Two 100% capability trains are provided.	
12. Check VA 72-507	Flow path to CS Pump B-B	Valve stuck closed	Mechanical failure	Pump discharge flow indicator FI-72-13	Train B is inoperable	None. Two 100% capability trains are provided.	

TABLE 6.2.2-3 (Sheet 3 of 5)

FAILURE MODES AND EFFECTS ANALYSIS

ITEM NO.	COMPONENT	FUNCTION	FAILURE MODE	POTENTIAL CAUSE	METHOD OF DETECTION	EFFECTS ON SYSTEM	EFFECT ON PLANT	REMARKS
13.	Check VA 72-524	CS Pump A-A discharge flow path	Valve stuck closed	Mechanical failure	Flow measurement via flow indicator FI-72-34	Train A is inoperable	None. Two 100% capability trains are provided.	
14.	Check VA 72-525	CS Pump B-B discharge flow path	Valve stuck closed	Mechanical failure	Flow measurement via flow indicator FI-72-13	Train B is inoperable	None. Two 100% capability trains are provided.	
15.	Check VA 72-548	Flow path to Train A spray nozzles	Valve stuck closed	Mechanical failure	Pump discharge via flow indicator FI-72-34	Train A is inoperable	None. Two 100% capability trains are provided.	
16.	Check VA 72-549	Flow path to Train B spray nozzles	Valve stuck closed	Mechanical failure	Pump discharge via flow indicator FI-72-13	Train B is inoperable	None. Two 100% capability trains are provided.	
17.	Relief VA 72-508	CS Pump A-A suction piping over-pressure protection	Valve fails open	Mechanical failure		Train A capacity will be diminished	None. Two 100% capacity trains provided.	
18.	Relief VA 72-509	CS Pump B-B suction piping over-pressure protection	Valve fails open	Mechanical failure		Train B capacity will be diminished	None. Two 100% capacity trains provided.	
RECIRC. MODE ACTIVE FAILURE								
19.	Valve FCV-72-44	Train A flow path from containment sump	Fails to open or recloses after opening	Mechanical failure or electrical failure	Valve position indication via switches HS-72-44A, HS-72-44B, HS-72-44C	Train "A" is inoperable	None. Two 100% capacity trains are provided.	
20.	Valve FCV-72-45	Train B flow path from containment sump	Fails to open or recloses after opening	Mechanical failure or electrical failure	Valve position indication via switches HS-72-45A, HS-72-	Train "B" is inoperable	None. Two 100% capacity trains are	

FAILURE MODES AND EFFECTS ANALYSIS

ITEM NO. COMPONENT	FUNCTION	FAILURE MODE	POTENTIAL CAUSE	METHOD OF DETECTION	EFFECTS ON SYSTEM	EFFECT ON PLANT	REMARKS
				45B, HS-72-45C		provided.	
21. Valve FCV-72-22	Isolate "RWST" during recirc. mode	Fails to close	Mechanical failure or electrical failure	Valve position indication via switch HS-72-22A	Train "A" is inoperable	None. Two 100% capacity trains are provided.	Interlock will prevent opening of FCV-72-44
22. Valve FCV-72-21	Isolate "RWST" during recirc. mode	Fails to close	Mechanical failure or electrical failure	Valve position indication via switch HS-72-21A	Train "B" is inoperable	None. Two 100% capacity trains are provided.	Interlock will prevent opening of FCV-72-45
23. Containment Spray Heat Exch. A	Cools containment spray water	ERCW flow lost	ERCW Train A system mechanical or electrical failure	High CS water temp. indication via ICS point T0168A	CS water in Train A can not be cooled.	None. 100% capacity Train B will be operated.	
24. Containment Spray Heat Exch. B	Cools containment spray water	ERCW flow lost	ERCW Train B system mechanical or electrical failure	High CS water temp. indication via ICS point T0169A.	CS water in Train B can not be cooled.	None. 100% capacity Train A will be operated.	
25. Check VA 72-506	Isolate RWST during recirc. mode	Valve stuck open	Mechanical failure		None. Valve FCV-72-22 will isolate RWST	None.	
26. Check VA 72-507	Isolate RWST during recirc. mode	Valve stuck open	Mechanical failure		None. Valve FCV-72-21 will isolate RWST	None.	
PASSIVE							
27. Containment Spray Heat Exch. A	Cools CS water during recirc. mode	Clogging/tube rupture	Impurities, particulate matter	High CS water Temp. indication via ICS point T0168A	Diminished heat transfer capability in Train A	None. Two 100% capacity trains including heat exchangers are provided.	Only one train is required to mitigate accident consequences.
28. Containment Spray Heat Exch. B	Cools CS water during recirc. mode	Clogging/tube rupture	Impurities, particulate matter	High CS water Temp. indication via ICS point	Diminished heat transfer capability in Train B	None. Two 100% capacity trains including heat exchangers	Only one train is required to mitigate accident consequences.

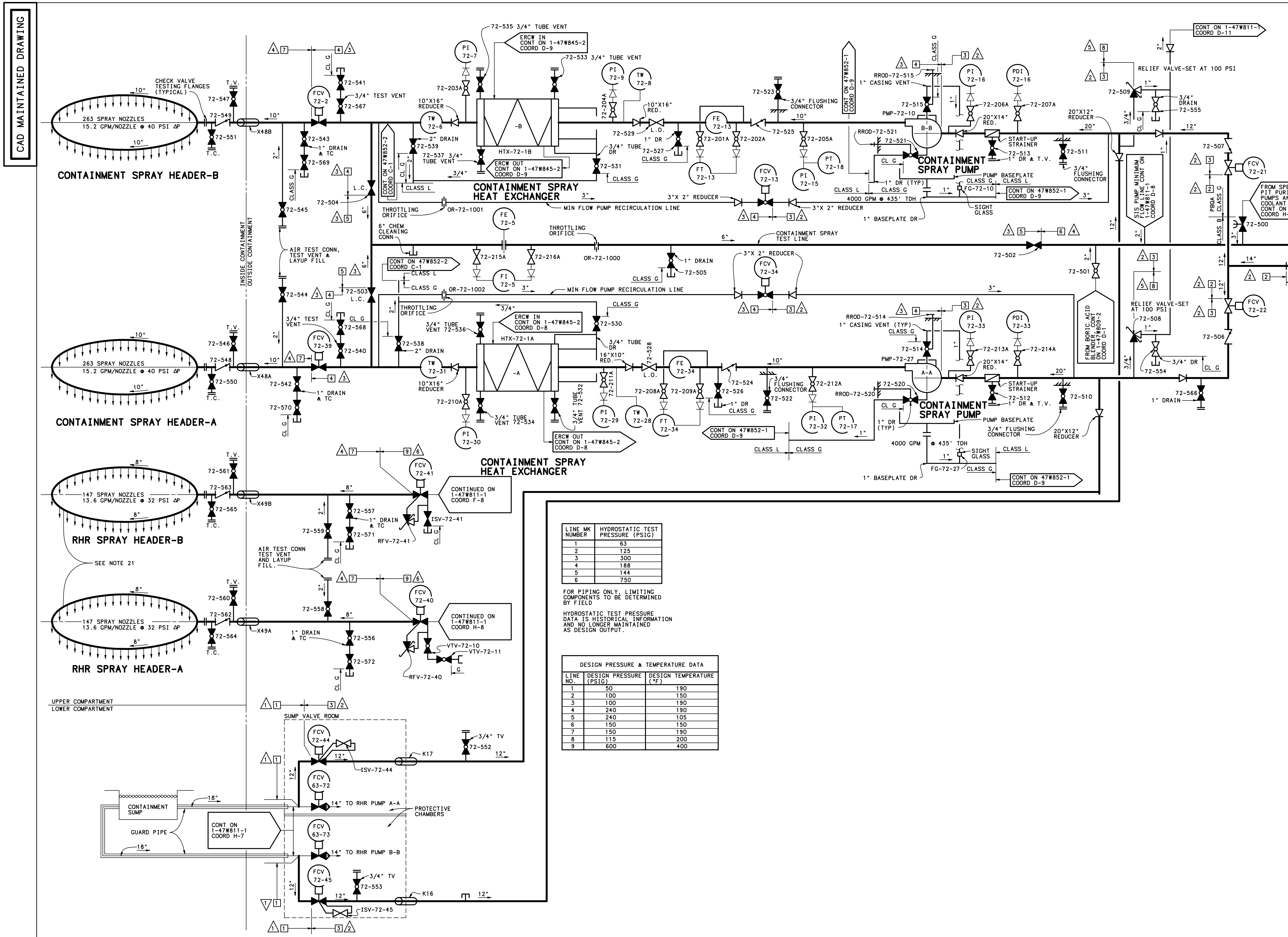
TABLE 6.2.2-3 (Sheet 5 of 5)

FAILURE MODES AND EFFECTS ANALYSIS

ITEM NO. COMPONENT	FUNCTION	FAILURE MODE	POTENTIAL CAUSE	METHOD OF DETECTION	EFFECTS ON SYSTEM	EFFECT ON PLANT	REMARKS
				T0169A.		are provided.	
29. Spray nozzles	Spray water in the containment	Clogged	Impurities, particulate matter.	Reduced pump flow	263 nozzles provided per CSS train. Will diminish cooling capacity slightly.	Redundant trains are provided to fulfill minimum plant requirements.	Nozzles are 3/8 inch dia and Containment Sump strainer perforation size is 0.085 inch, so possibility of clogging is remote.
30. Piping or valve body or pump casing or HX, shell		Ruptures or major leakage		In service Inspection	System capability diminished.	None. Two 100% capacity trains are provided.	Only one train is required to mitigate accident consequences.
31. All valves required for system operation		Disc separated from stem	Mechanical failure		System capability diminished	None. Two 100% capacity trains are provided.	

Note:

* Valve position indication for Valve FCV-72-02 may be derived from indicator lights on all three hand switches (HS-72-2A, HS-72-2B or HS-72-2C). This is unique in that 2B hand switches have been removed from nearly all other applied uses due to EQ considerations.



NOTES:

1. ALL PIPING ON THIS DRAWING IS TVA CLASS B UNLESS OTHERWISE INDICATED.
2. ALL VALVES ARE THE SAME SIZE AS PIPING, UNLESS OTHERWISE NOTED.
3. ALL INSTRUMENT VALVES ARE 3/4" GLOBE UNLESS OTHERWISE NOTED.
4. ALL VALVE NUMBERS AND EQUIPMENT SHALL BE PREFIXED WITH THE UNIT NUMBER, 1-72-525, 1A-A, ETC.
5. ALL PRESSURE GAUGE, TEST & SAMPLING CONNECTIONS ARE 3/4".
6. FOR DEFINITION OF SYMBOLS OF "L.C." OR "L.O.", REFER TO MEMO B26850402001.
7. T.V. & T.C. DENOTES "TEST VENT" & "TEST CONNECTIONS".
8. 1 ETC., INDICATES LINE NUMBER CORRESPONDING TO DESIGN PRESSURE & TEMPERATURE CHART ON THIS SHEET.
9. HYDROSTATIC TESTING SHALL BE IN ACCORDANCE WITH APPLICABLE CODES.
10. THE DESIGN PRESSURE & TEMPERATURE OF ALL DRAIN VENTS AND FLUSH CONNECTIONS THRU THE LAST ISOLATION VALVE SHALL BE THE SAME AS THE PROCESS LINE.
11. 1 ETC., INDICATES LINE NUMBER CORRESPONDING TO HYDROSTATIC TEST PRESSURE CHART ON THIS SHEET. HYDROSTATIC TESTING SHALL BE IN ACCORDANCE WITH APPLICABLE CODE CASES WITH CASE BY CASE APPLICATION REQUIRING PRIOR APPROVAL BY NUCLEAR ENGINEERING. CS & RHR SPRAY HEADERS AND CS PUMP TEST LINE MAY BE HYDROSTATICALLY TESTED IN ACCORDANCE WITH CODE CASE N-237.
12. ALL PIPING DOWNSTREAM OF THE ISOLATION VALVE ON LOCAL VENTS, DRAINS, AND TEST CONNECTIONS IS TVA CLASS G UNLESS OTHERWISE NOTED.
13. IS CONTAINMENT SPRAY PUMP START UP STRAINER SYMBOL. START UP STRAINERS TO BE REMOVED AFTER UNIT 1 CLEANUP AND USED FOR UNIT 2.
14. INDICATES EXTENSION VALVE STEM OPERATOR
15. DESIGN CRITERIA/SYSTEM DESCRIPTION REFERENCE DOCUMENTS (USE THE LATEST REVISION ON ALL WORK UNLESS OTHERWISE SPECIFIED. SEE THE LATEST REVISION OF THE 47B21 SERIES DRAWINGS "PIPING SYSTEM CLASSIFICATION."):
    N3-72-4001-----CONTAINMENT HEAT REMOVAL SPRAY SYSTEMS
16. NOT USED.
17. NOT USED.
18. TO PREVENT BONNET OVERPRESSURIZATION VALVES 1-FCV-72-44 AND 45 HAVE A RELIEF LINE INSTALLED. SEE VENDOR VALVE DRAWING FOR DETAILS.
19. TO PREVENT BONNET OVERPRESSURIZATION VALVES 1-FCV-72-40 AND -41 HAVE A RELIEF LINE INSTALLED. SEE VENDOR VALVE DRAWING FOR DETAILS.
20. VALVES DENOTED AS L.O. ARE LOCKED OPEN DURING NORMAL PLANT OPERATION. VALVES DENOTED AS L.T. ARE LOCKED IN A THROTTLED POSITION DURING NORMAL PLANT OPERATION.
21. SEE SYSTEM DESCRIPTION N3-72-4001 SECTION 3.2.4 FOR UNOBSTRUCTED NOZZLE QUANTITY DESIGN BASIS.

REFERENCE DRAWINGS:
47W800-1 ---------------- FLOW DIAG-GEN PLT SYS
47W610-72 SERIES -------- CONTROL DIAG
47W611-72 SERIES -------- LOGIC DIAGRAM
47W601-72 SERIES -------- INSTR TAB.
47W437 SERIES ----------- C.S. SYSTEM PIPING.
47W812-100 -------------- SERIES STRESS ANALYSIS PROBLEM BOUNDARY
92QL072-SERIES ---------- "Q" LIST
47B437-72X1,-72X2 ------- MASTER VALVE STATUS REPORT.
SD WAT/WBT 906-4 -------- WESTINGHOUSE SPECIFICATIONS.
113E803 ----------------- WESTINGHOUSE C.S. FLOW DIAGRAM.

LINE MK NUMBER	HYDROSTATIC TEST PRESSURE (PSIG)
1	63
2	125
3	300
4	188
5	144
6	750

FOR PIPING ONLY, LIMITING COMPONENTS TO BE DETERMINED BY FIELD

HYDROSTATIC TEST PRESSURE DATA IS HISTORICAL INFORMATION AND NO LONGER MAINTAINED AS DESIGN OUTPUT.

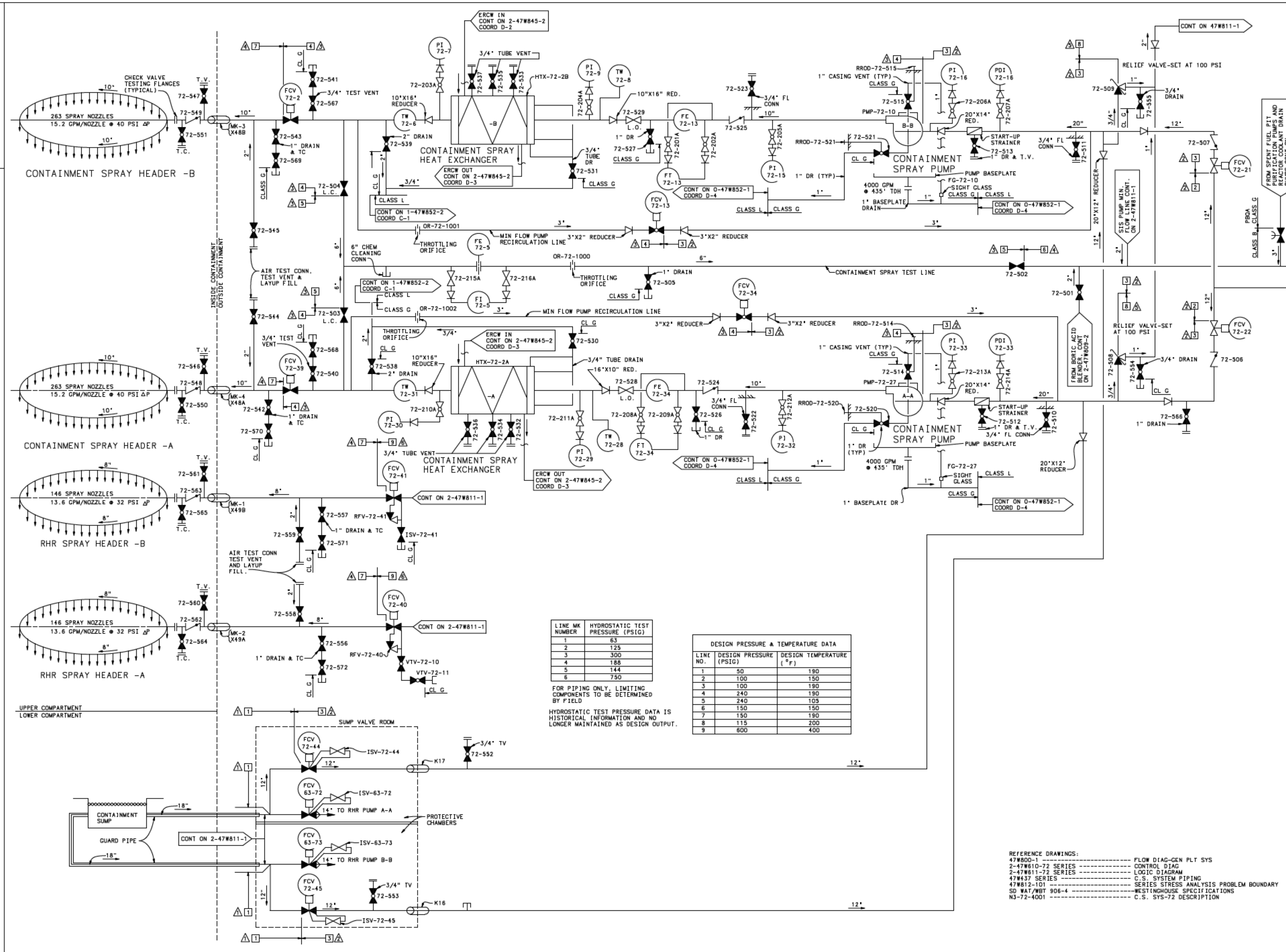
DESIGN PRESSURE & TEMPERATURE DATA		
LINE NO.	DESIGN PRESSURE (PSIG)	DESIGN TEMPERATURE (°F)
1	50	190
2	100	150
3	100	190
4	240	190
5	240	105
6	150	150
7	150	190
8	115	200
9	600	400

WATTS BAR
FINAL SAFETY
ANALYSIS REPORT

POWERHOUSE
UNIT 1

FLOW DIAGRAM
CONTAINMENT SPRAY SYSTEM
TVA DWG NO. 1-47W812-1 R29
FIGURE 6.2.2-1

CAD MAINTAINED DRAWING



LINE MK NUMBER	HYDROSTATIC TEST PRESSURE (PSIG)
1	63
2	125
3	300
4	188
5	144
6	750

FOR PIPING ONLY, LIMITING COMPONENTS TO BE DETERMINED BY FIELD

HYDROSTATIC TEST PRESSURE DATA IS HISTORICAL INFORMATION AND NO LONGER MAINTAINED AS DESIGN OUTPUT.

DESIGN PRESSURE & TEMPERATURE DATA		
LINE NO.	DESIGN PRESSURE (PSIG)	DESIGN TEMPERATURE (°F)
1	50	190
2	100	150
3	100	190
4	240	190
5	240	105
6	150	150
7	150	190
8	115	200
9	600	400

NOTES:
1. ALL PIPING ON THIS DRAWING IS TVA CLASS B UNLESS OTHERWISE INDICATED.
2. ALL VALVES ARE THE SAME SIZE AS PIPING, UNLESS OTHERWISE NOTED.
3. ALL INSTRUMENT VALVES ARE 3/4" GLOBE UNLESS OTHERWISE NOTED.
4. ALL VALVE NUMBERS AND EQUIPMENT SHALL BE PREFIXED WITH THE UNIT NUMBER, 2-72-525, 2A-A, ETC.
5. ALL PRESSURE GAUGE, TEST & SAMPLING CONNECTIONS ARE 3/4". UNLESS OTHERWISE NOTED.
6. FOR DEFINITION OF SYMBOLS OF "L.C." OR "L.O.", REFER TO MEMO B26850402001.
7. T.V. & T.C. DENOTES "TEST VENT" & "TEST CONNECTIONS".
8. 1 ETC., INDICATES LINE NUMBER CORRESPONDING TO DESIGN PRESSURE & TEMPERATURE CHART ON THIS SHEET.
9. HYDROSTATIC TESTING SHALL BE IN ACCORDANCE WITH APPLICABLE CODES.
10. THE DESIGN PRESSURE & TEMPERATURE OF ALL DRAIN VENTS AND FLUSH CONNECTIONS THRU THE LAST ISOLATION VALVE SHALL BE THE SAME AS THE PROCESS LINE.
11. 1 ETC., INDICATES LINE NUMBER CORRESPONDING TO HYDROSTATIC TEST PRESSURE CHART ON THIS SHEET. HYDROSTATIC TESTING SHALL BE IN ACCORDANCE WITH APPLICABLE CODE CASES WITH CASE BY CASE APPLICATION REQUIRING PRIOR APPROVAL BY NUCLEAR ENGINEERING. CS & RHR SPRAY HEADERS AND CS PUMP TEST LINE MAY BE HYDROSTATICALLY TESTED IN ACCORDANCE WITH CODE CASE N-237.
12. ALL PIPING DOWNSTREAM OF THE ISOLATION VALVE ON LOCAL VENTS, DRAINS, AND TEST CONNECTION AND FLUSHING CONNECTION LINES ARE TVA CLASS G UNLESS OTHERWISE NOTED.
13. TO PREVENT BONNET OVER PRESSURIZATION 2-FCV-72-40 AND -41 HAVE A RELIEF LINE INSTALLED. SEE VENDOR VALVE DRAWING FOR DETAILS.
14. TO PREVENT BONNET OVER PRESSURIZATION VALVES 2-FCV-72-44 AND 45 HAVE A RELIEF LINE INSTALLED. SEE VENDOR VALVE DRAWING FOR DETAILS.
15. IS CONTAINMENT SPRAY PUMP START UP STRAINER SYMBOL. START UP STRAINERS TO BE REMOVED AFTER UNIT 2 CLEANUP.
16. INDICATES EXTENSION VALVE STEM OPERATOR
17. DESIGN CRITERIA/SYSTEM DESCRIPTION REFERENCE DOCUMENTS (USE THE LATEST REVISION ON ALL WORK UNLESS OTHERWISE SPECIFIED. SEE THE LATEST REVISION OF THE 47B21 SERIES DRAWINGS "PIPING SYSTEM CLASSIFICATION."):
N3-72-4001-----CONTAINMENT HEAT REMOVAL SPRAY SYSTEMS
18. VALVES DENOTED AS L.O. ARE LOCKED OPEN DURING NORMAL PLANT OPERATION. VALVES DENOTED AS L.T. ARE LOCKED IN A THROTTLED POSITION DURING NORMAL PLANT OPERATION.
19. ALL PIPING ON THIS DRAWING IS TVA CLASS B UNLESS OTHERWISE INDICATED. SEE SYSTEM DESCRIPTION N3-72-4001 SECTION 3.2.4 FOR UNOBSTRUCTED NOZZLE QUANTITY DESIGN BASIS.

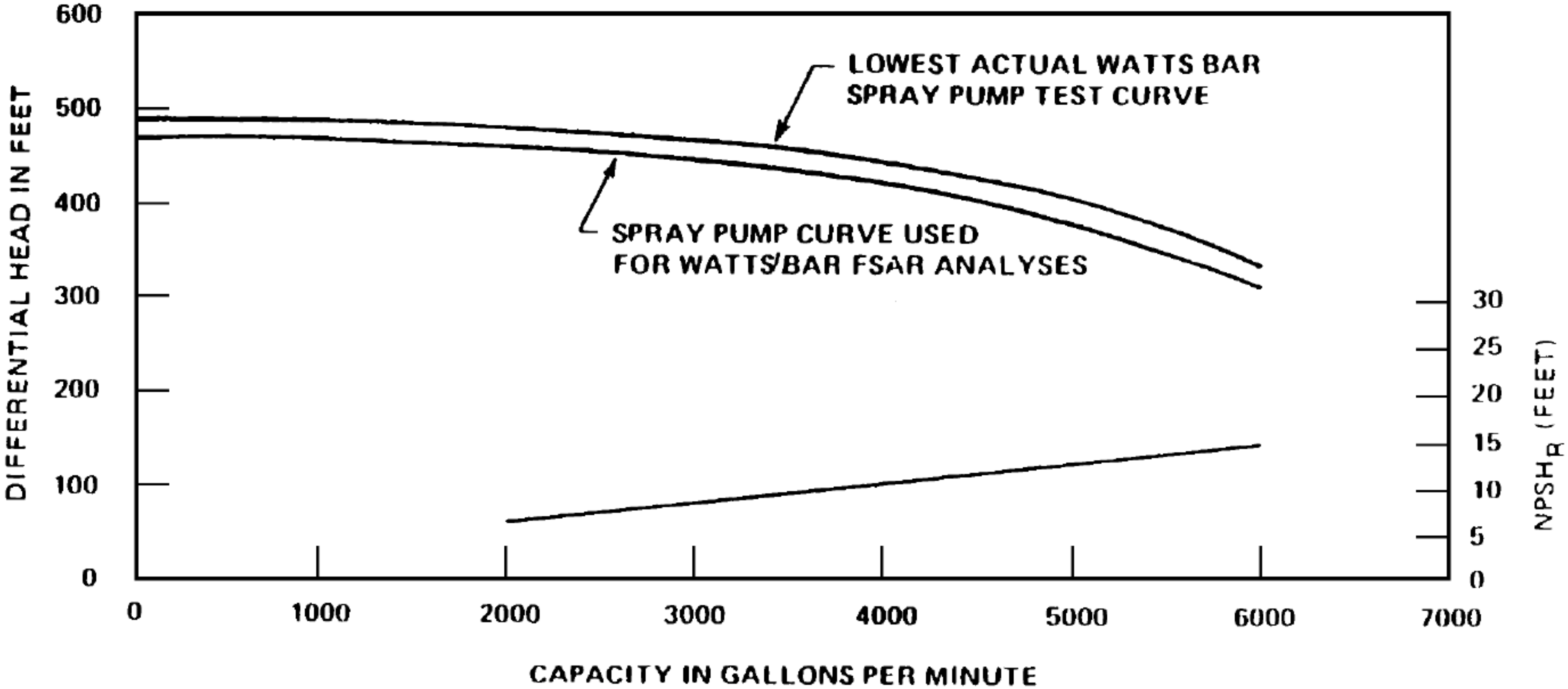
REFERENCE DRAWINGS:
47W800-1 ---------------------------- FLOW DIAG-GEN PLT SYS
2-47W610-72 SERIES ---------------- CONTROL DIAG
2-47W611-72 SERIES ---------------- LOGIC DIAGRAM
47W437 SERIES ---------------------- C.S. SYSTEM PIPING
47W812-101 -------------------------- SERIES STRESS ANALYSIS PROBLEM BOUNDARY
SD WAT/WBT 906-4 -------------------WESTINGHOUSE SPECIFICATIONS
N3-72-4001 -------------------------- C.S. SYS-72 DESCRIPTION

UFSAR AMENDMENT 2

WATTS BAR
FINAL SAFETY
ANALYSIS REPORT

POWERHOUSE
UNIT 2

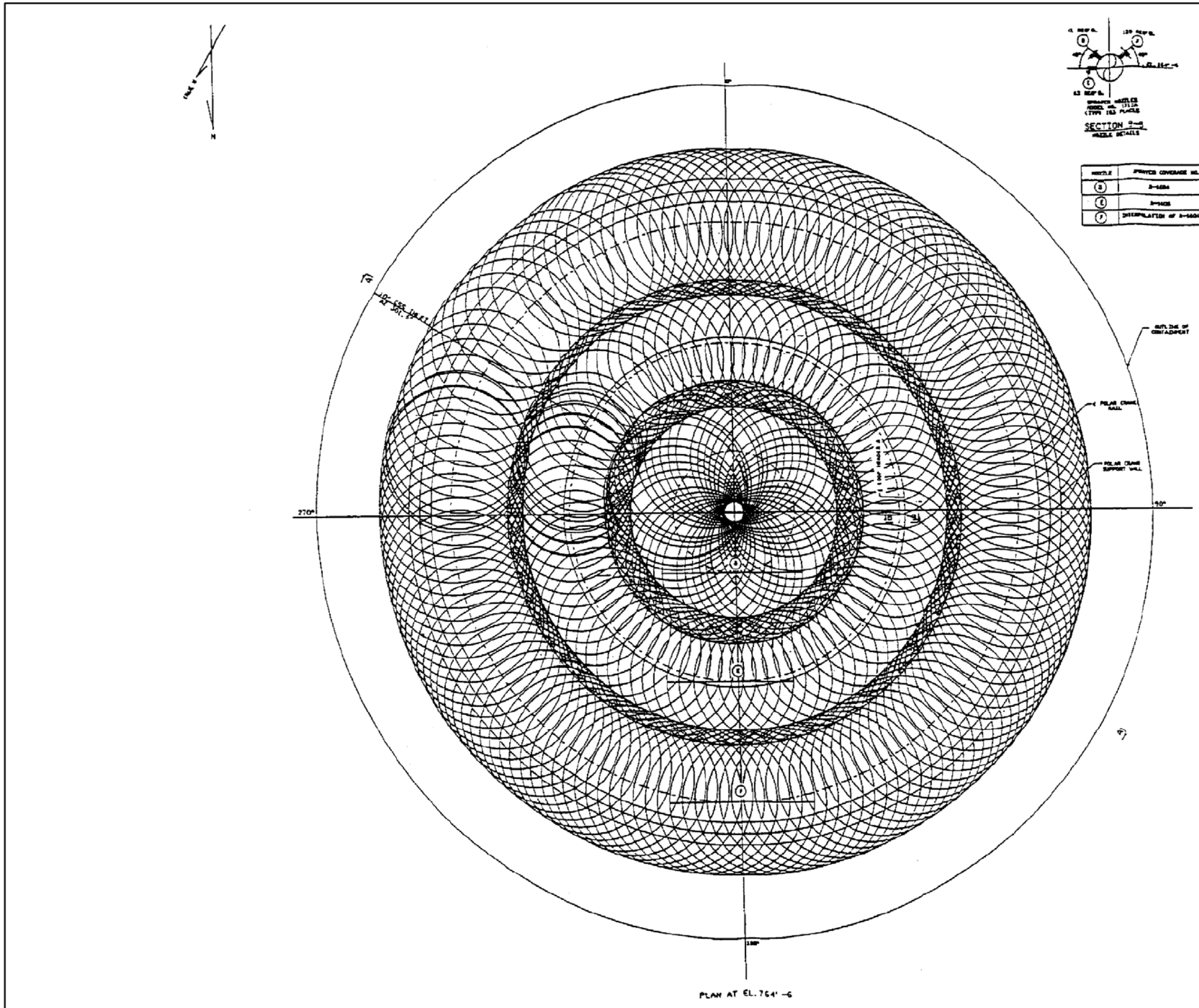
FLOW DIAGRAM
CONTAINMENT SPRAY SYSTEM
TVA DWG NO. 2-47W812-1 R36
FIGURE 6.2.2-1(U2)



WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

Containment Spray Pump
Performance Curves

FIGURE 6.2.2-2



WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

Reactor Buildings Unit 1 & 2
Mechanical Containment Spray System
Piping Plan of Spray Patterns from
C.S. Loop Header A
FIGURE 6.2.2-3

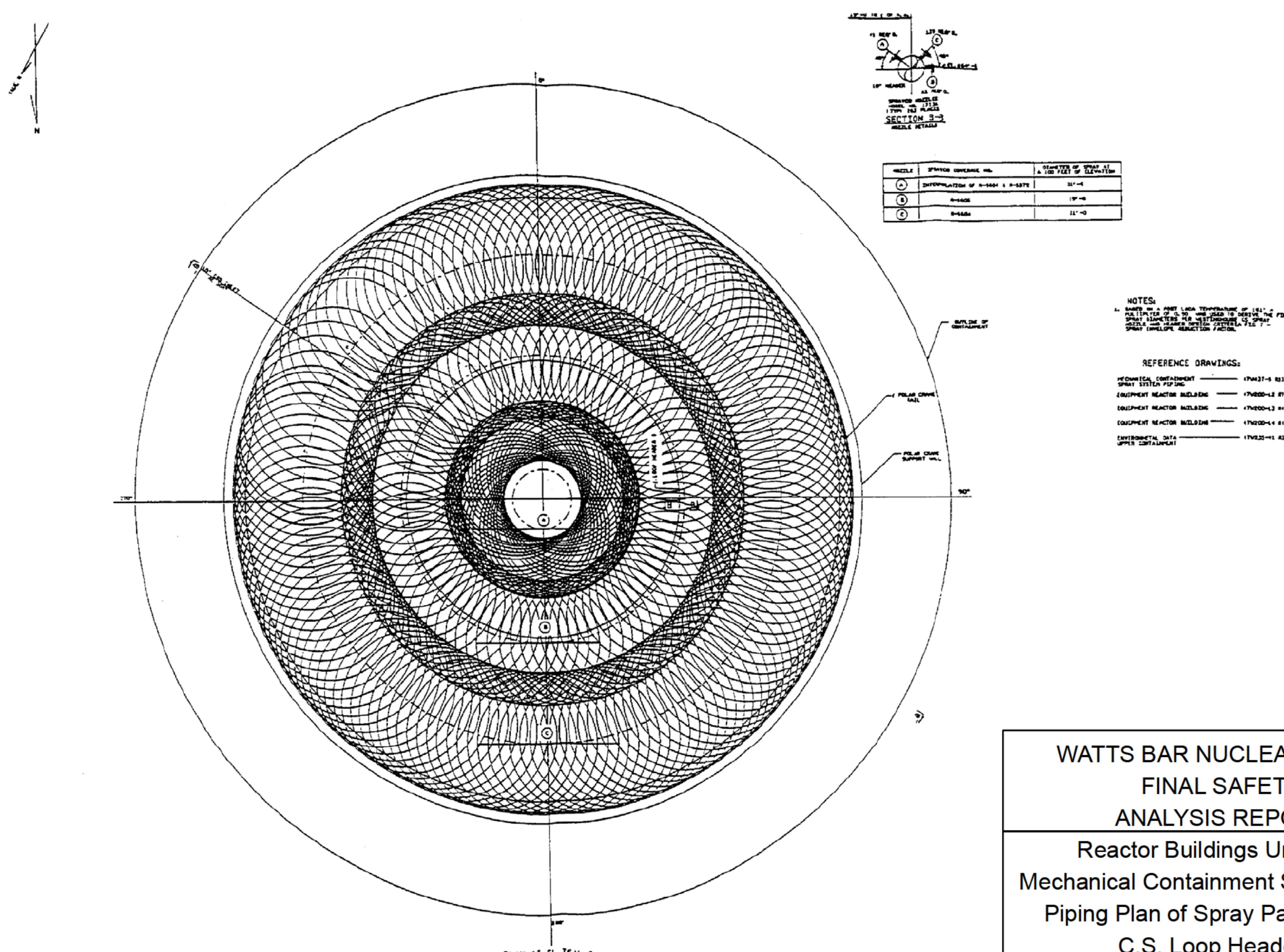
SECURITY-RELATED INFORMATION, WITHHELD UNDER 10CFR2.390

FIGURE 6.2.2-4

SECURITY-RELATED INFORMATION, WITHHELD UNDER 10CFR2.390

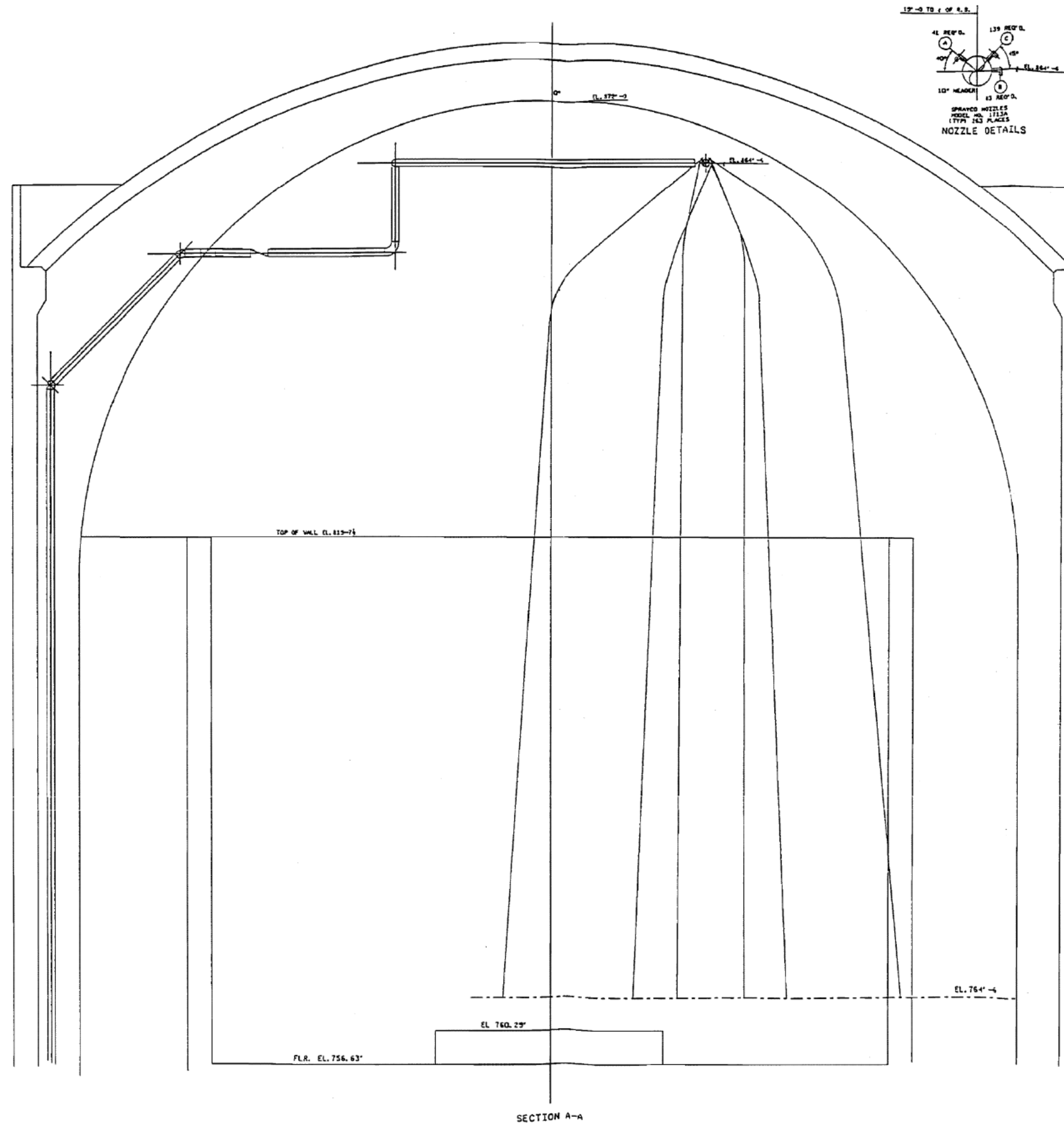
FIGURE 6.2.2-4

UNIT 2



WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

Reactor Buildings Unit 1 & 2
Mechanical Containment Spray System
Piping Plan of Spray Patterns from
C.S. Loop Header B
FIGURE 6.2.2-5



NOTE:

1. FOR NOTES AND REFERENCE DRAWINGS SEE FIG. 6.2.2-5

WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

Reactor Buildings Unit 1 & 2
Mechanical Containment Spray System
Piping Section of Spray Patterns from
C.S. Loop Header B
FIGURE 6.2.2-6

6.2.3 SECONDARY CONTAINMENT FUNCTIONAL DESIGN

Structures included as part of the secondary containment system are the Shield Building of each reactor unit, the Auxiliary Building, the Condensate Demineralizer Waste Evaporator (CDWE) Building, and the essential raw cooling water (ERCW) pipe tunnels adjacent to the Auxiliary Building. Depending on the configuration of the plant, the Primary Containment Building(s) may also be included as a structure which is part of the secondary containment system. This condition exists when the primary containment is open to the Auxiliary Building. The emergency gas treatment system (EGTS) is provided for ventilation control and cleanup of the atmosphere inside the annulus between the Shield Building and the Primary Containment Building. The Reactor Building purge air system is also available for cleaning up the atmosphere inside the Shield Building annulus. Refer to Section 9.4.6 for further details relating to the purge air system. The Auxiliary Building gas treatment system (ABGTS) provides a similar contamination control capability in the Auxiliary Building Secondary Containment Enclosure (ABSCE), which includes all of the areas listed above.

The Interim ABSCE boundary supporting Unit 2 construction does not include Unit 2 primary containment and annulus spaces in the ABSCE. With the Interim ABSCE boundary supporting Unit 2 construction, the sections of the Unit 2 Shield Building that are part of the secondary containment system are identified by Figures 6.2.3-04 through 6.2.3-07. (Unit 1 Only)

The Interim ABSCE boundary supporting Unit 2 construction does not include the spaces within Auxiliary Building Rooms 713-A21 and 757-A14. (Unit 1 Only)

The Interim ABSCE boundary normally includes the space within Auxiliary Building Room 757-A15 except when the Unit 2 Reactor Building Blast Doors are open and Door R002 is closed, and Door R003 is open. This exception is an alternate configuration for moving equipment through the hatch. (Unit 1 Only)

As identified by Figure 9.4-8, the configuration of Auxiliary Building General Ventilation supply branch ducting transiting Auxiliary Building Room 757 A-14 provides support for the Interim ABSCE boundary functions of the floor and east wall of Room 757-A14. (Unit 1 Only)

6.2.3.1 Design Bases

6.2.3.1.1 Secondary Containment Enclosures

Design bases for the secondary containment structures were devised to assure that an effective barrier exists for airborne fission products that may leak from the primary containment or the Auxiliary Building fuel handling area, during a loss-of-coolant accident (LOCA). The original design credited the secondary containment enclosures to mitigate the consequences of a fuel handling accident (FHA). Based on the use of the Regulatory Guide 1.183, Revision 0 (Alternate Source Term) methodology for a FHA, the structures are no longer required for mitigation of a postulated FHA. Within the scope of these design bases are requirements that influence the size, structural integrity, and leak tightness of the secondary containment enclosure. Specifically, these include a capability to: (a) maintain an effective barrier for gases and vapors that may leak from the primary containment during all normal and abnormal events; (b) delay the release of any gases and vapors that may leak from the primary containment during accidents; (c) allow gases and vapors that may leak through the primary containment

during accidents to flow into the contained air volume within the secondary containment where they are diluted, held up, and purified prior to being released to the environs; (d) bleed to the secondary containment each air-filled containment penetration enclosure which extends beyond the Shield Building and which is formed by automatically actuated isolation valves; (e) maintain an effective barrier for airborne radioactive contaminants, gases, and vapors originating in the ABSCE during normal and abnormal events.

Refer to Sections 3.8.1 and 3.8.4 for further details relating to the design of the Shield Building and the Auxiliary Building.

6.2.3.1.2 Emergency Gas Treatment System (EGTS)

The design bases for the EGTS are:

1. To keep the air pressure within each Shield Building annulus below atmospheric pressure at all times in which the integrity of that particular containment is required.

2. To reduce the concentration of radioactive nuclides in annulus air that is released to the environs during a LOCA in either reactor unit to levels sufficiently low to keep the site boundary and low population zone (LPZ) dose rates below the 10 CFR 100 values.

3. To withstand the safe shutdown earthquake.

4. To provide for initial and periodic testing of the system capability to function as designed.

6.2.3.1.3 Auxiliary Building Gas Treatment System (ABGTS)

The design bases for the ABGTS are:

1. To establish and keep an air pressure that is below atmospheric within the portion of the buildings serving as a secondary containment enclosure during accidents.

2. To reduce the concentration of radioactive nuclides in air releases from the secondary containment enclosures to the environs during accidents to levels sufficiently low to keep the site boundary and LPZ dose rates below the 10 CFR 100 guideline values.

3. To minimize the spreading of airborne radioactivity within the Auxiliary Building following an accidental release in the fuel handling and waste packaging areas. ABGTS is not required to mitigate the consequences of a fuel handling accident.

4. To withstand the safe shutdown earthquake.

5. To provide for initial and periodic testing of the system capability to function as designed. (See Chapter 14.0 for information on initial testing of systems).

6.2.3.2 System Design

6.2.3.2.1 Secondary Containment Enclosures

1. <u>Shield Building</u>

The principal components that function collectively to form a secondary containment barrier around the steel primary containment vessel are the Shield Building itself, the Shield Building penetration seals, the isolation valves installed in the penetrations to the Shield Building, and the Shield Building penetration leakoff facilities.

With the Interim ABSCE boundary supporting Unit 2 construction, the sections of the Unit 2 Shield Building that are part of the secondary containment system are identified by Figures 6.2.3-04 through 6.2.3-07. (Unit 1 Only)

Structure

The Shield Building is a reinforced concrete structure that encloses the reactor's steel primary containment structure; it has a circular horizontal cross section and a shallow domed roof. The vertical center line of this building is also the vertical center line of the steel primary containment vessel. The inside diameter of this building was sized to provide an annular shaped air space between the two reactor enclosures that is approximately five feet wide. The total enclosed free air space between the two enclosures is approximately 396,000 cubic feet.

Penetrations

To ensure that the Shield Building provides a nearly leak tight enclosure for the primary containment structure all openings in the shield building penetrations are sealed. Typical mechanical piping or ventilation penetrations are equipped with a flexible membrane seal as shown in Figure 6.2.3-1. Mechanical penetrations are designed for a leakage rate no greater than 0.0052 cfm per square inch when secondary containment is at a minus 0.5 inch water gauge. The primary containment personnel hatch passes through the Shield Building and opens directly to the Auxiliary Building. This opening in the Shield Building wall is handled as an ordinary piping penetration and is provided with a flexible, double membrane seal as shown in Figure 3.8.2-5 (see Section 3.8.2). Personnel and equipment access doors to the secondary containment are treated as special cases and are provided with resilient seals as shown in Figures 3.8.4-21 through 3.8.4-23. (See Section 3.8.4 for descriptions of the personnel access doors and the equipment access doors.) The allowable leakage for each personnel access door is 0.5 cfm when secondary containment is at minus 1 inch water gauge, and 10 cfm for the equipment access hatch when secondary containment is at minus 0.75 inch water gauge.

With the Interim ABSCE boundary supporting Unit 2 construction, the sections of the Unit 2 Shield Building that are part of the secondary containment system are identified by Figures 6.2.3-04 through 6.2.3-07. Corresponding mechanical and electrical penetrations are qualified to support ABSCE boundary functions. (Unit 1 Only)

Air filled lines which must be isolated by automatic valve actuation and which penetrate both the primary containment and the shield building are considered more likely to pass airborne radioactivities than other lines. Therefore these lines are provided with a third

isolation valve outside the secondary containment for additional leak protection. This single, third valve receives both Train A and B actuation signals. Buffering prevents an electrical failure in one train from affecting the performance of the other train. To enhance the effectiveness of the third isolation valve as a barrier to leakage, the enclosed volume between the second and third isolation valves is open to the annulus at all times. Opening this enclosed space to the annulus is accomplished with leak off facilities as shown in Figure 6.2.3-2. This allows the negative pressure in the annulus to include this small volume, and leakage from the primary containment outward or leakage from outside the Shield Building inward to be drawn into the annulus for processing. The lines provided with this feature are those for the primary containment purge supply and exhaust. The 8-inch lower compartment pressure relief line is excluded from this consideration since it terminates in the annulus through a filter train and does not go outside the Shield Building.

Electrical penetrations are of either a cable tray/cable slot type or a conduit type. Typical seals for these penetrations are shown in Figure 6.2.3-3. For cable tray/cable slot penetrations, silicone room temperature vulcanizing (RTV) foam is used as the sealant around cables within the wall opening over a portion of the length of the cable slot penetration. In conduit penetrations, the interstitial spaces between cables and conduit or condulet walls are filled with RTV silicone rubber as the sealant over a portion of the length of the penetration. Electrical penetrations are designed for a leakage rate not greater than 0.014 cfm per square inch when secondary containment is at a minus 0.5 inch water gauge.

The total expected infiltration rate across all leakage paths into the annulus is 250 cfm at the post accident annulus control setpoint for a postulated single failure of one EGTS train (see section 6.2.3.3.2 for a discussion of infiltration rate associated with a postulated single failure of one pressure control train). During normal operation, the annulus pressure is maintained at or more negative than 5.0 inches water gauge with respect to the outside atmosphere. Periodic tests demonstrate that inleakage is less than this value.

The fraction of primary containment leakage which may bypass the Shield Building and go directly to the Auxiliary Building is specified to be no greater than 25% of the total primary containment out-leakage. Permitting this leakage fraction results in acceptable site boundary and LPZ doses for the LOCA condition as described in Chapter 15. There are no significant paths by which primary containment leakage may bypass both the Shield Building and the Auxiliary Building, and result in exceeding the offsite dose limits.

Information concerning isolation features utilized in support of the secondary containment is presented in Section 6.2.4. Potential leakage paths by which primary containment leakage could bypass the secondary containment, and measures utilized to prevent such leakage are also discussed in Section 6.2.4.

2. <u>Auxiliary Building</u>

Structure

The Auxiliary Building provides an isolation barrier during certain postulated accidents

involving airborne radioactive contamination. Certain of the buildings interior and exterior walls, floor slabs, and a part of its roof form the isolation barrier as shown in Figures 6.2.3-4 through 6.2.3-9. The enclosed volume of this isolation barrier, the ABSCE, is approximately 6.9×10^6 cubic feet. The ABSCE is discussed in more detail in Section 3 below. Additional data on the Auxiliary Building is provided in Section 3.8.4 and Table 6.2.3-1.

3. Auxiliary Building Secondary Containment Enclosure (ABSCE)

The ABSCE is that portion of the Auxiliary Building and CDWE Building (and for certain configurations, the annulus and primary containment, as discussed below) which serves to maintain an effective barrier for airborne radioactive contaminants released in the auxiliary building during abnormal events. Mechanical and electrical penetrations of this enclosure are provided with seals to minimize infiltration. Piping penetrations are either analyzed to pressure boundary retention requirements, or the effects of their failure are demonstrated not to impair the ability of the ABGTS system to maintain the ABSCE under the required negative pressure of 0.25 inches w.g., or they are isolated by physical means (e.g., locked-closed valves, etc). Airlock-type doors are provided at portals where needed by the frequency of use. The negative pressure is maintained within the ABSCE to ensure that no contaminated air is released to the environs following an abnormal event without first being processed by the ABGTS. The ABSCE is shown in Figures 6.2.3-4 though 6.2.3-9.

Entrances and exits to those portions of the Auxiliary Building within the primary containment barrier for both equipment and personnel are through air locks. The air lock locations are shown in Figures 1.2-3 and 1.2-5. The doors in each air lock are electrically interlocked such that only one side of the air lock can be opened at a time. A control room alarm is provided should both sides of an air lock ever be opened simultaneously. As a safety precaution, an interlock defeat switch is mounted on the containment side of each air lock to allow emergency egress should either side of the air lock be blocked open in an accident.

A special case is the interlock system for the large exterior door to the railway loading area. When the large railroad access door is open, ABSCE airlock doors, such as the doors to the Post Accident Sampling Facility (PASF) and nitrogen storage area nor the access hatches above can be opened, and when either of these two doors and either of the access hatches above are open, the large railway access door cannot be opened. Administrative controls are provided to control doors and hatches during normal operation. The railway access doors and hatches are described in Section 3.8.4.

A special case is the interim ABSCE air lock located in the Unit 2 Reactor Building Equipment Hatch. When ABGTS is required to be operable, at least one of the two fabric roll-up doors is in the closed, sealed position. This interim ABSCE configuration is described in Section 3.8.4. (Unit 1 Only)

The total permissible leakage rate for the ABSCE at a pressure of -0.25 inches water gauge with respect to the outside is 9900 cfm maximum. This represents 165.5% of the ABSCE free volume per day. Periodic tests demonstrate that inleakage is less than the design value. Any improvement in the inleakage recorded during subsequent tests,

shall be used to establish the allowable margin for breaching permits.

UNIT 1

During periods when the primary containment and/or annulus are open to the Auxiliary Building, the ABSCE also includes these areas. During fuel handling operations in this configuration, a High Radiation in Refueling Area (HRRA) signal from spent fuel pool radiation monitors, Containment Isolation Phase A (SI signal), high temperature in the Auxiliary Building (AB) air intake, or manual Auxiliary Building Isolation (ABI) will result in a Containment Ventilation Isolation (CVI) and ABGTS start in addition to the HRRA. In addition, with the unit in refuel mode, an ABI signal or HRRA will initiate shutdown and isolation of the containment purge system independently of the CVI signal. Similarly, a CVI signal, including a CVI signal generated by a high radiation signal from the containment purge air exhaust radiation monitors, will initiate a HRRA and start of ABGTS. These actions will ensure proper operation of the ABSCE. Both doors of the containment vessel personnel airlocks may be open at the same time during refueling activities while the purge air ventilation system is operating. Under special administrative controls, one of the airlock doors at each location will be closed and the purge air ventilation system will be shutdown and isolated in a timely manner subsequent to a fuel handling accident to ensure ABSCE boundary integrity. Although the ABGTS is available to minimize the consequences of a fuel handling accident, it is not required to function in order to meet control room and offsite dose limits based on the use of the Regulatory Guide 1.183, Revision 0 (Alternate Source Term) methodology. The Interim ABSCE boundary supporting Unit 2 construction does not include Unit 2 primary containment and annulus spaces in the ABSCE. Installed Unit 2 Containment Purge components are configured to support the interim ABSCE boundary. The Interim ABSCE boundary supports the ABGTS Train B exhaust path via the Unit 2 Shield Building Vent.

UNIT 2

During periods when the primary containment and/or annulus of both units are open to the Auxiliary Building, the ABSCE also includes these areas. A Containment Isolation Phase A (SI Signal) from the operating unit or high temperature in the Unit 1 or Unit 2 Auxiliary Building air intake, or manual ABI will cause a CVI signal in the refueling unit. These actions will ensure proper operation of the ABSCE. Both doors of the containment vessel personnel airlocks may be open at the same time during refueling activities while the purge air ventilation system is operating. During fuel handling operations in this configuration, a high radiation signal from spent fuel pool radiation monitors will result in a Containment Ventilation Isolation (CVI) in addition to an Auxiliary Building isolation and ABGTS start. Similarly, a CVI signal, including a CVI signal generated by a high radiation signal from the containment purge air exhaust radiation monitors, will initiate an Auxiliary Building isolation and start of ABGTS. These are not required functions for the ABGTS and the Reactor Building Purge Systems filters or for Purge System isolation as no credit for these features is in mitigating a fuel handling accident. In the case where containment of both units is open to the Auxiliary Building spaces, a CVI in one unit will initiate a CVI in the other unit in order to maintain those spaces open to the ABSCE.

Doors and penetrations of the ABSCE perimeter are provided with seals to reduce infiltration. Doors and hatches entering the area are either locked or under

administrative control.

Automatic redundant isolation dampers are provided in ducts which pass from areas inside the ABSCE to areas outside of the enclosure. These permit isolation of the ABSCE and allow the ABGTS to maintain a negative pressure in the area following an abnormal event.

As identified by Figure 9.4-8, the configuration of Auxiliary Building General Ventilation supply branch ducting transiting Auxiliary Building Room 757-A14 provides support for the Interim ABSCE boundary functions of the floor and east wall of Room 757-A14. The duct section does not include any intake air flow path from external to the ABSCE. The duct section does not include any exhaust air flow from a space within ABSCE to the external environment. Consistent with Section 9.4.3.3, the application of the duct section to support Interim ABSCE introduces no requirement for any installation or application of any isolation dampers. (Unit 1 Only)

The ABGTS maintains a negative pressure with respect to the outside in the ABSCE during emergency operation and processes the Auxiliary Building exhaust. Either train of the ABGTS may be used to maintain the negative pressure and treat air exhausted from the ABSCE.

The annulus vacuum control subsystem continues to operate whenever the ABSCE is isolated, except for a Phase A containment isolation signal. A Phase A containment isolation signal which is generated by a LOCA starts the air cleanup subsystem of the emergency gas treatment system. Calculations have shown that this condition does not result in exceeding the limits given in 10 CFR 100. For additional description of the annulus vacuum control subsystem of the EGTS, see Section 6.2.3.2.2.

Proper actuation of the isolation dampers associated with the ABSCE and operation of the ABGTS is confirmed periodically.

6.2.3.2.2 Emergency Gas Treatment System (EGTS)

The EGTS is shown schematically in Figure 6.2.3-11. The logic and control diagrams for this system are shown in Figures 6.2.3-12 to 6.2.3-15a. This system has two subsystems; the annulus vacuum control subsystem and the air cleanup subsystem. The portions of the EGTS necessary to ensure that the system performs its functions during post-accident operation are classified Seismic Category I. Portions of the system which are not necessary for post-accident operation are seismically qualified to the extent that the system does not adversely affect any safety-related system(s)/components(s) as a result of its failure due to the seismic event; that is, qualified Seismic Category I(L).

Annulus Vacuum Control Subsystem

The annulus vacuum control subsystem is a fan and duct network used to establish and keep a negative pressure level within the annular space between the primary and secondary containment structures. It is utilized during normal operations in which containment integrity is required. In emergencies in which containment isolation is required, this subsystem is isolated and shut down. This subsystem performs no safety-related function. Because of this, the

annulus vacuum control subsystem is not classified as an engineered safety feature.

This subsystem has two independently controlled branches. Each branch serves one reactor unit. These branches draw air from their assigned annuli and release it into the Auxiliary Building Fuel Handling Area exhaust duct system. The air inlet for each branch is centrally located in its respective secondary containment volume above the steel containment dome. The Unit 2 annulus vacuum control subsystem is isolated by means of blank-off plates located in the common portions of ductwork to ensure the independent operation of branch serving the Unit 1 annulus.

Air pressure control in each secondary containment annulus is achieved with one of the two, 100% capacity redundant fans, a differential pressure sensor, motor operated damper and control circuitry installation incorporated into each branch. This equipment provides a capability to vary the volumetric flow rate drawn from the annulus to keep the pressure at a predetermined negative pressure level. This control function is accomplished with a modulating damper under control of a differential pressure sensor that adjusts the amount of outside air introduced upstream of a constant capacity fan in the proper manner to keep the annulus pressure within a designated narrow range. One of the two redundant units starts automatically in the event the operating control unit fails to function in the proper manner.

The fans and flow control dampers serving the reactor secondary containment annulus are installed in the EGTS room at Elevation 757.0' adjacent to the Unit 2 Shield Building.

The nominal negative pressure for the annulus vacuum control equipment installation is 5-inches of water gauge. The controller setpoint is established such that the annulus pressure is maintained at, or more negative than, 5 inches water gauge. This negative pressure level, chosen for normal operation, ensures that the annulus pressure will not reach positive values during the annulus pressure surge produced by a LOCA in the primary containment. Two 100% capacity fans are utilized to maintain this negative pressure. One fan is normally on standby.

Annulus vacuum control subsystem aids in containment pressure relief by exhausting the containment vent air after it goes through the containment vent air clean up units and is discharged into the annulus then into the Auxiliary Building exhaust stack (See Section 9.4.6).

Air Cleanup Subsystem

The air cleanup subsystem is a redundant, shared airflow network having the capability to perform two functions for the affected reactor secondary containment during a LOCA. One of these is to keep the secondary containment annulus air volume below atmospheric pressure. The second function is to remove airborne particulates and vapors that may contain radioactive nuclides from air drawn from the annulus. Each of these is accomplished by this subsystem without disturbing operation of the unaffected reactor unit.

Both of these functions are performed by processing and controlling a stream of air taken from the reactor unit secondary containment annulus. The air cleanup operation is conducted by drawing the air stream through a series of filters and adsorbers. Annulus air pressure control is accomplished by adjusting the fraction of the airstream that is returned to the annulus air space. The EGTS ductwork which is connected to the Unit 2 annulus is isolated from the air cleanup

units by means of locked-closed isolation valves and blank-off plates.

The negative pressure control setpoint chosen for post-accident operation is low enough that leakage across the boundary is into the annulus from both the primary containment and areas adjacent to the Shield Building. The pressure differentials produced by wind effects are also overcome by appropriate selection of the annulus negative pressure level.

The rated capacity of each redundant air cleanup unit in the subsystem is 4000 cfm. This subsystem of the EGTS is classified as an engineered safety feature.

The air flow network for the air cleanup subsystem was designed to provide the redundant services needed for either reactor secondary containment annulus. The suction-side ductwork in this network which is used to bring annulus air to the EGTS room on Elevation 757.0 in the Auxiliary Building is also used by the annulus vacuum control subsystem. The intake is centrally located within the Shield Building above the steel containment dome. Within the EGTS room the network branches out in a manner to supply two air cleanup unit installations that can be aligned with flow control dampers to serve either annulus air volume. After the air is processed through the air cleanup subsystem, it is routed to the redundant damper-controlled flow dividers in the annulus. Inside each annulus, the air flow network contains two air flow paths; one leading to the Shield Building vent and the other to a manifold that distributes and releases the air uniformly around the bottom of the annulus.

The vertical separation between the intake above the dome and the exhaust ports in the manifold is approximately 168.0 feet. Butterfly valves, rather than dampers, are installed in the ducts just above the flow distribution manifold to minimize the outside air inleakage from the reactor unit vents into the annulus.

Another feature incorporated into the air cleanup subsystem air flow network is the capability to cool the filters and adsorbers in an inactive air cleanup unit that is loaded with radioactive material. This is accomplished with two cross-over flow ducts that can draw air at a minimum of 200 cfm from the active air cleanup unit through the inactive air cleanup unit. (Such an air flow is sufficient to keep the temperature rise through a fully loaded inactive air cleanup unit to less than 75°F.) Two butterfly valves in series are installed in each cross-over air flow path to assure sufficient isolation to perform accurate removal efficiency tests on the HEPA filter and carbon adsorber banks. This feature is provided in the event excessive adsorber bed temperature occurs following the failure of an operating EGTS train. Adsorber bed temperature is indicated and supplied to the plant process computer. Status indication of each EGTS train is also provided. Upon failure of an operating EGTS train, adsorber bed temperature is monitored to detect subsequent temperature rise. After a Phase A containment isolation signal has initiated EGTS operation, the control room operator will shut one of the two EGTS trains down and align the appropriate butterfly valves for automatic operation. In addition, the associated suction valve is remotely opened from the main control room to establish a flow path from the annulus through the air cleanup unit.

The two air cleanup units in the air cleanup subsystem have stainless steel housings containing air treatment equipment, sampling ports, heaters, drains, test fittings, and access facilities for maintenance. See Section 6.5.1.2.1 for a description of the air cleanup units and information related to their design.

A centrifugal fan is provided outside each air cleanup unit housing. These fans were designed to function in process air flow streams at temperatures up to 200°F. See Table 6.2.3-1 for additional information on these fans.

Two air flow control modules are also included in the air cleanup subsystem. Each consists of a differential pressure sensor and transmitter, control circuitry, a damper actuator, and two modulating dampers. The single damper actuator adjusts the dampers simultaneously in opposite directions, i.e., one is closed as the other is opened.

A pressure control system with a station located in the main control room, modulates pressure control dampers in the annulus to maintain the differential pressure setpoint. Two sets of independent pressure control dampers installed in the secondary containment annulus provide the capability to adjust the amount of air recirculated to the reactor unit annulus or discharged to the shield building exhaust vent. Annulus pressures that are more positive than the pressure controller setpoint produce a signal causing the damper actuator to begin closing the damper controlling the air flow to the annulus and to start opening the damper controlling the air flow to the Shield Building exhaust vent. Annulus pressures that are more negative than the pressure controller setpoint initiate the opposite kind of damper motions.

Four isolation valves (two valves per train) installed in the secondary containment annulus provide isolation of the pressure control dampers. Each train has a handswitch in the control room positioned so that both trains of isolation valves are in A AUTO. A containment isolation Phase A signal causes the valves to open. The open isolation valves provide a flow path for both trains of pressure control dampers which modulate to control annulus pressure. For further details, see Figures 6.2.3-14 and 6.2.3-15.

Operation of the air cleanup subsystem during accidents is initiated by the Phase A containment isolation signal. Both the A and the B trains start and damper alignment initiated by this signal. A capability is also provided to start each fan with a hand switch in the main control room. Damper alignment is also initiated by the same signal. Another adjustment of hand switches in the main control room changes the operating mode to the single train operation with the redundant fan in a standby status. Employment of this operating mode is required after 1 hour and before 2 hours of operation. The control room operator can select either train to remain in operation.

6.2.3.2.3 Auxiliary Building Gas Treatment System (ABGTS)

The ABGTS is a fully redundant air cleanup network provided to reduce radioactive nuclide releases from the secondary containment enclosure during accidents. It does this by drawing air from the fuel handling and waste packaging areas through ducting normally used for ventilation purposes to air cleanup equipment, and then directing this air to the reactor unit vent. In doing so, this system draws air from all parts of the ABSCE to establish a negative pressure region in which virtually no unprocessed air passes from this secondary containment enclosure to the atmosphere.

UNIT 1

During periods when the primary containment and/or annulus are open to the Auxiliary Building, the ABSCE also includes these areas. (The Interim ABSCE boundary supporting Unit 2

construction does not include Unit 2 primary containment and annulus spaces in the ABSCE.) The ABGTS has been designed to establish a negative pressure in these additional areas for this configuration. During fuel handling operations in this configuration, a High Radiation in Refueling Area (HRRA) signal from the spent fuel pool radiation monitors, Containment Isolation Phase A (SI signal), high temperature in the Auxiliary Building (AB) air intake, or manual Auxiliary Building Isolation (ABI) will result in a Containment Ventilation Isolation (CVI) and ABGTS start in addition to the HRRA. In addition, with the unit in refuel mode, an ABI signal or HRRA will initiate shutdown and isolation of the containment purge system independent of the CVI signal. Similarly, a CVI signal, including a CVI signal generated by a high radiation signal from the containment purge air exhaust radiation monitors, will initiate a HRRA and start of ABGTS. These actions will ensure proper operation of the ABSCE. However, as an added precaution to protect the ABGTS operational boundary, operator action is needed to ensure the closure of the containment purge exhaust isolation valves which are controlled by hand switches.

UNIT 2

During periods when the primary containment and/or annulus of both units are open to the Auxiliary Building, the ABSCE also includes these areas. The ABGTS has been designed to establish a negative pressure in these additional areas for this configuration. During fuel handling operations in this configuration, a high radiation signal from the spent fuel pool radiation monitors will result in a Containment Ventilation Isolation (CVI) in addition to an Auxiliary Building isolation and ABGTS start. Similarly, a CVI signal, including a CVI signal generated by a high radiation signal from the containment purge air exhaust radiation monitors, will initiate an Auxiliary Building isolation and start of ABGTS. Likewise, a Containment Isolation Phase A (SI Signal) from the operating unit or high temperature in the Unit 1 or Unit 2 Auxiliary Building air intake, or manual ABI will cause a CVI signal in the refueling unit. These actions will ensure proper operation of the ABSCE. However, as an added precaution to protect the ABGTS operational boundary, operational action is needed to ensure the closure of the containment purge exhaust isolation valves (system valves not containment isolation valves) which are controlled by hand switches. In the case where containment of both units is open to the Auxiliary Building spaces, a CVI in one unit will initiate a CVI in the other unit in order to maintain those spaces open to the ABSCE.

Although the ABGTS is available to minimize the consequences of a fuel handling accident, it is not required to function in order to meet control room and offsite dose limits based on the use of the Regulatory Guide 1.183, Revision 0 (Alternate Source Term) methodology.

The portions of the ABGTS that are required to ensure that the system functions properly are classified Seismic Category I. Certain portions of the system, which are not required for post-accident operation of the ABGTS, are seismically qualified to the extent that system operation is not adversely affected should they fail due to the seismic event (i.e., classified according to Seismic Category I(L)).

The rated capacity of each redundant ABGTS air cleanup unit (ACU) is 9000 cfm. ACUs are designed in accordance with engineered safety feature standards.

The unique portions of the ABGTS are shown schematically in Figures 6.2.3-16 and 9.4-8. Logic and control diagrams for the ABGTS are shown in Figures 9.4-10 and 9.4-17. The airflow network for this system consists of the exhaust ductwork, which normally serves the fuel handling and waste packaging areas, connected to the suction-side of the Train A ACU, and a

cross-over duct which connects the ductwork with the suction-side of the Train B ACU. The discharges from the Train A and Train B ABGTS exhaust fans separately go to the Unit 1 and Unit 2 Shield Building exhaust vents, respectively.

The air flow ductwork that is not unique to this system consists of some of the normal ventilation ducting installed in the ABSCE. When the ABSCE is isolated, this duct network provides a flow path for the ABGTS to draw down and maintain a negative pressure in ABSCE. In some instances, air is drawn in the opposite direction to the normal air flow pattern during operation of the ABGTS. Each ACU includes air treatment components instrumentation, test fittings, and other equipment required for proper operation and testing of the system. Refer to Section 6.5.1.2.2 for a description of the ABGTS air cleanup units and information related to their design. Information on these fans is given in Table 6.2.3-1.

The air flow control modules utilized in the ABGTS contain a differential pressure sensor and transmitter, control circuitry, and a modulating damper. These air flow control modules provide the capability for keeping the pressure within the ABSCE at a minimum of 1/4 inch water gauge below atmospheric with the differential pressure transmitter and differential pressure controller circuit adjusting the amount of outside air admitted into the duct network by the vacuum relief line. Such action brings in sufficient outside air to keep the fan flow rate at its rated flow at all times.

The negative pressure level chosen for post-accident operation is sufficiently low to ensure that airborne contamination present in the Auxiliary Building is not released to the environs without being processed by an air cleanup assembly. External pressure gradients produced by outside temperature and wind loadings on the building do not adversely affect the ability of the ABGTS to maintain the negative pressure in the ABSCE.

The controls for the ABGTS are designed to provide two basic control modes. One control mode has either one of the air cleanup units in operation and the other in a state in which the redundant unit can automatically come into operation in the event the operating unit fails. Less than adequate pressure in the ABSCE is utilized in this control mode to make this failure determination. This operational redundancy is achieved with spatially separated power and control circuitry having different independent power sources to prevent a loss of function from any single system component failure. The term "Train A" is used to identify one complete set of full capacity equipment and the term "Train B" is used to identify the other set of full capacity equipment. Power for both equipment trains is supplied by the emergency power system.

Operation of the ABGTS begins automatically upon initiation of a high radiation signal from the spent fuel pool accident radiation monitors, or an ABI signal which is generated from one of the following signals:

1. Phase A containment isolation signal, or

2. High temperature signal from the Unit 1 and Unit 2 Auxiliary Building air intakes.

Additionally, during refueling operations when containment and/or the annulus is open to the Auxiliary Building ABSCE spaces, a containment vent isolation signal from either the operating unit or refueling unit will automatically start the ABGTS.

A capability is also provided to start each train with a hand switch (one hand switch per train) in the main control room. Another adjustment capability provided in the hand switches in the main

control room changes the operating mode to the single train operation with the redundant train in a standby status. Employment of this operating mode is expected after the first 30 minutes of operation. In this instance, the main control room operator has the capability to select either train to remain in operation. The standby unit selected automatically starts in the event the operating unit does not adequately maintain negative pressure in the ABSCE.

6.2.3.3 Design Evaluation

6.2.3.3.1 Secondary Containment Enclosures

The secondary containment enclosures are designed to provide a positive barrier to all potential primary containment leakage pathways during a LOCA or a FHA inside containment and to radioactive contaminants released in accidental spills and fuel handling accidents that may occur in the ABSCE. In a LOCA or a FHA inside containment, the Shield Building containment enclosure provides the barrier to all airborne primary containment leakage, and the ABSCE provides a barrier to through-the-line leakage from containment which can potentially become airborne. The ABSCE also maintains an effective barrier for airborne radioactive contaminants originating inside the ABSCE during normal and abnormal events. The original design credited the secondary containment enclosures to mitigate the consequences of a FHA. Although these enclosures are available to minimize the consequences of a FHA, based on the use of the Regulatory Guide 1.183, Revision 0 (Alternate Source Term) methodology for a FHA, the structures are no longer required for mitigation of a postulated FHA.

1. Shield Building

   Structure

   The Shield Building provides the physical barrier for airborne primary containment leakage during a LOCA or a FHA inside containment. Because the Shield Building completely encloses the free standing primary containment, most of the airborne leakage from primary containment passes into the annular region provided by this arrangement.

   The building construction employs monolithic pours of concrete. This approach for structures of this type produces a very low leakage barrier. The low leakage characteristics of this barrier help to reduce the rate at which purified annulus air must be released to maintain the enclosed volume at a negative pressure. This factor contributes significantly to keeping the site boundary and the low population zone (LPZ) dosage levels within 10 CFR 100 guidelines.

   The design of the EGTS system and the size of the annular region between the primary containment and the Shield Building assure a residence time for leakage into the annulus.

   Penetrations

   The Shield Building wall is provided with many penetrations to accommodate mechanical equipment piping, cable trays, and electrical conduit which leave and enter the Shield Building. Due to the low leakage characteristics of the building, leakage through the Shield Building wall is restricted almost entirely to openings in these penetrations. The design assures that penetration leakage does not exceed predetermined quantities. Such a capability ensures that the inleakage is sufficiently

low to keep the dose contributions at the site boundary and to the LPZ within 10 CFR 100 guidelines.

Openings in mechanical piping penetrations are sealed principally as shown in Figure 6.2.3-1. The seals are a flexible membrane type of single gaskets which incorporate fire resistant materials and are designed to withstand the combinations of Shield Building and piping movements in the SSE and retain their functional integrity. In addition, seals at or below the probable maximum flood elevation are designed to be water tight for flood static head and surge forces. All seals, where possible, are installed outside the Shield Building such that whether during normal operation, accidents, or flood, the differential pressures will tend to enhance the tightness of the seal. The penetration seal materials have been selected to be compatible with the maximum design integrated dose for the Shield Building during a LOCA.

Cables routed in cable trays pass through the Shield Building wall through rectangular cable slot penetrations as shown in Figure 6.2.3-3. The sealant material installed around cables over a portion of the length of the cable slot is silicone RTV (room temperature vulcanizing) foam and is RTV silicone rubber installed around cables within conduits. The seals are typically shown in Figure 6.2.3-3 and are designed to withstand the SSE and retain their integrity. Electrical penetration seals are allowed twice the leakage of mechanical seals to provide sufficient margin in meeting the total allowable Shield Building leakage requirements.

The personnel and equipment access doors to the Shield Building are designed with heat resistant, resilient seals which reduce their leakage to the allowable values as stated in Section 6.2.3.2. These doors are designed to retain their structural integrity and leak tightness during a SSE as described in Sections 3.8.1 and 3.8.2.

To allow personnel access to the annulus during operation, the annulus personnel access doors form an airlock. The doors are electrically interlocked such that only one of the pair may be opened at a time, but an electric interlock defeat switch is provided inside the annulus to provide for emergency egress from the annulus should the door on the Auxiliary Building side of the lock be blocked open during an accident. Therefore, a continuous secondary containment barrier is provided while allowing personnel movement.

The fuel transfer tubes penetrate the primary and secondary containment on their way to the Auxiliary Building. Each transfer tube has a blind flange on the inboard side of primary containment, equipped with double seals and a pressure test connection between the seals. The valve in the Auxiliary Building end of the transfer tube serves as the secondary containment isolation valve. The inner space between the primary containment flange and the isolation valve is bled to the annulus so that any leakage into the tube from primary containment or the Auxiliary Building flows into the annulus. The bleed line is routed above the maximum refueling pool water level to preclude accidental spills of refueling water.

2. Auxiliary Building

Structure

The entire Auxiliary Building including walls, roof, and interior partitions is constructed by consecutive monolithic pours of concrete. This method of assembly produces a structure with very low leakage characteristics. The portions of the building chosen to constitute the isolation barrier were selected such that sources of potential contamination are completely enclosed. Therefore, the structure utilized to form the Auxiliary Building containment envelope functions effectively as a barrier to the environs. This same structure also helps to reduce inleakage into the Auxiliary Building containment envelope during accidents to levels easily accommodated by the ABGTS.

Penetrations

Seals for mechanical penetrations are a flexible membrane type or single gaskets. The seals are designed to withstand Auxiliary Building and piping movements in the SSE and retain their structural integrity. The materials chosen for the seals are fire resistant. The seals, where possible, are designed such that whether during normal operation or accidents, the differential pressures tend to enhance the tightness of the seal. Sealing methods for electrical penetrations are similar to those for the Shield Building electrical penetrations.

Each normal ventilation duct penetrating the ABSCE is equipped with two isolation dampers in series. The dampers have resilient blade end and blade edge seals which are designed to retain their functional characteristics. The motor operators for these dampers have been sized to tightly close the damper blades against their resilient seals. The damper and motor operator assemblies are designed to operate during and after the SSE.

Piping penetrations are analyzed to pressure boundary retention requirements. The effects of piping penetration failures are demonstrated to not impair the ability of the ABGTS system to maintain the ABSCE pressure boundary (under the required negative pressure of 0.25 inches w.g.), or the penetrations are isolated by physical means (e.g., locked-closed valves).

The Interim ABSCE boundary supporting Unit 2 construction does not include Unit 2 primary containment and annulus spaces in the ABSCE. With the interim ABSCE boundary supporting Unit 2 construction, the sections of the Unit 2 Shield Building that are part of the secondary containment system are identified by Figures 6.2.3-04 through 6.2.3-07. (Unit 1 Only)

The Interim ABSCE boundary supporting Unit 2 construction does not include the spaces within Auxiliary Building Rooms 713-A21, 757-A14. (Unit 1 Only)

The Interim ABSCE boundary normally includes the space within Auxiliary Building Room 757-A15 except when the Unit 2 Reactor Building Blast Doors are open, and Door R002 is closed and Door R003 is open. This exception is an alternate configuration for moving equipment through the hatch. (Unit 1 Only)

Corresponding mechanical and electrical penetrations in applicable Unit 2 shield building wall sections, Auxiliary Building walls, and Auxiliary Building floor/ceiling slabs are qualified to support ABSCE boundary functions. Where these penetrations also provide penetrations through fire barriers, the corresponding configurations also provide qualified Appendix R fire barriers. (Unit 1 Only)

As identified by Figure 9.4-8, the configuration of Auxiliary Building General Ventilation supply branch ducting transiting Auxiliary Building Room 757-A14 provides support for the Interim ABSCE boundary functions of the floor and east wall of Room 757-A14. The duct section does <u>not</u> include any intake air flow path from external to the ABSCE. The duct section does <u>not</u> include any exhaust air flow from a space within ABSCE to the external environment. Consistent with Section 9.4.3.3, the application of the duct section to support Interim ABSCE introduces no requirement for any installation or application of any isolation dampers. (Unit 1 Only)

6.2.3.3.2 <u>Emergency Gas Treatment System (EGTS</u>)

The EGTS has the capabilities needed to preserve safety in accidents as severe as the design basis LOCA. To verify that the proper features are provided, functional analyses were conducted which consist of failure modes and effects analysis of the system, reviews of Regulatory Guide 1.52 sections to assure licensing requirement conformance, and performance analyses to verify that the system has the desired accident mitigation capabilities. A detailed failure modes and effects analysis is presented in Table 6.2.3-2. The system is shown schematically in Figure 6.2.3-11.

The functional analyses conducted on the EGTS have shown that:

1. Adequate isolation of the annulus vacuum control subsystem during accidents is provided. The two low leakage valves in series upstream of the annulus vacuum control subsystem fans used to isolate the two subsystems--one operated by each subsystem train--give assurance that the annulus vacuum control subsystem will be isolated during accidents. These valves fail closed.

2. The air flow control dampers in the air cleanup subsystem align to service the affected reactor units. The network was designed to have the air flow control dampers shown in Figures 6.2.3-18 and 6.2.3-19 needed to service a particular reactor unit responsive to only the containment isolation signal from that particular reactor unit.

3. The system intake and recirculation air outlets, shown on Figures 6.2.3-18 and 6.2.3-19, within the Shield Building annulus are positioned to promote mixing and dilution of primary containment leakage. Positioning the recirculation air manifold and the air outlets almost completely around the base of the annulus below the level of the containment penetrations assures a clean air flow past most of the penetrations. This air, warmed by the relative humidity heater, flows upward past these likely sources of leakage. In doing so, the flow impediments (i.e., penetrations, and structures within the annulus) tend to redirect this air flow to induce mixing and dilution. Substantial amounts of mixing and dilution are likely in the vertical rise of over 168 feet to the system air intake above the steel containment dome.

4. System startup reliability is very high. The practice of automatically starting up both full capacity trains in the system simultaneously gives greater assurance that one train of equipment functions promptly upon receipt of an accident signal.

5. The use of a single actuator in each equipment train to adjust dampers controlling the air flow recirculated and vented improves train reliability and minimizes the possibility of annulus pressure instability. Simultaneous adjustment that closes one damper and opens the other eliminates the hunting problems that could arise from nonsimultaneous operation of separately actuated dampers.

6. The Train A and Train B air cleanup units are adequately protected from each other to eliminate the possibility of a single failure destroying the capability to process annulus air during emergencies. The 13.5 feet high and 27 inch thick concrete wall built between the two units protects each from missiles originating in the other unit.

The EGTS, designed prior to issuance of Regulatory Guide 1.52, is in general agreement with requirements in the guide. Details on this compliance with Regulatory Guide 1.52 are given in Table 6.5-1.

The performance analyses conducted to verify that the EGTS has the required accident mitigation capabilities were conducted in three basic parts. One of these was concerned with the capability for keeping the Shield Building annulus below atmospheric pressure at all times during a LOCA. The second part was an analysis of the cooling capabilities provided to keep temperatures within filters and adsorbers fully loaded with radioactive nuclides at safe levels. The third part was concerned with the site boundary and LPZ dosage contribution from radioactive nuclides present in annulus air releases during the design basis LOCA. These three analyses are discussed under the respective headings below.

Annulus Negative Pressure Control Capability

The capability of the EGTS to keep the Shield Building annulus below atmospheric pressure during a design basis LOCA was established with a time iteration analysis performed by a computer. Energy and mass balances were accomplished successively in accordance with mass and volume changes calculated to take place during each time increment. Such a methodology allowed sufficient freedom to account for:

1. Steel containment vessel growth from internal pressure,

2. Steel containment vessel growth from thermal expansion,

3. Outside air inleakage into the Shield Building annulus, and

4. Heat transfer from the steel containment structure to the annulus air mass.

To assure that this analysis was valid and conservative:

1. Heat transfer from the primary containment atmosphere to the primary containment vessel was assumed to be convective. An air-steam mixture convective heat transfer coefficient was chosen to maximize heat transfer to the secondary containment atmosphere. The constant value of 400 Btu/hr-ft^2-°F given in Table 6.2.3-1 compares conservatively to the integrated transient heat transfer coefficients recommended in Branch Technical Position CSB 6-1. Heat transfer from the primary containment vessel to the annulus atmosphere and from the atmosphere to the secondary containment wall was assumed to be convective. Heat transfer from the primary containment vessel to the secondary containment wall was assumed to be by radiation. Forms of the transient convective heat transfer coefficients and values for the constant radiative heat transfer coefficients are given in Table 6.2.3-1. Consideration was given to the heat capacity of both the primary and secondary containment structures. The thermal conductivity and capacitance for these walls, as given in Table 6.2.3-1, agree closely with those obtained from Branch Technical Position CSB 6-1.

2. The thermal growth of the steel vessel was based on linear expansion which was applied to the transient containment vessel temperature increases above the initial steady-state values to obtain the transient radial expansion (see Table 6.2.3-1 for total containment expansion). Temperature gradients were calculated for three regions: upper compartment, ice condenser, and lower compartment. The radial expansions in each of these three regions were converted to volume changes which were summed to yield a total annulus volume change due to primary containment vessel thermal expansion.

3. Table 6.2.3-1 presents the characteristics of the internal pressure effects on the containment vessel. This model uses linear elastic thin shell theory to determine the expansion. The cylindrical portion of the vessel is assumed to act as a cylindrical shell with capped ends. The hemispherical dome expands uniformly as a simple sphere and includes the axial expansion of the cylinder. External vertical and circumferential stiffeners are assumed not to be present so that conservative results are obtained. This pressure-induced growth was assumed to occur instantaneously at the start of the LOCA.

4. Air leakage into the Shield Building annulus was assumed to be 250 cfm at the post accident annulus control setpoint for a postulated single failure of one EGTS train. A more conservative air inleakage value of 957 cfm (Unit 1) and 832 cfm (Unit 2) was assumed for an alternate single failure scenario which results in one pressure control train in full exhaust to the shield building exhaust stack while the other train remains functional. This higher inleakage value is based upon the higher negative pressure that will exist within the annulus during this scenario.

5. The air temperature in the annulus was assumed to be a thermally mixed average.

6. Only one train of the EGTS was assumed to operate during the first single failure scenario, allowing for a possible single failure in the other. An alternate single failure scenario was analyzed which postulates one pressure control train failing which results in full exhaust to the shield building exhaust stack while the other train remains functional.

The initial steady state conditions used in this analysis were as follows (refer also to Table 6.2.3-1):

	Pressure	Temperature	Relative Humidity
Containment upper compartment	atm.	110°F	0%
Ice condenser compartment	atm.	15°F	0%
Containment lower compartment	atm.	120°F	0%
Shield Building annulus	-5 in. w.g	50°F	0%
Outside	atm.	0°	0%

These initial values were chosen to maximize the secondary containment pressure after the LOCA. The initial pressure of minus 5.0 inches of water gauge with respect to the outside is the pressure maintained by the annulus vacuum control subsystem during normal operation. The initial temperature of 50°F is the estimated minimum temperature which was assumed for maximum annulus air density. Similarly, the initial relative humidity of 0% was assumed for maximum annulus air density for conservatism.

The results obtained from this analysis for the first single failure scenario are shown in Figure 6.2.3-17. This annulus pressure and EGTS exhaust rate vs. time curve indicates that, after the initial containment pressure induced step increase, the pressure rises to a peak value of approximately minus 1.06 inch of water in about 79 seconds after the LOCA begins. The annulus pressure is then restored and maintained at or below the EGTS setpoint value as shown in Figure 6.2.3-17.

The results obtained from the analysis for the alternate single failure scenario are shown in figure 6.2.3-17A. This annulus pressure and EGTS exhaust rate vs. time curve indicates that, after the initial containment pressure induced step increase, the pressure rises to a peak value of approximately minus 1.75 inch of water in about 44 seconds after the LOCA begins for Unit 1 and approximately minus 1.52 inch of water in about 50 seconds after the LOCA begins for Unit 2. The annulus pressure is then restored and maintained at or below the EGTS setpoint value as shown in figure 6.2.3-17A.

The expansion of approximately 1,234 ft^3 due to internal temperature summed with the expansion of 766 ft^3 from internal pressure yields a total primary containment vessel expansion of approximately 2,000 ft^3. Such results indicate that:

1. The negative pressure level of 5 inches of water below atmospheric in the Shield Building annulus maintained by the annulus vacuum control subsystem before an accident minimizes the amount of unfiltered radioactive nuclides potentially released to the environment before the air cleanup subsystem becomes operational.

2. The rated flow rate of 4000 $\pm$10% cfm for each train of the air cleanup subsystem is adequate to keep the annulus pressure below the negative pressure setpoint throughout the remaining period of the LOCA.

Inactive Air Cleanup Unit Cooling Capabilities

The second performance analysis conducted to show that the EGTS can cope with circumstances that may occur in a LOCA was concerned with temperature control capabilities provided for air filters and adsorbers loaded with radioactive material. The analysis conducted assumed accident releases in accordance with Regulatory Guide 1.89 (100% noble gas activity, 50% iodines, and 1% particulates) and, containment leakages of 0.25% per day for the first day and 0.125% per day from one to 100 days in accordance with Regulatory Guide 1.4. An additional assumption made was that the gamma and beta energy releases were transformed into heat within the filters and adsorbers.

This occurs a few days after the LOCA takes place. The design objective is to assure that the air cleanup unit component temperatures do not exceed 200°F; it was found that a cooling air flow rate of 120 cfm for Unit 1 and 112 cfm or Unit 2 is required. Such results indicate that the cooling air flow rate of 200 cfm provided for this purpose should keep the temperature within the carbon adsorber bank well below the 620°F carbon ignition temperature.

Site Boundary and LPZ Dosage Contributions

The last performance analysis conducted to show that the EGTS has the capability to perform in the required manner to preserve safety during a LOCA was concerned with the site boundary and LPZ dosage contributions arising from annulus air releases to the environs. This analysis is described and evaluated in Chapter 15.

6.2.3.3.3 <u>Auxiliary Building Gas Treatment System (ABGTS)</u>

The ABGTS has the capabilities needed to preserve safety in accidents as severe as a LOCA. This was determined by conducting functional analyses of the system to verify that the system has the proper features for accident mitigation which consist of a failure modes and effects analysis, a review of Regulatory Guide 1.52 sections to assure licensing requirement conformance, and a performance analysis to verify that the system has the desired accident mitigation capabilities. A detailed failure modes and effects analysis is presented in Table 6.2.3-3.

The functional analyses conducted on the ABGTS have shown that:

1. The air intakes for the system are properly located to minimize accident effects. The use of the air intakes provided in the fuel handling and waste disposal areas minimizes the spread of airborne contamination that may be accidentally released at these positions in which the probability of an accidental release, e.g., a fuel handling accident, is more likely. This localization effect is provided without reducing the effectiveness of the system to cope with multiple activity released throughout the ABSCE that may occur during a LOCA. Such coverage is accomplished by utilizing the normal ventilation ducting to draw outside air inleakage from any point along the secondary containment enclosure to the fuel handling and waste disposal areas.

2. Accident indication signals are utilized to bring the ABGTS into operation to assure that the system functions when needed to mitigate accident effects. Accidents in which this system is needed to preserve safety are automatically detected by at least one of the three instrumentation sets used to generate accident signals that result in system startup.

3. System startup reliability is very high. The practice of allowing the automatic startup of both full capacity trains in the system gives greater assurance that one train of equipment functions upon receipt of an accident signal.

4. The method adopted to establish and keep the negative pressure level within this secondary containment enclosure minimizes the time needed to reach the desired pressure level. Initially, the full capacity of the ABGTS fans is utilized for this purpose. After reaching the desired operating level, the system control module allows outside air to enter the air flow network just upstream of the fan at a rate to keep the fans operating at full capacity with the enclosed volume at the desired negative pressure level. In this situation, the amount of air withdrawn from the enclosed volume is equal to the amount of outside air inleakage through the ABSCE. In addition, two vacuum breaker dampers in series are provided to admit outside air in case the modulating dampers fail.

5. The ABSCE is maintained at a slightly negative pressure to reduce the amount of unprocessed air escaping from this secondary containment enclosure to the atmosphere to insignificant quantities. In addition, this negative pressure level is less than that which is maintained within the annulus; such that, any air leakage between the Auxiliary Building and the Shield Building is from the Auxiliary Building into the Shield Building.

6. The Train A and Train B air cleanup units are sufficiently separated from each other to eliminate the possibility of a single failure destroying the capability to process Auxiliary Building air prior to its release to the atmosphere. Two concrete walls and a distance of more than 80 feet separate the two trains. The use of separate trains of the emergency power system to drive the air cleanup trains gives further assurance of proper equipment separation.

The review of the ABGTS conducted to determine its conformance with Regulatory Guide 1.52 has shown that this system, designed prior to issuance of the guide, is in general agreement with its requirements. Details on compliance with Regulatory Guide 1.52 are given in Table 6.5-2.

The performance analysis conducted to verify that the ABGTS has the required accident mitigation capabilities has shown that the system flow rate is sized properly to handle all expected outside air inleakage at a ¼-inch water gauge negative pressure differential. This indicates that the nominal flow rate of 9000 cfm is sufficient to assure an adequate margin above the expected ABSCE inleakage (ACU filters are replaced as needed to maintain a minimum flow capability of 9300 cfm under surveillance instructions).

The performance analysis evaluated the capability of the ABGTS to reach and maintain a negative pressure of ¼-inch water gauge with respect to the outside within the boundaries of the ABSCE. The following was utilized in the analysis:

1. Leakage into the ABSCE is proportional to the square root of the pressure differential.

2. Only one air cleanup unit in the ABGTS operates at the rated capacity.

3. The air cleanup unit fan begins to operate 30 seconds after the initiation of an ABI signal (See Section 6.2.3.2.3).

4. The initial static pressure inside the ABSCE is conservatively considered to be atmospheric pressure, although the ABSCE is under a negative pressure during normal operation.

5. The effective pressure head due to wind equals 1/8 inch water gauge.

6. Initial average air temperature inside the ABSCE equals 140°F.

7. Atmospheric temperature and pressure are 70°F and 14.4 psia, respectively.

8. ABSCE isolation dampers/valves close within 30 seconds after receiving an ABI signal.

9. The non-safety-related general ventilation and fuel handling area exhaust fans are designed to shut down automatically following a LOCA. Each fan is provided with a safety related Class 1E primary circuit breaker and a safety related Class 1E shunt trip isolation switch which is tripped by a signal of the opposite train from that for the primary circuit breaker to ensure that power is isolated from the fan.

The analysis utilizes the first law of thermodynamics and perfect gas relations in an iterative approach to determine temperature and pressure changes in the ABSCE. Heat sources and sinks (ESF equipment room coolers) are considered.

The results obtained indicate that the ABGTS has the capability to reach and maintain a negative pressure differential of 0.25 inches water gauge within four minutes of the receipt of an ABI signal.

The system contains sufficient air cleanup facilities to keep the contributions to the site boundary and LPZ dosage arising from Auxiliary Building air releases to small fractions of the 10 CFR 100 guideline values. This part of the analysis is presented and evaluated in Chapter 15.

6.2.3.3.4 <u>Tritium Production Core Evaluation (Unit 1 Only)</u>

The effects of converting WBN to a Tritium Production Core (TPC) on the Secondary Containment design has been evaluated. The evaluation includes the effect of adding a leading edge flow meter (LEFM) into the main feedwater piping and the effect of uprating the plant nominal power level by 1.4%.

The secondary containment is designed to have the capability to control the atmosphere within the secondary containment and contiguous areas during normal operation and during plant upset conditions. As explained in Section 6.2.1.3.12, the current licensing basis mass and energy release bounds that of the TPC; therefore, the conversion to a TPC will not have any effect on the secondary containment functional design.

6.2.3.4 Test and Inspections

6.2.3.4.1 Emergency Gas Treatment System (EGTS)

Preoperational testing of the EGTS was conducted to verify that the Shield Building and the EGTS had the capabilities needed to keep LOCA generated activity releases from the affected reactor unit at or below limits specified in 10 CFR 100. Included in the scope of testing were functional tests on system instrumentation, controls, and alarms. The tests were structured to accomplish the following:

1. Verification that Shield Building infiltration was less than or equal to the design value at the design negative pressure level for post-accident conditions.

2. Verification of the system capability to establish and maintain the proper negative pressure level in the annulus.

3. Verification that the air cleanup units met requirements specified in Regulatory Guide 1.52. Refer to Section 6.5.1.4.1 for further information related to tests applicable to the air cleanup units.

4. Verification of proper operation of all system components, instrumentation, alarms, and data displays.

The periodic test program for the EGTS fans and air cleanup units is described in the Technical Specifications. A periodic test is performed once every 18 months to verify that the EGTS can maintain the annulus at a negative pressure within the instrument deadband immediately above and below the nominal design value. This test also verifies that the Shield Building inleakage rate to the annulus is less than or equal to 250 cfm at the nominal design value. A verification of system flow capacity and Shield Building inleakage rates at the specified negative pressure is adequate to confirm that the calculated depressurization time is conservative. The EGTS fans start within 30 seconds following the initiation of a containment isolation phase A signal.

6.2.3.4.2 Auxiliary Building Gas Treatment System (ABGTS)

Preoperational testing of the ABGTS was conducted to verify that the ABGTS has the capabilities needed to reduce radioactive releases from the ABSCE to the environment during an accident to levels sufficiently low to keep the site boundary dose rates below the requirements of 10 CFR 100. Included in the test scope were functional tests on all system instrumentation, controls, and alarms. The tests were structured to accomplish the following:

1. Verification of the system startup and control capabilities, considering a single operating component failure.

2. Verification of the capability of the air flow control modules to create and maintain a negative pressure within the ABSCE.

3. Verification that ABSCE infiltration rate was less than or equal to the design value at the design negative pressure level considering a postulated failure of a non-safety related component.

4. Verification that the air cleanup units met the requirements specified in Regulatory Guide 1.52. Refer to Section 6.5.1.4.2 for further information related to tests applicable to the air cleanup units.

The periodic test program for the ABGTS fans and air cleanup units is described in the Technical Specifications. A periodic test is performed to verify that the ABGTS can maintain the ABSCE at a negative pressure between -0.25 and -0.5 inches of water with respect to atmospheric pressure. This test also verifies that the ABSCE vacuum relief flow rate is greater than or equal to 1370 cfm while the ABSCE is being maintained at the negative pressure described above. A verification of system flow capacity and ABSCE vacuum relief flow rate at the specified negative pressure is adequate to confirm that the calculated depressurization time is conservative.

6.2.3.5 Instrumentation Requirements

6.2.3.5.1 Emergency Gas Treatment System (EGTS)

The air flow control instrumentation requirements for the EGTS are described in Section 6.2.3.2.2. Instrumentation associated with the air cleanup units is discussed in Section 6.5.1.5.1. The logic, controls, and instrumentation of this engineered safety feature system are such that a single failure of any component does not result in the loss of functional capability for the system.

6.2.3.5.2 Auxiliary Building Gas Treatment System (ABGTS)

Instrumentation required for the air flow control modules and air cleanup units are discussed in Section 6.2.3.2.3. Instrumentation associated with the air cleanup units is discussed in Section 6.5.1.5.2. The logic, controls, and instrumentation of this engineered safety feature system are such that a single failure of any component does not result in the loss of functional capability for the system.

REFERENCES

None

TABLE 6.2.3-1 (Sheet 1 of 2)

DUAL CONTAINMENT CHARACTERISTICS

I. Secondary Containment Design Information

	Shield Bldg.	ABSCE (Incl. 2x Shield Bldg., 2 x Contmt., AB, and CDWE Bldg.)l
A. Free Volume (ft^3)	3.96×10^5	6.9×10^{6}+
B. Pressure (in. wg)#		
Normal Operation	-5.0	-0.25
Post-Accident	-0.5	-0.25
C. Leak Rate at Post-Accident Pressure (%/day)	91	165.5
D. Exhaust Fans		
Normal Operation		
Number	4 (2/reactor unit)*	6**
Type	centrifugal	centrifugal
Post-Accident Operation		
Number	2***	2****
Type	centrifugal	centrifugal

E. Filters: Refer to Table 6.5-5

II. Transient Analysis

A. Initial Conditions

1. Pressure = 14.4 psig
2. Annulus temperature = 50°F
3. Outside air temperature = 0°F
4. Thickness of secondary containment wall = 36 in.
5. Thickness of steel containment vessel = ranging from 0.8125 to 1.50 inches

* Annulus vacuum control subsystem
** Auxiliary Building general exhaust (2/unit) and fuel handling area exhaust (2)
*** EGTS
**** ABGTS
\# Due to instrument locations and inaccuracies, the actual setpoints are more negative than the required values shown.
\+ This value is an applied maximum value in calculations demonstrating the drawdown capabilities of ABGTS. It is conservative in relation to an ABSCE configuration that includes the free volume of only a single Reactor Building.

TABLE 6.2.3-1 (Sheet 2 of 2)

DUAL CONTAINMENT CHARACTERISTICS

II. Transient Analysis (continued)

B. Thermal Characteristics

1. Primary containment wall

   a. Total expansion = 2000 ft^3
      Pressure expansion = 766 ft^3
      Temperature expansion = 1234 ft^3

   b. Thermal conductivity = 31 Btu/hr-ft-°F

   c. Heat capacity = 0.111 Btu/lb-°F

2. Secondary containment wall

   a. Thermal conductivity = 1.6 Btu/hr-ft-°F

   b. Heat capacity = 0.22 Btu/lb-°F

3. Heat transfer coefficients

   a. Primary containment atmosphere to primary containment wall = 400 Btu/hr-ft^2-°F

   b. Primary containment wall to secondary containment atmosphere = 0.19 $(\Delta T)^{1/3}$ Btu/hr-ft^2-°F

   c. Secondary containment wall to secondary containment atmosphere = 0.19 $(\Delta T)^{1/3}$ Btu/hr-ft^2-°F

   d. Primary containment emissivity = 0.90

   e. Secondary containment emissivity = 0.90

TABLE 6.2.3-2 (Sheet 1 of 15)

FAILURE MODES AND EFFECTS ANALYSIS
EMERGENCY GAS TREATMENT SYSTEM

	Component Identification	Function	Failure Mode	Potential Cause	Method of Failure Detection	Effect on System	Effect on Plant	Remarks
1.	EGTS ACU Fans A-A & B-B	Draws air from annulus to maintain negative pressure in the annulus during design basis events	Stops running Does not run	Electrical failure Mechanical failure	Low flow alarm in the MCR	Reduction of flow from annulus and temporary reduction in annulus negative pressure	None. (See Remarks)	Redundant fan starts on low flow signal from failed fan and train. Both trains of the EGTS system start on receipt of a Phase A isolation signal. However, operators will align the system to single train operation after a period of time.
2.	EGTS ACU A-A & B-B	Filter air to remove airborne particulates and vapors from the annulus during design basis events	Filters leak Filters become loaded	Defective filters High levels of radioactivity contamination Large quantities of particulates	High radiation levels indicated in the MCR from shield building exhaust vent	High radiation in shield building exhaust vent.	None. (See Remarks)	Periodic testing of EGTS ACUs is conducted in accordance with R.G. 1.52 to verify leak tightness of HEPA and charcoal bank efficiencies. High radiation levels are indicated in the MCR for operators to start the redundant ACU, if necessary.
3.	Containment annulus vacuum control fan suction isolation valves 1-FCV-65-52	Isolate annulus vacuum control fans from EGTS during ACU operation	Open	Electrical failure Mechanical failure	Valve position indicating light in the MCR	None. (See Remarks)	None. (See Remarks)	Redundant valve in series with failed valve provides isolation function.

FAILURE MODES AND EFFECTS ANALYSIS
EMERGENCY GAS TREATMENT SYSTEM

	Component Identification	Function	Failure Mode	Potential Cause	Method of Failure Detection	Effect on System	Effect on Plant	Remarks
	1-FCV-65-53 2-FCV-65-4 2-FCV-65-5							
4.	B train isolation valves at EGTS Train A suction 1-FCV-65-8 (Unit 1 in LOCA)	Provide decay heat removal cooling flow path for A-A ACU when B-B ACU is operating for Unit 1 (valve open by operator action)	Open (ACU A-A is in operation)	Electrical failure Mechanical failure	Valve position indication light in the MCR	Parallel flow path to ACU A-A is open	None. (See Remarks)	Additional flow path is available which causes no adverse effect.
			Open (ACU A-A has not operated and B-B is operating)	Electrical failure Mechanical failure	Valve position indication light in the MCR	If valves 0-FCV-65-28A and -28B are open, there will be negative pressure on ACU A-A by suction ACU fan B-B.	None (See Remarks)	Valve on Fan A-A discharge side (0-FCV-65-24) closes when ACU A-A is in standby and will prevent backflow.
			Closed (ACU A-A has operated and is on stand-by and B-B is operating)	Electrical failure Mechanical failure	Valve position indication light in the MRC	None (See Remarks)	None (See Remarks)	Operator will return to using ACU A-A. An additional flow path is available by opening 1-FCV-65-10, 0-FCV-65-28A and 0-FCV-65-28B.
4a.	B train isolation valves at EGTS Train A suction 2-FCV-65-7 (Unit 2 in	Provide decay heat removal cooling flow path for A-A ACU when B-B ACU is operating for Unit 2 (Valve	Open (ACU A-A is in operation)	Electrical failure Mechanical failure	Valve position indication light in the MCR	Parallel flow path to ACU A-A is open	None (see Remarks)	Additional flow path is available which causes no adverse effect.
			Open (ACU A-A has not operated and	Electrical failure	Valve position indication light	If valves 0-FCV-65-28A and -28B are open,	None (See Remarks)	Valve on Fan A-A discharge side (0-FCV-65-24) closes when

FAILURE MODES AND EFFECTS ANALYSIS
EMERGENCY GAS TREATMENT SYSTEM

	Component Identification	Function	Failure Mode	Potential Cause	Method of Failure Detection	Effect on System	Effect on Plant	Remarks
	LOCA)	opened by operator action)	B-B is operating)	Mechanical failure	in the MCR	there will be negative pressure on ACU A-A by suction ACU fan B-B.		ACU A-A is in standby and will prevent backflow.
			Closed (ACU A-A has operated and is on stand-by and B-B is operating)	Electrical failure Mechanical failure	Valve position indication light in the MCR	None (See Remarks)	None (See Remarks)	Operator will return to using ACU A-A. Additional flow path is available by opening 2-FCV-65-9, 0-FCV-65-28A and 0-FCV-65-28B.
5.	A train isolation valves at EGTS Train B suction 1-FCV-65-51 (Unit 1 in LOCA)	Provide decay heat cooling path for B-B ACU when A-A ACU is operating for Unit 1 (valve opened by operator action)	Open (ACU B-B is in operation)	Electrical failure Mechanical failure	Valve position indication light in the MCR	Parallel flow path to ACU B-B is open	None. (See Remarks)	Additional flow path is available which causes no adverse effect. .
			Open (ACU B-B has not operated and A-A is operating)	Electrical failure Mechanical failure	Valve position indication light in the MCR	If valves 0-FCV-65-47A and -47B are open, there will be negative pressure on ACU B-B by suction ACU fan A-A.	None. (See Remarks)	Valve on Fan B-B discharge side 0-FCV-65-43 closes when ACU B-B is in standby and will prevent backflow
			Closed (ACU	Electrical	Valve position	None. (See	None. (See	Operator will return to

TABLE 6.2.3-2 (Sheet 4 of 15)

FAILURE MODES AND EFFECTS ANALYSIS
EMERGENCY GAS TREATMENT SYSTEM

	Component Identification	Function	Failure Mode	Potential Cause	Method of Failure Detection	Effect on System	Effect on Plant	Remarks
			B-B has operated and is on Stand-by and A-A is operating)	failure Mechanical failure	indication light in the MCR	Remarks)	Remarks)	using ACU B-B. An additional flow path is available by opening 1-FCV-65-30, 0-FCV-65-47A, and 0-FCV-65-47B
5a.	A train isolation valves at EGTS Train B suction 2-FCV-65-50 (Unit 2 in LOCA)	Provide decay heat removal cooling path for B-B ACU when A-A ACU is operating for Unit 2 (valve opened by operator action)	Open (ACU B-B is in operation)	Electrical failure Mechanical failure	Valve position indication light in the MCR	Parallel flow path to ACU B-B is open	None. (See Remarks)	Additional flow path is available which causes no adverse effect.
			Open (ACU B-B has not operated and A-A is operating)	Electrical failure Mechanical failure	Valve position indication light in the MCR	If valves 0-FCV-65-47A and -47B are open, there will be negative pressure on ACU B-B by suction ACU fan A-A.	None. (See Remarks)	Valve on Fan B-B discharge side 0-FCV-65-43 closes when ACU B-B is in standby and will prevent backflow.
			Closed (ACU B-B has operated and is on Stand-by and A-A is operating)	Electrical failure Mechanical failure	Valve position indication light in the MCR	None (See Remarks)	None. (See Remarks)	Operator will return to using ACU B-B. An additional flow path is available by opening 2-FCV-65-29, 0-FCV65-47A and 0-FCV-65-47B.
6.	Isolation valves at EGTS Train A suction 1-FCV-65-10	Isolation control valve that opens on a containment isolation signal so that the	Closed (ACU A-A is in operation)	Electrical failure Mechanical failure	Low flow alarm and valve position indication light in the MCR	Momentary decrease in flow	None (See Remarks)	Redundant ACU Fan B-B starts on low flow signal from Fan A-A and Valve 1-FCV-65-30 is open. Fan starting

FAILURE MODES AND EFFECTS ANALYSIS
EMERGENCY GAS TREATMENT SYSTEM

	Component Identification	Function	Failure Mode	Potential Cause	Method of Failure Detection	Effect on System	Effect on Plant	Remarks
	(Unit 1 in LOCA)	EGTS can exhaust air from the annulus. It also isolates Train A ACU during normal plant operation.						signal is independent of valve failure.
			Closed (both Trains A & B ACUs are operating)	Electrical failure Mechanical failure	Low flow alarm and valve position indication light in the MCR	Momentary decrease in flow	None (See Remarks)	Train B ACU continues to operate with Valve 1-FCV-65-30 open
			Open. (Train B-B is operating. A-A is not operating and has never operated or has operated and stopped.)	Electrical failure Mechanical failure	Low flow alarm and valve position indication light in the MCR	Open backflow path to ACU A-A	None (See Remarks)	Flow Control Valve 0-FCV-65-24 and Backdraft Damper 0-65-524 close when ACU A-A is in standby and will prevent back flow.
6a.	Isolation valve at EGTS Train A suction, 2-FCV-65-9 (Unit 2 in LOCA)	Isolation control valve that opens on a containment isolation signal so that the EGTS can exhaust air from the annulus. It also isolates	Closed (ACU A-A is in operation)	Electrical failure Mechanical failure	Low flow alarm and valve position indication light in the MCR	Momentary decrease in flow	None (See Remarks)	Redundant ACU Fan B-B starts on low flow signal from Fan A-A and Valve 2-FCV-65-29 is opened. The fan starting signal is independent of valve failure.

TABLE 6.2.3-2 (Sheet 6 of 15)

FAILURE MODES AND EFFECTS ANALYSIS
EMERGENCY GAS TREATMENT SYSTEM

	Component Identification	Function	Failure Mode	Potential Cause	Method of Failure Detection	Effect on System	Effect on Plant	Remarks
		Train A ACU during normal plant operation.	Closed (both Trains A-A and B-B ACUs are operating)	Electrical failure Mechanical failure	Low flow alarm and valve position indication light in the MCR	Momentary decrease in flow	None (See Remarks)	Train B ACU continues to operate with Valve 2-FCV-65-29 open.
			Open (Train B-B is operating. A-A is not operating and has never operated or has operated and stopped.)	Electrical failure Mechanical failure	Low flow alarm and valve position indication light in the MCR	Open backflow path to ACU A-A	None (See Remarks)	Flow Control Valve 0-FCV-65-24 and Backdraft Damper 0-65-523 close when ACU A-A is in standby and will prevent backflow.
7.	Isolation valves at EGTS Train B suction 1-FCV-65-30 (Unit 1 in LOCA)	Isolation control valve opens on a containment isolation signal so that EGTS can exhaust air from the annulus. It also isolates Train B ACUs during normal plant operation.	Closed (ACU B-B is in operation)	Electrical failure Mechanical failure	Low flow alarm and valve position indication light in the MCR	Momentary decrease in flow	None (See Remarks)	Redundant ACU Fan A-A starts on low flow signal from Fan B-B and Valve 1-FCV-65-10 is opened. The fan starting signal is independent of valve failure.
			Closed (both Trains A-A and B-B ACUs are operating)	Electrical failure Mechanical failure	Low flow alarm and valve position indication light in the MCR	Momentary decrease in flow	None (See Remarks)	Train A ACU continues to operate with Valve 1-FCV-65-10 is open.
			Open (Train A-A is	Electrical failure	Low flow alarm and valve	Open backflow path to ACU B-	None (See Remarks)	Flow Control Valve 0-FCV-65-43 and

FAILURE MODES AND EFFECTS ANALYSIS
EMERGENCY GAS TREATMENT SYSTEM

	Component Identification	Function	Failure Mode	Potential Cause	Method of Failure Detection	Effect on System	Effect on Plant	Remarks
			operating. B-B is not operating and has never operated or has operated and stopped.)	Mechanical failure	position indication light in the MCR	B		backdraft damper 0-65-526 close when ACU B-B is in standby and will prevent backflow. Also the additional flow paths will not cause an adverse effect.
7a.	Isolation valves at EGTS Train B suction, 2-FCV-65-29 (Unit 2 in LOCA)	Isolation control valve that opens on a containment isolation signal so that the EGTS can exhaust air from the annulus. It also isolates Train B ACU during normal plant operation.	Closed (ACU B-B is in operation)	Electrical failure Mechanical failure	Low flow alarm and valve position indication light in the MCR	Momentary decrease in flow	None (See Remarks)	Redundant ACU Fan A-A starts on low flow signal from Fan B-B and Valve 2-FCV-65-9 is opened. The fan starting signal is independent of valve failure.
			Closed (both Trains A-A and B-B ACUs are operating)	Electrical failure Mechanical failure	Low flow alarm and valve position indication light in the MCR	Momentary decrease in flow	None (See Remarks)	Train A ACU continues to operate with Valve 2-FCV-65-9 open.
			Open (Train B-B is operating. A-A is not operating and has never operated or has operated and	Electrical failure Mechanical failure	Valve position indication light in the MCR	Open backflow path to ACU A-A	None (See Remarks)	Flow Control Valve 0-FCV-65-24 and backdraft damper 0-65-525 close when ACU B-B is in standby and will prevent backflow. Also, the additional flow paths will not cause an adverse effect.

FAILURE MODES AND EFFECTS ANALYSIS
EMERGENCY GAS TREATMENT SYSTEM

	Component Identification	Function	Failure Mode	Potential Cause	Method of Failure Detection	Effect on System	Effect on Plant	Remarks
			stopped).					
8.	EGTS fan discharge isolation valves 0-FCV-65-24 0-FCV-65-43	Isolates EGTS fan from duct distribution system when EGTS fan is on standby	One damper fails closed. (EGTS fan is operating)	Electrical failure Mechanical failure	Low flow alarm and valve position indication light in the MCR	Loss of flow through the affected EGTS train	None. (See Remarks)	Redundant fan starts on low flow signal from the affected train. Fan starting signal is independent of valve failure.
9.	Shield building exhaust isolation dampers 1-FCO-65-26 1-FCO-65-27 2-FCO-65-45 2-FCO-65-46	Open air path for EGTS exhaust to be discharged to either shield building vent and recirculated air flow to either annulus	One damper fails closed	Electrical failure Mechanical failure	Damper position indication light in the MCR	None. (See Remarks)	None. (See Remarks)	Damper in parallel flow path is open.
10.	EGTS inlet flow elements abandoned in place (1-FE-65-54 and 2-FE-65-3).							
11.	Shield Building Exhaust flow elements abandoned in place (1-FE-65-84 and -85, 1-FE-65-78, 2-FE-65-84 and -85, 2-FE-65-78)							
12.	Back draft dampers 0-65-523 0-65-524	Prevent backflow through Train A ACU when Train B ACU is in operation	Damper closed (Train A ACU is in operation and Train B ACU is in standby).	Mechanical failure	Low flow alarm indication light in the MCR	Momentary decrease in flow from annulus	None. (See Remarks)	Redundant ACU Fan B-B starts on low flow from Train A-A. Fan starting signal is independent of damper failure.
			Damper closed. (Both Train A & Train B are operational.)	Mechanical failure	Low flow alarm indication light in the MCR	None. (See Remarks)	None. (See Remarks)	Train B ACU continues to operate and Train A will be turned off by operator.

TABLE 6.2.3-2 (Sheet 9 of 15)

FAILURE MODES AND EFFECTS ANALYSIS
EMERGENCY GAS TREATMENT SYSTEM

	Component Identification	Function	Failure Mode	Potential Cause	Method of Failure Detection	Effect on System	Effect on Plant	Remarks
13.	Back draft dampers 0-65-525 0-65-526	Prevent backflow through Train B ACU when Train A ACU is in operation	Damper closed (Train B ACU is in operation and Train A ACU is in standby).	Mechanical failure	Low flow alarm indication light in the MCR	Momentary decrease in flow from annulus	None. (See Remarks)	Redundant ACU Fan A-A starts on low flow from Train B-B. Fan starting signal is independent of damper failure.
			Damper closed (Train A and Train B ACUs are operational)	Mechanical failure	Low flow alarm indication light in the MCR	None. (See Remarks)	None. (See Remarks)	Train A ACU continues to operate and Train B will be turned off by operator.
14.	Modulating dampers 1-PCO-65-80 1-PCO-65-82 1-PCO-65-88 1-PCO-65-89 2-PCO-65-80 2-PCO-65-82 2-PCO-65-88 2-PCO-65-89	Modulates EGTS flow released to outside atmosphere, controlling to the required annulus negative pressure.	Damper closed, open or improper modulation. The worst case PCO single failure is such that either 1(2)-PCO-65-80 or -82 fail open to the shield building stack, and the opposite train PCOs still modulate & attempt to maintain annulus ΔP	Mechanical Linkage Failure Control Loop Failure of one Train of PCOs.	Pressure differential is indicated in MCR to detect the -2.1" (-1.88" for U2) w.g. annulus ΔP. However, the worst case PCO failure has analyzed to exist with both trains of EGTS fans running for the first one to two hours after event initiation; and then one fan running thereafter due	With the worst case PCO single failure, the EGTS maintains annulus ΔP at -2.1"(-1.88" for U2) w.g. (steady State) during the short term, which is acceptable since slightly more negative than the -1.45" w.g. controller set point. When one EGTS fan is stopped between one to	None. The elevated 957 cfm (832 cfm for U2) short term / 694 cfm (604 cfm for U2) long term EGTS exhaust flow associated with the worst case PCO single failure mode is addressed in the Remarks.	Off-site and control room radiological dose analyses demonstrate that the offsite and control room dose results remain below the regulatory limits, when the elevated EGTS exhaust flows associated with the worst case POC single failure mode are considered.

FAILURE MODES AND EFFECTS ANALYSIS
EMERGENCY GAS TREATMENT SYSTEM

	Component Identification	Function	Failure Mode	Potential Cause	Method of Failure Detection	Effect on System	Effect on Plant	Remarks
			at the -1.45" w.g. control setpoint. This single failure scenario takes the short term annulus ΔP to -2.1"(-1.88" w.g. for U2) w.g.(i.e., failed PCO train with open flow path to the shield building stack and non-failed PCO train with open recirculation flow path to the annulus and two fans operating.)		to operator action to shut off one EGTS fan.	two hours by operator action, the resulting configuration maintains annulus ΔP at -1.60" (-1.48" for U2) w.g., which is still slightly more negative than -1.45" w.g.		
15.	Isolation valves 1-PCV-65-81 1-PCV-65-83 1-PCV-65-86 1-PCV-65-87	Isolates EGTS ductwork from outside atmosphere and ring header during normal	One valve closed when EGTS is in operation	Electrical failure Mechanical failure	Valve position indication light in the MCR	One of the two parallel flow paths is lost	None. (See Remarks)	Redundant flow path is available.

TABLE 6.2.3-2 (Sheet 11 of 15)

FAILURE MODES AND EFFECTS ANALYSIS
EMERGENCY GAS TREATMENT SYSTEM

	Component Identification	Function	Failure Mode	Potential Cause	Method of Failure Detection	Effect on System	Effect on Plant	Remarks
	2-PCV-65-81 2-PCV-65-83 2-PCV-65-86 2-PCV-65-87	plant operation						
16.	Isolation valves 0-FCV-65-28A 0-FCV-65-28B	Valves open to remove decay heat in the idle train ACU	One valve opens when bypass cooling is not in operation	Electrical failure Mechanical failure	Valve position indication light in the MCR	None (See Remarks)	None. (See Remarks)	Redundant isolation valve
			One valve closed when bypass cooling is in operation	Electrical failure Mechanical failure	Valve position indication light in the MCR	None. (See remarks)	None. (See Remarks)	Bypass cooling provision will not be used unless ACU fails and enough heat is generated by radioactivity collected on HEPA and charcoal adsorber to raise the charcoal bed temperature significantly. Therefore, a second failure (closed isolation valve) need not be postulated.
17.	Isolation valves 0-FCV-65-47A 0-FCV-65-47B	Valves open to remove decay heat in the idle train ACU	One valve opens when bypass cooling is not in operation	Electrical failure Mechanical failure	Valve position indication light in the MCF	None (See Remarks)	None (See Remarks)	Redundant isolation valves
			One valve closed when	Electrical failure	Valve position indication light	None (See Remarks)	None (See Remarks)	Bypass cooling provision will not be

TABLE 6.2.3-2 (Sheet 12 of 15)

FAILURE MODES AND EFFECTS ANALYSIS
EMERGENCY GAS TREATMENT SYSTEM

	Component Identification	Function	Failure Mode	Potential Cause	Method of Failure Detection	Effect on System	Effect on Plant	Remarks
			bypass cooling is in operation	Mechanical failure	in the MCR			used unless ACU fails and enough heat is generated by radioactivity collected on HEPA and charcoal adsorber to raise the charcoal bed temperature significantly. Therefore, a second failure (closed isolation valve) need not be postulated.
18.	Flow elements 0-FS-65-31A/B 0-FS-65-55A/B	Opens decay heat removal isolation valves to the operating ACU when no flow is sensed at the idle ACU	No signal sent	Electrical failure	Valve position indication light in the MCV	None (See Remarks)	None (See Remarks)	Bypass cooling provision will not be used unless ACU fails and enough heat is generated by radioactivity collected on HEPA and charcoal bed temperature significantly. Therefore, a second failure (closed isolation valve) need not be postulated.
19.	Flow elements 0-FS-65-31B/A 0-FS-65-55B/A	Starts EGTS standby ACU upon loss of flow in normally operating ACU	No signal sent	Electrical failure	Valve position indication light in the MCV	Momentary decrease in flow from annulus	None. (See Remarks)	The EGTS fan can be stopped either by operator's action or fan failure, which is a single failure; the other EGTS fan is available to function. The heater

TABLE 6.2.3-2 (Sheet 13 of 15)

FAILURE MODES AND EFFECTS ANALYSIS
EMERGENCY GAS TREATMENT SYSTEM

	Component Identification	Function	Failure Mode	Potential Cause	Method of Failure Detection	Effect on System	Effect on Plant	Remarks
								requires two signals to operate: on from flow switch and one from temperature switch. Therefore, heater not receiving the signal need not be assumed.
20.	Flow elements 0-FS-65-25A/B 0-FS-65-44A/B	Shuts off relative humidity heater on low air flow and alarm in MCR	No signal sent	Electrical failure	Low flow and Hi-temperature alarms in the MCR	Humidity heater may stay on after EGTS fan stops	None. (See Remarks)	The EGTS fan can be stopped either by operator action or fan failure, which is a single failure. The other EGTS fan is available to function. The heater requires two signals to operate, on from flow switch, and one from temperature switch; therefore, the spurious signal of the flow element has no effect.
21.	Flow elements 0-FS-65-25B/A 0-FS-65-44B/A	Opens decay heat removal isolation valves on idle ACU when high flow is sensed at the operating ACU	No signal sent	Electrical failure	Valve position indication light in the MCR	None (See Remarks)	None (See Remarks)	Bypass cooling provision will not be used unless ACU fails and enough heat is generated by radioactivity collected on HEPA and charcoal adsorber to raise the charcoal bed temperature significantly. Therefore, a second

TABLE 6.2.3-2 (Sheet 14 of 15)

FAILURE MODES AND EFFECTS ANALYSIS
EMERGENCY GAS TREATMENT SYSTEM

	Component Identification	Function	Failure Mode	Potential Cause	Method of Failure Detection	Effect on System	Effect on Plant	Remarks
								failure (closed isolation valve because signal not sent) need not be postulated.
22.	Train A Emergency Power	Provides Class 1E diesel-backed power supply to active components of Train A of EGTS	Loss of or inadequate Voltage	Diesel generator failure; bus fault (Train A); operator error	Alarm and indication in the MCR	Loss of redundancy in EGTS exhaust flow paths	None (See Remarks)	Redundant train of EGTS is turned on by automatic and manual actions on loss of fan airflow, or annulus ΔP, which ensures that the EGTS safety functions are continued to be performed.
23.	Train B Emergency Power	Provides Class 1E diesel-backed power supply to active components of Train B of EGTS	Loss of or inadequate Voltage	Diesel generator failure; bus fault (Train B); operator error	Alarm and indication in the MCR	Loss of redundancy in EGTS exhaust flow paths	None (See Remarks)	Redundant train of EGTS is turned on by automatic and manual actions on loss of fan airflow, or annulus ΔP, which ensures that the EGTS safety functions are continued to be performed.
24.	Deluge Valve 0-FCV-26-175	Floods the carbon adsorbers in the event of fire	Spurious actuation	Electrical failure Mechanical failure	Local and MCR alarms on water discharge	None (See Remarks)	None (See Remarks)	ACU will not be flooded since the fusible links on the spray heads will be closed due to absence of heat.
24a.	Deluge System Piping/Heads on Train A	Floods the carbon adsorbers in the event of fire	Spurious actuation	Fusible Link Failure Piping Failure	None	Loss of redundancy in EGTS	None (See Remarks)	Opposite train ACU is unaffected and remains available.

TABLE 6.2.3-2 (Sheet 15 of 15)

FAILURE MODES AND EFFECTS ANALYSIS
EMERGENCY GAS TREATMENT SYSTEM

	Component Identification	Function	Failure Mode	Potential Cause	Method of Failure Detection	Effect on System	Effect on Plant	Remarks
25.	Deluge Valve 0-FCV-26-179	Floods the carbon adsorbers in the event of fire	Spurious actuation	Electrical failure Mechanical failure	Local and MCR alarms on water discharge	None (See Remarks)	None (See Remarks)	ACU will not be flooded since the fusible links on the spray heads will be closed due to absence of heat.
25a.	Deluge System Piping/Heads on Train B	Floods the carbon adsorbers in the event of fire	Spurious actuation	Fusible Link Failure Piping Failure	None	Loss of redundancy in EGTS	None (See Remarks)	Opposite train ACU is unaffected and remains available.

TABLE 6.2.3-3 (Sheet 1 of 37)

FAILURE MODES AND EFFECTS ANALYSIS FOR THE ABGTS

Item No.	Component	Function	Failure Mode	Potential Cause	Method of Detection	Effect on System	Effect on Plant	Remarks
1.	Auxiliary Building Isolation (ABI) signal Train A	De-energizes solenoid valves to close associated dampers and establish AB secondary containment enclosure; stops AB general ventilation fans; starts various ESF room coolers; starts ABGTS fans to maintain negative pressure in the ABSCE and remove contaminants from the ABSCE air prior to discharge to atmosphere.	Signal fails.	Train A vital ac bus failure; Relay VKA1 failure; Train A initiating signal (Phase A containment isolation, high rad in refueling area, high temp. in Aux. Building general supply duct) failure.	MCR indication of only one train ABGTS fan starting and one train of ABSCE dampers closing.	Loss of redundancy in ABSCE isolation and in ABGTS until operator starts Train A ABGTS manually from MCR, after ascertaining that Train B ABI signal is not spurious.	None. (See Remarks)	Train A and Train B ABI initiating signals are derived from independent (train-separated) qualified devices. Either train signal is sufficient to stop all AB general vent supply and exhaust fans and fuel handling area exhaust fans. Since ABSCE dampers are arranged in series, either train's actuation is adequate to establish the ABSCE boundary.
		Starts Train A ABGTS, closes Train A ABSCE isolation dampers, stops Train A containment purge air supply and exhaust fans. (Ref. 5.28)	Same as above.	Same as above.	Same as above.	Same as above.	None. (See Remarks).	Only Train B ABGTS will start and only Train B ABSCE isolation dampers will close. Train A containment purge air supply fan and exhaust fan may be running, and the Incore Instrument Room supply and exhaust fans may be running. For conservatism, it is

TABLE 6.2.3-3 (Sheet 2 of 37)

FAILURE MODES AND EFFECTS ANALYSIS FOR THE ABGTS

Item No.	Component	Function	Failure Mode	Potential Cause	Method of Detection	Effect on System	Effect on Plant	Remarks
								assumed that all exhaust fans are stopped. Preoperational and surveillance tests have shown that, in this configuration, the ABGTS is able to fulfill its safety functions.
		Same as above	Spuriously starts.	Operator error, spurious initiating signal (initiating signals listed above)	MCR indicating lights on ABGTS start and ABSCE dampers' closure.	*Disrupts the normally operating ventilation systems, and starts safety-related systems and components.	None. (See Remarks)	Unnecessary isolation of ABSCE and actuation of ABGTS, as well as disrupting the normally operating ventilation systems and starting the ESF pump room and area coolers.
2.	Auxiliary Building Isolation (ABI) signal Train B	De-energizes solenoid valves to close associated dampers and establish AB secondary containment enclosure; stops AB general ventilation fans; starts various ESF room coolers; starts ABGTS fans to maintain negative pressure in ABSCE and remove contaminants from	Signal fails.	Train B vital ac bus failure; Relay VKA1 failure; Train B initiating signal (Phase A containment isolation, high rad in refueling area, high temp. in Aux. Building general supply duct) failure.	MCR indication of only one train ABGTS fan starting and one train of ABSCE dampers closing.	Loss of redundancy in ABSCE isolation and in ABGTS until operator starts Train B ABGTS manually from MCR, after ascertaining that Train A ABI signal is not spurious. .	None. (See Remarks)	Train A and Train B ABI initiating signals are derived from independent (train-separated) qualified devices. Either train signal is sufficient to stop all AB general vent supply and exhaust fans and fuel handling area exhaust fans. Since ABSCE dampers are arranged in series, either train's actuation is sufficient to establish the ABSCE

FAILURE MODES AND EFFECTS ANALYSIS FOR THE ABGTS

Item No.	Component	Function	Failure Mode	Potential Cause	Method of Detection	Effect on System	Effect on Plant	Remarks
		the ABSCE air prior to discharge to atmosphere.						boundary. Also see Table 9.4-8 and 9.4-8A.
		Starts Train B ABGTS, closes Train B ABSCE isolation dampers, stops Train B containment purge air supply and exhaust fans and stops the Incore Instrument Room supply and exhaust fans. (Ref.5.28)	Same as above.	Same as above.	Same as above.	Same as above.	None. (See Remarks).	Only Train B ABGTS will start and only Train B ABSCE isolation dampers will close. Train B containment purge air supply fan and exhaust fan may be running, and the Incore Instrument Room supply and exhaust fans may be running. For conservatism, it is assumed that all Exhaust fans are Stopped. Preoperational and surveillance tests have shown that, in this configuration, the ABGTS is able to fulfill its safety functions. (See Section 8.3 for detail).
		Same as above.	Spuriously start.	Operator error, spurious initiating signal (initiating signals listed above).	MCR indicating lights on ABGTS start and ABSCE dampers' closure.	*Disrupts the normally operating ventilation systems and starts safety-related systems and	None. (See Remarks)	Unnecessary isolation of ABSCE and actuation of ABGTS, as well as disrupting the normally operating ventilation systems and starting the ESF pump room and area coolers.

TABLE 6.2.3-3 (Sheet 4 of 37)

FAILURE MODES AND EFFECTS ANALYSIS FOR THE ABGTS

Item No.	Component	Function	Failure Mode	Potential Cause	Method of Detection	Effect on System	Effect on Plant	Remarks
						components.		
3.	ABGTS Exhaust Fan A-A	Draws a portion of air in the ABSCE through an air cleanup unit (ACU) to remove radioactive contaminants and discharge into the shield building exhaust vent to maintain a negative pressure in the ABSCE relative to the outside.	Fails to start or fails to run.	Mechanical failure; Train A power failure; Train A ABI signal (HS in A-Auto).	Indicating light in MCR.	Loss of redundancy in ABGTS.	None. ABGTS Fan B-B can perform the functions of maintaining the ABSCE at a negative pressure and removing contaminant	Handswitches for ABGTS Fans A-A and B-B in the MCR should normally be in the A-Auto position. On an ABI signal, both fans start and the operator may stop one fan and place its handswitch in the P-Auto (pull-out) position. This mode of operation is expected to occur after 30 minutes of two fan operation. During an ABI, the fan in the P-Auto mode will start automatically on insufficient negative pressure in the ABSCE relative to the outside. An alarm is provided for the condition when flow is inadequate 45 seconds after fan start.
			Starts spuriously	Spurious Train A ABI signal (HS in A-Auto); spurious low flow signal from Fan B-B after valid ABI signal (HS in P-Auto).	See "Remarks" column.	Vacuum relief line dampers may open to prevent excessive negative	None. (See Remarks)	Indicating Light in MCR provide indication to operator that fan is running. However, if only one ABGTS train starts when both fans are in A-Auto or if the

FAILURE MODES AND EFFECTS ANALYSIS FOR THE ABGTS

Item No.	Component	Function	Failure Mode	Potential Cause	Method of Detection	Effect on System	Effect on Plant	Remarks
						pressure in ABSCE by admitting outside air.		fan in P-Auto starts, the operator cannot determine whether the signal is valid or spurious (no detection of spurious operation). This is acceptable since there is no impact on plant safety without a second failure (e.g., failure of a vacuum relief damper). See end of Table 6.2.3-3 for the effects of the potential ABGTS fan failures on the EGTS's ability to maintain the annulus -P more negative than that of the ABSCE.
4.	ABGTS Exhaust Fan B-B	Draws a portion of air in the ABSCE through an air cleanup unit (ACU) to remove radioactive contaminants and discharge into the shield building exhaust vent to maintain a negative pressure in the ABSCE relative to the outside.	Fails to start or fails to run.	Mechanical failure; Train B power failure; Train B ABI signal failure (HS in A-Auto).	Indicating light in MCR.	Loss of redundancy in ABGTS.	None. ABGTS Fan A-A can perform the functions of maintaining the ABSCE at a negative pressure and removing contaminant	Handswitches for ABGTS Fans A-A and B-B, in the MCR should normally be in the A-Auto position. On an ABI signal, both fans start and the operator may stop one fan and place its handswitch in the P-Auto (pull-out) position. This mode of operation is expected to occur after 30 minutes of two fan operation.

TABLE 6.2.3-3 (Sheet 6 of 37)

FAILURE MODES AND EFFECTS ANALYSIS FOR THE ABGTS

Item No.	Component	Function	Failure Mode	Potential Cause	Method of Detection	Effect on System	Effect on Plant	Remarks
								During an ABI, the fan in the P-Auto mode will start automatically on insufficient negative pressure in the ABSCE relative to the outside. An alarm is provided for the condition when flow is inadequate 45 seconds after fan start.
			Starts spuriously	Spurious Train B ABI signal (HS in A-Auto); spurious low flow signal from Fan A-A after valid ABI signal (HS in P-Auto).	See "Remarks" column	Vacuum relief line dampers may open to prevent excessive negative pressure in ABSCE by admitting outside air.	None.	Indicating light in MCR provide indication to operator that fan is running. However, if only one ABGTS train starts when both fans are in A-Auto or if the fan in P-Auto starts, the operator cannot determine whether the signal is valid or spurious (no detection of spurious operation). This is acceptable since there is no impact on plant safety without a second failure (e.g., failure of a vacuum relief damper). See end of Table 6.2.3-3 for the effects of the potential ABGTS fan failures on the EGTS's ability to maintain the

FAILURE MODES AND EFFECTS ANALYSIS FOR THE ABGTS

Item No.	Component	Function	Failure Mode	Potential Cause	Method of Detection	Effect on System	Effect on Plant	Remarks
								annulus ΔP more negative than that of the ABSCE.
5.	ABGTS Fan A-A Inlet Damper 1-FCO-30-146B	Provides flowpath for ABGTS Exhaust Fan A-A.	Fails to open or stuck closed.	Mechanical failure.	Alarm in MCR from 1-FS-30-146.	Loss of redundancy in ABGTS.	None. (See Remarks) .	ABGTS Fan A-A Inlet and Outlet Dampers 1-FCO-30-146A and 1-FCO-30-146B open on starting of associated fan and close when the fan stops; ABGTS Fan B-B provides redundancy.
		Provides isolation of ACU A-A when the Fan A-A is not running.	Fails to close, or stuck open.	Mechanical failure.	Alarm in MCR from 1-FS-30-146.	None. (See Remarks)	None. (See Remarks)	Damper 146A provides redundancy (Both dampers fail closed).
6.	ABGTS Fan A-A Outlet Damper 1-FCO-30-146A	Provides flowpath for ABGTS Exhaust Fan A-A.	Fails to open or stuck closed.	Mechanical failure.	Alarm in MCR from 1-FS-30-146.	Loss of redundancy in ABGTS.	None. (See Remarks)	ABGTS Fan A-A Inlet and Outlet Dampers 1-FCO-30-146A and 1-FCO-30-146B open on starting of associated fan and close when the fan stops.
		Provides isolation of ACU A-A when the Fan A-A is not running.	Fails to close, or stuck open.	Mechanical failure.	Alarm in MCR from 1-FS-30-146.	None. (See Remarks)	None. (See Remarks)	Damper 146B provides redundancy (both dampers fail closed).
7.	ABGTS Fan B-B Inlet Damper 2-FCO-30-157B	Provides flowpath for ABGTS Exhaust Fan B-B.	Fails to open or stuck closed.	Mechanical failure.	Alarm in MCR from 2-FS-30-165.	Loss of redundancy in ABGTS.	None. (See Remarks).	ABGTS Fan B-B Inlet and Outlet Dampers 2-FCO-30-157A and 2-FCO-30-157B open on starting of associated

TABLE 6.2.3-3 (Sheet 8 of 37)

FAILURE MODES AND EFFECTS ANALYSIS FOR THE ABGTS

Item No.	Component	Function	Failure Mode	Potential Cause	Method of Detection	Effect on System	Effect on Plant	Remarks
								fan and close when the fan stops; ABGTS Fan A-A provides redundancy.
		Provides isolation of ACU B-B when the Fan B-B is not running.	Fails to close or stuck open.	Mechanical failure.	Alarm in MCR from 2-FS-30-165.	Loss of redundancy in ABGTS.	None. (See Remarks).	Damper 157A provides redundancy (both dampers fail closed).
8.	ABGTS Fan B-B Outlet Damper 2-FCO-30-146A	Provides flowpath for ABGTS Exhaust Fan B-B.	Fails to open or stuck closed.	Mechanical failure.	Alarm in MCR from 2-FS-30-165.	Loss of redundancy in ABGTS.	None. (See Remarks).	ABGTS Fan B-B Inlet and Outlet Dampers 2-FCO-30-157A and 2-FCO-30-157B open on starting of associated fan and close when the fan stops; ABGTS Fan A-A provides redundancy.
		Provides isolation of ACU B-B when the Fan B-B is not running.	Fails to close or stuck open.	Mechanical failure.	Alarm in MCR from 2-FS-30-165.	Loss of redundancy in ABGTS.	None. (See Remarks).	Damper 157B provides redundancy (both dampers fail closed).
9.	Isolation Damper 0-FCO-30-137 Train A	Closes on ABI or high rad in refueling area signal to isolate Fuel Handling Area Exhaust Fan A-A and to establish boundary for ABGTS.	Stuck open, fails to close, or spuriously opens.	Mechanical failure; hot short in control wiring; Train A ABI or high rad in refueling area signal failure; HS failure to spring return from open to A-Auto.	Indicating light in MCR.	Loss of redundancy in isolation of Fuel Handling Area Exhaust Fan A-A.	None. Train B Damper 0-FCO-30-138 provides isolation and maintains the ABSCE.	Damper fails closed on loss of Train A 125 Vdc power. Train A (0-FCO-30-137) and Train B (0-FCO-30-138) dampers, in series, are provided with non-safety control air. Loss of the non-safety related control air does not prevent the damper from closing. Independence

FAILURE MODES AND EFFECTS ANALYSIS FOR THE ABGTS

Item No.	Component	Function	Failure Mode	Potential Cause	Method of Detection	Effect on System	Effect on Plant	Remarks
								of Train A and Train B isolation signals is discussed in "Remarks" under Items 1 and 2 of this table. Damper failure to open is not listed since the damper has no safety function to open for DBE mitigation.
10.	Isolation Damper 0-FCO-30-138 Train B	Closes on ABI or high rad in refueling area signal to isolate Fuel Handling Area Exhaust Fan A-A and to establish boundary for ABGTS.	Stuck open, fails to close, or spuriously opens.	Mechanical failure; hot short in control wiring; Train B ABI or high rad in refueling area signal failure; HS failure to spring return from open to A-Auto.	Indicating light in MCR.	Loss of redundancy in isolation of Fuel Handling Area Exhaust Fan A-A.	None. Train A Damper 0-FCO-30-137 provides isolation and maintains the ABSCE.	Damper fails closed on loss of Train B 125 Vdc power. Train A (0-FCO-30-137) and Train B (0-FCO-30-138) dampers, in series, are provided with non-safety control air. Loss of the non-safety related control air does not prevent the damper from closing. Independence of Train A and Train B isolation signals is discussed in "Remarks" under Items 1 and 2 of this table. Damper failure to open is not listed since the damper has no safety function to open for DBE mitigation.
11.	Isolation	Closes on ABI or	Stuck	Mechanical	Indicating light	Loss of	None. Train	Damper fails closed on

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FAILURE MODES AND EFFECTS ANALYSIS FOR THE ABGTS

Item No.	Component	Function	Failure Mode	Potential Cause	Method of Detection	Effect on System	Effect on Plant	Remarks
	Damper 0-FCO-30-1 40 Train A	high rad in refueling area signal to isolate Fuel Handling Area Exhaust Fan B-B and to establish boundary for ABGTS.	open, fails to close, or spuriously opens.	failure; hot short in control wiring; Train A ABI or high rad in refueling area signal failure; HS failure to spring return from open to A-Auto.	in MCR.	redundancy in isolation of Fuel Handling Area Exhaust Fan B-B.	B Damper 0-FCO-30 -141 provides isolation and maintains the ABSCE.	loss of Train A 125 Vdc power. Train A (0-FCO-30-140) and Train B (0-FCO-30-141) dampers, in series, are provided with non-safety control air. Loss of the non-safety related control air does not prevent the damper from closing. Independence of Train A and Train B isolation signals is discussed in "Remarks" under Items 1 and 2 of this table. Damper failure to open is not listed since the damper has no safety function to open for DBE mitigation.
12.	Isolation Damper 0-FCO-30-1 41 Train B	Closes on ABI or high rad in refueling area signal to isolate Fuel Handling Area Exhaust Fan B-B and to establish boundary for ABGTS.	Stuck open, fails to close, or spuriously opens.	Mechanical failure; hot short in control wiring; Train B ABI or high rad in refueling area signal failure; HS failure to spring return from open to A-Auto.	Indicating light in MCR.	Loss of redundancy in isolation of Fuel Handling Area Exhaust Fan B-B.	None. Train A Damper 0-FCO-30 -140 provides isolation and maintains the ABSCE.	Damper fails closed on loss of Train B 125 Vdc power. Train A (0-FCO-30-140) and Train B (0-FCO-30-141) dampers, in series, are provided with non-safety control air. Loss of the non-safety related control air does not prevent the damper from closing. Independence of Train A and Train B

FAILURE MODES AND EFFECTS ANALYSIS FOR THE ABGTS

Item No.	Component	Function	Failure Mode	Potential Cause	Method of Detection	Effect on System	Effect on Plant	Remarks
								isolation signals is discussed in "Remarks" under Items 1 and 2 of this table. Damper failure to open is not listed since the damper has no safety function to open for DBE mitigation.
13.	Isolation Damper 2-FCO-30-21 Train A	Closes on ABI or high rad in refueling area signal to isolate AB Gen Supply Fans 2A-A and 2B-B and to establish boundary for ABGTS.	Stuck open, fails to close, or spuriously opens.	Mechanical failure; hot short in control wiring; Train A ABI or high rad in refueling area signal failure; HS failure to spring return from open to A-Auto.	Indicating light in MCR.	Loss of redundancy in isolation of part of ductwork on Unit 2 side of AB.	None. Train B Damper 2-FCO-30-22 provides isolation and maintains the ABSCE.	Damper fails closed on loss of Train A 125 Vdc power. Train A (2-FCO-30-21) and Train B (2-FCO-30-22) dampers, in series, are provided with non-safety control air and both dampers fail closed on loss of control air. Independence of Train A and Train B isolation signals is discussed in "Remarks" under Items 1 and 2 of this table. Damper failure to open is not listed since the damper has no safety function to open for DBE mitigation.
14.	Isolation Damper 2-FCO-30-22 Train B	Closes on ABI or high rad in refueling area signal to isolate AB Gen	Stuck open, fails to close, or spuriously	Mechanical failure; hot short in control wiring; Train B ABI or	Indicating light in MCR.	Loss of redundancy in isolation of part of	None. Train A Damper 2-FCO-30-21 provides	Damper fails closed on loss of Train B 125 Vdc power. Train A (2-FCO-30-21) and Train

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FAILURE MODES AND EFFECTS ANALYSIS FOR THE ABGTS

Item No.	Component	Function	Failure Mode	Potential Cause	Method of Detection	Effect on System	Effect on Plant	Remarks
		Supply Fans 2A-A and 2B-B and to establish boundary for ABGTS.	opens.	high rad in refueling area signal failure; HS failure to spring return from open to A-Auto.		ductwork on Unit 2 side of AB.	isolation and maintains the ABSCE.	B (2-FCO-30-22) dampers, in series, are provided with non-safety control air and both dampers fail closed on loss of control air. Independence of Train A and Train B isolation signals is discussed in "Remarks" under Items 1 and 2 of this table. Damper failure to open is not listed since the damper has no safety function to open for DBE mitigation.
15.	Isolation Damper 1-FCO-30-86 Train A	Closes on ABI or high rad in refueling area signal to isolate AB Gen Supply Fans 1A-A and 1B-B and to establish boundary for ABGTS.	Stuck open, fails to close, or spuriously opens.	Mechanical failure; hot short in control wiring; Train A ABI or high rad in refueling area signal failure; HS failure to spring return from open to A-Auto.	Indicating light in MCR.	Loss of redundancy in isolation of part of ductwork on Unit 1 side of AB.	None. Train B Damper 1-FCO-30-87 provides isolation and maintains the ABSCE.	Damper fails closed on loss of Train A 125 Vdc power. Train A (1-FCO-30-86) and Train B (1-FCO-30-87) dampers, in series, are provided with non-safety control air and both dampers fail closed on loss of control air. Independence of Train A and Train B isolation signals is discussed in "Remarks" under Items 1 and 2 of this table. Damper failure to open is not listed since the damper has no safety

FAILURE MODES AND EFFECTS ANALYSIS FOR THE ABGTS

Item No.	Component	Function	Failure Mode	Potential Cause	Method of Detection	Effect on System	Effect on Plant	Remarks
								function to open for DBE mitigation.
16.	Isolation Damper 1-FCO-30-87 Train B	Closes on ABI or high rad in refueling area signal to isolate AB Gen Supply Fans 1A-A and 1B-B and to establish boundary for ABGTS.	Stuck open, fails to close, or spuriously opens.	Mechanical failure; hot short in control wiring; Train B ABI or high rad in refueling area signal failure; HS failure to spring return from open to A-Auto.	Indicating light in MCR.	Loss of redundancy in isolation of part of ductwork on Unit 1 side of AB.	None. Train A Damper 1-FCO-30-86 provides isolation and maintains the ABSCE.	Damper fails closed on loss of Train B 125 Vdc power. Train A (1-FCO-30-86) and Train B (1-FCO-30-87) dampers, in series, are provided with non-safety control air and both dampers fail closed on loss of control air. Independence of Train A and Train B isolation signals is discussed in "Remarks" under Items 1 and 2 of this table. Damper failure to open is not listed since the damper has no safety function to open for DBE mitigation.
17.	Isolation Damper 1-FCO-30-106 Train A	Closes on ABI or high rad in refueling area signal to isolate AB Gen Supply Fans 1A-A and 1B-B and to establish boundary for ABGTS.	Stuck open, fails to close, or spuriously opens.	Mechanical failure; hot short in control wiring; Train A ABI or high rad in refueling area signal failure; HS failure to spring return from open to A-Auto.	Indicating light in MCR.	Loss of redundancy in isolation of part of ductwork on Unit 1 side of AB.	None. Train B Damper 1-FCO-30-107 provides isolation and maintains the ABSCE.	Damper fails closed on loss of Train A 125 Vdc power. Train A (1-FCO-30-106) and Train B (1-FCO-30-107) dampers, in series, are provided with non-safety control air and both dampers fail closed on loss of control air. Independence of Train A and Train B isolation

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FAILURE MODES AND EFFECTS ANALYSIS FOR THE ABGTS

Item No.	Component	Function	Failure Mode	Potential Cause	Method of Detection	Effect on System	Effect on Plant	Remarks
								signals is discussed in "Remarks" under Items 1 and 2 of this table. Damper failure to open is not listed since the damper has no safety function to open for DBE mitigation.
18.	Isolation Damper 1-FCO-30-107 Train B	Closes on ABI or high rad in refueling area signal to isolate AB Gen Supply Fans 1A-A and 1B-B and to establish boundary for ABGTS.	Stuck open, fails to close, or spuriously opens.	Mechanical failure; hot short in control wiring; Train B ABI or high rad in refueling area signal failure; HS failure to spring return from open to A-Auto.	Indicating light in MCR.	Loss of redundancy in isolation of part of ductwork on Unit 1 side of AB.	None. Train A Damper 1-FCO-30-106 provides isolation and maintains the ABSCE.	Damper fails closed on loss of Train B 125 Vdc power. Train A (1-FCO-30-106) and Train B (1-FCO-30-107) dampers, in series, are provided with non-safety control air and both dampers fail closed on loss of control air. Independence of Train A and Train B isolation signals is discussed in "Remarks" under Items 1 and 2 of this table. Damper failure to open is not listed since the damper has no safety function to open for DBE mitigation.
19.	Isolation Damper 2-FCO-30-108 Train A	Closes on ABI or high rad in refueling area signal to isolate AB Gen Supply Fans 2A-A	Stuck open, fails to close, or spuriously opens.	Mechanical failure; hot short in control wiring; Train A ABI or high rad in	Indicating light in MCR.	Loss of redundancy in isolation of part of ductwork on	None. Train B Damper 2-FCO-30-109 provides	Damper fails closed on loss of Train A 125 Vdc power. Train A (2-FCO-30-108) and Train B (2-FCO-30-109)

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FAILURE MODES AND EFFECTS ANALYSIS FOR THE ABGTS

Item No.	Component	Function	Failure Mode	Potential Cause	Method of Detection	Effect on System	Effect on Plant	Remarks
		and 2B-B and to establish boundary for ABGTS.		refueling area signal failure; HS failure to spring return from open to A-Auto.		Unit 2 side of AB.	isolation and maintains the ABSCE.	dampers, in series, are provided with non-safety control air and both dampers fail closed on loss of control air. Independence of Train A and Train B isolation signals is discussed in "Remarks" under Items 1 and 2 of this table. Damper failure to open is not listed since the damper has no safety function to open for DBE mitigation.
20.	Isolation Damper 2-FCO-30-109 Train B	Closes on ABI or high rad in refueling area signal to isolate AB Gen Supply Fans 2A-A and 2B-B and to establish boundary for ABGTS.	Stuck open, fails to close, or spuriously opens.	Mechanical failure; hot short in control wiring; Train B ABI or high rad in refueling area signal failure; HS failure to spring return from open to A-Auto.	Indicating light in MCR.	Loss of redundancy in isolation of part of ductwork on Unit 2 side of AB.	None. Train A Damper 2-FCO-30-108 provides isolation and maintains the ABSCE.	Damper fails closed on loss of Train B 125 Vdc power. Train A (2-FCO-30-108) and Train B (2-FCO-30-109) dampers, in series, are provided with non-safety control air and both dampers fail closed on loss of control air. Independence of Train A and Train B isolation signals is discussed in "Remarks" under Items 1 and 2 of this table. Damper failure to open is not listed since the damper has no safety function to open for DBE mitigation.

FAILURE MODES AND EFFECTS ANALYSIS FOR THE ABGTS

Item No.	Component	Function	Failure Mode	Potential Cause	Method of Detection	Effect on System	Effect on Plant	Remarks
								mitigation.
21.	Isolation Damper 1-FCO-30-160 Train A	Closes on ABI or high rad in refueling area signal to isolate AB Gen Exhaust Fan 1A-A suction and to establish boundary for ABGTS.	Stuck open, fails to close, or spuriously opens.	Mechanical failure; hot short in control wiring; Train A ABI or high rad in refueling area signal failure; HS failure to spring return from open to A-Auto.	Indicating light in MCR.	Loss of redundancy in isolation of AB Gen Exhaust Fan 1A-A.	None. Train B Damper 1-FCO-30-161 provides isolation and maintains the ABSCE.	Damper fails closed on loss of Train A 125 Vdc power. Train A (1-FCO-30-160) and Train B (1-FCO-30-161) dampers, in series, are provided with non-safety control air. Loss of the non-safety related control air does not prevent the damper from closing. Independence of Train A and Train B isolation signals is discussed in “Remarks” under Items 1 and 2 of this table. Damper failure to open is not listed since the damper has no safety function to open for DBE mitigation.
22.	Isolation Damper 1-FCO-30-161 Train B	Closes on ABI or high rad in refueling area signal to isolate AB Gen Exhaust Fan 1A-A suction and to establish boundary for ABGTS.	Stuck open, fails to close, or spuriously opens.	Mechanical failure; hot short in control wiring; Train B ABI or high rad in refueling area signal failure; HS failure to spring return from open to A-Auto.	Indicating light in MCR.	Loss of redundancy in isolation of AB Gen Exhaust Fan 1A-A.	None. Train A Damper 1-FCO-30-160 provides isolation and maintains the ABSCE.	Damper fails closed on loss of Train B 125 Vdc power. Train A (1-FCO-30-160) and Train B (1-FCO-30-161) dampers, in series, are provided with non-safety control air. Loss of the non-safety related control air does not prevent the damper from closing. Independence of Train A

FAILURE MODES AND EFFECTS ANALYSIS FOR THE ABGTS

Item No.	Component	Function	Failure Mode	Potential Cause	Method of Detection	Effect on System	Effect on Plant	Remarks
								and Train B isolation signals is discussed in "Remarks" under Items 1 and 2 of this table. Damper failure to open is not listed since the damper has no safety function to open for DBE mitigation.
23.	Isolation Damper 1-FCO-30-166 Train A	Closes on ABI or high rad in refueling area signal to isolate AB Gen Exhaust Fan 1B-B suction and to establish boundary for ABGTS.	Stuck open, fails to close, or spuriously opens.	Mechanical failure; hot short in control wiring; Train A ABI or high rad in refueling area signal failure; HS failure to spring return from open to A-Auto.	Indicating light in MCR.	Loss of redundancy in isolation of AB Gen Exhaust Fan 1B-B.	None. Train B Damper 1-FCO-30-167 provides isolation and maintains the ABSCE.	Dampers fails closed on loss of Train A 125 Vdc power. Train A (1-FCO-30-166) and Train B (1-FCO-30-167) dampers, in series, are provided with non-safety control air. Loss of the non-safety related control air does not prevent the damper from closing. Independence of Train A and Train B isolation signals is discussed in "Remarks" under Items 1 and 2 of this table. Damper failure to open is not listed since the damper has no safety function to open for DBE mitigation.
24.	Isolation Damper 1-FCO-30-1	Closes on ABI or high rad in refueling area signal to	Stuck open, fails to close, or	Mechanical failure; hot short in control wiring;	Indicating light in MCR.	Loss of redundancy in isolation of	None. Train A Damper 1-FCO-30	Damper fails closed on loss of Train B 125 Vdc power. Train A

FAILURE MODES AND EFFECTS ANALYSIS FOR THE ABGTS

Item No.	Component	Function	Failure Mode	Potential Cause	Method of Detection	Effect on System	Effect on Plant	Remarks
	67 Train B	isolate AB Gen Exhaust Fan 1B-B suction and to establish boundary for ABGTS.	spuriously opens.	Train B ABI or high rad in refueling area signal failure; HS failure to spring return from open to A-Auto.		AB Gen Exhaust Fan 1B-B.	-166 provides isolation and maintains the ABSCE.	(1-FCO-30-166) and Train B (1-FCO-30-167) dampers, in series, are provided with non-safety control air. Loss of the non-safety related control air does not prevent the damper from closing. Independence of Train A and Train B isolation signals is discussed in “Remarks” under Items 1 and 2 of this table. Damper failure to open is not listed since the damper has no safety function to open for DBE mitigation.
25.	Isolation Damper 2-FCO-30-271 Train A	Closes on ABI or high rad in refueling area signal to isolate AB Gen Exhaust Fan 2A-A suction and to establish boundary for ABGTS.	Stuck open, fails to close, or spuriously opens.	Mechanical failure; hot short in control wiring; Train A ABI or high rad in refueling area signal failure; HS failure to spring return from open to A-Auto.	Indicating light in MCR.	Loss of redundancy in isolation of AB Gen Exhaust Fan 2A-A.	None. Train B Damper 2-FCO-30-272 provides isolation and maintains the ABSCE.	Damper fails closed on loss of Train A 125 Vdc power. Train A (2-FCO-30-271) and Train B (2-FCO-30-272) dampers, in series, are provided with non-safety control air. Loss of the non-safety related control air does not prevent the damper from closing. Independence of Train A and Train B isolation signals is discussed in “Remarks”

FAILURE MODES AND EFFECTS ANALYSIS FOR THE ABGTS

Item No.	Component	Function	Failure Mode	Potential Cause	Method of Detection	Effect on System	Effect on Plant	Remarks
								under Items 1 and 2 of this table. Damper failure to open is not listed since the damper has no safety function to open for DBE mitigation.
26.	Isolation Damper 2-FCO-30-272 Train B	Closes on ABI or high rad in refueling area signal to isolate AB Gen Exhaust Fan 2A-A suction and to establish boundary for ABGTS.	Stuck open, fails to close, or spuriously opens.	Mechanical failure; hot short in control wiring; Train B ABI or high rad in refueling area signal failure; HS failure to spring return from open to A-Auto.	Indicating light in MCR.	Loss of redundancy in isolation of AB Gen Exhaust Fan 2A-A.	None. Train A Damper 2-FCO-30-271 provides isolation and maintains the ABSCE.	Damper fails closed on loss of Train B 125 Vdc power. Train A (2-FCV-30-271) and Train B (2-FCV-30-272) dampers, in series, are provided with non-safety control air. Loss of non-safety related control air does not prevent the damper from closing. Independence of Train A and Train B isolation signals is discussed in "Remarks" under Items 1 and 2 of this table. Damper failure to open is not listed since the damper has no safety function to open for DBE mitigation.
27.	Isolation Damper 2-FCO-30-275 Train A	Closes on ABI or high rad in refueling area signal to isolate AB Gen Exhaust Fan 2B-B	Stuck open, fails to close, or spuriously opens.	Mechanical failure; hot short in control wiring; Train A ABI or high rad in	Indicating light in MCR.	Loss of redundancy in isolation of AB Gen Exhaust Fan	None. Train B Damper 2-FCO-30-276 provides	Damper fails closed on loss of Train A 125 Vdc power. Train A (2-FCO-30-275) and Train B (2-FCO-30-276)

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FAILURE MODES AND EFFECTS ANALYSIS FOR THE ABGTS

Item No.	Component	Function	Failure Mode	Potential Cause	Method of Detection	Effect on System	Effect on Plant	Remarks
		suction and Exhaust Fan B-B and to establish boundary for ABGTS.		refueling area signal failure; HS failure to spring return from open to A-Auto.		2B-B.	isolation and maintains the ABSCE.	dampers, in series, are provided with non-safety control air. Loss of the non-safety related control air does not prevent the damper from closing. Independence of Train A and Train B isolation signals is discussed in “Remarks’ under items 1 and 2 of this Table. Damper failure to open is not listed since the damper has no safety function to open for DBE mitigation.
28.	Isolation Damper 2-FCO-30-276 Train B	Closes on ABI or high rad in refueling area signal to isolate AB Gen Exhaust Fan 2B-B suction area and to establish boundary for ABGTS.	Stuck open, fails to close, or spuriously opens.	Mechanical failure; hot short in control wiring; Train B ABI or high rad in refueling area signal failure; HS failure to spring return from open to A-Auto.	Indicating light in MCR.	Loss of redundancy in isolation of AB Gen Exhaust Fan 2B-B.	None. Train A Damper 2-FCO-30-275 provides isolation and maintains the ABSCE.	Damper fails closed on loss of Train B 125 Vdc power. Train A (2-FCV-30-275) and Train B (2-FCV-30-276) dampers, in series, are provided with non-safety control air. Loss of non-safety related control air does not prevent the damper from closing. Independence of Train A and Train B isolation signals is discussed in "Remarks" under Items 1 and 2 of this table. Damper failure to open

FAILURE MODES AND EFFECTS ANALYSIS FOR THE ABGTS

Item No.	Component	Function	Failure Mode	Potential Cause	Method of Detection	Effect on System	Effect on Plant	Remarks
								is not listed since the damper has no safety function to open for DBE mitigation.
29.	Isolation Damper 1, 2-FCO-30-294 Train A**	Closes on ABI or high rad in refueling area signal to isolate Purge Air Supply Fan inlet duct and to establish boundary for ABGTS.	Stuck open, fails to close, or spuriously opens.	Mechanical failure; hot short in control wiring; Train A ABI or high rad in refueling area signal failure; HS failure to spring return from open to A-Auto.	Indicating light in MCR.	Loss of redundancy in isolation of Purge Air Supply Fan inlet duct.	None. Train B Damper 1,2-FCO-30-295 provides isolation and maintains the ABSCE.	Damper fails closed on loss of Train A 125 Vdc power. Train A (1,2-FCV-30-294) and Train B (1,2-FCV-30-295) dampers, in series, are provided with non-safety control air and both dampers fail closed on loss of control air. Independence of Train A and Train B isolation signals is discussed in "Remarks" under Items 1 and 2 of this table. Solenoid circuits for 1,2-FCO-30-294 and 1,2-FCO-30-295 are independent, with trained Class 1E power and trained ABI signal. Damper failure to open is not listed since the damper has no safety function to open for DBE mitigation.
30.	Isolation Damper 1,2-FCO-30	Closes on ABI or high rad in refueling area signal to	Stuck open, fails to close, or	Mechanical failure; hot short in control wiring;	Indicating light in MCR.	Loss of redundancy in isolation of	None. Train A Damper 1,2-FCO-30	Damper fails closed on loss of Train B 125 Vdc power. Train A

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FAILURE MODES AND EFFECTS ANALYSIS FOR THE ABGTS

Item No.	Component	Function	Failure Mode	Potential Cause	Method of Detection	Effect on System	Effect on Plant	Remarks
	-295 Train B**	isolate Purge Air Supply Fan inlet duct and to establish boundary for ABGTS.	spuriously opens.	Train B ABI or high rad in refueling area signal failure; HS failure to spring return from open to A-Auto.		Purge Air Supply Fan inlet duct.	-294 provides isolation and maintains the ABSCE.	(1,2-FCV-30-294) and Train B (1,2-FCV-30-295) dampers, in series, are provided with non-safety control air and both dampers fail closed on loss of control air. Independence of Train A and Train B isolation signals is discussed in "Remarks" under Items 1 and 2 of this table. Solenoid circuits for 1,2-FCO-30-294 and 1,2-FCO-30-295 are independent, with trained Class 1E power and trained ABI signal. Damper failure to open is not listed since the damper has no safety function to open for DBE mitigation.
31.	Isolation Damper 0-FCO-30-122 Train A	Closes on ABI or high rad in refueling area signal to isolate the cask loading area exhaust and to establish boundary for ABGTS.	Stuck open, fails to close, or spuriously opens.	Mechanical failure; hot short in control wiring; Train A ABI or high rad in refueling area signal failure; HS failure to spring return from open to A-Auto.	Indicating light in MCR.	Loss of redundancy in isolation of cask loading area exhaust.	None. Train B Damper 0-FCO-30-123 provides isolation and maintains the ABSCE.	Damper fails closed on loss of Train A 125 Vdc power. Train A (0-FCO-30-122) and Train B (0-FCO-30-123) dampers, in series, are provided with non-safety control air and both dampers fail closed on loss of control air.

FAILURE MODES AND EFFECTS ANALYSIS FOR THE ABGTS

Item No.	Component	Function	Failure Mode	Potential Cause	Method of Detection	Effect on System	Effect on Plant	Remarks
				Damper re-opens on loss (or RESET) of ABI signal.				Independence of Train A and Train B isolation signals is discussed in "Remarks" under Items 1 and 2 of this table. Damper failure to open is not listed since the damper has no safety function to open for DBE mitigation.
32.	Isolation Damper 0-FCO-30-123 Train B	Closes on ABI or high rad in refueling area signal to isolate the cask loading area exhaust and to establish boundary for ABGTS.	Stuck open, fails to close, or spuriously opens.	Mechanical failure; hot short in control wiring; Train B ABI or high rad in refueling area signal failure; HS failure to spring return from open to A-Auto. Damper re-opens on loss (or RESET) of ABI signal.	Indicating light in MCR.	Loss of redundancy in isolation of cask loading area exhaust.	None. Train A Damper 0-FCO-30-122 provides isolation and maintains the ABSCE.	Damper fails closed on loss of Train B 125 Vdc power. Train A (0-FCO-30-122) and Train B (0-FCO-30-123) dampers, in series, are provided with non-safety control air and both dampers fail closed on loss of control air. Independence of Train A and Train B isolation signals is discussed in "Remarks" under Items 1 and 2 of this table. Damper failure to open is not listed since the damper has no safety function to open for DBE mitigation.
33.	Isolation Damper 0-FCO-30-1	Closes on ABI or high rad in refueling area signal to	Stuck open, fails to close, or	Mechanical failure; hot short in control wiring;	Indicating light in MCR.	Loss of redundancy in isolation of	None. Train B Damper 0-FCO-30	Damper fails closed on loss of Train A 125 Vdc power. Train A

FAILURE MODES AND EFFECTS ANALYSIS FOR THE ABGTS

Item No.	Component	Function	Failure Mode	Potential Cause	Method of Detection	Effect on System	Effect on Plant	Remarks
	29 Train A	isolate the cask loading area supply and to establish boundary for ABGTS.	spuriously opens.	Train A ABI or high rad in refueling area signal failure; HS failure to spring return from open to A-Auto.		cask loading area supply.	-130 provides isolation and maintains the ABSCE.	(0-FCO-30-129) and Train B (0-FCO-30-130) dampers, in series, are provided with non-safety control air and both dampers fail closed on loss of control air. Independence of Train A and Train B isolation signals is discussed in "Remarks" under Items 1 and 2 of this table. Damper failure to open is not listed since the damper has no safety function to open for DBE mitigation.
34.	Isolation Damper 0-FCO-30-130 Train B	Closes on ABI or high rad in refueling area signal to isolate the cask loading area supply and to establish boundary for ABGTS.	Stuck open, fails to close, or spuriously opens.	Mechanical failure; hot short in control wiring; Train B ABI or high rad in refueling area signal failure; HS failure to spring return from open to A-Auto.	Indicating light in MCR.	Loss of redundancy in isolation of cask loading area supply.	None. Train A Damper 0-FCO-30-129 provides isolation and maintains the ABSCE.	Damper fails closed on loss of Train B 125 Vdc power. Train A (0-FCO-30-129) and Train B (0-FCO-30-130) dampers, in series, are provided with non-safety control air and both dampers fail closed on loss of control air. Independence of Train A and Train B isolation signals is discussed in "Remarks" under Items 1 and 2 of this table. Damper failure to open is not listed since the

FAILURE MODES AND EFFECTS ANALYSIS FOR THE ABGTS

Item No.	Component	Function	Failure Mode	Potential Cause	Method of Detection	Effect on System	Effect on Plant	Remarks
								damper has no safety function to open for DBE mitigation.
35.	Isolation Damper 0-FCO-31-350 Train A	Closes on ABI or high rad in refueling area signal to isolate PASF outside air intake and to establish boundary for ABGTS.	Stuck open, fails to close, or spuriously opens.	Mechanical failure; hot short in control wiring; Train A ABI or high rad in refueling area signal failure; HS failure to spring return from open to A-Auto.	Indicating light in MCR.	Loss of redundancy in isolation of PASF outside air intake.	None. Train B Damper 0-FCO-31-365 provides isolation and maintains the ABSCE.	Damper fails closed on loss of Train A 125 Vdc power. Train A (0-FCO-31-350) and Train B (0-FCO-31-365) dampers are in series. Independence of Train A and Train B isolation signals is discussed in "Remarks" under Items 1 and 2 of this table. Damper failure to open is not listed since the damper has no safety function to open for DBE mitigation.
36.	Isolation Damper 0-FCO-31-365 Train B	Closes on ABI or high rad in refueling area signal to isolate PASF outside air intake and to establish boundary for ABGTS.	Stuck open, fails to close, or spuriously opens.	Mechanical failure; hot short in control wiring; Train B ABI or high rad in refueling area signal failure; HS failure to spring return from open to A-Auto.	Indicating light in MCR.	Loss of redundancy in isolation of PASF outside air intake.	None. Train A Damper 0-FCO-31-350 provides isolation and maintains the ABSCE.	Damper fails closed on loss of Train B 125 Vdc power. Train A (0-FCV-31-350) and Train B (0-FCV-31-365) dampers are in series. Independence of Train A and Train B isolation signals is discussed in "Remarks" under Items 1 and 2 of this table. Damper failure to open is not listed since the damper has no safety

FAILURE MODES AND EFFECTS ANALYSIS FOR THE ABGTS

Item No.	Component	Function	Failure Mode	Potential Cause	Method of Detection	Effect on System	Effect on Plant	Remarks
								function to open for DBE mitigation.
37.	Isolation Damper 1-FCO-31-342 Train A	Closes on ABI or high rad in refueling area signal to isolate PASF Room No. 1 exhaust and to establish boundary for ABGTS.	Stuck open, fails to close, or spuriously opens.	Mechanical failure; hot short in control wiring; Train A ABI or high rad in refueling area signal failure; HS failure to spring return from open to A-Auto.	Indicating light in MCR.	Loss of redundancy in isolation of PASF Room No. 1 exhaust.	None. Train B Damper 1-FCO-31-343 provides isolation and maintains the ABSCE.	Damper fails closed on loss of Train A 125 Vdc power. Train A (1-FCO-31-342) and Train B (1-FCO-31-343) dampers are in series. Independence of Train A and Train B isolation signals is discussed in "Remarks" under Items 1 and 2 of this table. Damper failure to open is not listed since the damper has no safety function to open for DBE mitigation.
38.	Isolation Damper 1-FCO-31-343 Train B	Closes on ABI or high rad in refueling area signal to isolate PASF Room No. 1 exhaust and to establish boundary for ABGTS.	Stuck open, fails to close, or spuriously opens.	Mechanical failure; hot short in control wiring; Train B ABI or high rad in refueling area signal failure; HS failure to spring return from open to A-Auto.	Indicating light in MCR.	Loss of redundancy in isolation of PASF Room No. 1 exhaust.	None. Train A Damper 1-FCO-31-342 provides isolation and maintains the ABSCE.	Damper fails closed on loss of Train B 125 Vdc power. Train A (1-FCO-31-342) and Train B (1-FCO-31-343) dampers are in series. Independence of Train A and Train B isolation signals is discussed in "Remarks" under Items 1 and 2 of this table. Damper failure to open is not listed since the damper has no safety function to open for DBE mitigation.

TABLE 6.2.3-3 (Sheet 27 of 37)

FAILURE MODES AND EFFECTS ANALYSIS FOR THE ABGTS

Item No.	Component	Function	Failure Mode	Potential Cause	Method of Detection	Effect on System	Effect on Plant	Remarks
								function to open for DBE mitigation.
39.	Modulating Damper 0-FCO-30-149 Train A	Regulates the amount of outside air to maintain ABSCE at -0.25 in w.g.	Allows more outside air than required (stuck open or spuriously excessive)	Train A vital ac power failure; Train A aux. control air failure; spurious high dP signal; failure of E/I or I/P converter; positioner mechanical failure.	"Hi Press. in Aux. Bldg." alarm.	See "Remarks" column.	None. See "Remarks" column.	ABI signal stops AB gen supply fans automatically. The Aux. Bldg. is designed for minimum leakage and, with at least one ABGTS exhaust fan running, failure of the modulating damper can cause pressure in the Aux. Bldg. to approach outside pressure, but the AB pressure cannot become positive with respect to the outside. The DPIS used for alarm in the control room is separate and independent from the DPIS used for modulating control of the damper. If the (non-safety) alarm functions, the operator can either close the associated Isolation Damper 0-FCO-30-280 or, if only one ABGTS exhaust fan is in operation, can start the redundant fan to maintain the -0.25 in

FAILURE MODES AND EFFECTS ANALYSIS FOR THE ABGTS

Item No.	Component	Function	Failure Mode	Potential Cause	Method of Detection	Effect on System	Effect on Plant	Remarks
								w.g. pressure.
			Does not allow the required amount of outside air (stuck closed or spurious inadequate opening).	Train A vital ac power failure; Train A aux. control air failure; spurious low dP signal from 0-FCO-30-149; failure of E/I or I/P converter; positioner mechanical failure.		None. Train B Modulating Damper 0-FCO-30-148 will open to allow sufficient outside air to control the negative pressure.	None.(See note on items 3 and 4)	Modulating Dampers 0-FCO-30-148 and 0-FCO-30-149 are provided with train-separated, safety-grade auxiliary control air.

TABLE 6.2.3-3 (Sheet 29 of 37)

FAILURE MODES AND EFFECTS ANALYSIS FOR THE ABGTS

Item No.	Component	Function	Failure Mode	Potential Cause	Method of Detection	Effect on System	Effect on Plant	Remarks
40.	Modulating Damper 0-FCO-30-148 Train B	Regulates the amount of outside air to maintain ABSCE at -0.25 in w.g.	Allows more outside air than required (stuck open or spuriously excessive opening).	Train B vital ac power failure; Train B aux. control air failure; spurious high dP signal; failure of E/I or I/P converter; positioner mechanical failure.	"Hi Press. in Aux. Bldg." alarm.	See "Remarks" column.	None. See "Remarks" column.	ABI signal stops AB gen supply fans automatically. The Aux. Bldg. is designed for minimum leakage and, with no air coming in and with at least one ABGTS exhaust fan running, failure of the modulating damper can cause pressure in the Aux. Bldg. to approach outside pressure, but the AB pressure cannot become positive with respect to the outside. The DPIS used for alarm in the control room is separate and independent from the DPIS used for modulating control of the damper. If the (non-safety) alarm functions, the operator can either close the associated Isolation Damper 0-FCO-30-280 or, if only one ABGTS exhaust fan is in operation, can start the redundant fan to maintain -0.25 in w.g.

TABLE 6.2.3-3 (Sheet 30 of 37)

FAILURE MODES AND EFFECTS ANALYSIS FOR THE ABGTS

Item No.	Component	Function	Failure Mode	Potential Cause	Method of Detection	Effect on System	Effect on Plant	Remarks
			Does not allow the required amount of outside air (stuck closed or spurious inadequate opening)	Train B vital ac power failure; Train B aux. control air failure; spurious low ΔP signal from 0-FCO-30-149; failure of E/I or I/P converter; positioner mechanical failure.		None. Train A Modulating Damper 0-FCO-30-149 will open to allow sufficient outside air to control the negative pressure.	None.(see note on items 3 and 4)	Modulating Dampers 0-FCO-30-148 and 0-FCO-30-149 are provided with train-separated, safety-grade auxiliary control air.
41.	ABGTS Vacuum Relief Line Isolation Damper 0-FCO-30-280 Train A	Provides flow path for outside air.	Fails to open, stuck closed, or spuriously closes.	Mechanical failure; Train A power failure; Train A aux. control air failure; operator error (HS in wrong position).	Indicating light in MCR.	Aux. Bldg. at more negative pressure (lower absolute pressure) than required to prevent leakage from outside.	None. (See Remarks)	Dampers 0-FCO-30-279 and 0-FCO-30-280 are provided with train-separated, safety-grade auxiliary control air. In addition, there are two vacuum breaker dampers, 0-DMP-30-1128 and 0-DMP-30-1129 in series which will admit outside air into the Bldg in case of increasing vacuum (more negative pressure).
			Fails to close, stuck open, or spuriously opens.	Mechanical failure; operator error (HS in wrong position).	Indicating light in MCR	None. Modulating Damper 0-FCO-30-149 can independentl	None. (See Remarks)	The modulating damper cannot shut off outside air completely. However, the amount of air passed will be insignificant to cause

FAILURE MODES AND EFFECTS ANALYSIS FOR THE ABGTS

Item No.	Component	Function	Failure Mode	Potential Cause	Method of Detection	Effect on System	Effect on Plant	Remarks
						y control amount of outside air.		pressure control problem.
42.	ABGTS Vacuum Relief Line Isolation Damper 0-FCO-30-279 Train B	Provides flow path for outside air.	Fails to open, stuck closed, or spuriously closes.	Mechanical failure; Train B power failure; Train B aux. control air failure; operator error (HS in wrong position).	Ind. light in MCR.	Aux. Bldg. At more negative pressure (lower absolute pressure) than required to prevent leakage from outside.	None. (See Remarks).	Dampers 0-FCO-30-279 and 0-FCO-30-280 are provided with train-separated, safety-grade auxiliary control air. In addition, there are two vacuum breaker dampers, 0-DMP-30-1128 and 0-DMP-30-1129 in series which will admit outside air into the Bldg in case of increasing vacuum.
			Fails to close, stuck open, or spuriously opens.	Mechanical failure; operator error (HS in wrong position).	Indicating light in MCR.	None. Modulating Damper 0-FCO-30-148 can independently control amount of outside air.	None	The modulating damper cannot shut off outside air completely. However, the amount of air passed will be insignificant to cause pressure control problem.
43.	Train A Emergency Power	Provides Class 1E diesel-backed power supply to active components of Train A of ABGTS.	Loss of or inadequate voltage.	Diesel generator failure; bus fault (Train A); operator error.	Alarm and indication in MCR.	Loss of redundancy in ABGTS exhaust flow paths.	None. Redundant Train B exhaust fan can maintain required negative	Train A isolation dampers are not directly affected since damper solenoids and control circuits are supplied either battery power or battery-backed vital ac power. Loss of power to

TABLE 6.2.3-3 (Sheet 32 of 37)

FAILURE MODES AND EFFECTS ANALYSIS FOR THE ABGTS

Item No.	Component	Function	Failure Mode	Potential Cause	Method of Detection	Effect on System	Effect on Plant	Remarks
							pressure.	the damper control circuits does not result in loss of redundancy since circuits are such that isolation dampers fail closed.
44.	Train B Emergency Power	Provides Class 1E diesel-backed power supply to active components of Train B of ABGTS.	Loss of or inadequate voltage.	Diesel generator failure; bus fault (Train B); operator error.	Alarm and indication in MCR.	Loss of redundancy in ABGTS exhaust flow paths.	None. Redundant Train A exhaust fan can maintain required negative pressure.	Train B isolation dampers are not directly affected since damper solenoids and control circuits are supplied either battery power or battery-backed vital ac power. Loss of power to the damper control circuits does not result in loss of redundancy since circuits are such that isolation dampers fail closed.
45.	Aux. Bldg. vacuum relief damper 0-DMP-30-1 128	Provides flow path for outside air.	Fails to open; stuck closed.	Mechanical Failure	Visual	Aux. Bldg at more negative press. (lower absolute press.) than req'd to prevent leakage to outside.	None.	This damper will only be used in the event that isolation damper 1-DMP-30-279 or 0-DMP-30-280 fail closed. Therefore, for this damper to fail closed, and one of the isolation dampers to fail closed at the same time would constitute a double failure.
			Fails to	Mechanical	Visual	None.		Vacuum relief dampers

TABLE 6.2.3-3 (Sheet 33 of 37)

FAILURE MODES AND EFFECTS ANALYSIS FOR THE ABGTS

Item No.	Component	Function	Failure Mode	Potential Cause	Method of Detection	Effect on System	Effect on Plant	Remarks
			close; stuck open.	Failure		Vacuum relief damper 0-DMP-30-1 129 can close independentl y and eliminate flow path from outside air.		0-DMP-30-1128 and 0-DMP-30-1129 are installed in series.
46.	Aux. Bldg. vacuum relief damper 0-DMP-30-1 129	Provides flow path for outside air.	Fails to open; stuck closed.	Mechanical Failure	Visual	Aux. Bldg. at more negative press. (lower absolute press.) than required to prevent leakage to outside.	None.	This damper will only be used in the event that isolation damper 0-DMP-30-279 or 0-DMP-30-280 fail closed. Therefore, for this damper to fail closed, and one of the isolation dampers to fail closed at the same time would constitute a double failure.
			Fails to close; stuck open.	Mechanical Failure	Visual	None. Vacuum relief damper 0-DMP-30-1 128 can close independentl y and eliminate flow path from outside	None. (See Remarks.)	Vacuum relief damper 0-DMP-30-1128 and 0-DMP-30-1129 are installed in series.

FAILURE MODES AND EFFECTS ANALYSIS FOR THE ABGTS

Item No.	Component	Function	Failure Mode	Potential Cause	Method of Detection	Effect on System	Effect on Plant	Remarks
						air.		
47.	Fire Dampers 0-ISV-31-38 34 and 0-ISV-31-38 45	Provide air flow path for common duct between ABGTS fans.	Spurious closure.	Failure of fusible link.	Low flow alarm	Loss of Train B	None. (See Remarks).	Fire dampers are safety-related; their failure constitutes a single failure. Therefore, both ABGTS trains would be available.
48.	Deluge System (Spray Heads and piping inside ACU A-A or B-B)	Floods the carbon adsorbers in event of fire.	Spurious opening of spray heads, or piping failure.	Failure of fusible link. Loss of piping pressure boundary.	---	None. (See Remarks).	None. (See Remarks).	Opposite train ACU is independent and remains available, or deluge valve remains closed in the absence of heat detection.
48b.	Deluge Valve 1-FCV-26-163	Discharge water in event of fire in adsorbers.	Spurious actuation.	Mechanical failure. or Electrical failure.	Local and MCR alarms on water discharge	None. (See Remarks).	None. (See Remarks).	ACU will not be flooded since the fusible links on the spray heads will be intact due to absence of heat.
48c.	Deluge Valve 1-FCV-26-171	Discharge water in event of fire in adsorbers.	Spurious actuation.	Mechanical failure. or Electrical failure.	Local and MCR alarms on water discharge	None. (See Remarks).	None. (See Remarks).	ACU will not be flooded since the fusible links on the spray heads will be intact due to absence of heat.
49.	Ductwork in the ABSCE	Provides containment for air flow path.	Leakage.	Cracks.	---	Minimal localized reduction of negative pressure.	None. (See Remarks).	Only small cracks are postulated due to seismic qualification. Minimal localized reduction of negative pressure will not affect the ABSCE. Loss of fluid (air) is not a

FAILURE MODES AND EFFECTS ANALYSIS FOR THE ABGTS

Item No.	Component	Function	Failure Mode	Potential Cause	Method of Detection	Effect on System	Effect on Plant	Remarks
								concern since the system is submerged in the same fluid.
50.	ABGTS Air Cleanup Unit A Heater	Controls humidity of exhaust air.	Fails to turn on or fails to operate.	Train A power failure; temperature sensing error.	Hi rad alarm in MCR for air to Shield Bldg. vent.	Loss of redundancy in ABGTS. Failure of heater will allow humid air into carbon filter reducing its efficiency (see "Remark #3").	None. (See Remarks)	1. Failure to cutout is not considered in this table since this is the safe position for controlling air humidity. 2. The air temperature for the switch, 0-TS-30-143, for this heater is sensed by a non-safety related sensor, 0-TE-30-143. DCN M-21750 has been initiated to resolve this problem. Section 8.2 address this further. 3. Heater operation is tested every 31 days per procedure.
51.	ABGTS Air Cleanup Unit B Heater	Controls humidity of exhaust air.	Fails to turn on or fails to operate.	Train B power failure; temperature sensing error.	Hi rad alarm in MCR for air to Shield Bldg. vent.	Loss of redundancy in ABGTS. Failure of heater will allow humid air into carbon filter reducing its efficiency	None. (See Remarks)	1. Failure to cutout is not considered in this table since this is the safe position for controlling air humidity. 2. The air temperature for the switch, 0-TS-30-143, for this heater is sensed by a

FAILURE MODES AND EFFECTS ANALYSIS FOR THE ABGTS

Item No.	Component	Function	Failure Mode	Potential Cause	Method of Detection	Effect on System	Effect on Plant	Remarks
						(see "Remark #3").		non-safety related sensor, 0-TE-30-143. DCN M-2175 has been initiated to resolve this problem. Section 8.2 address this further. 3. Heater operation is tested every 31 days per procedure.

* Interim U1 configuration: As a result of the changes made by DCN 56414, a spurious ABI signal would also result in a Containment Vent Isolation (CVI) signal. Dual Unit configuration: EDCR 55801 provides for a functional cross-tie between the two units which would cause a refueling unit CVI upon spurious ABI Signal.

** Unit 2 component, 2-FCO-30-294, was added for dual unit operation.

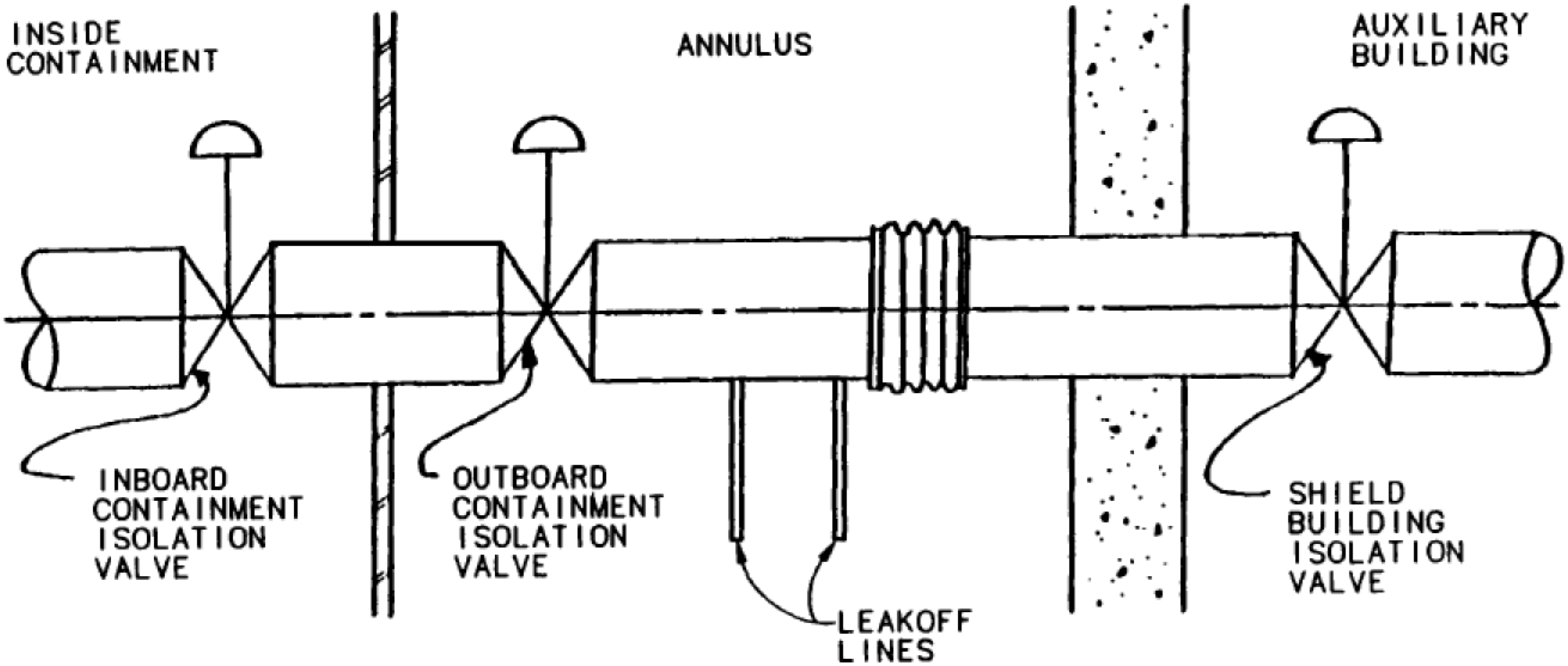
FAILURE MODES AND EFFECTS ANALYSIS FOR THE ABGTS

NOTES FOR ABGTS FAN FAILURE MODES AND THEIR IMPACT ON THE EGTS

POST-ACCIDENT TIME INTERVAL	EGTS	ABGTS	REMARKS
0-30 minutes	2 fans (auto start)	2 fans (auto start)	2 EGTS fans together produced - 2.3” w.g. DP in the annulus more than the DP of -0.69” w.g. effected by the ABGTS fans in ABSCE.
	1 fan operating 1 fan failed	2 fan (auto start)	2 ABGTS fans together pulled a vacuum of -0.69” w.g. @ Elevation 763.0 of the ABSCE, whereas, 1 EGTS fan drew down the annulus to -1.15” w.g., thus attesting that the annulus DPs are more negative than ABSCE.
	2 fans (auto start)	2 fans (auto start)	2 ABGTS fans together drew down the ABSCE to -1.09” w.g. @ Elevation 763.0, whereas, the 2 EGTS fans together produced -2.3” w.g. in the annulus. Therefore, the annulus DP remains more negative.
	1 fan operating	2 fans (auto start)	The failure of two safety-related components (i.e., fan and a damper affecting the same parameter, namely the annulus dP, is not postulated.) Therefore, this failure mode is invalid.
> 30 minutes	1 fan (operating)	1 fan (operating)	1 EGTS fan produces more negative dP in the annulus than 1 ABGTS fan in the ABSCE.
	1 fan (shutdown by operator)	1 fan (shutdown by operator)	
	1 fan (operating)	1 fan (operating)	2 ABGTS fans together produced a dP of -0.69” w.g. @ Elevation 763.0 of AB; whereas, 1 EGTS fan drew down the annulus to -1.15” w.g. @ Elevation 783.0. Therefore, the annulus is more negative than ABSCE.
	1 fan (operating) 1 fan (shutdown by operator)	2 fans (operator is yet to shutdown one of the fans) 1 vacuum relief damper failed	2 ABGTS fans together drew down the ABSCE to -1.09” w.g. @ Elevation 763.0, whereas, 1 EGTS fan was able to maintain -1.15” w.g. @ Elevation 783.0 of the annulus. However, in a later test, the same EGTS fan maintained -1.05” w.g.. However, the dP in the ABSCE remained less negative than that of the annulus.

SECURITY-RELATED INFORMATION, WITHHELD UNDER 10CFR2.390

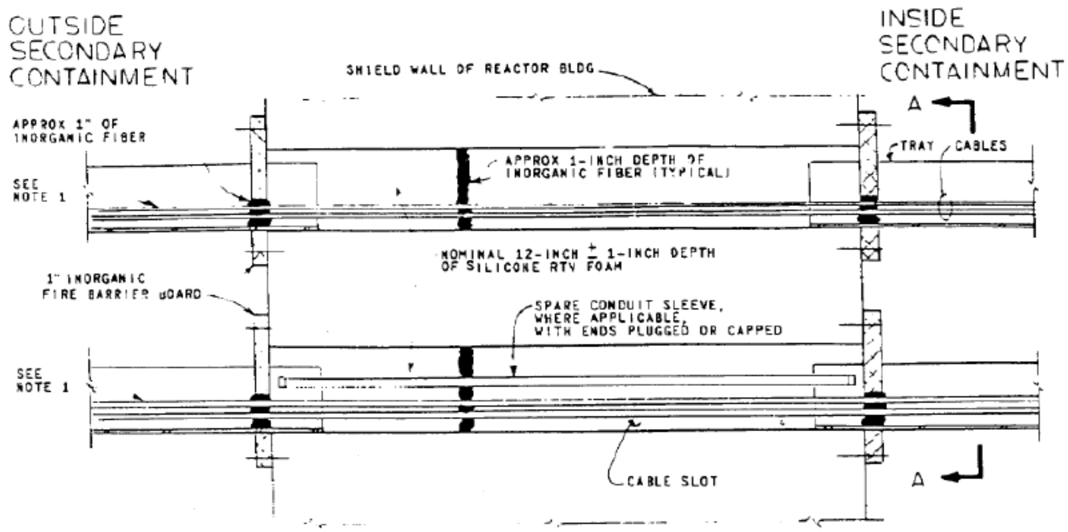
FIGURE 6.2.3-1



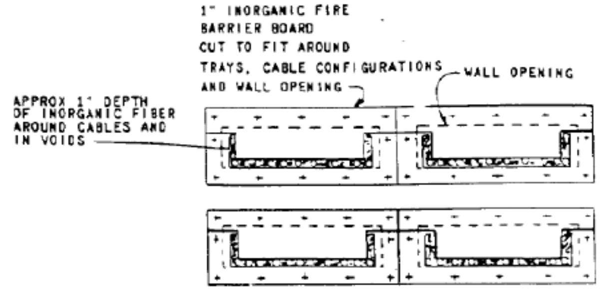
WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

Typical Purge Penetration
Arrangement

FIGURE 6.2.3-2



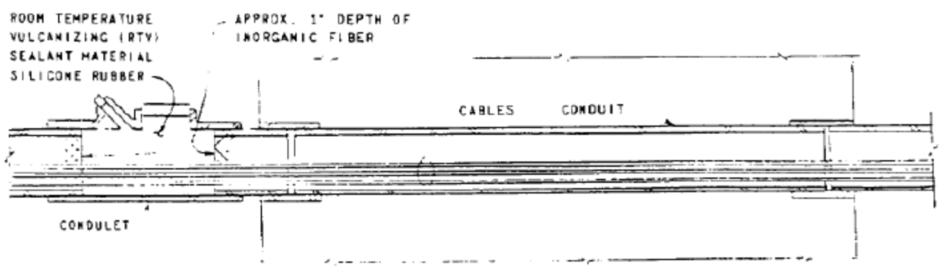
CABLE TRAY PENETRATION



SECTION A-A

NOTES:

1. AFTER FIRE BARRIER BOARD IS IN PLACE, COAT EXPOSED SURFACES OF CABLES FOR A MINIMUM DISTANCE OF 5 FEET FROM FIRE BARRIER BOARD OR TO NEAREST ELECTRICAL PANEL OR ENCLOSURE WITH A 3/16" ± 1/16" (WET DEPTH) OF ABLATIVE MATERIAL (BOTH SIDES OF WALL).



CONDUIT PENETRATION

WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

Typical Electrical
Penetrations

FIGURE 6.2.3-3

SECURITY-RELATED INFORMATION, WITHHELD UNDER 10 CFR 2.390

FIGURE 6.3.3-4

SECURITY-RELATED INFORMATION, WITHHELD UNDER 10CFR2.390

FIGURE 6.2.3-5

SECURITY-RELATED INFORMATION, WITHHELD UNDER 10CFR2.390

FIGURE 6.2.3-6

SECURITY-RELATED INFORMATION, WITHHELD UNDER 10CFR2.390

FIGURE 6.2.3-7

SECURITY-RELATED INFORMATION, WITHHELD UNDER 10CFR2.390

FIGURE 6.2.3-8

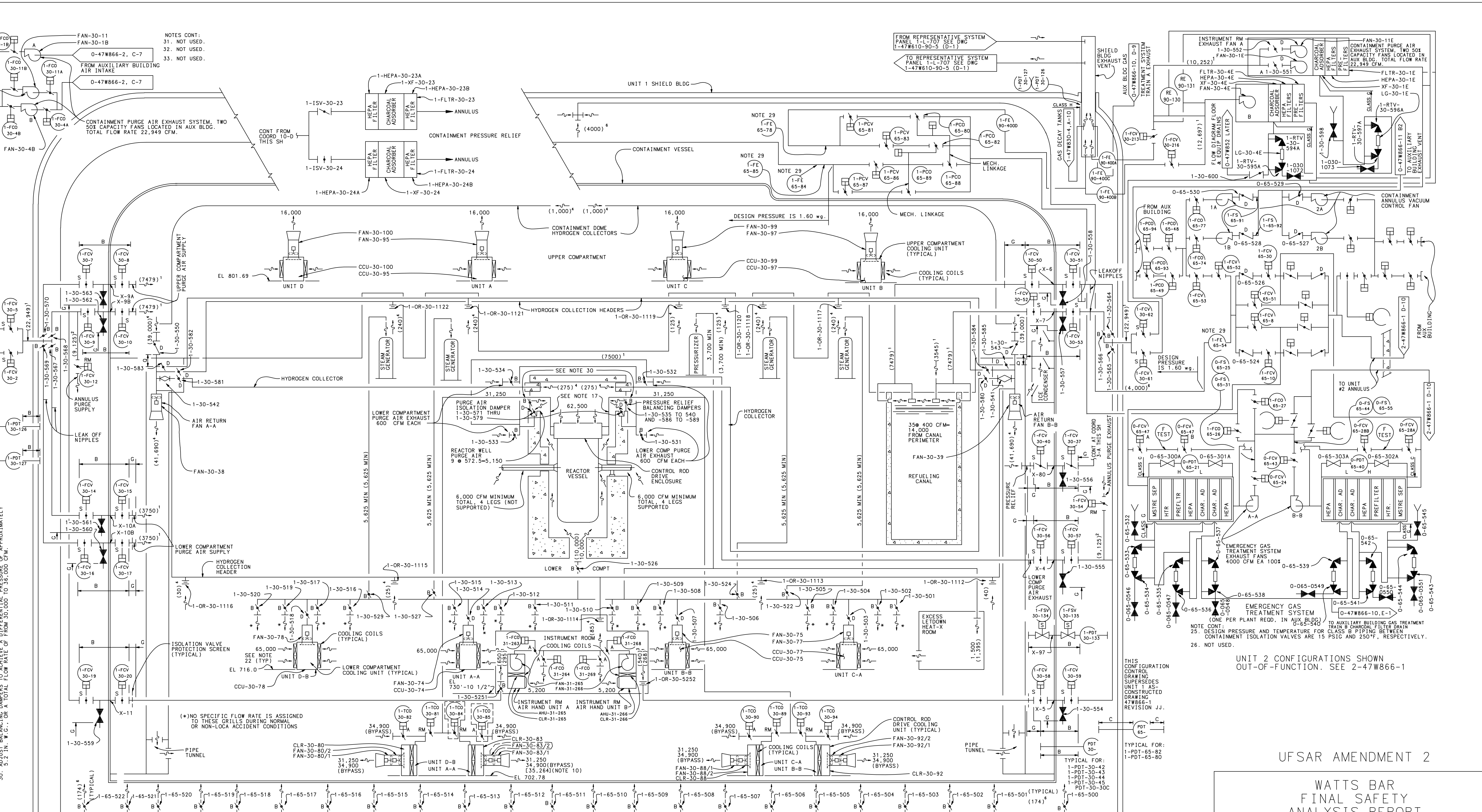
SECURITY-RELATED INFORMATION, WITHHELD UNDER 10CFR2.390

FIGURE 6.2.3-9

FIGURE 6.2.3-10

Deleted

CAD MAINTAINED DRAWING



NOTES CONT:
31. NOT USED.
32. NOT USED.
33. NOT USED.

NOTE CONT:
27. NOT USED.
28. NOT USED.
29. LOOP IS DISABLED. EQUIPMENT IS RETAINED FOR PRESSURE BOUNDARY.
30. ADJUST BALANCING DAMPERS TO ACHIEVE A DIFFERENTIAL PRESSURE OF APPROXIMATELY 5.2 IN. W.G., OR A TOTAL FLOW RATE OF FROM 30,000 TO 36,000 CFM.

NOTE CONT:
25. DESIGN PRESSURE AND TEMPERATURE FOR CLASS B PIPING BETWEEN CONTAINMENT ISOLATION VALVES ARE 15 PSIG AND 250°F, RESPECTIVELY.
26. NOT USED.

UNIT 2 CONFIGURATIONS SHOWN OUT-OF-FUNCTION. SEE 2-47W866-1

THIS CONFIGURATION CONTROL DRAWING SUPERSEDES UNIT 1 AS-CONSTRUCTED DRAWING 47W866-1 REVISION JJ.

NOTES:
1. NUMBERS SHOWN THUS —∿→ 4800 INDICATE AIR FLOW IN CFM.
2. FOR IDENTIFICATION OF SYMBOLS SEE 30B617-1 THRU -5.
3. FLOWS SHOWN WITHOUT PARENTHESIS ARE FOR NORMAL OPERATION.
4. ( )[1] INDICATES MINIMUM FLOW IN PURGE MODE.
5. ( )[2] INDICATES ALTERNATE MODE OF DIVERTING SOME PURGE FLOW THRU ANNULUS.
6. ( )[3] INDICATES MINIMUM SUPPLY AND MAXIMUM EXHAUST FLOW TO MAINTAIN A POSITIVE PRESSURE WHEN THE INSTRUMENT ROOM ONLY IS BEING PURGED.
7. ( )[4] INDICATES MINIMUM FLOW WHEN AIR RETURN FANS ARE OPERATED.
8.
9. ( )[6] INDICATES FLOW FROM ANNULUS, OR ENGINEERED SAFEGUARDS WHEN EMERGENCY GAS TREATMENT SYSTEM IS OPERATED.
10. FLOW RATE IN BRACKETS, [35,264], WAS MEASURED WITH ALL FOUR LCC'S OPERATING. IT IS OUTSIDE THE G-37 ACCEPTANCE CRITERIA BUT HAS BEEN EVALUATED AND APPROVED AS DOCUMENTED IN CALCULATIONS WBNOSG4130 AND EPMAMP122289.
11. FLOWS SHALL BE BALANCED TO THE RATES SHOWN ON THE DIAGRAM PER CONSTRUCTION SPECIFICATION NO. G-37. TESTING AND BALANCING OF HV AND AC SYSTEMS.
12. DESIGN PRESSURE IS 6 IN-WG IN ACCORDANCE WITH SMACNA HIGH VELOCITY DUCT CONSTRUCTION STANDARDS FOR MEDIUM PRESSURE (EXCEPT AS NOTED). DUCTWORK SHALL BE TESTED IN ACCORDANCES WITH TVA GENERAL CONSTRUCTION SPECIFICATION NO. G-37.
13. FOR VALVE MARKER TABULATIONS SEE MEL.
14. ALL DUCTWORK IS CLASS Q (ROUND) OR S (RECTANGULAR), EXCEPT WHERE NOTED AS B OR C. ALL PIPING ON THIS DRAWING IS CLASSIFIED AS TVA CLASS B, C, G, OR H AS NOTED.
15. ALL DUCTWORK CONNECTED TO THE SHIELD BUILDING EXHAUST VENT SHALL BE LEAK TESTED TO THE MOST STRINGENT REQUIREMENTS SPECIFIED IN SECTION 4.12.1 OF ANSI-N509.
16. PURGE AIR FILTER HOUSINGS SHALL BE LEAK-TESTED PER ANSI N509-1976. ALLOWABLE LEAKAGE RATES FOR HOUSINGS 1A AND 1B SHALL BE 7.3 CFM AND 6.7 CFM, RESPECTIVELY (REF. NCR 4234R).
17. DUCT WORK IN THIS AREA REMOVABLE TO ALLOW ACCESS TO REACTOR VESSEL HEAD.
18. DESIGN CRITERIA/SYSTEM DESCRIPTION REFERENCE DOCUMENTS (USE THE LATEST REVISION ON ALL WORK UNLESS OTHERWISE SPECIFIED. SEE THE LATEST REVISION OF THE 47B21 SERIES DRAWINGS PIPING SYSTEM CLASSIFICATION."):
N3-30RB-4002-------REACTOR BUILDING VENTILATION SYSTEM
19. ALL DUCTWORK SERVING THE AIR RETURN FANS AND LCC COOLERS IS SAFETY CLASS 2b AND SEISMIC CATEGORY I.
20. ALL PURGE DUCTWORK INSIDE THE CONTAINMENT UP TO THE INBOARD CONTAINMENT ISOLATION VALVES IS NON-SAFETY CLASS AND SEISMIC CATEGORY I(L). ALSO, THE SUPPLY DUCT BETWEEN THE DOWNSTREAM ABSCE ISOLATION VALVE AND SHIELD BUILDING ISOLATION VALVES IS NON-SAFETY CLASS AND SEISMIC CATEGORY I(L).
21. ALL PURGE EXHAUST DUCTWORK DOWNSTREAM OF THE OUTBOARD CONTAINMENT ISOLATION VALVES IS SAFETY CLASS 2b AND SEISMIC CATEGORY I.
22. MINIMUM TOTAL FLOWRATE DURING NORMAL OPERATION WITH 3 LOWER COMPARTMENT COOLER UNITS IS 146,000 CFM. MINIMUM TOTAL FLOWRATE DURING NON-LOCA ACCIDENT CONDITIONS IS 112,000 CFM WITH 2 COOLERS OPERATING. DUCTWORK AND GRILLE FLOWRATES FOR NON-LOCA ACCIDENT CONDITIONS ARE SHOWN IN PARENTHESIS ().
23. LEAK TESTING/LEAK CHECKING OF DUCTWORK INSIDE CONTAINMENT AND ANNULUS SHALL BE IN ACCORDANCE WITH G-37, EXCEPT AS NOTED ON DRAWINGS.
24. BASED ON G-37 PARAGRAPHS 4.3.3.2.1, 4.3.3.2.2, 4.3.3.2.3, 4.3.3.2.5, 4.3.3.2.7.1, 4.3.3.2.8, AND 4.3.3.2.9, WHICH PROVIDE EXEMPTIONS FROM LEAK-TESTING OF DUCTWORK, AND EXCEPTION G-37-WBN-1, DUCTWORK FOR LOWER COMPARTMENT COOLING, PURGE AIR, AND CRDM IS EXEMPTED FROM LEAK-TESTING/LEAK-CHECKING REQUIREMENTS, IF TESTED AIRFLOW RATES SATISFY DESIGN VALUES.

DAMPER LEGEND:
A -AUTOMATIC OPERATING FOR NORMAL SERVICE
RM-REMOTE MANUAL OPERATION
S -AUTOMATIC OPERATING FOR ISOLATION
M MANUAL (FOR OPEN-CLOSE OPERATION)
B BALANCING
D BACKDRAFT
F FIRE
O FAILS OPEN
X FAILS CLOSED
◇ FAILS AS IS
L.O. LOCKED OPEN
V VACUUM RELIEF

REFERENCE DRAWINGS:
1-47W610-30-1 THRU 7
1-47W610-65-1
47W611-30-1 THRU 4
47W611-65-1, 2 & 3
47W844-4
47W865-5
0-47W866-2, 10 & 11
47W852 SERIES
47W830-4
47B920-30X1 THRU 30X4
2-47W866-1
47W915 SERIES
47B920 SERIES
47B601-30 SERIES
47B601-65 SERIES
47B915-65 X 1 THRU -65 X 4
47W866-101 STRESS ANALYSIS PROBLEM BOUNDARY

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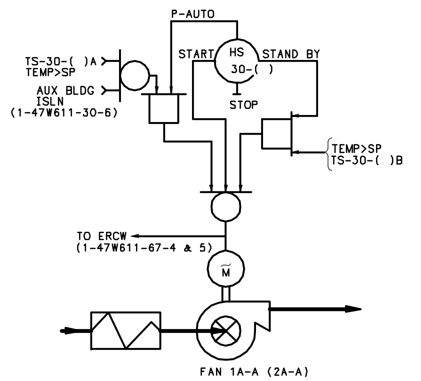
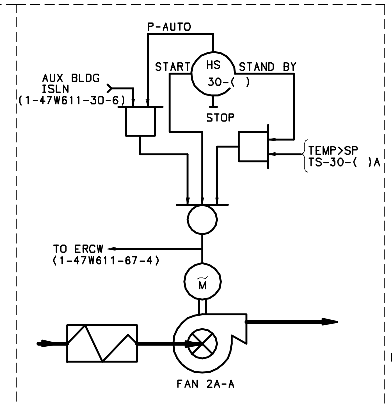
REACTOR BUILDING
UNIT 1
FLOW DIAGRAM
HEATING AND VENTILATION
AIR FLOW
TVA DWG NO. 1-47W866-1 R69
FIGURE 6.2.3-11

CAD MAINTAINED DRAWING



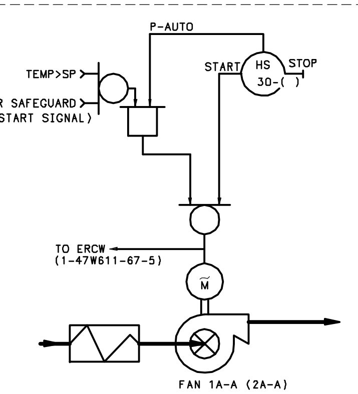
CONTROLS FOR FAN 1B-B (2B-B
TYPICAL OF FAN 1A-A (2A-A)

FANS	UNITS	INSTRUMENT NO.
PENETRATION ROOM COOLER, EL 692	1, 2	HS-30-186, -187 TS-30-186A, -186B, -187A, -187B
PENETRATION ROOM COOLER, EL 713	1, 2	HS-30-196, -197 TS-30-196A, -196B, -197A, -197B
PENETRATION ROOM COOLER, EL 737	1, 2	HS-30-194, -195 TS-30-194A, -194B, -195A, -195B
CCS PUMPS & AUX FW PUMPS SPACE COOLER	1	HS-30-190, -191 TS-30-190A, -190B, -191A, -191B
BORIC ACID TRANSFER PUMPS & AUX FW PUMPS SPACE COOLER	2	HS-30-184, -185 TS-30-184A, -184B, -185A, -185B
SPENT FUEL PIT PUMP & THERMAL BARRIER BSTR PUMP SPACE COOLER	0 (COMMON)	HS-30-192, -193 TS-30-192A, -192B, -193A, -193B
PIPE CHASE COOLER	1, 2	HS-30-201, -202 TS-30-201A, -201B, -202A, -202B



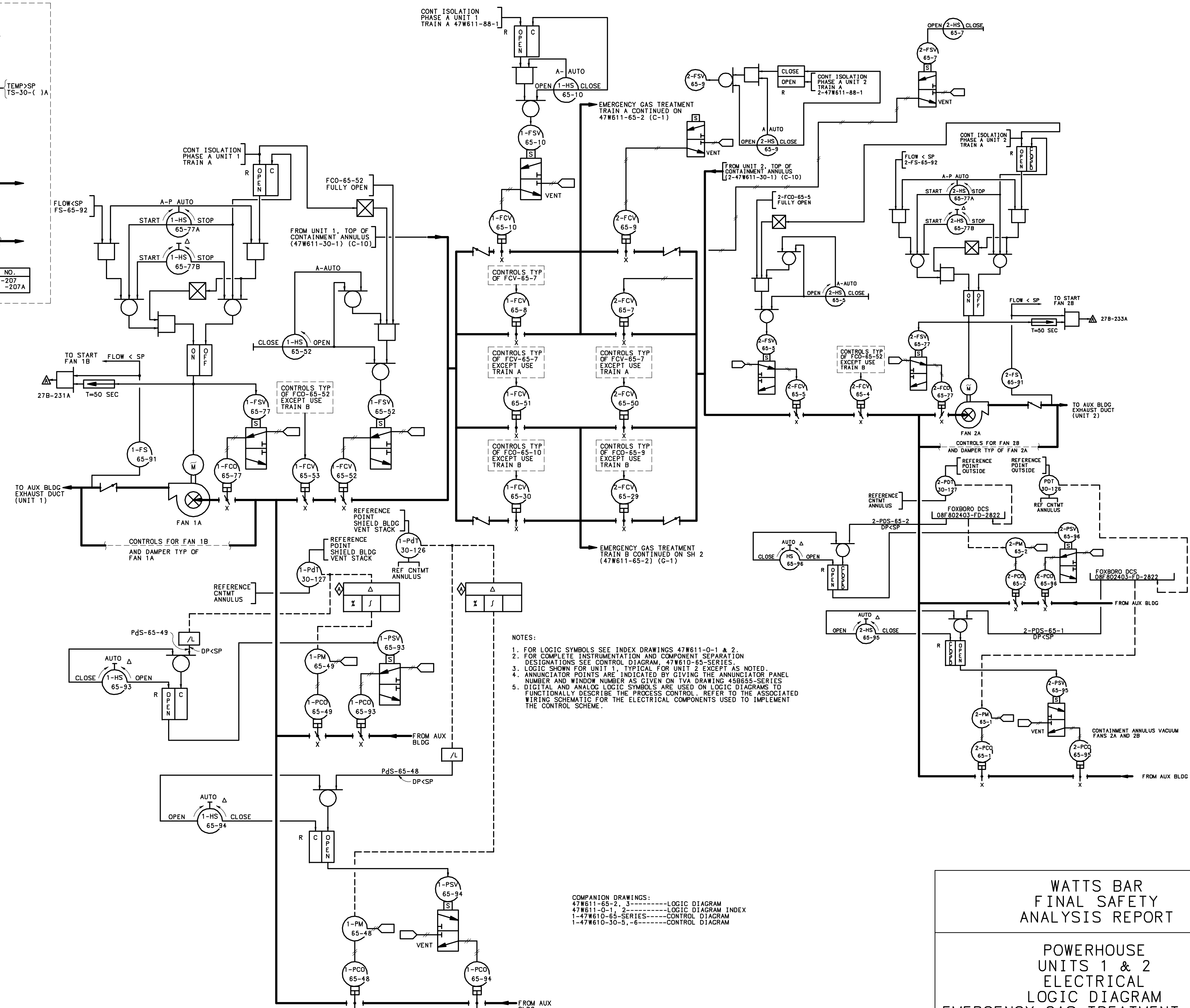
CONTROLS FOR FAN 2B-B
TYPICAL OF FAN 2A-A

FANS	UNIT	INSTRUMENT NO.
EGTS ROOM COOLER	2	HS-30-200, -207 TS-30-200A, -207A



CONTROLS FOR FAN 1B-B (2B-B)
TYPICAL OF FAN 1A-A (2A-A)

FANS	UNITS	INSTRUMENT NO.
SIS PUMP ROOM COOLERS, FAN 1A-A & 1B-B	1, 2	HS-30-179, -180 TS-30-179, -180
RHR PUMP ROOM COOLERS, FAN 1A-A & 1B-B	1, 2	HS-30-175, -176 TS-30-175, -176
CNTMT SPRAY PUMP ROOM COOLERS, FAN 1A-A & 1B-B	1, 2	HS-30-177, -178 TS-30-177, -178
CHARGING PUMP ROOM COOLERS, FAN 1A-A, 1B-B, 1C (CONTROLS FOR FAN 1C ARE TYP FOR FAN 1A-A)	1, 2	HS-30-181, -182, -183 TS-30-181, -182, -183



NOTES:

1. FOR LOGIC SYMBOLS SEE INDEX DRAWINGS 47W611-0-1 & 2.
2. FOR COMPLETE INSTRUMENTATION AND COMPONENT SEPARATION DESIGNATIONS SEE CONTROL DIAGRAM, 47W610-65-SERIES.
3. LOGIC SHOWN FOR UNIT 1, TYPICAL FOR UNIT 2 EXCEPT AS NOTED.
4. ANNUNCIATOR POINTS ARE INDICATED BY GIVING THE ANNUNCIATOR PANEL NUMBER AND WINDOW NUMBER AS GIVEN ON TVA DRAWING 45B655-SERIES
5. DIGITAL AND ANALOG LOGIC SYMBOLS ARE USED ON LOGIC DIAGRAMS TO FUNCTIONALLY DESCRIBE THE PROCESS CONTROL. REFER TO THE ASSOCIATED WIRING SCHEMATIC FOR THE ELECTRICAL COMPONENTS USED TO IMPLEMENT THE CONTROL SCHEME.

COMPANION DRAWINGS:
47W611-65-2, 3----------LOGIC DIAGRAM
47W611-0-1, 2-----------LOGIC DIAGRAM INDEX
1-47W610-65-SERIES-----CONTROL DIAGRAM
1-47W610-30-5,-6--------CONTROL DIAGRAM

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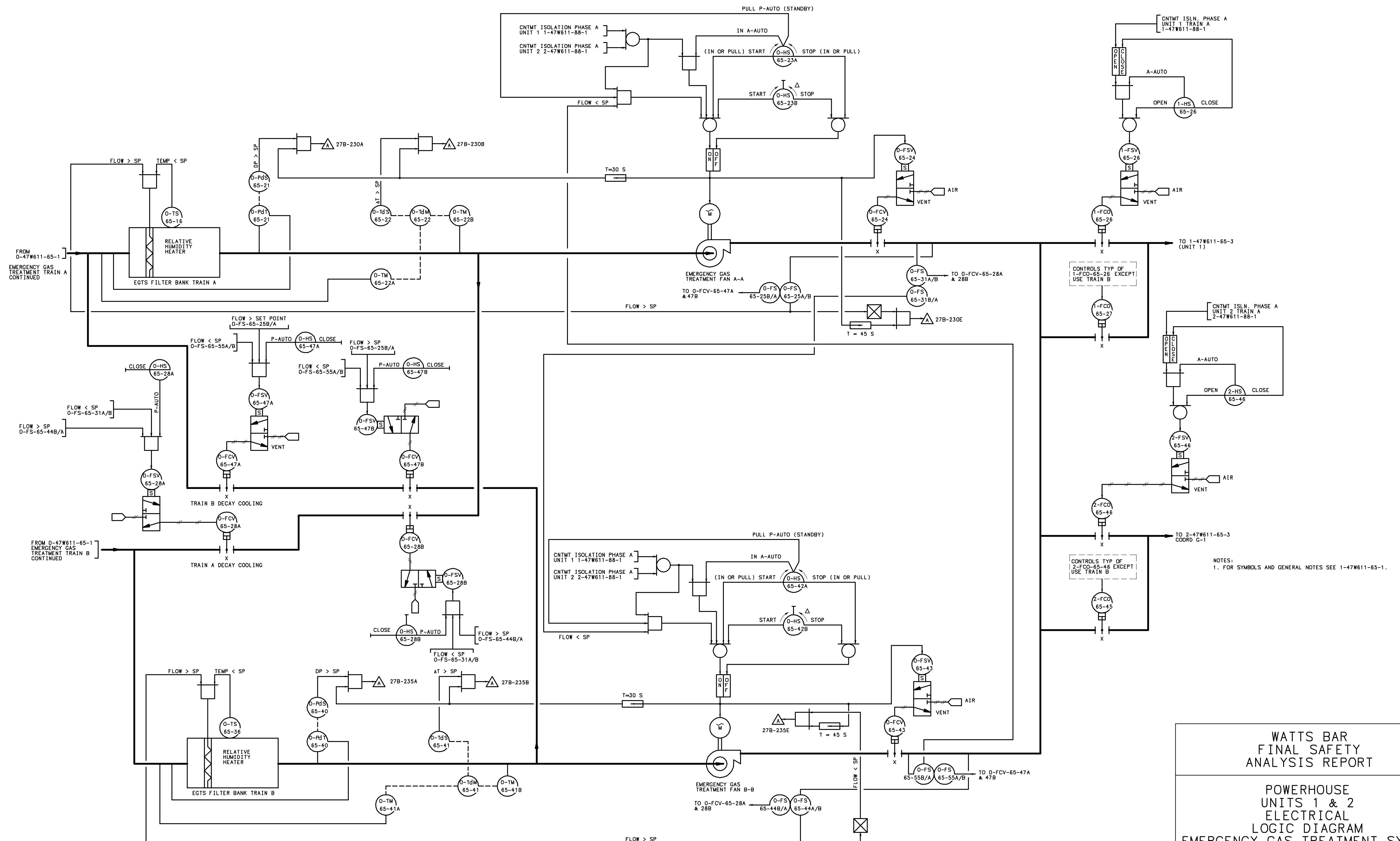
POWERHOUSE
UNITS 1 & 2
ELECTRICAL
LOGIC DIAGRAM
EMERGENCY GAS TREATMENT SYSTEM
TVA DWG NO. 0-47W611-65-1 R0
FIGURE 6.2.3-12

CAD MAINTAINED DRAWING



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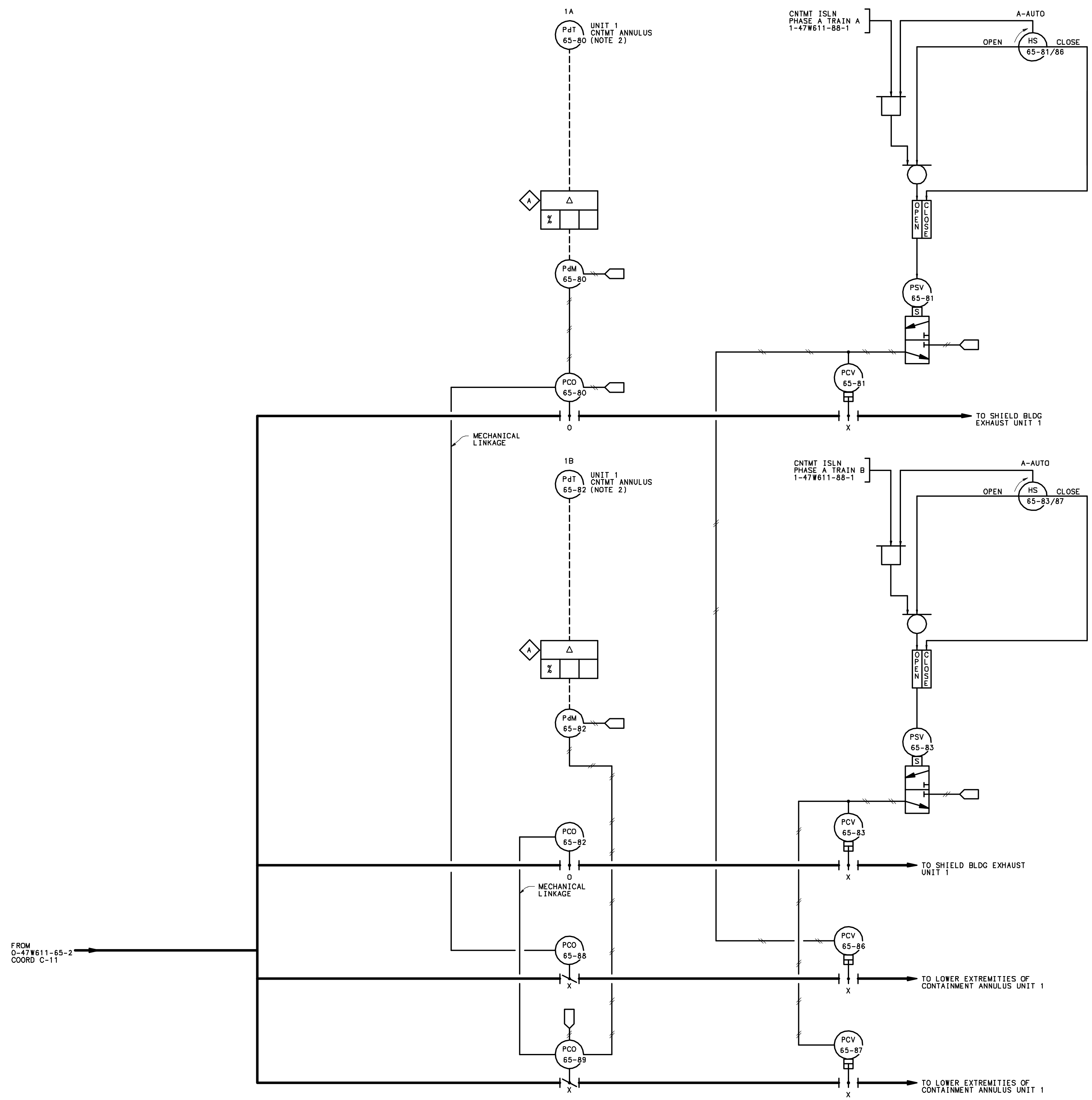
POWERHOUSE
UNITS 1 & 2
ELECTRICAL
LOGIC DIAGRAM
EMERGENCY GAS TREATMENT SYSTEM
TVA DWG NO. 0-47W611-65-2 R0
FIGURE 6.2.3-13

CAD MAINTAINED DRAWING



NOTES:
1. FOR SYMBOLS AND GENERAL NOTES, SEE 1-47W611-65-1.
2. PdT'S MEASURE PRESSURE DIFFERENTIAL FROM CONTAINMENT ANNULUS TO OUTSIDE ATMOSPHERE.

UFSAR AMENDMENT 1

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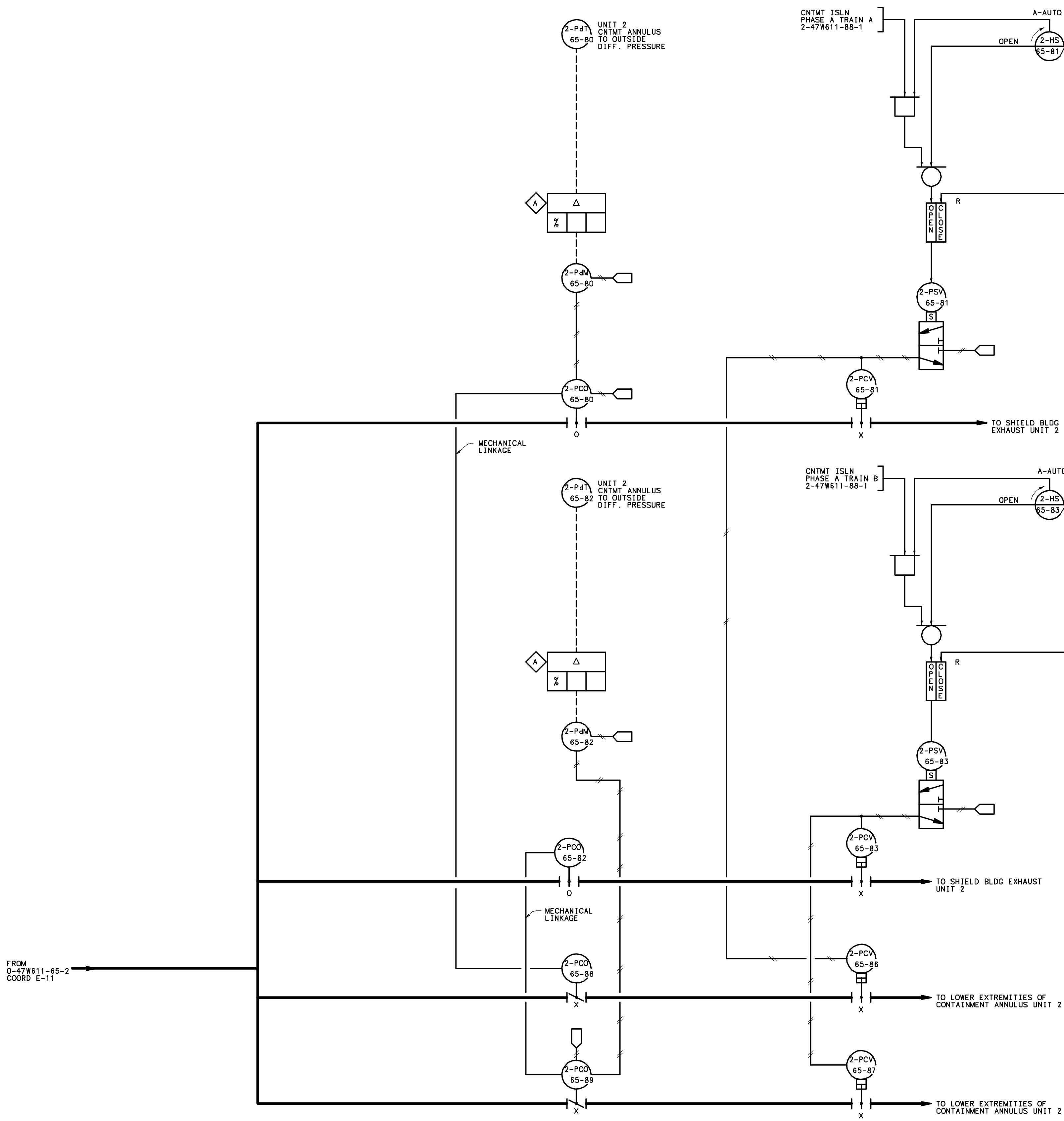
POWERHOUSE
UNIT 1
ELECTRICAL
LOGIC DIAGRAM
EMERGENCY GAS TREATMENT
TVA DWG NO. 1-47W611-65-3 R11
FIGURE 6.2.3-14

CAD MAINTAINED DRAWING



NOTES:
1. FOR SYMBOLS AND GENERAL NOTES, SEE 2-47W611-65-1.
2. NOT USED.
3. PDT'S MEASURE PRESSURE DIFFERENTIAL FROM CONTAINMENT ANNULUS TO OUTSIDE ATMOSPHERE.

COMPANION DWG.'S 2-47W611-65-1 & 2

UFSAR AMENDMENT 1

WATTS BAR
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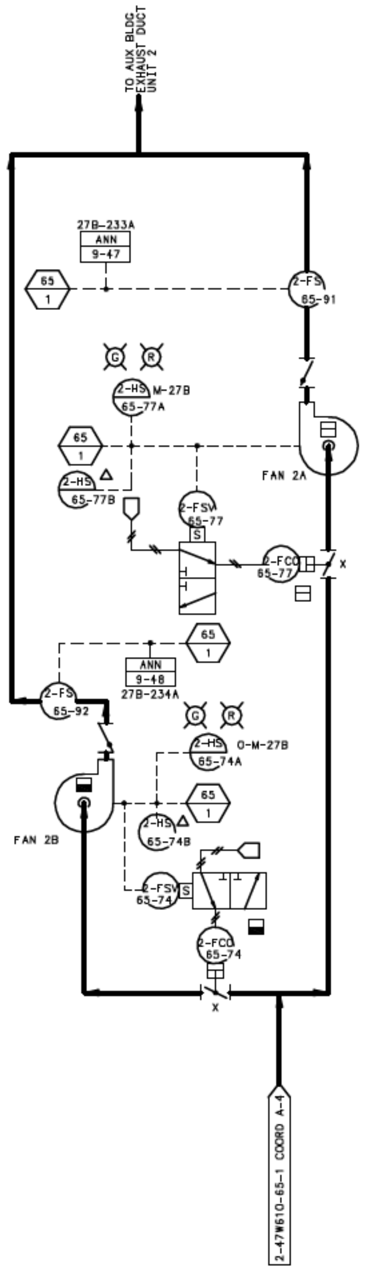
POWERHOUSE
UNIT 2
ELECTRICAL
LOGIC DIAGRAM
EMERGENCY GAS TREATMENT SYSTEM
TVA DWG NO. 2-47W611-65-3 R4
FIGURE 6.2.3-14(U2)

CAD MAINTAINED DRAWING

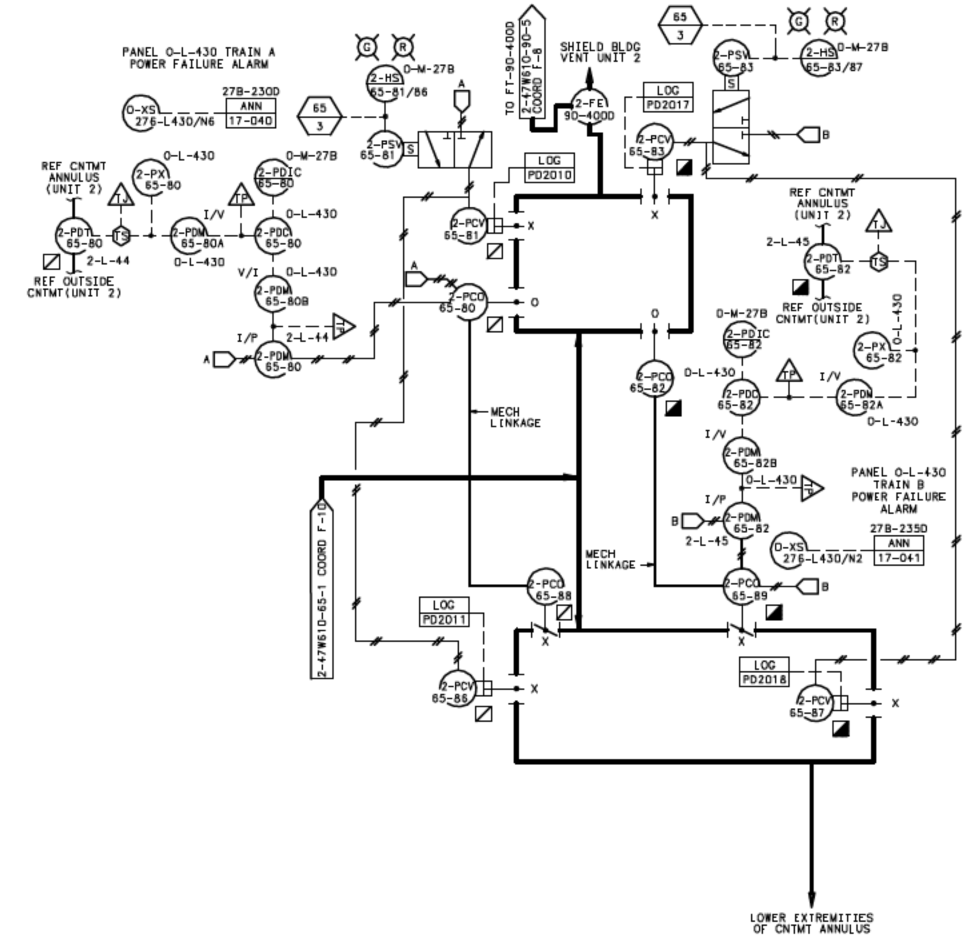


CONTAINMENT ANNULUS
VACUUM FANS. UNIT 2

TABLE A1A
DIGITAL COMPUTER POINTS

INSTRUMENT ID	POINT ID	DESCRIPTION
0-FAN-065-0023	XD2014	EGTS FAN 1A-A
0-FAN-065-0023	XD2013	EGTS FAN 1A-A, FCV-65-24
0-FAN-065-0042	XD2065	EGTS FAN B-B
0-FAN-065-0042	XD2064	EGTS FAN B-B, FCV-65-43
1-FCO-065-0026	FD2021	U-1 SHLD BLDG EXH DMPR
1-FCO-065-0027	FD2369	U-1 SHLD BLDG EXH DMPR
1-FCV-065-0010	FD2015	EGTS TR-A UNIT 1 SUCT DMPR
1-FCV-065-0030	FD2363	EGTS TRAIN B UNIT 1 SUCT DMPR
1-HS-065-0008	HD2054	EGTS TR-A SUCT VLV SW
1-HS-065-0010	FD2016	EGTS TR-A UNIT 1 SUCT DMPR HS
1-HS-65-0010	HD2019	EGTS TR-A HS-10A, AND -26A
0-HS-065-0023A	HD2021	EGTS FAN 1A-A SW
1-HS-065-0026	FD2022	U-1 SHLD BLDG EXH DMPR SW
1-HS-065-0027	FD2370	U-1 SHLD BLDG EXH DMPR SW
1-HS-065-0027	HD2053	EGTS TR-B HS-27A, AND -30A
1-HS-065-0030	FD2364	EGTS TRAIN B UNIT 1 SUCT DMPR SW
0-HS-065-0042A	HD2055	EGTS FAN B-B SW
1-HS-065-0051	HD2020	EGTS TR-B SUCT VLV SW
1-HS-065-0081	HD2071	SHLD BLDG VENT & ANNS HS-81A, -83B
1-HS-065-0081	PD2009	SHLD & ANNS ISO DMPR SW
1-HS-065-0083	PD2016	SHLD & ANNS ISO DMPR SW
1-PCO-065-0080	PD2006	SHLD BLDG VENT & ANNS DMPR -80,-88
1-PCO-065-0082	PD2013	SHLD BLDG VENT & ANNS DMPR -82, -89
1-PCV-065-0081	PD2007	SHLD & ANNS ISO DMPR -81, -86 AC PWR
1-PCV-065-0081	PD2008	SHLD & ANNS ISO DMPR -81, -86 DC PWR
1-PCV-065-0083	PD2014	SHLD & ANNS ISO DMPR -83, -87 AC PWR
1-PCV-065-0083	PD2015	SHLD & ANNS ISO DMPR -83, -87 DC PWR
2-FCO-65-46	FD2021	U-2 SHLD BLDG EXH DMPR
2-FCO-65-45	FD2369	U-2 SHLD BLDG EXH DMPR
2-FCV-65-9	FD2015	EGTS TR A UNIT 2 SUCT DMPR
2-FCV-65-29	FD2363	EGTS TR B UNIT 2 SUCT DMPR
2-HS-65-7	HD2054	EGTS TR-A SUCT VLV SW
2-HS-65-9	FD2016	EGTS TRAIN B UNIT 2 SUCT DMPR SW
2-HS-65-9	HD2019	EGTS TR-A HS-09A, AND -46A
2-HS-65-46	FD2022	U-2 SHLD BLDG EXH DMPR SW
2-HS-65-45	FD2370	U-2 SHLD BLDG EXH DMPR SW
2-HS-65-46	HD2053	EGTS TR-B HS-45A, AND -29A
2-HS-65-29	FD2364	EGTS TRAIN B UNIT 2 SUCT DMPR SW
2-HS-65-81	HD2071	SHLD BLDG VENT & ANNS HS-81A, -83B
2-HS-65-81	PD2009	SHLD & ANNS ISO DMPR SW
2-HS-65-83	PD2016	SHLD & ANNS ISO DMPR SW
2-PCV-65-81	PD2008	SHLD & ANNS ISO DMPR -81, -86 DC PWR
2-PCV-65-83	PD2015	SHLD & ANNS ISO DMPR -83, -87 DC PWR
2-HS-65-50	HD2020	EGTS TR-B SUCT VLV SW

SEE SHEET 1 FOR ASSOCIATED INSTRUMENTATION



NOTES:

1. FOR GENERAL NOTES SEE 0-47W610-65-1.

WATTS BAR
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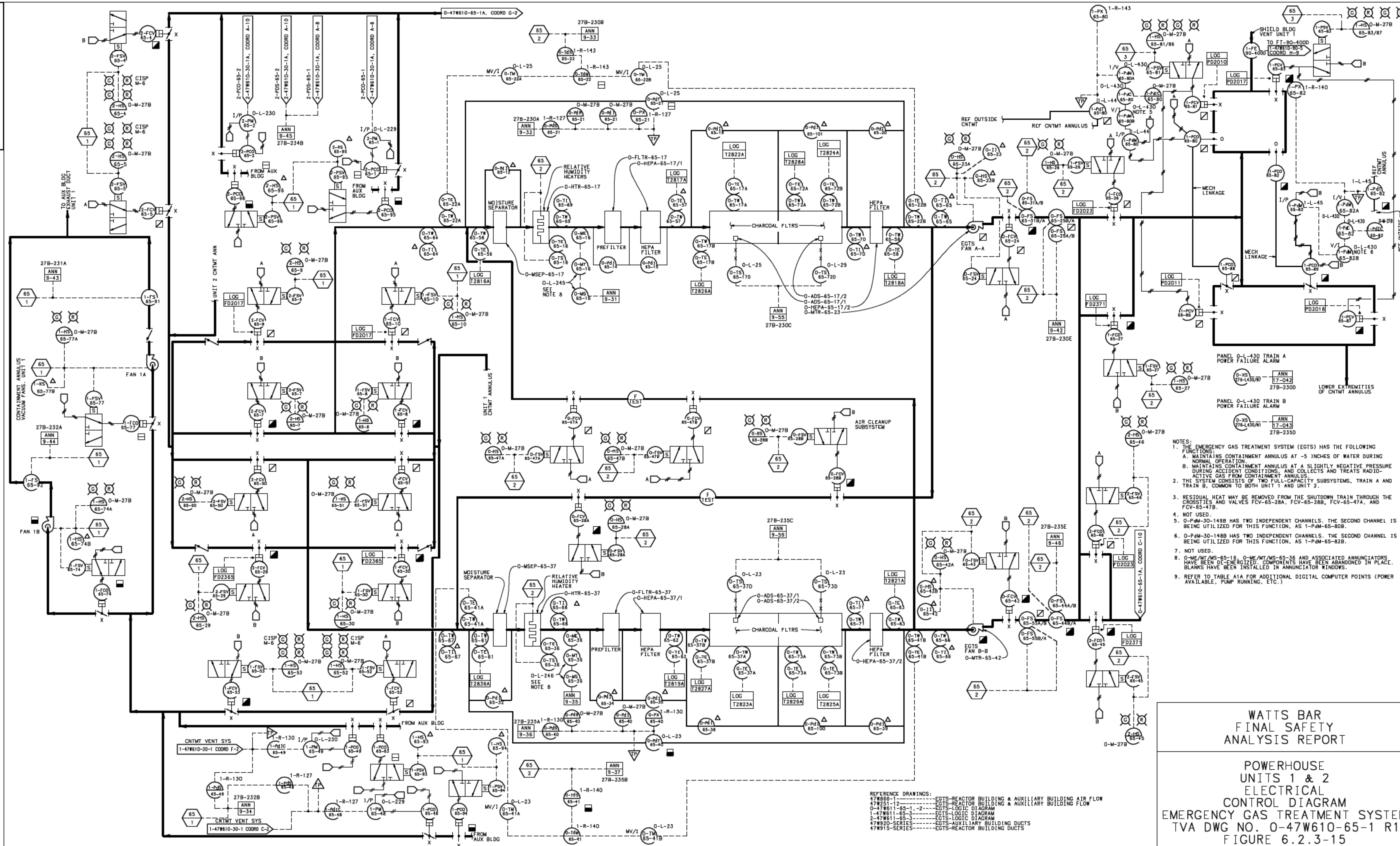
POWERHOUSE
UNITS 1 & 2
ELECTRICAL
CONTROL DIAGRAM
EMERGENCY GAS TREATMENT SYSTEM
TVA DWG NO. 0-47W610-65-1A R0
FIGURE 6.2.3-15 SH A

CAD MAINTAINED DRAWING



NOTES:

1. THE EMERGENCY GAS TREATMENT SYSTEM (EGTS) HAS THE FOLLOWING FUNCTIONS:
   A. MAINTAINS CONTAINMENT ANNULUS AT -5 INCHES OF WATER DURING NORMAL OPERATION.
   B. MAINTAINS CONTAINMENT ANNULUS AT A SLIGHTLY NEGATIVE PRESSURE DURING ACCIDENT CONDITIONS, AND COLLECTS AND TREATS RADIO-ACTIVE GAS FROM CONTAINMENT ANNULUS.
2. THE SYSTEM CONSISTS OF TWO FULL-CAPACITY SUBSYSTEMS, TRAIN A AND TRAIN B, COMMON TO BOTH UNIT 1 AND UNIT 2.
3. RESIDUAL HEAT MAY BE REMOVED FROM THE SHUTDOWN TRAIN THROUGH THE CROSSTIES AND VALVES FCV-65-28A, FCV-65-28B, FCV-65-47A, AND FCV-65-47B.
4. NOT USED.
5. 0-PdM-30-149B HAS TWO INDEPENDENT CHANNELS. THE SECOND CHANNEL IS BEING UTILIZED FOR THIS FUNCTION, AS 1-PdM-65-80B.
6. 0-PdM-30-148B HAS TWO INDEPENDENT CHANNELS. THE SECOND CHANNEL IS BEING UTILIZED FOR THIS FUNCTION, AS 1-PdM-65-82B.
7. NOT USED.
8. 0-ME/MT/MS-65-16, 0-ME/MT/MS-65-36 AND ASSOCIATED ANNUNCIATORS HAVE BEEN DE-ENERGIZED. COMPONENTS HAVE BEEN ABANDONED IN PLACE. BLANKS HAVE BEEN INSTALLED IN ANNUNCIATOR WINDOWS.
9. REFER TO TABLE A1A FOR ADDITIONAL DIGITAL COMPUTER POINTS (POWER AVAILABLE, PUMP RUNNING, ETC.)

REFERENCE DRAWINGS:
47W866-1------------EGTS-REACTOR BUILDING & AUXILIARY BUILDING AIR FLOW
47W251-12-----------EGTS-REACTOR BUILDING & AUXILIARY BUILDING FLOW
0-47W611-65-1,-2----EGTS-LOGIC DIAGRAM
1-47W611-65-3-------EGTS-LOGIC DIAGRAM
2-47W611-65-3-------EGTS-LOGIC DIAGRAM
47W920-SERIES-------EGTS-AUXILIARY BUILDING DUCTS
47W915-SERIES-------EGTS-REACTOR BUILDING DUCTS

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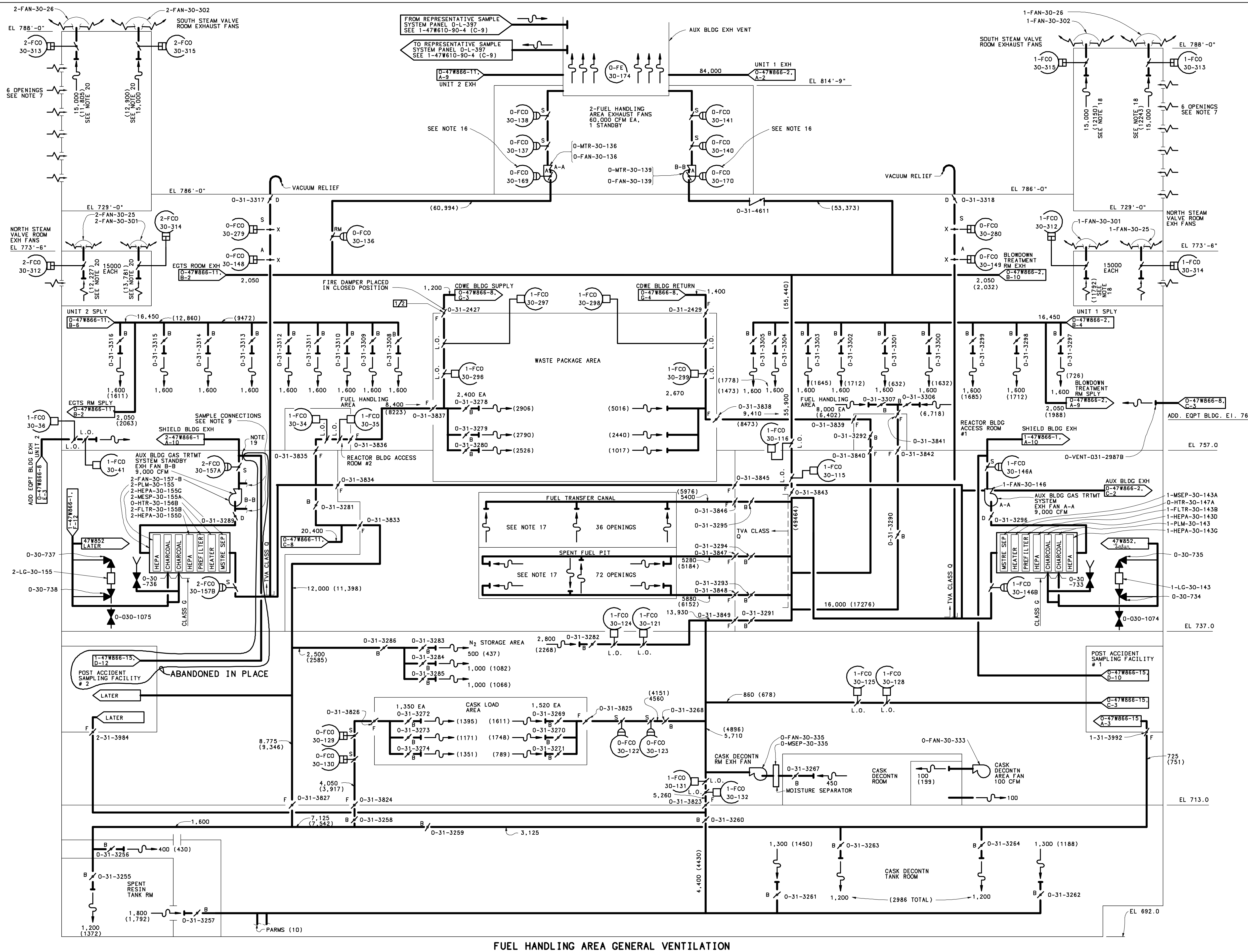
POWERHOUSE
UNITS 1 & 2
ELECTRICAL
CONTROL DIAGRAM
EMERGENCY GAS TREATMENT SYSTEM
TVA DWG NO. 0-47W610-65-1 R1
FIGURE 6.2.3-15

CAD MAINTAINED DRAWING



NOTES:

1. FOR NOTES, COMPANION DRAWINGS, AND REFERENCE DRAWINGS SEE 47W866-2.
2. DESIGN PRESSURE ABOVE ELEVATION 690.0 IS 6 IN-WG IN ACCORDANCE WITH SMACNA HIGH VELOCITY DUCT CONSTRUCTION STANDARDS FOR MEDIUM PRESSURE. DESIGN PRESSURE FOR DUCT WORK BELOW ELEVATION 690.0 IS 2 IN-WG IN ACCORDANCE WITH SMACNA LOW VELOCITY DUCT CONSTRUCTION STANDARDS. DUCT WORK SHALL BE TESTED IN ACCORDANCE WITH TVA GENERAL CONSTRUCTION SPECIFICATION NO. G-37.
3. PARMS (X) INDICATES A TEMPORARY SAMPLE POINT FOR THE PORTABLE AIRBORNE RADIOACTIVITY MONITORING SYSTEMS, PARMS. THERE ARE 18 POINTS. POINT 16 HAS THE RETURN LINE ABANDONED IN PLACE. THE SAMPLE LINE IS NOT INSTALLED. POINT 17 HAS THE DUCT CONNECTION FOR THE RETURN LINE CAPPED. THE SAMPLE LINE IS NOT INSTALLED. THE REMAINING 16 POINTS HAVE THE SAMPLE LINE AND THE RETURN LINE ABANDONED IN PLACE. SEE SHEETS 2 AND 11 FOR THE REMAINDER OF THE POINTS AND 47W600-108 FOR CONNECTION DETAILS.
4. FOR VALVE MARKER TABULATIONS SEE MEL.
5. ALL DUCTWORK IN THE AUXILIARY BUILDING IS TVA CLASS Q (ROUND) OR CLASS S (RECTANGULAR).
6. FOR SEISMIC REQUIREMENT FOR PIPING AND DUCT WORK SEE DETAILED CONSTRUCTION DRAWINGS (47W920 SERIES).
7. EACH OF 6 OPENINGS EQUIPPED WITH BLANK-OFF PLATE DAMPERS. EACH DAMPER SHALL BE CLOSED ANYTIME THE REACTOR IS IN A SHUTDOWN MODE AND THERE IS A POSSIBILITY OF THE OUTSIDE AIR TEMPERATURE DROPPING BELOW 35°F. ALL DAMPERS MUST BE OPENED JUST PRIOR TO EACH UNIT STARTUP.
8. FLOWRATES SHOWN WITHOUT PARENTHESES ARE THE DESIGN FLOW RATES. CONSTRUCTION SPECIFICATION G-37 ALLOWS A TOLERANCE OF ± 10% TO BE APPLIED TO THE DESIGN FLOW RATES FOR SYSTEM BALANCING. FLOW RATES SHOWN IN PARENTHESES REFLECT PREOP AIR BALANCE TEST RESULTS. THESE FLOWRATES ARE DEEMED ACCEPTABLE BASED ON MEETING THE CRITERION THAT THE AIRFLOW DIRECTION BE FROM THE CLEANER AREAS TO AREAS OF PROGRESSIVELY GREATER CONTAMINATION POTENTIAL, AND THE FACT THAT COOLING OF SAFETY-RELATED EQUIPMENT AREAS IS ACCOMPLISHED BY THE THERMOSTATICALLY CONTROLLED ESF EQUIPMENT ROOM/AREA COOLERS (SEE EX-G-37-WBN-1 REV. 1).
9. GRAB SAMPLE CONNECTIONS FOR TEMPORARY ANALYSIS OF EXHAUST EMISSIONS TO UNIT 2 STACK. ROUTE TO SYSTEM 90 PROBE STATION NUMBER 9 (47W600-108).
10. FOR DESIGN REQUIREMENTS REFER TO DESIGN CRITERIA: WB-DC-40-36.1 "THE CLASSIFICATION OF HEATING, VENTILATING AND AIR CONDITIONING SYSTEMS".
11. FOR FUNCTION DESCRIPTION, REFER TO WATTS BAR NUCLEAR PLANT SYSTEM DESCRIPTION, N3-30AB-4001 "SYSTEM DESCRIPTION FOR AUXILIARY BUILDING-HEATING, VENTILATION AND AIR CONDITIONING SYSTEM".
12. ALL PIPING ON THIS DRAWING IS CLASS G, CAT. I(L).
13. SEISMIC CATEGORY I (L) DUCTWORK, ATTACHED TO THE CATEGORY I SUCTION-SIDE ABGTS DUCTWORK LOCATED IN THE SAME ROOM AS THE ABGTS FILTER TRAIN AND BOUNDED BY THE ISOLATION DAMPERS 1-FCO-30-116, 122, 124, 128, AND 132, SHALL BE ANALYZED TO SHOW THAT IT WILL NOT ADVERSELY AFFECT THE SAFETY FUNCTION OF THE ABGTS SYSTEM (REF. WB-DC-40-36.1, TABLE 3.3-1, NOTE 7).
14. SECONDARY CONTAINMENT ISOLATION DAMPERS 1-FCO-30-34, -35, -36, -41, -115, -116, -121, -124, -125, -128, -131, -132, -296, -297, -298, -299 ARE LOCKED IN THE FULLY OPEN POSITION.
15. 1 2 & 1/2 DENOTE UNIT 1 & UNIT 2 INTERFACE POINTS. THE STRUCTURAL BOUNDARY IS THE FIRST ANCHORED EQUIPMENT OR PIPE ANCHOR ON THE UNIT 2 SIDE OF THE INTERFACE POINT. BECAUSE OF TEES, SOME INTERFACE POINTS WILL HAVE MORE THAN ONE STRUCTURAL TERMINATION.
16. THIS SHAFT SHOULD BE EXTENDED AS NECESSARY TO PREVENT THE MODULATING DAMPER FROM CLOSING COMPLETELY.
17. WHEN PERFORMING MAINTENANCE ACTIVITIES IN EITHER FTC OR SFP SEE SD N3-30AB-4001, SECTION 4.26 FOR SPECIAL OPERATION FOR ALTERNATE CONFIGURATION REQUIREMENTS.
18. FLOWRATES IN PARENTHESES ARE MEASURED DATA OBTAINED DURING RFO5 (WO # 03-010366-001, -002 & -003). FOR ACCEPTANCE REFER TO EDC 51619A. THE TOTAL ACCEPTABLE AIR FLOW RATE (FROM BOTH FANS) IS 87% OF THE TOTAL DESIGN VALUE FOR THE NORTH MSVV ROOM AND 81.3% FOR THE SOUTH MSVV ROOM.
19. 2-PL-31-426 BLANK-OFF PLATE INSTALLED TO ISOLATE ABANDONED PASF UNIT 2.
20. FLOW RATES SHOWN IN PARENTHESES WERE RECORDED IN CONSTRUCTION TEST G-37 (TEST PACKAGE 0-03-AB-BT-TVA9C AND 2-30-01207-M05-000) AND APPROVED BY ENGINEERING. FLOW RATES NOT IN PARENTHESES ARE DESIGN FLOW RATES.

UFSAR AMENDMENT 1

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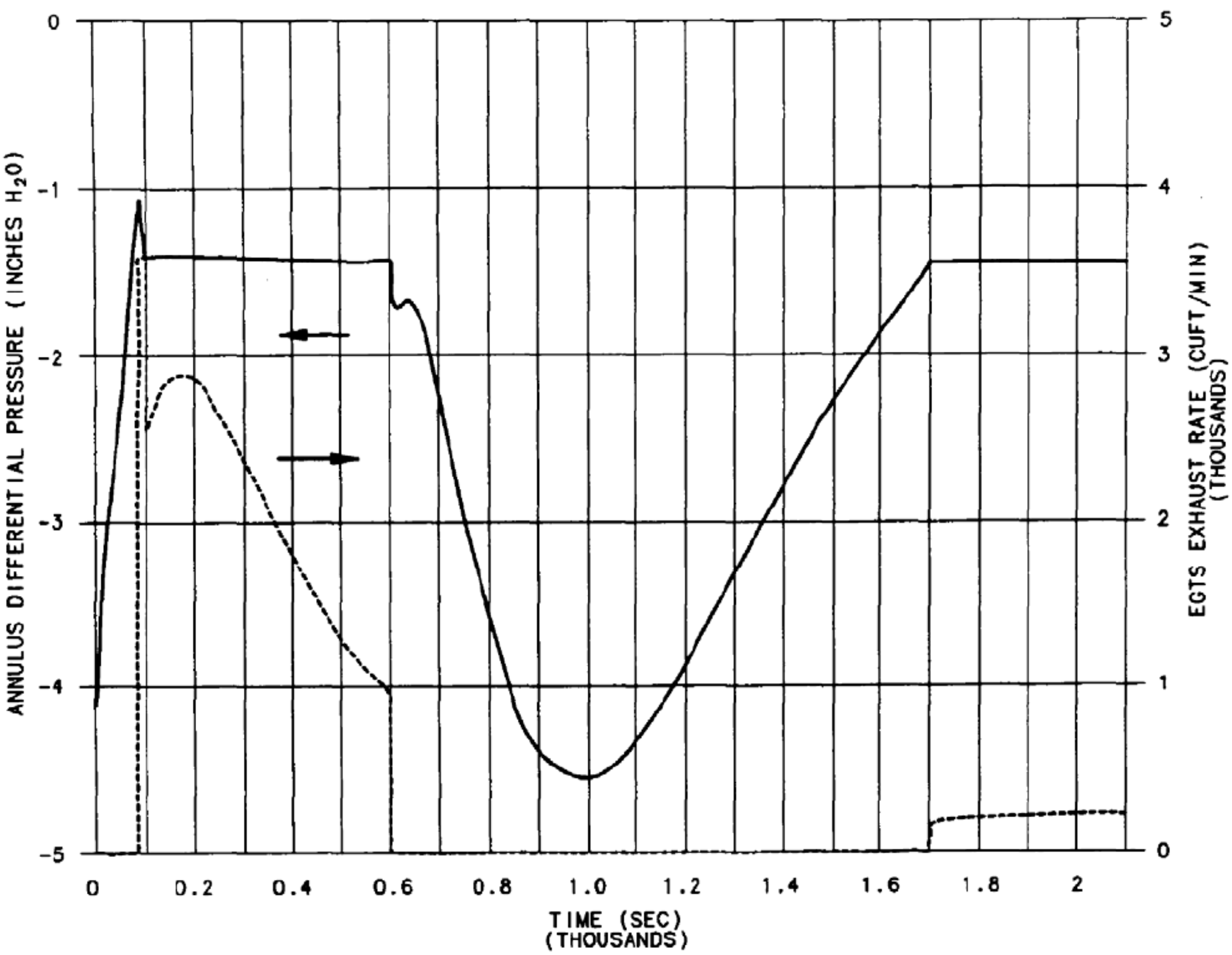
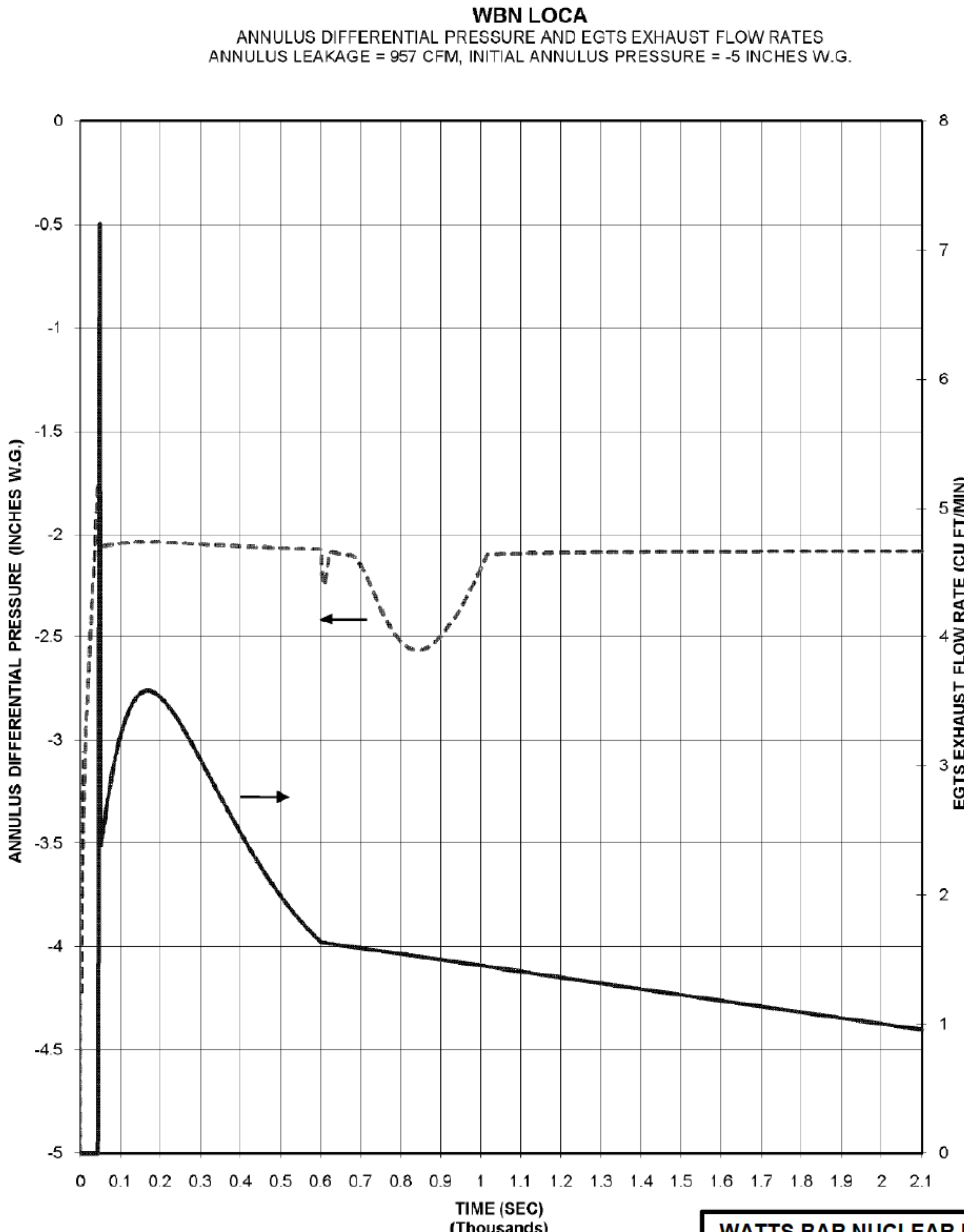
AUXILIARY BUILDING
UNITS 1 & 2
FLOW DIAGRAM
HEATING & VENTILATING
AIR FLOW
TVA DWG NO. 0-47W866-10 R1
FIGURE 6.2.3-16

WBN LOCA

ANNULUS DIFFERENTIAL PRESSURE AND EGTS EXHAUST RATES
ANNULUS LEAKAGE = 250 CFM, INITIAL ANNULUS PRESSURE = -5 INCHES H_2O



WATTS BAR NUCLEAR PLANT FINAL SAFETY ANALYSIS REPORT
Post Accident Annulus Pressure and Reactor Unit Vent Flow Rate Transients Figure 6.2.3-17



WATTS BAR NUCLEAR PLANT
FINAL SAFETY
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Post Accident Annulus
Pressure and Reactor Unit
Vent Flow Rate Transients

Figure 6.2.3-17A

SECURITY-RELATED INFORMATION, WITHHELD UNDER 10CFR2.390

FIGURE 6.2.3-18

SECURITY-RELATED INFORMATION, WITHHELD UNDER 10CFR2.390

FIGURE 6.2.3-18

UNIT 2

SECURITY-RELATED INFORMATION, WITHHELD UNDER 10CFR2.390

FIGURE 6.2.3-19

SECURITY-RELATED INFORMATION, WITHHELD UNDER 10CFR2.390

FIGURE 6.2.3-19

UNIT 2

6.2.4 CONTAINMENT ISOLATION SYSTEMS

The containment isolation systems provide the means of isolating fluid systems that pass through containment penetrations so as to confine to the containment any radioactivity that may be released in the containment following a design basis event. The containment isolation systems are required to function following any design basis event that initiates a Phase A or Phase B containment isolation signal or releases radioactive materials into containment to isolate nonsafety-related fluid systems penetrating the containment. The Watts Bar Nuclear Plant does not have a particular system for containment isolation, but isolation design is achieved by applying common criteria to penetrations in many different fluid systems and by using ESF signals to actuate appropriate valves.

6.2.4.1 Design Bases

The main function of the containment isolation system is to provide containment integrity when needed. Containment integrity is defined to exist when:

1. The nonautomatic containment isolation valves and blind flanges are closed as required.

2. The containment equipment hatch is properly closed.

3. At least one door in each containment personnel air lock is properly closed.

4. Automatic containment isolation valves are operable or are deactivated in the closed position or at least one valve in each line having an inoperable valve is closed.

5. Requirements of the Technical Specification with regard to containment leakage and test frequency are satisfied.

Containment integrity is required if there is fuel in the reactor which has been used for power operation, except when the reactor is in the cold shutdown condition with the reactor vessel head installed, or when the reactor is in the refueling shutdown condition with the reactor vessel head removed. Containment isolation is not essential for design basis events, such as HELBs, outside containment which do not release radioactive materials into containment. The failure of containment isolation valves for such an event would not result in the release of radioactive fluids from inside containment.

In general, the containment isolation system is designed to the requirements of General Design Criteria 54, 55, 56, and 57 of 10 CFR 50, Appendix A. The following are alternate containment isolation provisions for certain classes of lines:

1. Fluid instrument lines penetrating the containment are designed to meet the referenced General Design Criteria except for the pressure sensor and reactor vessel level instrumentation system lines. Instrument lines which penetrate containment are listed in Table 6.2.4-4.

2. Remote-manual valves are used for isolation provisions on certain lines associated with engineered safety features (such as the ECCS) instead of automatic isolation valves.

3. A closed system outside the containment is acceptable as one of the two isolation barriers if designed to the following criteria:

   a. Does not communicate with the outside environment

   b. Meets Safety Class 2 design requirements

   c. Withstands the internal temperatures and pressures which occur as a result of the containment design basis events

   d. Withstands loss-of-coolant accident transients and environment

   e. Meets Seismic Category I design requirements

   f. Protected against missiles, pipe whip, and jet impingement.

4. The isolation function of an engineered safety feature or system required to test an engineered safety feature requires one barrier to remain functional after the occurrence of a single active failure. Normally, this is accomplished by providing two isolation valves in series. If it can be shown that a single active failure can be accommodated with only one valve in the line and that fluid system reliability is enhanced by having one valve rather than two valves in series, then one valve and a closed system both located outside of the containment are acceptable. The single valve and piping between the containment and the valve are enclosed in a protective leaktight housing to prevent leakage to the atmosphere in the event of external leakage.

5. Relief valves may be used as isolation valves provided the relief valve setpoint is greater than 1.5 times the containment design internal pressure.

The criteria for the number and location of containment isolation valves in each fluid system depend on the valves functions and whether they are open or closed to the containment atmosphere or reactor coolant system. Four isolation classes of fluid system penetrations are defined as follows:

1. Isolation Class I - Fluid lines which are open to the atmosphere outside the containment and are connected to the reactor coolant system or are open to the containment atmosphere. Each isolation Class I system has a minimum of two isolation valves in series. Where system design permits, one valve is located inside and one valve is located outside containment.

2. Isolation Class II - Fluid lines which are connected to a closed system outside the containment and are connected to the reactor coolant system or are open to the containment atmosphere. Also included in isolation Class II are fluid lines which are open to the atmosphere outside the containment and are separated from the reactor coolant system and the containment atmosphere by a closed system inside the containment. Each isolation Class II system has, as a minimum, one isolation valve.

3. Isolation Class III - Fluid lines which are connected to a closed system both inside and outside the containment. Isolation Class III systems have, as a minimum, one isolation valve.

4. Isolation Class IV - Fluid lines which must remain in service subsequent to a design basis event, such as lines serving ESF systems. Isolation valves on these lines are not automatically closed by the containment isolation signal. Each isolation Class IV system has, as a minimum, one isolation valve (remote-manual operation).

The following design requirements for containment isolation barriers apply:

1. The design pressure of all piping and connected equipment comprising the isolation boundary is equal to or greater than the design pressure of the containment.

2. All valves and equipment which are considered to be isolation barriers and designed in accordance with Seismic Category I criteria shall be protected against missiles and jet impingement, both inside and outside the containment.

3. All valves and equipment which are considered to be isolation barriers are designed, as a minimum, to ASME Section III Class 2 requirements except as noted in Item 1 of Section 6.2.4.2.1.

4. A system is closed inside the containment if it meets all of the following:

    a. It does not communicate with either the reactor coolant system or the reactor containment atmosphere.

    b. It will withstand external pressure and temperature equal to containment structural integrity test pressure and design temperature.

c. It will withstand accident temperature, pressure, and fluid velocity transients, and the resulting environment, including internal thermal expansion.

d. It is protected against missiles, pipe whip, and jet impingement.

5. A check valve inside the containment on the incoming line is considered an automatic isolation valve.

6. A pressure relief valve that relieves toward the inside of the containment is considered an automatic isolation valve.

7. A locked closed valve is considered an automatic isolation valve.

8. To qualify as an automatic isolation valve, an air-operated valve must fail closed on loss of air, power, etc.

9. Valves used for containment isolation shall be designed capable of tight shutoff in order to limit gas leakage to a specific maximum amount at containment design pressure.

The design bases for the containment isolation system include provision for the following:

1. A double barrier at the containment penetration in those fluid systems that are not required to function following a design basis event.

2. Automatic, fast, efficient closure of those valves required to close for containment integrity following a design bases event to minimize release of any radioactive material.

3. A means of leak testing barriers in fluid systems that serve as containment isolation.

4. The capability to periodically test the operability of containment isolation valves.

6.2.4.2 System Design

The containment isolation system meets the design bases presented in Section 6.2.4.1 with the exception of those cases which are discussed in detail in Section 6.2.4.3.

Containment isolation can be initiated by either Phase A or Phase B signals.

A Phase A signal is generated by either of the following:

1. Manual - either of two momentary controls

2. A safety injection signal, generated by one or more of the following:

   a. Low steamline pressure in any steamline

   b. Low pressurizer pressure

   c. High containment pressure

   d. Manual - either of two momentary controls.

A Phase B signal is generated by either of the following:

1. Manual - two sets (two switches per set) - actuation of both switches is necessary in either set for spray initiation

2. Automatic high-high containment pressure.

Containment isolation Phase A exists if containment isolation Phase B exists, when the Phase B signal is initiated by automatic instrumentation. Phase A containment isolation does not occur when the Phase B signal is initiated manually. The instrumentation circuits that generate both Phase A and Phase B signals are described in Chapter 7.

With exception of the cases identified below, the containment isolation system provides for automatic, fast, and efficient closure of those valves required to close for containment integrity following a design basis event to minimize the release of radioactive material. Closure times for isolation valves are included in Table 6.2.4-1.

With a single failure of containment isolation valves 1-FCV-32-110, 2-FCV-32-111, FCV-67-107, concurrent with an HELB inside containment such as a LOCA, Main Steam Line Break, or Feedwater Line Break, etc., manual actions are required to prevent the possibility of process fluid (air or water) from entering containment (see Tables 9.2-2, 9.2-9, and 9.3-7). Manual actions are required in these cases since a check valve serves as an inboard containment isolation valve, and pressure boundary integrity of the lines in containment past the check valves cannot be assured.

6.2.4.2.1 Design Requirements

Containment isolation barrier design includes the following requirements:

1. As a minimum, containment barriers are designed to ASME Section III Class 2 requirements. This design meets the requirements of Regulatory Guide 1.26 for the containment isolation systems, except that the four auxiliary feedwater lines incorporate safety-grade Quality Group C (ASME Section III, Class 3) valves outside containment for isolation. This has been documented in NUREG 0847 as acceptable to the NRC. Valves and equipment which are considered to be isolation barriers are designed to Seismic Category I requirement which is the intent of Regulatory Guide 1.29.

2. Isolation barriers either inside or outside of the containment are protected against missiles, pipe whip, and jet impingement during a LOCA.

3. Power operated isolation valves are tested for operability by the manufacturer and preoperationally after installation. Those automatic isolation valves with air or motor operators that do not restrict normal plant operation are periodically tested to ensure operability.

Additional design information is included in Table 6.2.4-1.

6.2.4.2.2 Containment Isolation Operation

A containment isolation signal initiates closing of automatic isolation valves in those lines which must be isolated immediately following a design basis event. The containment isolation valves will close within the time specified in Table 6.2.4-1. However, on loss of ac power, the diesel will have to be started prior to closure. It is estimated that the time required to start the diesel is 10 seconds. The logic diagram for this system is shown in Figure 6.2.4-21.

Check valves are used under conditions where differential pressure will close the valves to maintain containment integrity. Lines which, for safety reasons, must remain in service subsequent to a design basis event are provided with at least one isolation valve.

Each automatic isolation valve required to operate subsequent to an accident is additionally provided with a manual control switch for operation. The position of these automatic isolation valves is indicated by status lights in the main control room. Primary and secondary modes of valve actuation are shown in Table 6.2.4-1.

Redundant isolation barriers are used to prevent any single failure from causing an open path from the containment. If two power operated valves are used in series in a line for isolation purposes, one valve is supplied with one train of control and power and the other valve is supplied by the other train. Redundancy in power, signals, and barriers is provided to assume isolation.

Provisions for detecting leakage from remote manually controlled systems (such as the ECCS) include the use of pressure and flow meters, and inspection of the systems during normal plant operation. Details for leak detection are given in the appropriate system descriptions. Piping systems penetrating the containment have been provided with test vents and test connections or have other provisions to allow periodic leak testing (see Section 6.2.6).

For Unit 2, test connections, vents and drains located within the containment boundary (between containment isolation valves) of penetrations subject to 10CFR50, Appendix J, Type C testing also have the capability of being leak tested or are one inch in size or less, administratively secured closed and consist of a double barrier (e.g. two valves in series, one valve with nipple and cap, one valve and a blind flange).

The manufacturers of isolation system components perform tests to demonstrate the ability of mechanical and electrical components located inside the containment to perform as required in the containment environment following the design basis accident. Accident conditions which are considered in the design of isolation components are pressure, humidity, radiation, and

temperature. Section 3.11 gives information concerning the environmental conditions used in the design of the containment isolation system including more detail on qualification testing of ESF components.

The description and design requirements for the instrumentation and control portions of the containment isolation system are discussed in Chapter 7.

6.2.4.2.3 Penetration Design

The penetrations are classified into 24 different types. These are shown in Figures 6.2.4-1 through 6.2.4-17E.

The locations of these penetrations through the steel containment and the Shield Building are shown in Figures 6.2.4-18 and 6.2.4-19, respectively. The penetrations are tabulated in Table 6.2.4-1. The different types of penetrations are discussed below and the various possible leakage paths, as tabulated in Table 6.2.4-1 and shown in Figure 6.2.4-20, are also described below.

Penetration Types I and II - Main Steam and Feedwater

The main steam and feedwater line penetrations, shown in Figures 6.2.4-1 and 6.2.4-2, are the "hot" type in which the penetrations must accommodate thermal movement. Each "hot" process line where it passes through the containment penetration is enclosed in a guard pipe that is attached to the process line through a multiple fluid fitting. The guard pipe protects the bellows should the process line fail within the annulus between the containment vessel and the Shield Building, thereby precluding the discharge of fluids into the annulus. The inner end of the guard pipe is fitted with an impingement ring which protects the bellows from jets originating from pipe breaks inside containment. In addition, the guard pipe for this type of penetration extends through and is supported by the crane wall. This avoids transmitting loads to the containment vessel. Also, in the event of a pipe rupture it discharges fluid into the reactor compartment rather than smaller rooms outside the crane wall, thus preventing, overpressurization of these smaller rooms.

For each of these penetrations the penetration sleeve is welded to the containment vessel. The process line which passes through the penetration is allowed to move both axially and laterally. A two-ply bellows expansion joint is provided to accommodate any movement between the containment vessel and the Shield Building, under any conditions. The bellows is designed to withstand containment design pressure. When an embedded anchor is not utilized, a low-pressure flexible closure will seal the process line to the sleeve in the Shield Building, which will not impose significant stress on the penetration.

The flexible closure described above is located outdoors and serves to contain leakage from the flued head so that the leakage is routed back to the annulus, and to seal the annulus from the outdoors.

Guides and anchors limit movement of pipes such that design limits on the containment penetration and bellows are not exceeded during conditions of plant operation, test, or postulated accidents.

Penetration Type III - Residual Heat Removal Pump Supply and Return

The RHR pump supply and return penetrations, shown in Figure 6.2.4-3, are also the "hot" type. For these penetrations, the guard pipe does not penetrate the crane wall. This type of penetration is anchored at the Shield Building wall in addition to being supported from the internal concrete structure to minimize loads transmitted to the steel containment vessel.

The Shield Building sleeves have embedded anchors and the flued heads are in the Auxiliary Building. There is no need for low-pressure flexible closures as used in penetrations types I and II, since any leakage from the flued head will be processed by the auxiliary building gas treatment system.

Penetration Types IV and V

Types IV and V penetrations are also thermally "hot" with insulation and bellows, as shown in Figure 6.2.4-4. Any leakage through the flued heads or through the bellow will be into the annulus and thereby processed by the emergency gas treatment system. The two types differ by only the weld ends.

Penetration Types VI, VII, and VIII

Penetrations types VI through X and XIII through XVIII are "cold" penetrations.

For "cold" piping penetrations, a low-pressure flexible closure will seal the cold pipe to the sleeve penetrating the Shield Building. The piping configuration and supports on either side of the penetration will be designed to preclude overstressing the containment vessel at the penetration under any conditions, including postulated accidents.

Relatively small thermal movement or stress is expected for the "cold" penetrations. The clearance space provided for the pipe going through the Shield Building wall is computed by the summation of the relative movements of the pipe and the Shield Building. Ample clearance space is provided so that the pipe will not be in contact with the Shield Building sleeve under any condition.

Penetration types VI and VII have provisions for dissimilar metal welding. The two types differ in their weld ends only. Penetration types VI and VII are illustrated in Figure 6.2.4-5. The flued heads of both types are located in the annulus.

Penetration type VIII is similar to that of penetrations types VI and VII, except that there is no dissimilar metal weld. Penetration type VIII is illustrated in Figure 6.2.4-6.

Penetration Type IX Containment Spray and RHR Spray Headers

There is no difference between penetration types VII and IX except that penetration type IX is located at the dome. Penetration type IX is illustrated in Figure 6.2.4-7. The flued heads are located in the annulus.

Penetration Type X - Multiple Line Sleeves

Type X penetrations are primarily for instrumentation lines such as sampling and monitor lines. Typical multiple line sleeves are shown in Figure 6.2.4-8.

Penetration Types XI and XII - Emergency Sump

During long-term post-accident conditions, containment sump water is recirculated through the RHR system and the containment spray system. The water collects on the floor of the containment and flows to the emergency sump. The water flows out of the containment through two type XII penetrations shown in Figure 6.2.4-9. Each line contains an isolation valve enclosed in a valve compartment. The valve compartments are designed for the same conditions as the containment except for leaktightness. The penetration between each valve compartment and the Auxiliary Building is a Type XI penetration illustrated in Figure 6.2.4-10.

The type XII penetration has a flued head located in the containment sump. The outer sleeve (guard pipe) of the flued head is welded directly to the containment liner which is completely embedded in the concrete.

The type XI penetration has the flued head located in the Auxiliary Building. The penetration is insulated because of the hot sump water which would pass through it in the event of a design bases event.

Penetration Type XIII - Ventilation

Heating and ventilation ducts utilize penetration type XIII, as shown in Figure 6.2.4-11. Process lines are welded directly to these penetrations. Additional information on ventilation duct penetrations is given in Section 6.2.4.3.1 on possible leakage paths.

Penetration Type XIV - Equipment Hatch

An equipment hatch fabricated from welded steel and furnished with a double-gasketed flange and bolted dished door is provided. A test connection to the space between the gaskets is provided to pressurize the space for leak rate testing, as shown in Figure 6.2.4-12.

Penetration Type XV - Personnel Access

Two personnel air locks are provided. Each personnel air lock, as shown in Figure 6.2.4-13, is a double door welded steel assembly. Quick-acting type equalizing valves are provided to equalize pressure in the air lock when personnel enter or leave the containment vessel. The doors are sealed with double gaskets. A test connection to the space between the gaskets is provided to pressurize the space for leak rate testing. The emergency air supply connection to the space between the double doors serves as a test connection to pressurize this space for leak rate testing. A special hold-down device is provided to secure the inner door in a sealed position during leak rate testing of the space between the doors.

The two doors in each personnel air lock are interlocked to prevent both being opened simultaneously and to ensure that one door and its equalizing valve are completely closed before the opposite door can be opened. Remote indicating lights and annunciators located in the main control room indicate the door is in operational status. Provision is made to permit bypassing the door interlocking system with a special tool to allow doors to be left, open during plant cold shutdown. Both doors may also be left open during fuel handling activities under special administrative controls as discussed in Section 6.2.3.2.1. Each lock door hinge is designed to be capable of adjustment to assure proper seating. A lighting and communication system is provided in the lock interior.

Penetration Type XVI - Fuel Transfer Tube

A 20-inch OD fuel transfer tube penetration is provided for fuel movement between the refueling canal in the containment and the spent fuel pool. The penetration consists of 20-in stainless steel pipe installed inside a 24-inch carbon steel pipe, as shown on Figure 6.2.4-14. The inner pipe acts as the transfer tube and is fitted with a double gasketed blind flange in the refueling canal and a standard gate valve in the spent fuel pool. The inner pipe is welded to the containment penetration sleeve. Bellows expansion joints are provided on the pipes to compensate for any differential movement between the two pipes or other structures.

Penetration Type XVII - Thimble Renewal (Unit 1 Only)

Incore instrumentation thimble renewal requires penetrations in both the steel containment and the Shield Building at the same elevation and azimuth. These are separate penetrations and are not connected in the annulus. The containment penetration is illustrated in Figure 6.2.4-15. Double O-ring gaskets and leak rate test connectors are provided for the containment penetration, and the Shield Building penetration utilizes a raised face blind flange with a single gasket.

Penetration Type XVII - Incore Instrumentation Thimble Assembly (IITA) Renewal (Unit 2 Only)

Incore instrumentation thimble assembly (IITA) renewal requires penetrations in both the steel containment and the Shield Building at the same elevation and azimuth. These are separate penetrations and are not connected in the annulus. The containment penetration, illustrated in Figure 6.2.4-15, has a blind flange with double O-rings and a leak rate test connector installed inside the annulus. The shield building penetration utilizes a raised face blind flange with a single gasket installed outside the shield building.

Penetration Type - XVIII - Ice Blowing

The ice blowing line penetration has a blind flange with double O-rings installed on the outside of the containment as shown in Figure 6.2.4-16. Sealing between the outside and the annulus penetration through the shield wall is provided by a blind flange fitted with a gasket installed on the inside and outside of the Shield Building penetration.

Penetration Type XIX - Electrical

The electrical penetration assemblies provide a means for the continuity of power, control, and signal circuits through the primary containment structure.

Each assembly consists of redundant pressure barriers through which the electrical conductors are passed, as shown in Figure 6.2.4-17.

Each penetration assembly is sized such that it may be inserted into and be compatible with the penetration nozzles which are furnished as a part of the containment structure. Unless otherwise specified, the assembly is designed to be inserted from the outboard-end of the primary containment nozzle.

The criteria and requirements for the design, construction, and installation of the modular type electrical penetrations conform to IEEE Standard 317-1976, "IEEE Standard for Electrical Penetration Assemblies in Containment Structures for Nuclear Fueled Power Generating Stations."

The criteria and requirements for the design, construction and installation of fiber optic feed-throughs in modular type electrical penetrations conform to IEEE Standard 317-1983, “IEEE Standard for Electric Penetration Assemblies in Containment Structures for Nuclear Power Generating Station.”

Penetration Type XX

The feedwater bypass line penetrations, shown in Figure 6.2.4-17A are the 'hot' type in which the penetrations must accommodate thermal movement. Each 'hot' process line where it passes through the containment penetration is enclosed in a guard pipe that is attached to the process line through a multiple fluid fitting. The guard pipe protects the bellows should the process line fail within the annulus between the containment vessel, thereby precluding the discharge of fluids into the annulus. The inner end of the guard pipe is fitted with an impingement ring which protects the bellows from jets originating from pipe breaks inside containment. In addition, the guard-pipe for this type of penetration extends through and is supported by the crane wall. This avoids transmitting loads to the containment vessel. Also, in the event of a pipe rupture it discharges fluid into the reactor compartment rather than smaller rooms outside the crane wall, thus preventing overpressurization of these smaller rooms.

For each of these penetrations, the penetration sleeve is welded to the containment vessel. The process line which passes through the penetration is allowed to move both axially and laterally. A two-ply bellows expansion joint is provided to accommodate movement between the containment vessel and the Shield Building. The bellows is designed to withstand containment design pressure. When an embedded anchor is not utilized, a low-pressure flexible closure will seal the process line to the sleeve in the Shield Building, which will not impose significant stress on the penetration.

The flexible closure described above is located outdoors and serves to contain any leakage from the flued head so that the leakage is routed back to the annulus, and to seal the annulus from the outdoors.

Guides and anchors limit movement of pipes such that design limits on the containment penetration and bellows are not exceeded during plant operation, testing, or postulated accidents.

Penetration Type XXI

The ERCW lines and several component cooling water lines employ penetration type XXI, as shown in Figure 6.2.4-17B. Process lines are welded directly to these penetrations.

Penetration Type XXII

The type XXII penetration is used for the multiple line nitrogen penetration. This penetration is shown in Figure 6.2.4-17C.

Penetration Type XXIII

This type of penetration is used for the chilled water lines and each penetration contains a single chilled water line. The penetration is illustrated in Figure 6.2.4-17D.

Penetration Type XXIV

Type XXIV penetrations are used for maintenance ports. These penetrations employ bellows as shown in Figure 6.2.4-17E. Any leakage through the flued heads or through the bellows will be into the annulus and thereby processed by the emergency gas treatment system.

The following codes, standards, and guides were applied in the design of the containment isolation system.

1. 10 CFR Part 50

2. ASME Boiler and Pressure Vessel Code Section III

3. Regulatory Guide 1.26

4. Regulatory Guide 1.29

5. ANSI N18.2 - August 1970 draft

6. IEEE Standard 317-1976

6.2.4.3 Design Evaluation

The containment isolation systems are designed to present a double barrier to any flow path from the inside to the outside of the containment using the double-barrier approach to meet the single-failure criterion.

When permitted by fluid system design, diverse modes of actuation are used for automatic isolation valves. In addition to diverse modes of operation, channel separation is also maintained. This also ensures that the single-failure criterion is met.

Adequate protection is provided for piping, valves, and vessels against dynamic effects and missiles which might result from plant equipment failures, including a LOCA.

Isolation valves inside the containment are located between the crane wall and the inside containment wall. The crane wall serves as the main missile barrier. Other missile barriers are discussed in Section 3.5.

The requirements and intent of NRC General Design Criteria 54, 55, 56, and 57 have been met with five exceptions.

a. Primary containment monitoring instrument systems shall be designed to maintain the integrity of the containment isolation boundary in the event of a DBE. The instrument systems consist of pressure sensors (e.g., transmitters) located outside containment and associated sense lines that connect to the containment penetration nozzles. The sensors should be located as close as practical to the associated penetration nozzle. Any drain or test line used shall meet the double isolation barrier by use of two normally closed manual valves in series.

   The instrument system shall be designed to Seismic Category I requirements and evaluated for effects of possible missiles, pipe whip, and jet impingement.

b.1 The reactor vessel level indication system (RVLIS) is required post accident for continual indication of the water level in the reactor vessel. The capillary sensing lines which transmit pressure from the reactor vessel to instruments in the Auxiliary Building are armored and designed to withstand DBE conditions. Any containment isolation valves installed in the RVLIS capillary lines will jeopardize the performance of the system. For this reason, isolation of these capillary lines is accomplished by a sealed sensor located inside containment and an isolator located outside containment. These devices utilize a type of bellows which transmits pressure while preventing mixing of the fluids on either side of the isolation devices. The capillary line is armored stainless steel tubing and is filled with demineralized water and sealed. A postulated shear of this capillary line on either side of the containment would not allow a leak to develop through the containment boundary.

b.2 The RCS wide range pressure transmitter (PT-68-70) is required post accident for continual indication of the pressure in the reactor vessel. The capillary sensing lines which transmit pressure from the reactor vessel to instruments in the Auxiliary Building are armored and designed to withstand DBE conditions. Any containment isolation valves installed in the RCS wide range pressure transmitter capillary lines will jeopardize the performance of the system. For this reason, isolation of these capillary lines is accomplished by a sealed sensor located inside containment and an isolator located outside containment. These devices utilize a type of bellows which transmits pressure while preventing mixing of the fluids on either side of the steel tubing and is filled and sealed. A postulated shear of this capillary line on either side of the containment would not allow a leak to develop through the containment boundary.

c. Containment isolation for each RHR sump line penetration consists of:

1. A closed system outside containment.

2. A containment isolation valve outside containment in each of the two lines after the penetrating line branches in the RHR sump valve room. Both of these valves are remotely controlled from the main control room.

An enclosure of the RHR sump lines and isolation valves is provided from the containment out to and including the isolation valves. However, this enclosure is not designed to be leaktight after an accident for the following reasons:

1. The maximum pressure which will be experienced inside the RHR sump line will only be about 25 psig.

2. One of the isolation valves, the containment sump valve, is qualified to 600 psig. The other isolation valve, the containment spray valve, is qualified to 200 psig.

3. This portion of the system only operates post accident and, therefore, only a limited leak passive failure need be postulated (and this would be at the valve). However, based on the above two statements and the fact that deadweight loading (i.e., normal operation) should not exceed the MELB criteria, the over-design should preclude any problem.

Thus, the penetration has such over conservatism in its design that an external leaktight enclosure around the valves is not necessary.

d. The pressure boundary valve leak rate test line containment isolation valves (63-158, 63-112, 63-111, 63-167, 63-174, 63-21, and 63-121) are remote manually actuated from the main control room and do not receive a containment isolation signal. These valves are open for short periods of time during normal operation for the performance of SIS and RHR system venting. Thus, these valves do not automatically close when the containment isolation or safety injection signal is initiated during the venting of the SIS and RHR system. This exception is acceptable because administrative controls exist in the test document to assure valve closure after testing and containment integrity is not compromised during pump operation (i.e., during testing at accident conditions) since flow is being maintained into containment.

e. The design configuration for penetrations X-79A (ice blowing) and X-79B (negative return) is temporarily modified in operating Modes 5 and 6 and when the reactor is defueled to support ice blowing activities. The normally closed blind flange on each penetration will be opened and temporary piping will be installed in the penetrations. A 12-inch silicone seal will be installed between the piping segment and the penetration. Manual isolation valves will be connected to the piping on the inboard and outboard side of the penetrations. Administrative controls will ensure timely closure of the valves, if necessary, subsequent to a LOCA in the other unit. The penetrations will be returned to their normal design configuration prior to entry into Mode 4 operation.

6.2.4.3.1 Possible Leakage Paths

Possible leakage paths from the containment are defined below. The leakage paths are defined on the basis that the annulus pressure is less than outdoor ambient, the Auxiliary Building, and the containment pressures. Therefore, whenever containment is required, leakage is into the annulus. The possible leakage paths considered do not include containment leakage through the steel plates or through the full penetration welds in the containment vessels. The possible leakage paths also do not include shield building embedments. This is acceptable, as leakage through any of these paths will be into the annulus and the leakage will be processed by the EGTS.

The more probable sources of containment and Shield Building leakage, such as elastomer seals, bellows, and through lines are considered as possible leak path types. Each penetration that contains elastomer seals or a bellows has at least one leakage path defined in Table 6.2.4-1. Penetrations not open to the annulus are considered as possible paths for through-line containment leakage and have one or more isolation valves. Thus every pipe penetration has at least one type of leak path listed in Table 6.2.4-1. The five different types of possible leakage paths are shown in Figure 6.2.4-20, tabulated in Table 6.2.4-1 and are discussed separately below.

Type A - Leakage Path

Type A leakage is leakage from the Auxiliary Building into the annulus. Type A penetration leakage includes the following:

1. Equipment hatch Shield Building sleeve leak (see Figure 6.2.4-12).

2. Annulus access door leak.

3. Containment purge supply and exhaust isolation valves outside Shield Building leak. The possible leakage is through the valves and the leakoff (see Figure 6.2.3-2) into the annulus.

4. Shield Building penetration seal leakage.

Type B - Leakage Path

Type B leakage paths are from the containment to the annulus. Type B leakage includes the following:

1. Equipment hatch double O-ring through-line leak (see Figure 6.2.4-12).

2. Ice blowing line O-ring and blind flange through line leak (see Figure 6.2.4-16 and Figure 6.2.4-23).

3. Penetration bellows leak.

4. Containment purge supply and exhaust inboard and outboard valves through line leak. The leakage will pass through the leak off (see Figure 6.2.3-2) into the annulus.

5. Containment thimble (Unit 1) or Incore Instrumentation Thimble Assembly (Unit 2) renewal line double O-ring through-line leak (see Figure 6.2.4-15).

Type C - Leakage Path

Type C leakage is leakage from the out-of-doors into the annulus and includes the following:

1. Shield Building thimble (Unit 1) or Incore Instrumentation Thimble Assembly (Unit 2) renewal line flange single gasket through-line leak.

2. Main steam and feedwater lines annulus seal leak.

3. Ice blowing line Shield Building blind flange leak (See Figure 6.2.4-16).

Type D Leakage Path

Type D leakage path covers the through-line leakage from the containment to the Auxiliary Building (see Table 6.2.4-2). Included in this type of leakage are the lines associated with the safety systems required for post-LOCA operation, such as containment spray, RHR spray, high-head SIS, low-head SIS, SIS pump discharge, charging pump discharge, and containment emergency sump. For "closed" systems inside the Auxiliary Building, the through line leakage will stay within the closed system. The component cooling water system is basically a closed system, except for the vent header at the surge tank. Any through line leakage into the Auxiliary Building through this vent will be processed by the auxiliary building gas treatment system. Radiation monitoring is provided as a signal to initiate the closing of the vent. The nitrogen supply lines to the pressurizer relief tank and to the accumulators are normally closed. The high pressure outside of the isolation valves serves to minimize through line leakage outward. The personnel lock is yet another possible source for through line leakage, but the leakage through the double O-ring (assuming one door open) is small, if any, and will be processed by the auxiliary building gas treatment system.

Type E - Leakage Path

Type E leakage paths are paths from the containment that bypass the annulus and leak directly past a cleanup system. These leakage paths were considered during the design of the Watts Bar Nuclear Plant. The design features utilized at Watts Bar eliminates Type E leakage paths. This is done by the following methods:

1. Portions of the Auxiliary Building are maintained at a negative pressure relative to the outside atmosphere for the duration of an accident. Section 6.2.3 describes the implementing system and its operation.

2. Leakoff lines to the secondary containment and a third outboard valve receiving an isolation signal are used in certain lines (such as the containment purge lines) to prevent bypass leakage.

3. A water seal(s) at greater than peak containment accident pressure or which has been shown by analysis to provide the necessary containment integrity is used to prevent bypass leakage in certain lines (such as the safety injection pump discharge or ERCW upper and lower compartment supply and return lines). The seals are available for at least 30 days after a design basis event (see Table 6.2.6-3).

4. The secondary side of the steam generator is kept at a higher pressure than the primary side soon after the LOCA occurs (see Section 10.4.9). Leakage between the primary and secondary sides of the steam generator is thus directed inward to the containment.

5. For the maintenance ports through penetrations X-108 and X-109, a blind flange with a double O-ring design that is not opened during power operation and has a zero leakage acceptance criteria is relied upon to prevent bypass leakage.

Table 6.2.4-3 lists potential bypass leakage paths to the atmosphere and the methods chosen to eliminate such leakage.

6.2.4.4 Tests and Inspections

Components of the containment isolation systems were designed, fabricated, and tested under quality assurance requirements in accordance with 10 CFR 50, Appendix B, as further described in Chapter 17. An alternative to visual examination during ASME Section III hydrostatic pressure testing was approved by Reference [1] for penetrations having inaccessible vendor welds.

Nondestructive examination was performed on the components of the system in accordance with the applicable codes described in Section 3.2.

Subsequent to initial plant operation, containment isolation systems are periodically tested under conditions of normal operation to determine that all systems are in constant readiness to perform the desired function.

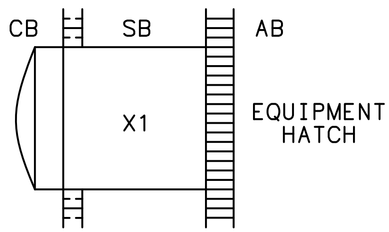
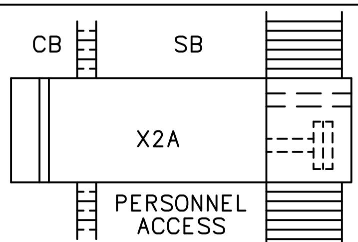
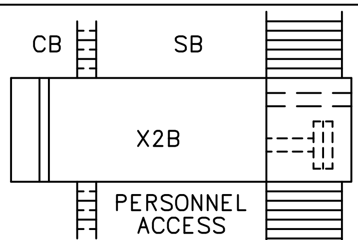
Automatic isolation valves that receive a containment isolation signal to close, where closure of the valve will not limit or restrict normal plant operation, are periodically functionally tested by the on-line testing capability described in Section 7.3. Other valves are periodically tested for CIS circuit electrical continuity. Other testing information is provided in Section 6.2.6.

REFERENCES

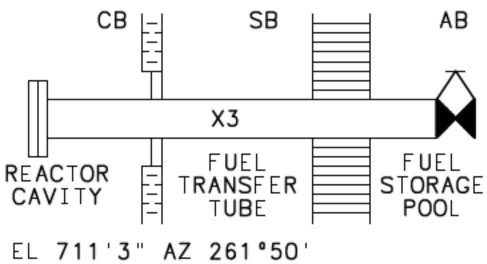
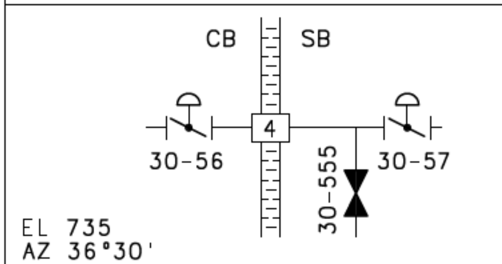
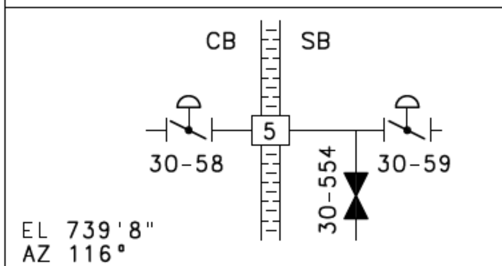
1. NRC Inspection Report Nos. 50-390/90-04 and 50-391/90-04, dated May 17, 1990.

MIS234

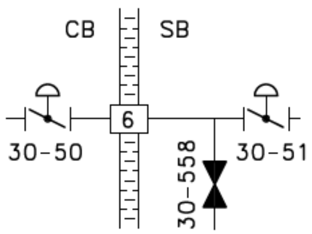
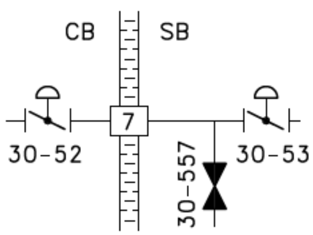
WATTS BAR NUCLEAR PLANT UNIT 1 CONTAINMENT PENETRATIONS AND BARRIERS-TABLE 6.2.4-1

PENETRATION DATA							VALVE DATA																		
DETAIL	DWG NUMBER	GEN DES CRITERION	PROCESS FLUID	FLUID STATE	POSS LEAK PATHS	SYSTEM NUMBER AND PENETRATION DESCRIPTION	VALVE LOCATION	VALVE NUMBER	ESF POWER TRAIN	VALVE TYPE	ACTUATOR	PRI ACT MODE	SEC ACT MODE	ISOLATION SIGNAL	STROKE TIME	VALVE STATUS NORMAL	VALVE STATUS SHUTDOWN	VALVE STATUS POST-ACCIDENT	VALVE STATUS POWER FAILURE	ILRT	POS IND IN MCR	ESF	APP J TEST	ESSENTIAL/NON-ESS	SHIELD BLDG PENETRATION	NOTES
 EL 764'7.5" AZ 285°	44N250	-	A	C	AB	EQUIPMENT HATCH	-	-	-	D	M	LM	-	-	-	C	V	C	-	C	-	N	AB	N	-	
 EL 719'3.63" AZ 62°	44N240	-	A	C	AD	PERSONNEL AIR LOCK	IN OUT	- -	- -	D D	M M	LM LM	- -	- -	- -	C C	O O	C C	- -	O C	Y Y	N N	AB AB	N N		MK31
 EL 760'3.63" AZ 255°	44N240	-	A	C	AD	PERSONNEL AIR LOCK	IN OUT	- -	- -	D D	M M	LM LM	- -	- -	- -	C C	O O	C C	- -	O C	Y Y	N N	AB AB	N N		MK31

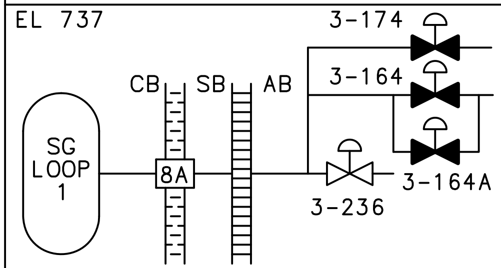
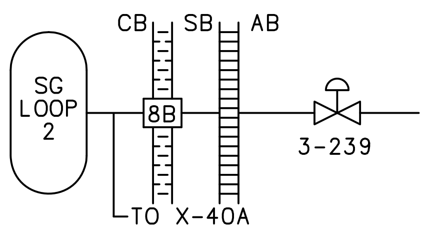
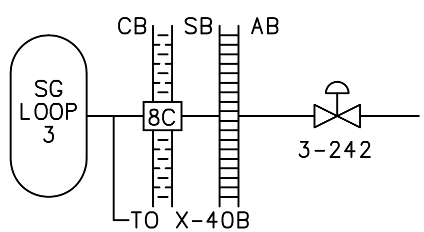
WATTS BAR NUCLEAR PLANT UNIT 1 CONTAINMENT PENETRATIONS AND BARRIERS-TABLE 6.2.4-1

PENETRATION DATA							VALVE DATA																			
DETAIL	DWG NUMBER	GEN DES CRITERION	PROCESS FLUID	FLUID STATE	POSS LEAK PATHS	SYSTEM NUMBER AND PENETRATION DESCRIPTION	VALVE LOCATION	VALVE NUMBER	ESF POWER TRAIN	VALVE TYPE	ACTUATOR	PRI ACT MODE	SEC ACT MODE	ISOLATION SIGNAL	STROKE TIME	VALVE STATUS: NORMAL	VALVE STATUS: SHUTDOWN	VALVE STATUS: POST-ACCIDENT	VALVE STATUS: POWER FAILURE	ILRT	POS IND IN MCR	ESF	APP J TEST	ESSENTIAL/NON-ESS	SHIELD BLDG PENETRATION	NOTES
	47W455-1 72-4333 -PS1 (CB & I CONTRACT #75320)	56	W	C	AB D	FUEL TRANSFER TUBE	AB	-	-	BL	M	LM	-	-	-	C	V	C	-	C	N	N	AB	N	MK62	
	47W866-1	56	A	C	AB CE	LOWER COMPARTMENT PURGE AIR EXHAUST (30)	CB SB SB	56 57 555	A B -	BF BF TC	AO AO -	AT AT LM	RM RM -	CV CV -	4.0 4.0 -	V V C	O O C	C C C	C C C	C C C	Y Y N	N N N	AC AC AC	N	F-005	3,6,14
	47W866-1	56	A	C	AB CE	INST RM PURGE AIR EXHAUST (30)	CB SB SB	58 59 554	B A -	BF BF TC	AO AO -	AT AT LM	RM RM -	CV CV -	4.0 4.0 -	V V C	O O C	C C C	C C C	C C C	Y Y N	N N N	AC AC AC	N	F-005	3,6,14

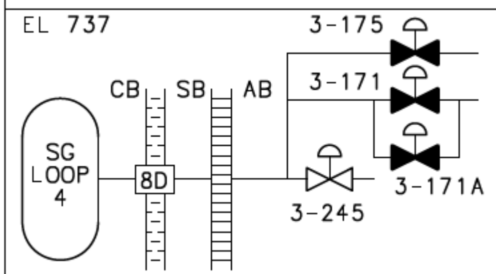
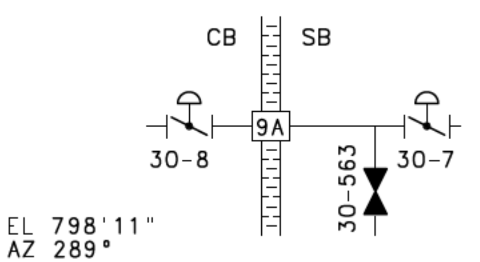
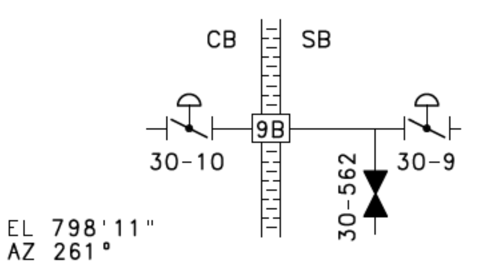
WATTS BAR NUCLEAR PLANT UNIT 1 CONTAINMENT PENETRATIONS AND BARRIERS-TABLE 6.2.4-1

PENETRATION DATA							VALVE DATA																			
DETAIL	DWG NUMBER	GEN DES CRITERION	PROCESS FLUID	FLUID STATE	POSS LEAK PATHS	SYSTEM NUMBER AND PENETRATION DESCRIPTION	VALVE LOCATION	VALVE NUMBER	ESF POWER TRAIN	VALVE TYPE	ACTUATOR	PRI ACT MODE	SEC ACT MODE	ISOLATION SIGNAL	STROKE TIME	VALVE STATUS NORMAL	VALVE STATUS SHUTDOWN	VALVE STATUS POST-ACCIDENT	VALVE STATUS POWER FAILURE	ILRT	POS IND IN MCR	ESF	APP J TEST	ESSENTIAL/NON-ESS	SHIELD BLDG PENETRATION	NOTES
 EL 749'3" AZ 293°	47W866-1	56	A	C	AB CE	UPPER COMPARTMENT PURGE AIR EXHAUST (30)	CB SB SB	50 51 558	B A -	BF BF TC	AO AO -	AT AT LM	RM RM -	CV CV -	4.0 4.0 -	V V C	O O C	C C C	C C C	C C C	Y Y N	N N N	AC AC AC	N	F-005	3,6,14
 EL 751'4" AZ 249°30'	47W866-1	56	A	C	AB CE	UPPER COMPARTMENT PURGE AIR EXHAUST (30)	CB SB SB	52 53 557	A B -	BF BF TC	AO AO -	AT AT LM	RM RM -	CV CV -	4.0 4.0 -	V V C	O O C	C C C	C C C	C C C	Y Y N	N N N	AC AC AC	N	F-005	3,6,14
X-8 EL 790'0" AZ 266°	48W406	-	-	-	-	SPARE	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	A	-	-	

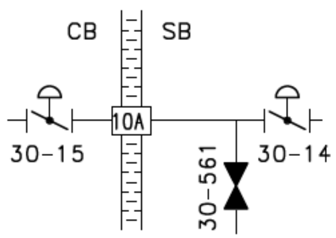
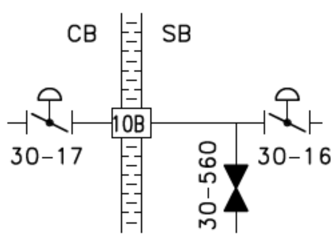
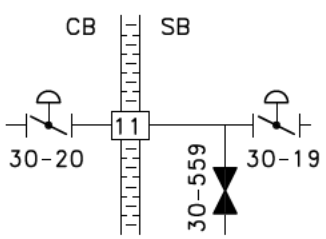
WATTS BAR NUCLEAR PLANT UNIT 1 CONTAINMENT PENETRATIONS AND BARRIERS-TABLE 6.2.4-1

PENETRATION DATA							VALVE DATA																		
DETAIL	DWG NUMBER	GEN DES CRITERION	PROCESS FLUID	FLUID STATE	POSS LEAK PATHS	SYSTEM NUMBER AND PENETRATION DESCRIPTION	VALVE LOCATION	VALVE NUMBER	ESF POWER TRAIN	VALVE TYPE	ACTUATOR	PRI ACT MODE	SEC ACT MODE	ISOLATION SIGNAL	STROKE TIME	VALVE STATUS NORMAL	VALVE STATUS SHUTDOWN	VALVE STATUS POST-ACCIDENT	POWER FAILURE	ILRT	POS IND IN MCR	ESF	APP J TEST	ESSENTIAL/NON-ESS	SHIELD BLDG PENETRATION	NOTES
EL 737 	47W803 -1,2	57	W	H	BC E	FEEDWATER BYPASS (3)	AB	236 174 164 164A	A,B B A A	GL GL GL A	AO AO AO AO	AT RM RM RM	LM LM LM LM	FW - - -	- - - -	O C C C	C C C C	C V V V	C C O C	C C C C	Y Y Y Y	N N N N	A A A A	N	MK112	22,23,25
EL 737 	47W803 -1,2	57	W	H	BC E	FEEDWATER BYPASS (3)	AB	239	A,B	GL	AO	AT	LM	FW	-	O	C	C	C	C	Y	N	A	N	MK113	22,25
EL 737 	47W803 -1,2	57	W	H	BC E	FEEDWATER BYPASS (3)	AB	242	A,B	GL	AO	AT	LM	FW	-	O	C	C	C	C	Y	N	A	N	MK114	22,25

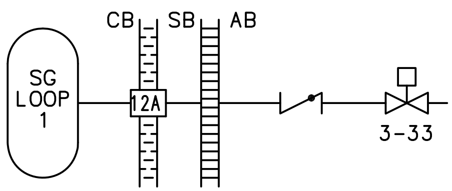
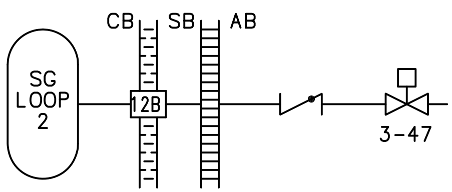
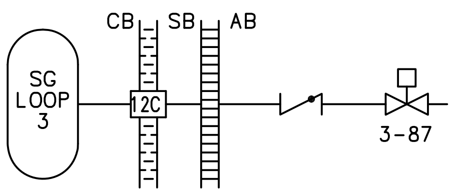
WATTS BAR NUCLEAR PLANT UNIT 1 CONTAINMENT PENETRATIONS AND BARRIERS–TABLE 6.2.4-1

PENETRATION DATA							VALVE DATA																			
																VALVE STATUS										
DETAIL	DWG NUMBER	GEN DES CRITERION	PROCESS FLUID	FLUID STATE	POSS LEAK PATHS	SYSTEM NUMBER AND PENETRATION DESCRIPTION	VALVE LOCATION	VALVE NUMBER	ESF POWER TRAIN	VALVE TYPE	ACTUATOR	PRI ACT MODE	SEC ACT MODE	ISOLATION SIGNAL	STROKE TIME	NORMAL	SHUTDOWN	POST-ACCIDENT	POWER FAILURE	ILRT	POS IND IN MCR	ESF	APP J TEST	ESSENTIAL/NON-ESS	SHIELD BLDG PENETRATION	NOTES
	47W803 -1,2	57	W	H	BC E	FEEDWATER BYPASS (3)	AB	245 175 171 171A	A,B A B B	GL GL GL A	AO AO AO AO	AT RM RM RM	LM LM LM LM	FW – – –	– – – –	O C C C	C C C C	C V V V	C C O C	C C C C	Y Y Y Y	N N N N	A A A A	N	MK115	22,23,25
	47W866-1	56	A	C	AB CE	UPPER COMPARTMENT PURGE AIR SUPPLY (30)	CB SB SB	8 7 563	B A –	BF BF TC	AO AO –	AT AT LM	RM RM –	CV CV –	4.0 4.0 –	V V C	O O C	C C C	C C C	C C C	Y Y N	N N N	AC AC AC	N	F-002	3,6,14
	47W866-1	56	A	C	AB CE	UPPER COMPARTMENT PURGE AIR SUPPLY (30)	CB SB SB	10 9 562	A B –	BF BF TC	AO AO –	AT AT LM	RM RM –	CV CV –	4.0 4.0 –	V V C	O O C	C C C	C C C	C C C	Y Y N	N N N	AC AC AC	N	F-002	3,6,14

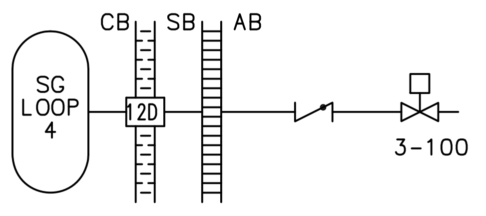
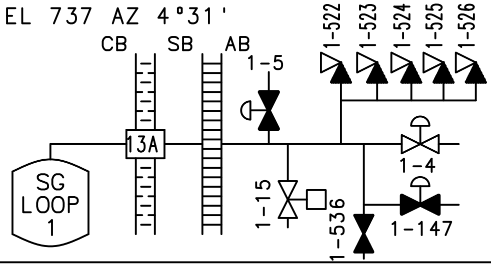
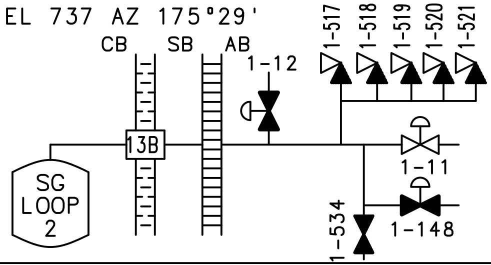
WATTS BAR NUCLEAR PLANT UNIT 1 CONTAINMENT PENETRATIONS AND BARRIERS-TABLE 6.2.4-1

PENETRATION DATA							VALVE DATA																			
DETAIL	DWG NUMBER	GEN DES CRITERION	PROCESS FLUID	FLUID STATE	POSS LEAK PATHS	SYSTEM NUMBER AND PENETRATION DESCRIPTION	VALVE LOCATION	VALVE NUMBER	ESF POWER TRAIN	VALVE TYPE	ACTUATOR	PRI ACT MODE	SEC ACT MODE	ISOLATION SIGNAL	STROKE TIME	VALVE STATUS NORMAL	VALVE STATUS SHUTDOWN	VALVE STATUS POST-ACCIDENT	VALVE STATUS POWER FAILURE	ILRT	POS IND IN MCR	ESF	APP J TEST	ESSENTIAL/NON-ESS	SHIELD BLDG PENETRATION	NOTES
 EL 737 AZ 301°	47W866-1	56	A	C	AB CE	LOWER COMPARTMENT PURGE AIR SUPPLY (30)	CB SB SB	15 14 561	B A –	BF BF TC	AO AO –	AT AT LM	RM RM –	CV CV –	4.0 4.0 –	V V C	O O C	C C C	C C C	C C C	Y Y N	N N N	AC AC AC	N	F-002	3,6,14
 EL 737 AZ 236°30'	47W866-1	56	A	C	AB CE	LOWER COMPARTMENT PURGE AIR SUPPLY (30)	CB SB SB	17 16 560	A B –	BF BF TC	AO AO –	AT AT LM	RM RM –	CV CV –	4.0 4.0 –	V V C	C C C	C C C	C C C	C C C	Y Y N	N N N	AC AC AC	N	F-002	3,6,14
 EL 728'6" AZ 57°	47W866-1	56	A	C	AB CE	INST RM PURGE AIR SUPPLY (30)	CB SB SB	20 19 559	A B –	BF BF TC	AO AO –	AT AT LM	RM RM –	CV CV –	4.0 4.0 –	V V C	O O C	C C C	C C C	C C C	Y Y N	N N N	AC AC AC	N	F-002	3,6,14

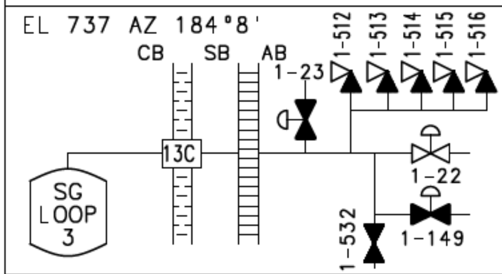
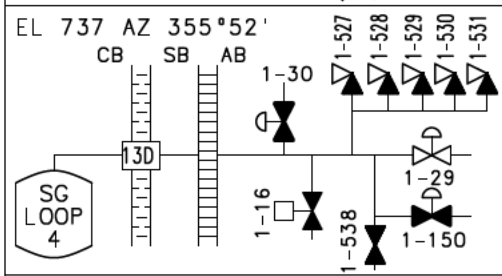
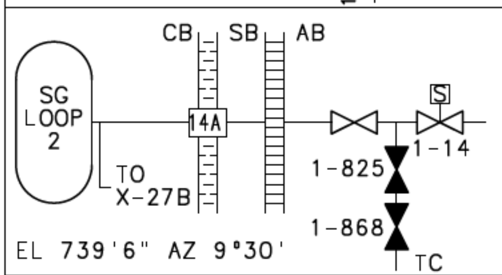
WATTS BAR NUCLEAR PLANT UNIT 1 CONTAINMENT PENETRATIONS AND BARRIERS-TABLE 6.2.4-1

PENETRATION DATA							VALVE DATA																			
DETAIL	DWG NUMBER	GEN DES CRITERION	PROCESS FLUID	FLUID STATE	POSS LEAK PATHS	SYSTEM NUMBER AND PENETRATION DESCRIPTION	VALVE LOCATION	VALVE NUMBER	ESF POWER TRAIN	VALVE TYPE	ACTUATOR	PRI ACT MODE	SEC ACT MODE	ISOLATION SIGNAL	STROKE TIME	VALVE STATUS NORMAL	VALVE STATUS SHUTDOWN	VALVE STATUS POST-ACCIDENT	VALVE STATUS POWER FAILURE	ILRT	POS IND IN MCR	ESF	APP J TEST	ESSENTIAL/NON-ESS	SHIELD BLDG PENETRATION	NOTES
 EL 731 AZ 7°40'	47W803-1	57	W	H	BCE	FEEDWATER (3)	AB	33	A	GA	MO	AT	LM	FW	-	O	C	C	AI	C	Y	N	A	N	MK70	22
 EL 731 AZ 172°20'	47W803-1	57	W	H	BCE	FEEDWATER (3)	AB	47	B	GA	MO	AT	LM	FW	-	O	C	C	AI	C	Y	N	A	N	MK69	22
 EL 731 AZ 187°40'	47W803-1	57	W	H	BCE	FEEDWATER (3)	AB	87	A	GA	MO	AT	LM	FW	-	O	C	C	AI	C	Y	N	A	N	MK68	22

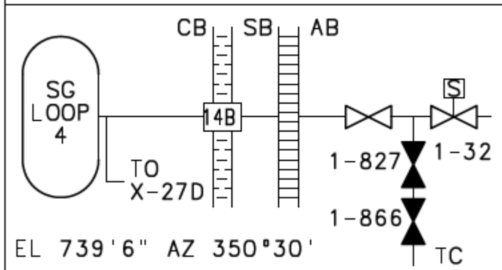
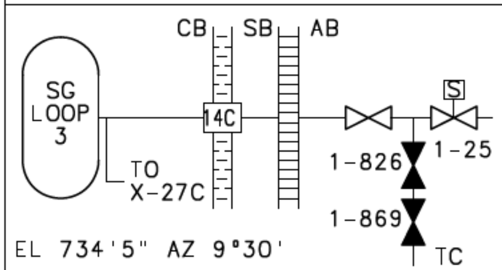
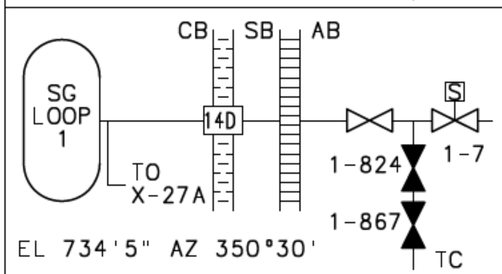
WATTS BAR NUCLEAR PLANT UNIT 1 CONTAINMENT PENETRATIONS AND BARRIERS-TABLE 6.2.4-1

PENETRATION DATA							VALVE DATA																		
DETAIL	DWG NUMBER	GEN DES CRITERION	PROCESS FLUID	FLUID STATE	POSS LEAK PATHS	SYSTEM NUMBER AND PENETRATION DESCRIPTION	VALVE LOCATION	VALVE NUMBER	ESF POWER TRAIN	VALVE TYPE	ACTUATOR	PRI ACT MODE	SEC ACT MODE	ISOLATION SIGNAL	STROKE TIME	VALVE STATUS: NORMAL	VALVE STATUS: SHUTDOWN	VALVE STATUS: POST-ACCIDENT	VALVE STATUS: POWER FAILURE	ILRT	POS IND IN MCR	ESF	APP J TEST	ESSENTIAL/NON-ESS	SHIELD BLDG PENETRATION	NOTES
 EL 731 AZ 352°20'	47W803-1	57	W	H	BCE	FEEDWATER (3)	AB	100	B	GA	MO	AT	LM	FW	-	O	C	C	AI	C	Y	N	A	N	MK67	22
EL 737 AZ 4°31' 	47W801-1 47W803-2	57	S	H	BCE	MAIN STEAM (1)	AB	4	A,B	GL	AO	AT	LM	MS	-	O	C	C	C	C	Y	N	A	N	MK66	22
							AB	5	A,B	RV	AO	AT	RM	-	-	C	C	C	C	C	Y	N	A			
							AB	15	A	GA	MO	AT	RM	-	-	O	C	V	AT	C	Y	N	A			
							AB	147	A	GA	AO	AT	RM	MS	-	C	V	C	C	C	Y	N	A			
							AB	522	-	RV	-	SA	-	-	-	C	C	C	-	C	N	N	A			
							AB	523	-	RV	-	SA	-	-	-	C	C	C	-	C	N	N	A			
							AB	524	-	RV	-	SA	-	-	-	C	C	C	-	C	N	N	A			
							AB	525	-	RV	-	SA	-	-	-	C	C	C	-	C	N	N	A			
							AB	526	-	RV	-	SA	-	-	-	C	C	C	-	C	N	N	A			
							AB	536	-	GL	M	LM	-	-		C	V	C	-	C	N	N	A			
EL 737 AZ 175°29' 	47W801-1	57	S	H	BCE	MAIN STEAM (1)	AB	11	A,B	GL	AO	AT	LM	MS	-	O	C	C	C	C	Y	N	A	N	MK65	22
							AB	12	A,B	RV	AO	AT	RM	-	-	C	C	C	C	C	Y	N	A			
							AB	148	B	GA	AO	AT	RM	MS	-	C	V	C	C	C	Y	N	A			
							AB	517	-	RV	-	SA	-	-	-	C	C	C	-	C	N	N	A			
							AB	518	-	RV	-	SA	-	-	-	C	C	C	-	C	N	N	A			
							AB	519	-	RV	-	SA	-	-	-	C	C	C	-	C	N	N	A			
							AB	520	-	RV	-	SA	-	-	-	C	C	C	-	C	N	N	A			
							AB	521	-	RV	-	SA	-	-	-	C	C	C	-	C	N	N	A			
							AB	534	-	GL	M	LM	-	-		C	V	C	-	C	N	N	A			

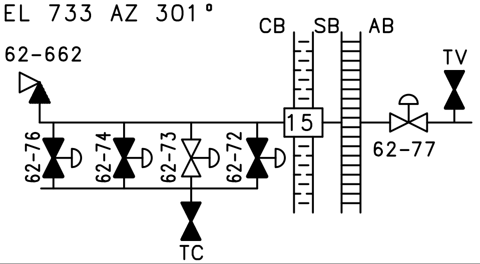
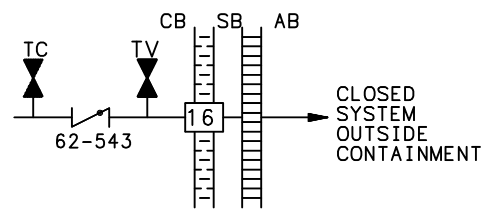
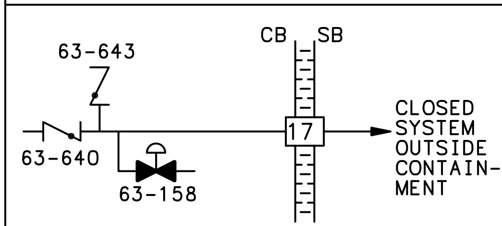
WATTS BAR NUCLEAR PLANT UNIT 1 CONTAINMENT PENETRATIONS AND BARRIERS-TABLE 6.2.4-1

PENETRATION DATA							VALVE DATA																			
																VALVE STATUS										
DETAIL	DWG NUMBER	GEN DES CRITERION	PROCESS FLUID	FLUID STATE	POSS LEAK PATHS	SYSTEM NUMBER AND PENETRATION DESCRIPTION	VALVE LOCATION	VALVE NUMBER	ESF POWER TRAIN	VALVE TYPE	ACTUATOR	PRI ACT MODE	SEC ACT MODE	ISOLATION SIGNAL	STROKE TIME	NORMAL	SHUTDOWN	POST-ACCIDENT	POWER FAILURE	ILRT	POS IND IN MCR	ESF	APP J TEST	ESSENTIAL/NON-ESS	SHIELD BLDG PENETRATION	NOTES
	47W801-1	57	S	H	BC E	MAIN STEAM (1)	AB	22	A,B	GL	AO	AT	LM	MS	-	O	C	C	C	C	Y	N	A	N	MK64	22
							AB	23	A,B	RV	AO	AT	RM	-	-	C	C	C	C	C	Y	N	A			
							AB	149	A	GA	AO	AT	RM	MS	-	C	V	C	C	C	Y	N	A			
							AB	512	-	RV	-	SA	-	-	-	C	C	C	-	C	N	N	A			
							AB	513	-	RV	-	SA	-	-	-	C	C	C	-	C	N	N	A			
							AB	514	-	RV	-	SA	-	-	-	C	C	C	-	C	N	N	A			
							AB	515	-	RV	-	SA	-	-	-	C	C	C	-	C	N	N	A			
							AB	516	-	RV	-	SA	-	-	-	C	C	C	-	C	N	N	A			
							AB	532	-	GL	M	LM	-	-	-	C	V	C	-	C	N	N	A			
	47W801-1 47W803-2	57	S	H	BC E	MAIN STEAM (1)	AB	29	A,B	GL	AO	AT	LM	MS	-	O	C	C	C	C	Y	N	A	N	MK63	22
								16	A	GA	MO	AT	RM	-	-	C	C	V	AI	C	Y	N	A			
								30	A,B	RV	AO	AT	RM	-	-	C	-	C	C	C	Y	N	A			
								150	B	GL	AO	AT	RM	MS	-	C	V	C	C	C	Y	N	A			
								527	-	RV	-	SA	-	-	-	C	C	C	-	C	N	N	A			
								528	-	RV	-	SA	-	-	-	C	C	C	-	C	N	N	A			
								529	-	RV	-	SA	-	-	-	C	C	C	-	C	N	N	A			
								530	-	RV	-	SA	-	-	-	C	C	C	-	C	N	N	A			
								531	-	RV	-	SA	-	-	-	C	C	C	-	C	N	N	A			
								538	-	GL	M	LM	-	-		C	V	C	-	C	N	N	A			
	47W801-2	57	W	H	BE	STEAM GENERATOR BLOWDOWN (1)	AB	14	A	GL	SO	AT	RM	PA	15.0	O	C	C	C	C	Y	N	A	N	AS14A	22

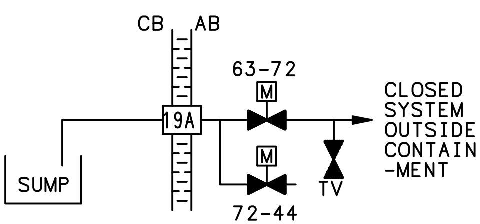
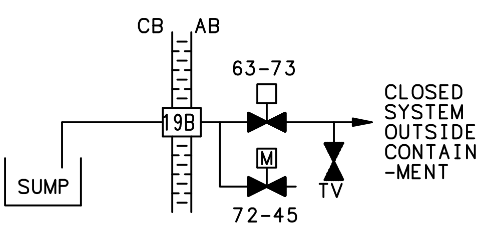
WATTS BAR NUCLEAR PLANT UNIT 1 CONTAINMENT PENETRATIONS AND BARRIERS-TABLE 6.2.4-1

PENETRATION DATA							VALVE DATA																		
															VALVE STATUS										
DETAIL	DWG NUMBER	GEN DES CRITERION	PROCESS FLUID	FLUID STATE	POSS LEAK PATHS	SYSTEM NUMBER AND PENETRATION DESCRIPTION	VALVE LOCATION	VALVE NUMBER	ESF POWER TRAIN	VALVE TYPE	ACTUATOR	PRI ACT MODE	SEC ACT MODE	ISOLATION SIGNAL	STROKE TIME	NORMAL	SHUTDOWN	POST-ACCIDENT	POWER FAILURE	ILRT	POS IND IN MCR	ESF	APP J TEST	ESSENTIAL/NON-ESS	SHIELD BLDG PENETRATION	NOTES
	47W801-2	57	W	H	BE	STEAM GENERATOR BLOWDOWN (1)	AB	32	A	GL	SO	AT	RM	PA	15.0	O	C	C	C	C	Y	N	A	N	AS14B	22
	47W801-2	57	W	H	BE	STEAM GENERATOR BLOWDOWN (1)	AB	25	B	GL	SO	AT	RM	PA	15.0	O	C	C	C	C	Y	N	A	N	AS14C	22
	47W801-2	57	W	H	BE	STEAM GENERATOR BLOWDOWN (1)	AB	7	B	GL	SO	AT	RM	PA	15.0	O	C	C	C	C	Y	N	A	N	AS14D	22

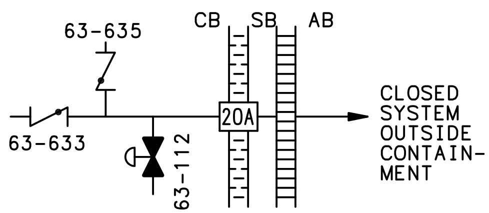
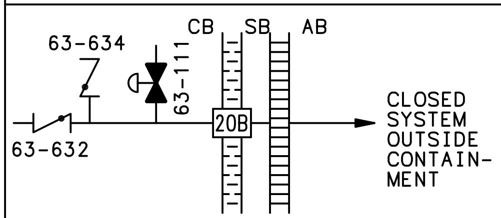
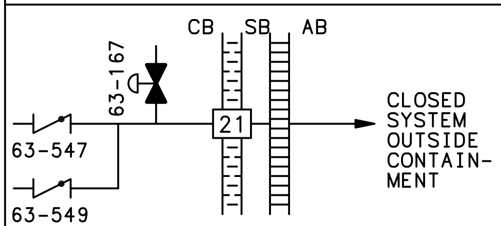
WATTS BAR NUCLEAR PLANT UNIT 1 CONTAINMENT PENETRATIONS AND BARRIERS-TABLE 6.2.4-1

PENETRATION DATA							VALVE DATA																			
DETAIL	DWG NUMBER	GEN DES CRITERION	PROCESS FLUID	FLUID STATE	POSS LEAK PATHS	SYSTEM NUMBER AND PENETRATION DESCRIPTION	VALVE LOCATION	VALVE NUMBER	ESF POWER TRAIN	VALVE TYPE	ACTUATOR	PRI ACT MODE	SEC ACT MODE	ISOLATION SIGNAL	STROKE TIME	VALVE STATUS NORMAL	VALVE STATUS SHUTDOWN	VALVE STATUS POST-ACCIDENT	POWER FAILURE	ILRT	POS IND IN MCR	ESF	APP J TEST	ESSENTIAL/NON-ESS	SHIELD BLDG PENETRATION	NOTES
EL 733 AZ 301° 	47W809-1	55	W	H	BD	CVCS LETDOWN (62)	CB	72	A	GL	AO	AT	RM	PA	10.0	C	V	C	C	C	Y	N	AC	N	AS15	4
							CB	73	A	GL	AO	AT	RM	PA	10.0	O	V	C	C	C	Y	N	AC			
							CB	74	A	GL	AO	AT	RM	PA	10.0	C	V	C	C	C	N	N	AC			
							CB	662	-	RV	-	SA	-	-	-	C	C	C	C	C	Y	N	AC			
							CB	76	A	GL	AO	AT	RM	PA	10.0	C	V	C	C	C	Y	N	AC			
							AB	77	B	GL	AO	AT	RM	PA	10.0	O	V	C	C	C	Y	N	AC			
 EL 730'6" AZ 299°30'	47W809-1	55	W	C	AD	CVCS NORMAL CHARGING (62)	CB	543	-	CK	-	SA	RM	-	-	O	V	C	-	-	N	N	A	N	MK19	
 EL 732 AZ 281°30'	47W811-1 47W810-1	55	W	C	BD	RHR HOT LEG INJECTION (63)	CB	640	-	CK	-	SA	-	-	-	C	V	C	-	-	N	Y	A	E	MK27	
							CB	643	-	CK	-	SA	-	-	-	C	V	C	-	-	N	Y	A			
							CB	158	-	GL	AO	RM	-	RM	-	V	V	C	C	C	Y	Y	A			

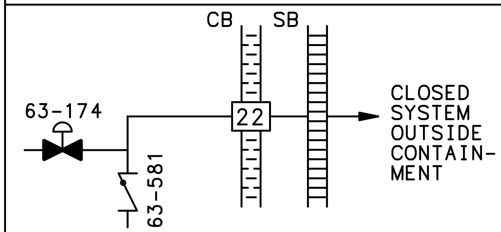
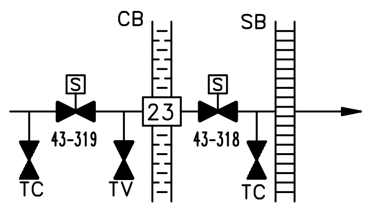
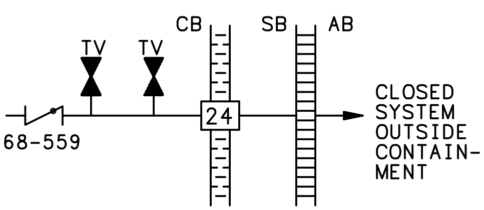
WATTS BAR NUCLEAR PLANT UNIT 1 CONTAINMENT PENETRATIONS AND BARRIERS-TABLE 6.2.4-1

PENETRATION DATA							VALVE DATA																			
DETAIL	DWG NUMBER	GEN DES CRITERION	PROCESS FLUID	FLUID STATE	POSS LEAK PATHS	SYSTEM NUMBER AND PENETRATION DESCRIPTION	VALVE LOCATION	VALVE NUMBER	ESF POWER TRAIN	VALVE TYPE	ACTUATOR	PRI ACT MODE	SEC ACT MODE	ISOLATION SIGNAL	STROKE TIME	VALVE STATUS NORMAL	SHUTDOWN	POST-ACCIDENT	POWER FAILURE	ILRT	POS IND IN MCR	ESF	APP J TEST	ESSENTIAL/NON-ESS	SHIELD BLDG PENETRATION	NOTES
X-18 EL 740 AZ 145	48W406	-	-	-	-	SPARE	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	A	-	-	
EL 689 AZ 318°  	47W811-1 47W812-1	-	W	H	D	SUMP SUCTION TO RHR PUMP 1A-A (63,72)	AB AB	72 44	A A	GA GA	MO MO	RM RM	LM LM	- -	- -	C C	C C	V V	AI AI	C C	Y Y	Y Y	A A	E	-	
EL 689 AZ 325°  	47W811-1 47W812-1	-	W	H	D	SUMP SUCTION TO RHR PUMP 1B-B (63,72)	AB AB	73 45	B B	GA GA	MO MO	RM RM	LM LM	- -	- -	C C	C C	V V	AI AI	C C	Y Y	Y Y	A A	E	-	

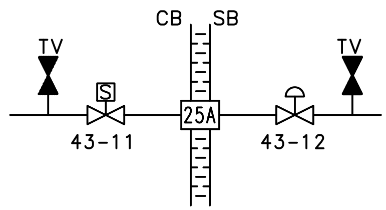
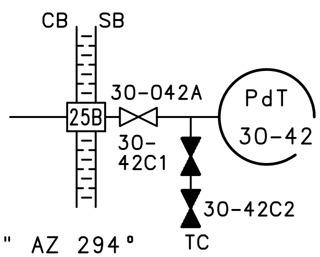
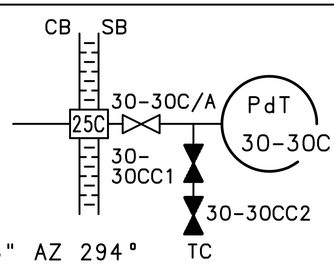
WATTS BAR NUCLEAR PLANT UNIT 1 CONTAINMENT PENETRATIONS AND BARRIERS-TABLE 6.2.4-1

PENETRATION DATA							VALVE DATA																			
DETAIL	DWG NUMBER	GEN DES CRITERION	PROCESS FLUID	FLUID STATE	POSS LEAK PATHS	SYSTEM NUMBER AND PENETRATION DESCRIPTION	VALVE LOCATION	VALVE NUMBER	ESF POWER TRAIN	VALVE TYPE	ACTUATOR	PRI ACT MODE	SEC ACT MODE	ISOLATION SIGNAL	STROKE TIME	VALVE STATUS: NORMAL	VALVE STATUS: SHUTDOWN	VALVE STATUS: POST-ACCIDENT	POWER FAILURE	ILRT	POS IND IN MCR	ESF	APP J TEST	ESSENTIAL/NON-ESS	SHIELD BLDG PENETRATION	NOTES
 EL 732 AZ 291°-30"	47W811-1	55	W	C	B D	RHR PUMP DISCHARGE TR B (63)	CB CB CB	112 635 633	- - -	GL CK CK	AO - -	RM SA SA	- - -	- - -	- - -	V C C	V C C	C V V	C - -	C - -	Y N N	N Y Y	A A A	E	AS20A	
 EL 732 AZ 287°30'	47W811-1	55	W	C	BD	RHR PUMP DISCHARGE TR A (63)	CB CB CB	111 632 634	- - -	GL CK CK	AO - -	RM SA SA	- - -	- - -	- - -	V C C	V C C	C V V	C - -	C - -	Y N N	N Y Y	A A A	E	AS20B	
 EL 728'6" AZ 289°30'	47W811-1	55	W	C	BD	SIS PUMP DISCHARGE TO HOT LEGS TR B (63)	CB CB CB	167 547 549	- - -	GL CK CK	AO - -	RM SA SA	- - -	- - -	- - -	C - -	C - -	V - -	C - -	C - -	Y N N	N Y Y	A A A	E	AS21	

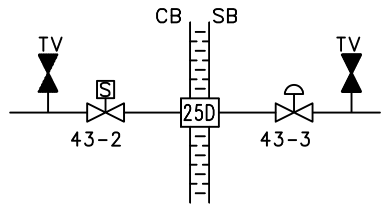
WATTS BAR NUCLEAR PLANT UNIT 1 CONTAINMENT PENETRATIONS AND BARRIERS-TABLE 6.2.4-1

PENETRATION DATA							VALVE DATA																			
																VALVE STATUS										
DETAIL	DWG NUMBER	GEN DES CRITERION	PROCESS FLUID	FLUID STATE	POSS LEAK PATHS	SYSTEM NUMBER AND PENETRATION DESCRIPTION	VALVE LOCATION	VALVE NUMBER	ESF POWER TRAIN	VALVE TYPE	ACTUATOR	PRI ACT MODE	SEC ACT MODE	ISOLATION SIGNAL	STROKE TIME	NORMAL	SHUTDOWN	POST-ACCIDENT	POWER FAILURE	ILRT	POS IND IN MCR	ESF	APP J TEST	ESSENTIAL/NON-ESS	SHIELD BLDG PENETRATION	NOTES
 EL 730'3" AZ 294°45'	47W811-1	55	W	C	BD	SIS CHARGING PUMP DIS-CHARGE (63)	CB CB	174 581	- -	GL CK	AO -	RM SA	- -	- -	- -	V -	V -	C -	C -	C -	Y N	N Y	A A	E	AS22	
 EL 729' AZ 283°	47W625-15	56	A	C	AB D	PAS CONT. AIR INTAKE TR.B (43)	CB SB	319 318	B B	GL GL	SO SO	RM RM	LM LM	- -	- -	C C	C C	V V	C C	C C	Y Y	N N	AC AC	E	MK61	19
 EL 727'6" AZ 301°	47W813-1 47W811-1 47W812-1	56	W	H	BD	RELIEF VALVE DISCHARGE (68)	CB	559	-	CK	-	SA	-	-	-	C	C	C	-	-	N	Y	-	E	AS24	25

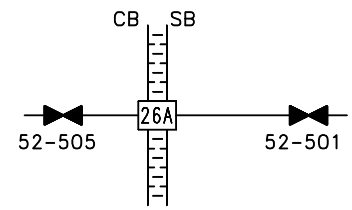
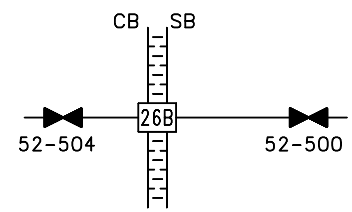
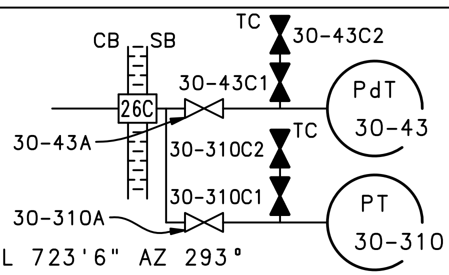
WATTS BAR NUCLEAR PLANT UNIT 1 CONTAINMENT PENETRATIONS AND BARRIERS-TABLE 6.2.4-1

PENETRATION DATA							VALVE DATA																		
DETAIL	DWG NUMBER	GEN DES CRITERION	PROCESS FLUID	FLUID STATE	POSS LEAK PATHS	SYSTEM NUMBER AND PENETRATION DESCRIPTION	VALVE LOCATION	VALVE NUMBER	ESF POWER TRAIN	VALVE TYPE	ACTUATOR	PRI ACT MODE	SEC ACT MODE	ISOLATION SIGNAL	STROKE TIME	VALVE STATUS NORMAL	VALVE STATUS SHUTDOWN	VALVE STATUS POST-ACCIDENT	VALVE STATUS POWER FAILURE	ILRT	POS IND IN MCR	ESF	APP J TEST	ESSENTIAL/NON-ESS	SHIELD BLDG PENETRATION	NOTES
 EL 723'6" AZ 294°	47W625-1	55	W	H	AD	PRESSURIZER LIQUID SAMPLE (43)	CB SB	11 12	B A	GL GL	SO AO	AT AT	RM RM	PA PA	5.0 5.0	V V	V V	C C	C C	C C	Y Y	N N	AC AC	N	MK44H	15
 EL 723'6" AZ 294°	47W600-89	–	A	C	B	dP SENSOR (30)	SB SB SB SB	SNSR. 42C1 42C2 42A	– – – –	– GL GL GL	– M M M	– LM LM LM	– – – –	– – – –	– – – –	O C C O	O V V V	O C C O	– – – –	– C C O	– N N N	N N N N	A A – A	E	–	12
 EL 723'6" AZ 294°	47W600-89	–	A	C	B	dP SENSOR (30)	SB SB SB SB	SNSR. 30CC1 30CC2 30C/A	– – – –	– GL GL GL	– M M M	– LM LM LM	– – – –	– – – –	– – – –	O C C O	O V V V	O C C O	– – – –	– C C O	– N N N	N N N N	A A – A	N	–	12

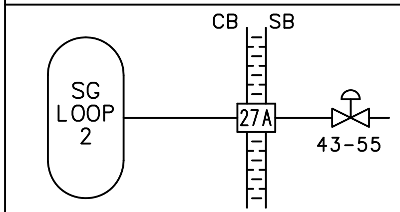
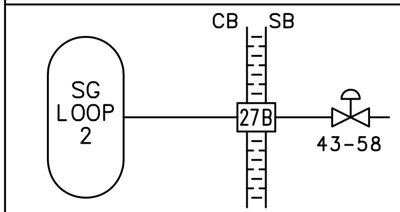
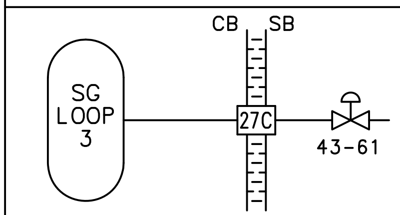
WATTS BAR NUCLEAR PLANT UNIT 1 CONTAINMENT PENETRATIONS AND BARRIERS-TABLE 6.2.4-1

PENETRATION DATA							VALVE DATA																				
DETAIL	DWG NUMBER	GEN DES CRITERION	PROCESS FLUID	FLUID STATE	POSS LEAK PATHS	SYSTEM NUMBER AND PENETRATION DESCRIPTION	VALVE LOCATION	VALVE NUMBER	ESF POWER TRAIN	VALVE TYPE	ACTUATOR	PRI ACT MODE	SEC ACT MODE	ISOLATION SIGNAL	STROKE TIME	VALVE STATUS NORMAL	VALVE STATUS SHUTDOWN	VALVE STATUS POST-ACCIDENT	VALVE STATUS POWER FAILURE	ILRT	POS IND IN MCR	ESF	APP J TEST	ESSENTIAL/NON-ESS	SHIELD BLDG PENETRATION	NOTES
 EL 723'6" AZ 294°	47W625-1	55	S	H	AD	PRESSURIZER STEAM SAMPLE (43)	CB SB	2 3	B A	GL GL	SO AO	AT AT	RM RM	PA PA	5.0 5.0	V V	V V	C C	C C	C C	Y Y	N N	AC AC	N	MK44D	15

WATTS BAR NUCLEAR PLANT UNIT 1 CONTAINMENT PENETRATIONS AND BARRIERS-TABLE 6.2.4-1

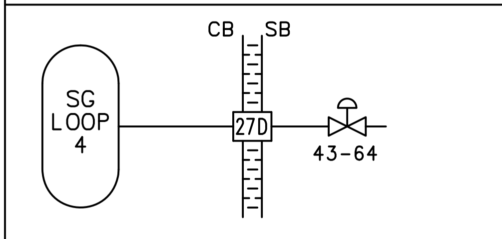
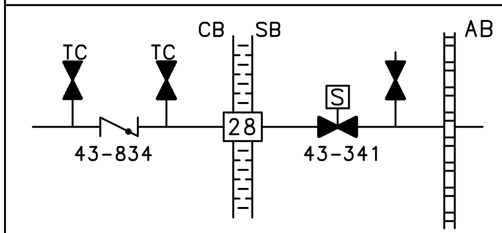
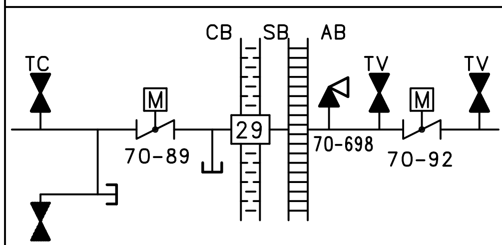
PENETRATION DATA							VALVE DATA										VALVE STATUS									
DETAIL	DWG NUMBER	GEN DES CRITERION	PROCESS FLUID	FLUID STATE	POSS LEAK PATHS	SYSTEM NUMBER AND PENETRATION DESCRIPTION	VALVE LOCATION	VALVE NUMBER	ESF POWER TRAIN	VALVE TYPE	ACTUATOR	PRI ACT MODE	SEC ACT MODE	ISOLATION SIGNAL	STROKE TIME	NORMAL	SHUTDOWN	POST-ACCIDENT	POWER FAILURE	ILRT	POS IND IN MCR	ESF	APP J TEST	ESSENTIAL/NON-ESS	SHIELD BLDG PENETRATION	NOTES
 EL 723'6" AZ 293°	47W331-3	56	A	C	B	ILRT SENSOR LINE (52) TEST INSTRUMENTATION	CB SB	505 501	- -	GL GL	M M	LM LM	- -	- -	- -	C C	V V	C C	- -	O O	N N	N N	AC AC	N	-	
 EL 723'6" AZ 293°	47W331-3	56	A	C	B	ILRT SENSOR LINE (52) TEST INSTRUMENTATION	CB SB	504 500	- -	GL GL	M M	LM LM	- -	- -	- -	C C	V V	C C	- -	CC	N N	N N	AC AC	N	-	
 EL 723'6" AZ 293°	47W600-89 47W331-3	-	A	C	B	dP SENSOR (30)	SB SB SB SB SB SB SB	SNSRS 43C1 43C2 310C1 310C2 43A 310A	- - - - - - -	- GL GL GL GL GL GL	- M M M M M M	- LM LM LM LM LM LM	- - - - - - -	- - - - - - -	- - - - - - -	O C C C C O O	- V V V V V V	O C C C C O O	- - - - - - -	C C C C C O O	N N N N N N N	N N N N N N N	A A - A - A A	E	-	12,13

WATTS BAR NUCLEAR PLANT UNIT 1 CONTAINMENT PENETRATIONS AND BARRIERS-TABLE 6.2.4-1

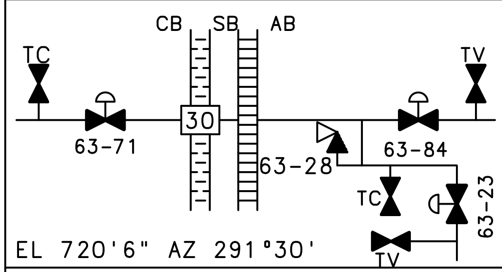
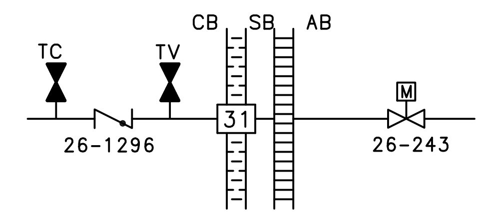
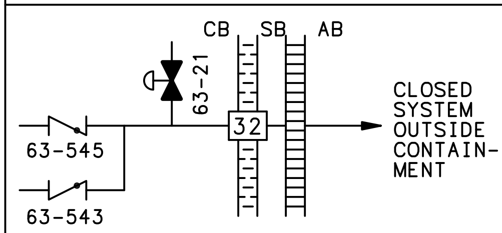
PENETRATION DATA							VALVE DATA																			
DETAIL	DWG NUMBER	GEN DES CRITERION	PROCESS FLUID	FLUID STATE	POSS LEAK PATHS	SYSTEM NUMBER AND PENETRATION DESCRIPTION	VALVE LOCATION	VALVE NUMBER	ESF POWER TRAIN	VALVE TYPE	ACTUATOR	PRI ACT MODE	SEC ACT MODE	ISOLATION SIGNAL	STROKE TIME	VALVE STATUS NORMAL	SHUTDOWN	POST-ACCIDENT	POWER FAILURE	ILRT	POS IND IN MCR	ESF	APP J TEST	ESSENTIAL/NON-ESS	SHIELD BLDG PENETRATION	NOTES
 EL 723'6" AZ 292°	47W625-2	56	W	H	A B D	STM GEN NO. 1 SAMPLE (43) SAMPLE AND WATER QUALITY	SB	55	A	GL	AO	AT	RM	PA	10.0	V	V	C	C	C	Y	N	A	N	MK046M	22
 EL 723'6" AZ 292°	47W625-2	56	W	H	AB D	STM GEN NO. 2 SAMPLE (43)	SB	58	A	GL	AO	AT	RM	PA	10.0	V	V	C	C	C	Y	N	A	N	MK46M	22
 EL 723'6" AZ 292°	47W625-2	56	W	H	AB D	STM GEN NO. 3 SAMPLE (43)	SB	61	A	GL	AO	AT	RM	PA	10.0	V	V	C	C	C	Y	N	A	N	MK46M	22

WATTS BAR NUCLEAR PLANT UNIT 1 CONTAINMENT PENETRATIONS AND BARRIERS-TABLE 6.2.4-1

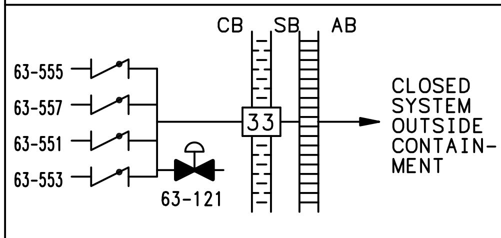
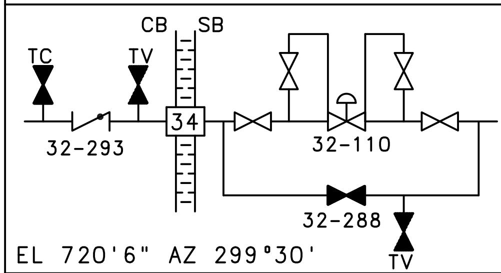
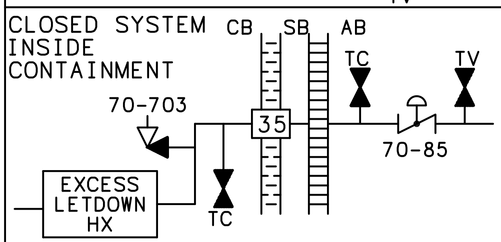
AMENDMENT 1

PENETRATION DATA							VALVE DATA																		
DETAIL	DWG NUMBER	GEN DES CRITERION	PROCESS FLUID	FLUID STATE	POSS LEAK PATHS	SYSTEM NUMBER AND PENETRATION DESCRIPTION	VALVE LOCATION	VALVE NUMBER	ESF POWER TRAIN	VALVE TYPE	ACTUATOR	PRI ACT MODE	SEC ACT MODE	ISOLATION SIGNAL	STROKE TIME	VALVE STATUS: NORMAL	VALVE STATUS: SHUTDOWN	VALVE STATUS: POST-ACCIDENT	VALVE STATUS: POWER FAILURE	ILRT	POS IND IN MCR	ESF	APP J TEST	ESSENTIAL/NON-ESS	SHIELD BLDG PENETRATION	NOTES
 EL 723'6" AZ 292°	47W625-2	56	W	H	AB D	STM GEN NO. 4 SAMPLE (43)	SB	64	A	GL	AO	AT	RM	PA	10.0	V	V	C	C	C	Y	N	A	N	MK46M	22
 EL 721 AZ 287°30'	47W625-15	56	W	C	AB D	PAS CONT SUMP RTRN (43)	CB SB	834 341	– B	CK GL	– SO	SA RM	– LM	– –	– –	C C	C C	V V	C C	– C	N Y	N N	AC AC	E	MK54	19
 EL 720 AZ 288°30'	47W859-2 -3	56	W	C	AD	CCS FROM RC PUMP COOLERS (70)	CB AB AB	89 92 698	B A –	BF BF RV	MO MO –	AT AT SA	– – –	PB PB –	66.0 66.0 –	O O C	O O C	C C V	AI AI –	C C –	Y Y N	N N N	AC AC AC	N	MK50	

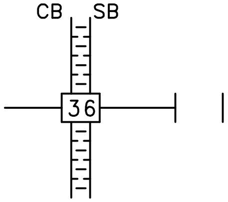
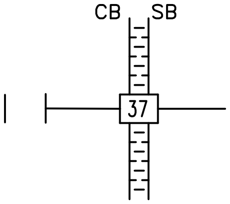
WATTS BAR NUCLEAR PLANT UNIT 1 CONTAINMENT PENETRATIONS AND BARRIERS-TABLE 6.2.4-1

PENETRATION DATA							VALVE DATA																			
DETAIL	DWG NUMBER	GEN DES CRITERION	PROCESS FLUID	FLUID STATE	POSS LEAK PATHS	SYSTEM NUMBER AND PENETRATION DESCRIPTION	VALVE LOCATION	VALVE NUMBER	ESF POWER TRAIN	VALVE TYPE	ACTUATOR	PRI ACT MODE	SEC ACT MODE	ISOLATION SIGNAL	STROKE TIME	VALVE STATUS NORMAL	VALVE STATUS SHUTDOWN	VALVE STATUS POST-ACCIDENT	VALVE STATUS POWER FAILURE	ILRT	POS IND IN MCR	ESF	APP J TEST	ESSENTIAL/NON-ESS	SHIELD BLDG PENETRATION	NOTES
 EL 720'6" AZ 291°30'	47W811-1	56	W	C	BD	ACCUM TO HOLDUP TANKS (63)	CB AB AB AB	71 84 23 28	A B B -	GL GL GL RV	AO AO AO -	AT AT AT SA	RM - - -	PA PA PA -	10.0 10.0 10.0 -	V V V C	V V V C	C C C V	C C C -	C C C C	Y Y Y N	N N N N	AC AC AC AC	N	AS30	9
 EL 722 AZ 289°30'	47W850-9	56	A	C	AB D	FIRE PROTECTION (26)	CB AB	1296 243	- A	CK GA	- MO	SA AT	- RM	- PA	- 20.0	C O	C O	C C	- C	C C	N Y	N N	AC AC	N	MK49	
 EL 720'6" AZ 282°30'	47W811-1	55	W	C	BD	SI TO HOT LEGS (63)	CB CB CB	545 543 21	- - -	CK CK GL	- - AO	SA SA RM	- - -	- - -	- - -	C C V	C C V	V V C	- - C	- - C	N N Y	Y Y N	A A A	E	AS32	

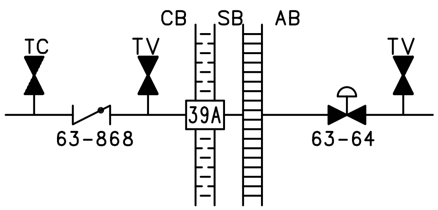
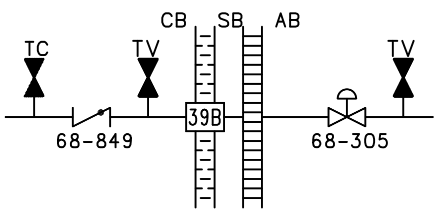
WATTS BAR NUCLEAR PLANT UNIT 1 CONTAINMENT PENETRATIONS AND BARRIERS-TABLE 6.2.4-1

PENETRATION DATA							VALVE DATA																			
DETAIL	DWG NUMBER	GEN DES CRITERION	PROCESS FLUID	FLUID STATE	POSS LEAK PATHS	SYSTEM NUMBER AND PENETRATION DESCRIPTION	VALVE LOCATION	VALVE NUMBER	ESF POWER TRAIN	VALVE TYPE	ACTUATOR	PRI ACT MODE	SEC ACT MODE	ISOLATION SIGNAL	STROKE TIME	VALVE STATUS NORMAL	VALVE STATUS SHUTDOWN	VALVE STATUS POST-ACCIDENT	VALVE STATUS POWER FAILURE	ILRT	POS IND IN MCR	ESF	APP J TEST	ESSENTIAL/NON-ESS	SHIELD BLDG PENETRATION	NOTES
 EL 720'6" AZ 277°30'	47W811-1	55	W	C	BD	SI TO LOW HEAD SI (63)	CB CB CB CB CB	553 551 557 555 121	- - - - -	CK CK CK CK GL	- - - - AO	SA SA SA SA RM	- - - - -	- - - - -	- - - - -	C C C C V	C C C C V	V V V V C	- - - - C	- - - - C	N N N N Y	Y Y Y Y N	A A A A A	E	AS33	
 EL 720'6" AZ 299°30'	47W848-1	56	A	C	AB D	CONTROL AIR I & C (32)	CB SB SB	293 110 288	- A -	CK GL GL	- AO M	SA AT LM	- RM -	- PB -	- 10.0 -	O O C	O O C	C C C	- C AI	- C C	N Y N	N N N	AC AC AC	N	MK43	10
 EL 720'6" AZ 301°30'	47W859-2, 3	57	W	C	AD	CCS FROM EXCESS LETDN HX (70)	CB AB	703 85	 B	RV BF	- AO	SA AT	- RM	- PA	- 10.0	V V	V C	V C	C C	C C	N Y	N N	AC AC	N	MK42	11

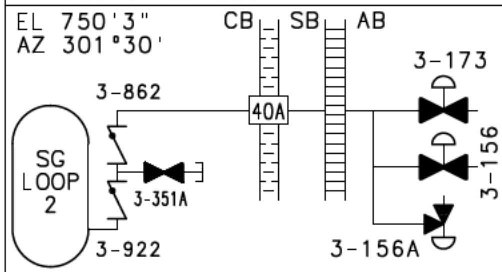
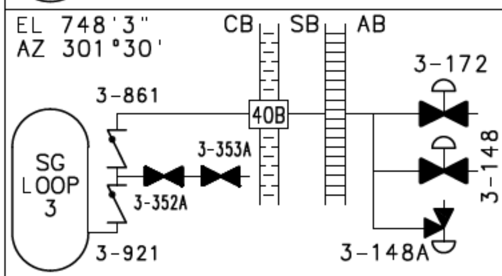
WATTS BAR NUCLEAR PLANT UNIT 1 CONTAINMENT PENETRATIONS AND BARRIERS-TABLE 6.2.4-1

PENETRATION DATA							VALVE DATA																		
																VALVE STATUS									
DETAIL	DWG NUMBER	GEN DES CRITERION	PROCESS FLUID	FLUID STATE	POSS LEAK PATHS	SYSTEM NUMBER AND PENETRATION DESCRIPTION	VALVE LOCATION	VALVE NUMBER	ESF POWER TRAIN	VALVE TYPE	ACTUATOR	PRI ACT MODE	SEC ACT MODE	ISOLATION SIGNAL	STROKE TIME	NORMAL	SHUTDOWN	POST-ACCIDENT	POWER FAILURE	ILRT	POS IND IN MCR	ESF	APP J TEST	ESSENTIAL/NON-ESS	SHIELD BLDG PENETRATION	NOTES
 EL 742'6" AZ 280°	72-4333-310	–	A	C	AB	SG. CHEM. CLEANING	SB	–	–	BL	M	LM	–	–	–	C	V	C	–	C	N	N	AB	N	MK20	
 EL 771'6" AZ 265°	47W301-1	–	A	C	AB	MAINT. PORT	CB	–	–	BL	M	LM	–	–	–	C	V	C	–	C	N	N	AB	N	MK14	
X-38 EL 771'6" AZ 268°	48N406	–	–	–	–	SPARE	–	–	–	–	–	–	–	–	–	–	–	–	–	–	–	–	A	–	–	

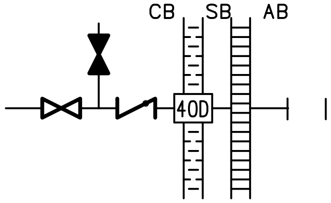
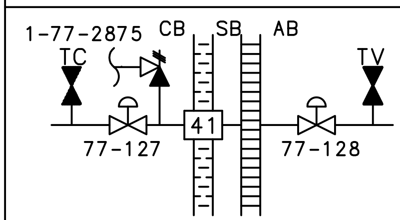
WATTS BAR NUCLEAR PLANT UNIT 1 CONTAINMENT PENETRATIONS AND BARRIERS-TABLE 6.2.4-1

PENETRATION DATA							VALVE DATA																			
																VALVE STATUS										
DETAIL	DWG NUMBER	GEN DES CRITERION	PROCESS FLUID	FLUID STATE	POSS LEAK PATHS	SYSTEM NUMBER AND PENETRATION DESCRIPTION	VALVE LOCATION	VALVE NUMBER	ESF POWER TRAIN	VALVE TYPE	ACTUATOR	PRI ACT MODE	SEC ACT MODE	ISOLATION SIGNAL	STROKE TIME	NORMAL	SHUTDOWN	POST-ACCIDENT	POWER FAILURE	ILRT	POS IND IN MCR	ESF	APP J TEST	ESSENTIAL/NON-ESS	SHIELD BLDG PENETRATION	NOTES
 EL 720'6" AZ 280°	47W830-6	56	N2	C	AD	N2 TO ACCUM (63)	CB AB	868 64	– A	CK GL	– AO	SA AT	– RM	– PA	– 10.0	O V	O O	C C	– C	– C	N Y	N N	AC AC	N	MK55M	
 EL 720'6" AZ 280°	47W830-6	56	N2	C	AD	N2 TO PRESS RELIEF TK (68)	CB AB	849 305	– A	CK DI	– AO	SA AT	– RM	– PA	– 10.0	O O	O O	C C	– C	– C	N Y	N N	AC AC	N	MK55M	
X-39C EL 720'6" AZ 280°	47W331-2	–	–	–	–	SPARE	–	–	–	–	–	–	–	–	–	–	–	–	–	–	–	–	A	–	–	

WATTS BAR NUCLEAR PLANT UNIT 1 CONTAINMENT PENETRATIONS AND BARRIERS-TABLE 6.2.4-1

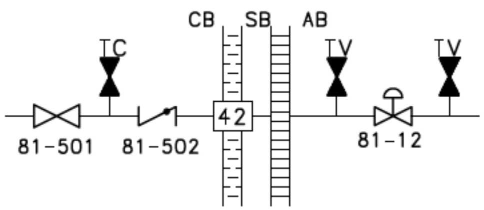
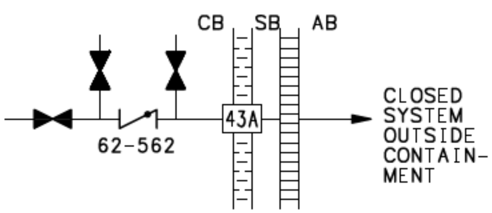
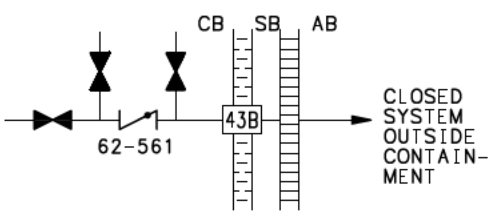
PENETRATION DATA							VALVE DATA																			
DETAIL	DWG NUMBER	GEN DES CRITERION	PROCESS FLUID	FLUID STATE	POSS LEAK PATHS	SYSTEM NUMBER AND PENETRATION DESCRIPTION	VALVE LOCATION	VALVE NUMBER	ESF POWER TRAIN	VALVE TYPE	ACTUATOR	PRI ACT MODE	SEC ACT MODE	ISOLATION SIGNAL	STROKE TIME	VALVE STATUS NORMAL	VALVE STATUS SHUTDOWN	VALVE STATUS POST-ACCIDENT	VALVE STATUS POWER FAILURE	ILRT	POS IND IN MCR	ESF	APP J TEST	ESSENTIAL/NON-ESS	SHIELD BLDG PENETRATION	NOTES
X-39D EL 720'6" AZ 280°	47W331-2	–	–	–	–	SPARE	–	–	–	–	–	–	–	–	–	–	–	–	–	–	–	–	A	–	–	
EL 750'3" AZ 301°30' 	47W803-2	57	W	C	BC E	AUX FW (3)	AB AB AB	156 156A 173	A A B	GL A GL	AO AO AO	RM RM RM	LM LM LM	– – –	– – –	C C C	C C C	V V V	O C C	C C C	Y Y Y	N N N	A A A	E	MK9	22
EL 748'3" AZ 301°30' 	47W803-2	57	W	C	BC E	AUX FW (3)	AB AB AB	148 148A 172	B B A	GL A GL	AO AO AO	RM RM RM	LM LM LM	– – –	– – –	C C C	C C C	V V V	O C C	C C C	Y Y Y	N N N	A A A	E	MK11	22

WATTS BAR NUCLEAR PLANT UNIT 1 CONTAINMENT PENETRATIONS AND BARRIERS-TABLE 6.2.4-1

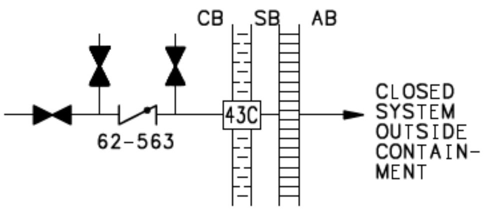
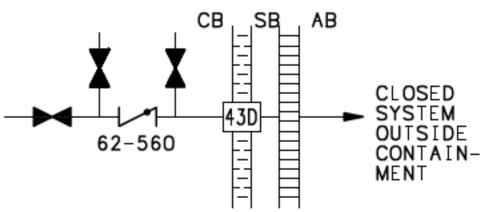
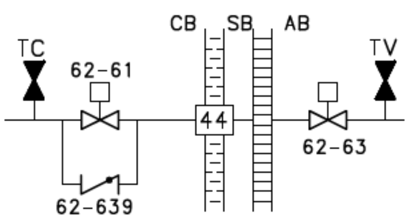
PENETRATION DATA							VALVE DATA																			
DETAIL	DWG NUMBER	GEN DES CRITERION	PROCESS FLUID	FLUID STATE	POSS LEAK PATHS	SYSTEM NUMBER AND PENETRATION DESCRIPTION	VALVE LOCATION	VALVE NUMBER	ESF POWER TRAIN	VALVE TYPE	ACTUATOR	PRI ACT MODE	SEC ACT MODE	ISOLATION SIGNAL	STROKE TIME	VALVE STATUS: NORMAL	VALVE STATUS: SHUTDOWN	VALVE STATUS: POST-ACCIDENT	VALVE STATUS: POWER FAILURE	ILRT	POS IND IN MCR	ESF	APP J TEST	ESSENTIAL/NON-ESS	SHIELD BLDG PENETRATION	NOTES
X-40C EL 750'3" AZ 299°30'	48N406	–	–	–	–	SPARE	–	–	–	–	–	–	–	–	–	–	–	–	–	–	–	–	A	–	–	
 EL 748'3" AZ 299°30'	47W846-2 47W492-2	–	A	C	AD	SERVICE AIR (33)	AB	–	–	BL	M	LM	–	–	–	C	C	C	–	C	–	N	AB	N	MK12	
 EL 719'6" AZ 294°	47W851-1	56	W	C	AD	FL SUMP PUMP DISCH (77)	CB AB CB	127 128 2875	B A –	BA BA RV	AO AO –	AT AT –	RM RM –	PA PA –	10.0 10.0 –	O O C	O O C	C C V	C C C	C C C	Y Y N	N N N	AC AC AC	N	MK47	

MIS259

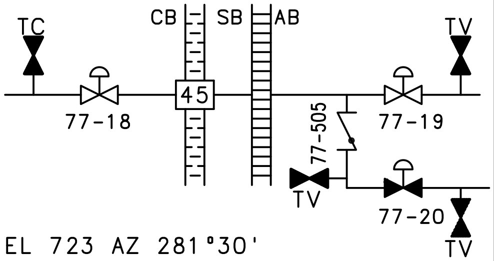
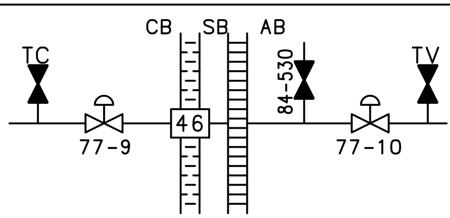
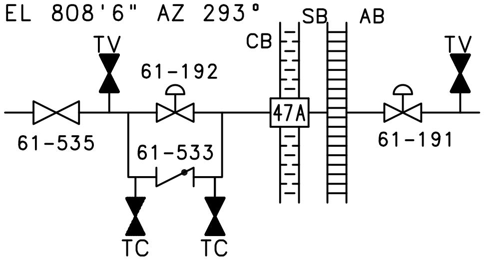
WATTS BAR NUCLEAR PLANT UNIT 1 CONTAINMENT PENETRATIONS AND BARRIERS-TABLE 6.2.4-1

DETAIL	DWG NUMBER	GEN DES CRITERION	PROCESS FLUID	FLUID STATE	POSS LEAK PATHS	SYSTEM NUMBER AND PENETRATION DESCRIPTION	VALVE LOCATION	VALVE NUMBER	ESF POWER TRAIN	VALVE TYPE	ACTUATOR	PRI ACT MODE	SEC ACT MODE	ISOLATION SIGNAL	STROKE TIME	VALVE STATUS: NORMAL	VALVE STATUS: SHUTDOWN	VALVE STATUS: POST-ACCIDENT	POWER FAILURE	ILRT	POS IND IN MCR	ESF	APP J TEST	ESSENTIAL/NON-ESS	SHIELD BLDG PENETRATION	NOTES
PENETRATION DATA							VALVE DATA																			
 EL 723'6" AZ 301°	47W819-1	56	W	C	ABD	PRESS RLF TK MAKE-UP (81)	CB AB	502 12	- A	CK DI	- AO	SA AT	- RM	- PA	- 10.0	V V	C C	C C	- C	- C	N Y	N N	AC AC	N	MK39	
 EL 728'6" AZ 293°30'	47W809-1	55	W	C	AD	TO RCP SEALS (62)	CB	562	-	CK	-	SA	-	-	-	O	C	C	-	-	N	N	A	N	MK31	
 EL 728'6" AZ 292°	47W809-1	55	W	C	AD	TO RCP SEALS (62)	CB	561	-	CK	-	SA	-	-	-	O	C	C	-	-	N	N	A	N	MK29	

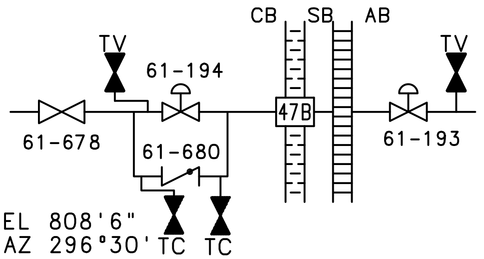
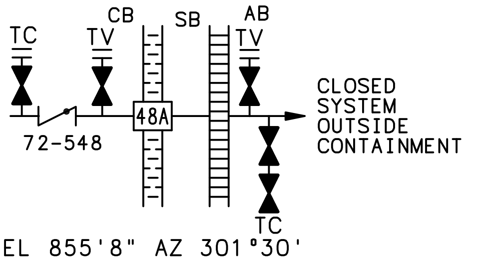
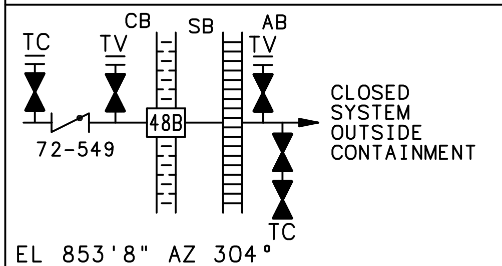
WATTS BAR NUCLEAR PLANT UNIT 1 CONTAINMENT PENETRATIONS AND BARRIERS-TABLE 6.2.4-1

PENETRATION DATA							VALVE DATA																			
DETAIL	DWG NUMBER	GEN DES CRITERION	PROCESS FLUID	FLUID STATE	POSS LEAK PATHS	SYSTEM NUMBER AND PENETRATION DESCRIPTION	VALVE LOCATION	VALVE NUMBER	ESF POWER TRAIN	VALVE TYPE	ACTUATOR	PRI ACT MODE	SEC ACT MODE	ISOLATION SIGNAL	STROKE TIME	VALVE STATUS NORMAL	VALVE STATUS SHUTDOWN	VALVE STATUS POST-ACCIDENT	VALVE STATUS POWER FAILURE	ILRT	POS IND IN MCR	ESF	APP J TEST	ESSENTIAL/NON-ESS	SHIELD BLDG PENETRATION	NOTES
 EL 727 AZ 293°30'	47W809-1	55	W	C	AD	TO RCP SEALS (62)	CB	563	-	CK	-	SA	-	-	-	O	C	C	-	-	N	N	A	N	MK28	
 EL 727 AZ 292°	47W809-1	55	W	C	AD	TO RCP SEALS (62)	CB	560	-	CK	-	SA	-	-	-	O	C	C	-	-	N	N	A	N	MK30	
 EL 727'6" AZ 281°30'	47W809-1	55	W	C	AD	FROM RCP SEALS (62)	CB AB CB	61 63 639	B A -	GA GA CK	MO MO -	AT AT SA	RM RM -	PA PA -	10.0 10.0 -	O O O	V V O	C C V	AI AI -	C C -	Y Y N	N N N	AC AC AC	N	MK37	

WATTS BAR NUCLEAR PLANT UNIT 1 CONTAINMENT PENETRATIONS AND BARRIERS-TABLE 6.2.4-1

PENETRATION DATA							VALVE DATA																			
DETAIL	DWG NUMBER	GEN DES CRITERION	PROCESS FLUID	FLUID STATE	POSS LEAK PATHS	SYSTEM NUMBER AND PENETRATION DESCRIPTION	VALVE LOCATION	VALVE NUMBER	ESF POWER TRAIN	VALVE TYPE	ACTUATOR	PRI ACT MODE	SEC ACT MODE	ISOLATION SIGNAL	STROKE TIME	VALVE STATUS: NORMAL	SHUTDOWN	POST-ACCIDENT	POWER FAILURE	ILRT	POS IND IN MCR	ESF	APP J TEST	ESSENTIAL/NON-ESS	SHIELD BLDG PENETRATION	NOTES
 EL 723 AZ 281°30'	47W830-1	56	N2	C	BD	RC DRAIN TK AND PRT TO VH (77)	CB AB AB	18 19 20	B A A	DI DI DI	AO AO AO	AT AT AT	RM RM RM	PA PA PA	10.0 10.0 10.0	O O V	O O O	C C C	C C C	C C C	Y Y Y	N N N	AC AC AC	N	AS45	
 EL 723 AZ 278°	47W830-1, 47W809-7	56	W	C	BD	RC DRAIN TK PUMP DISCHARGE (77/84)	CB AB AB	9 10 530	B A –	DI DI GL	AO AO M	AT AT LM	RM RM –	PA PA –	10.0 10.0 –	O O C	O O C	C C C	C C –	C C C	Y Y N	N N N	AC AC AC	N	AS46	18
EL 808'6" AZ 293° 	47W814-2	56	G	C	BD	GLYCOL SUPPLY (61)	CB AB CB	192 191 533	B A –	DI DI CK	AO AO –	AT AT SA	RM RM –	PA PA –	30.0 30.0 –	O O O	O O O	C C V	C C –	O O O	Y Y N	N N N	C C C	N	MK25	8

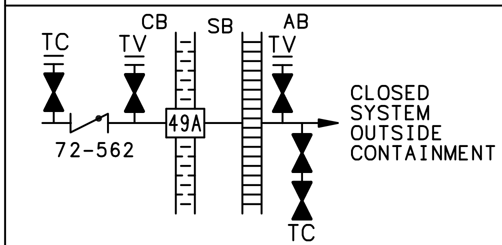
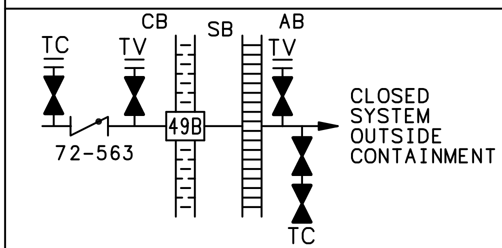
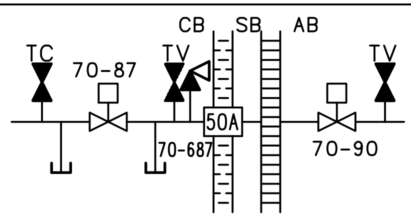
WATTS BAR NUCLEAR PLANT UNIT 1 CONTAINMENT PENETRATIONS AND BARRIERS-TABLE 6.2.4-1

PENETRATION DATA							VALVE DATA																			
DETAIL	DWG NUMBER	GEN DES CRITERION	PROCESS FLUID	FLUID STATE	POSS LEAK PATHS	SYSTEM NUMBER AND PENETRATION DESCRIPTION	VALVE LOCATION	VALVE NUMBER	ESF POWER TRAIN	VALVE TYPE	ACTUATOR	PRI ACT MODE	SEC ACT MODE	ISOLATION SIGNAL	STROKE TIME	VALVE STATUS NORMAL	VALVE STATUS SHUTDOWN	VALVE STATUS POST-ACCIDENT	VALVE STATUS POWER FAILURE	ILRT	POS IND IN MCR	ESF	APP J TEST	ESSENTIAL/NON-ESS	SHIELD BLDG PENETRATION	NOTES
	47W814-2	56	G	C	BD	GLYCOL RETURN (61)	CB AB CB	194 193 680	B A –	DI DI CK	AO AO –	AT AT SA	RM RM –	PA PA –	30.0 30.0 –	O O O	O O O	C C V	C C –	O O O	Y Y N	N N N	C C C	N	MK26	8
	47W812-1	56	W	C	AD	CONT. SPRAY (72)	CB	548	–	CK	–	SA	–	–	–	C	C	V	–	–	N	Y	A	E	MK4	26
	47W812-1	56	W	C	AD	CONT. SPRAY (72)	CB	549	–	CK	–	SA	–	–	–	C	C	V	–	–	N	Y	A	E	MK3	26

WATTS BAR NUCLEAR PLANT UNIT 1 CONTAINMENT PENETRATIONS AND BARRIERS-TABLE 6.2.4-1

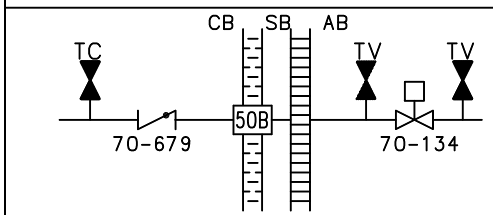
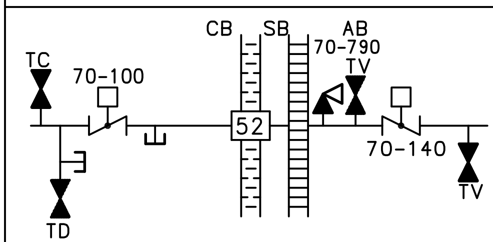
AMENDMENT 1

PENETRATION DATA / VALVE DATA

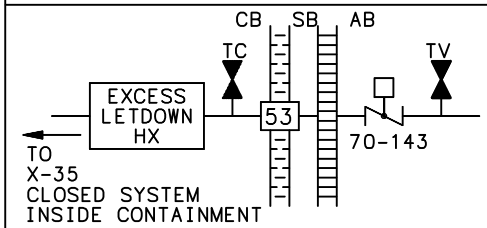
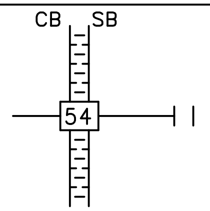
DETAIL	DWG NUMBER	GEN DES CRITERION	PROCESS FLUID	FLUID STATE	POSS LEAK PATHS	SYSTEM NUMBER AND PENETRATION DESCRIPTION	VALVE LOCATION	VALVE NUMBER	ESF POWER TRAIN	VALVE TYPE	ACTUATOR	PRI ACT MODE	SEC ACT MODE	ISOLATION SIGNAL	STROKE TIME	VALVE STATUS: NORMAL	VALVE STATUS: SHUTDOWN	VALVE STATUS: POST-ACCIDENT	VALVE STATUS: POWER FAILURE	ILRT	POS IND IN MCR	ESF	APP J TEST	ESSENTIAL/NON-ESS	SHIELD BLDG PENETRATION	NOTES
 EL 855'8" AZ 306°	47W812-1	56	W	C	AD	RHR SPRAY (72)	CB	562	–	CK	–	SA	–	–	–	C	C	V	–	–	N	Y	A	E	MK2	26
 EL 853'8" AZ 308°	47W812-1	56	W	C	AD	RHR SPRAY (72)	CB	563	–	CK	–	SA	–	–	–	C	C	V	–	–	N	Y	A	E	MK1	26
 EL 734'6" AZ 300°30'	47W859-3, -2	56	W	C	AD	RCP THERM BARRIER RETURN (70)	CB AB CB	87 90 687	B A –	GA GA RV	MO MO –	AT AT SA	RM RM –	PB PB –	66.0 66.0 –	O O C	O O C	C C V	AI AI –	C C –	Y Y N	N N N	AC AC AC	N	MK16	

WATTS BAR NUCLEAR PLANT UNIT 1 CONTAINMENT PENETRATIONS AND BARRIERS-TABLE 6.2.4-1

AMENDMENT 1

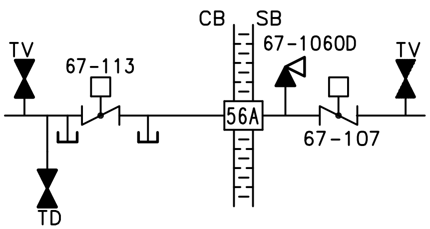
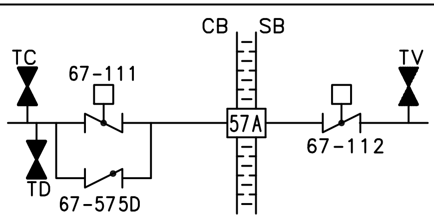
PENETRATION DATA							VALVE DATA																			
																VALVE STATUS										
DETAIL	DWG NUMBER	GEN DES CRITERION	PROCESS FLUID	FLUID STATE	POSS LEAK PATHS	SYSTEM NUMBER AND PENETRATION DESCRIPTION	VALVE LOCATION	VALVE NUMBER	ESF POWER TRAIN	VALVE TYPE	ACTUATOR	PRI ACT MODE	SEC ACT MODE	ISOLATION SIGNAL	STROKE TIME	NORMAL	SHUTDOWN	POST-ACCIDENT	POWER FAILURE	ILRT	POS IND IN MCR	ESF	APP J TEST	ESSENTIAL/NON-ESS	SHIELD BLDG PENETRATION	NOTES
 EL 734'6" AZ 299°30'	47W859-3, -2	56	W	C	AD	RCP THERM BARRIER SUPPLY (70)	CB AB	679 134	- B	CK GA	- MO	SA AT	- RM	- PB	- 66.0	O O	O O	C C	- AI	- C	N Y	N N	AC AC	N	MK17	
X-51 EL 728'6" AZ 286°30'	48W406	-	-	-	-	SPARE	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	A	-	-	
 EL 722'6" AZ 299°30'	47W859-2 -3	56	W	C	AD	CCS TO RCP COOLERS (70)	AB CB AB	140 100 790	B A -	BF BF RV	MO MO -	AT AT SA	RM RM -	PB PB -	66.0 66.0 -	O O C	O O C	C C V	AI AI -	C C -	Y Y N	N N N	AC AC AC	N	MK40	

WATTS BAR NUCLEAR PLANT UNIT 1 CONTAINMENT PENETRATIONS AND BARRIERS-TABLE 6.2.4-1

PENETRATION DATA							VALVE DATA																			
DETAIL	DWG NUMBER	GEN DES CRITERION	PROCESS FLUID	FLUID STATE	POSS LEAK PATHS	SYSTEM NUMBER AND PENETRATION DESCRIPTION	VALVE LOCATION	VALVE NUMBER	ESF POWER TRAIN	VALVE TYPE	ACTUATOR	PRI ACT MODE	SEC ACT MODE	ISOLATION SIGNAL	STROKE TIME	VALVE STATUS NORMAL	VALVE STATUS SHUTDOWN	VALVE STATUS POST-ACCIDENT	POWER FAILURE	ILRT	POS IND IN MCR	ESF	APP J TEST	ESSENTIAL/NON-ESS	SHIELD BLDG PENETRATION	NOTES
 EL 721'6" AZ 301°	47W859-2, 3	57	W	C	AD	CCS TO EXCESS LETDN HX (70)	AB	143	A	BF	MO	AT	RM	PA	66.0	V	C	C	AI	C	Y	N	AC	N	MK41	11
 EL 740 AZ 90°	72-4333-315	–	A	C	BC	THIMBLE RENEWAL	SB	–	–	BL	M	LM	–	–	–	C	V	C	–	O	N	N	AB	N	MK72	16
X-55 EL 771'6" AZ 262°	48w406	–	–	–	–	SPARE	–	–	–	–	–	–	–	–	–	–	–	–	–	–	–	–	A	–	–	

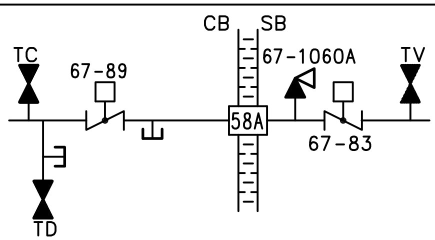
WATTS BAR NUCLEAR PLANT UNIT 1 CONTAINMENT PENETRATIONS AND BARRIERS-TABLE 6.2.4-1

AMENDMENT 1

PENETRATION DATA							VALVE DATA																			
DETAIL	DWG NUMBER	GEN DES CRITERION	PROCESS FLUID	FLUID STATE	POSS LEAK PATHS	SYSTEM NUMBER AND PENETRATION DESCRIPTION	VALVE LOCATION	VALVE NUMBER	ESF POWER TRAIN	VALVE TYPE	ACTUATOR	PRI ACT MODE	SEC ACT MODE	ISOLATION SIGNAL	STROKE TIME	NORMAL (VALVE STATUS)	SHUTDOWN (VALVE STATUS)	POST-ACCIDENT (VALVE STATUS)	POWER FAILURE	ILRT	POS IND IN MCR	ESF	APP J TEST	ESSENTIAL/NON-ESS	SHIELD BLDG PENETRATION	NOTES
 EL 720 AZ 353°	47W845-3	56	W	C	AB DE	LWR CONT ERCW SUPPLY (67)	SB CB SB	107 113 1060D	A B –	BF BF RV	MO MO –	AT AT SA	RM RM –	PB PB –	66.0 66.0 –	O O C	O O C	C C V	AI AI –	C C C	Y Y N	N N N	AC AC AC	N	MK80	
X-56B EL 720 AZ 355°30'	48W406	–	–	–	–	SPARE	–	–	–	–	–	–	–	–	–	–	–	–	–	–	–	–	A	–	–	
 EL 720 AZ 351°30'	47W845-3	56	W	C	AB DE	LWR CONT ERCW RETURN (67)	CB SB CB	111 112 575D	B A –	BF BF CK	MO MO –	AT AT SA	RM RM –	PB PB –	66.0 66.0 –	O O O	O O O	C C V	AI AI –	C C –	Y Y N	N N N	AC AC AC	N	MK81	8

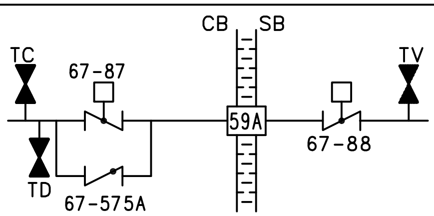
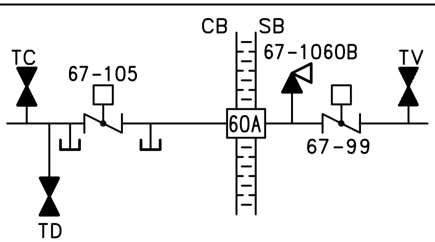
WATTS BAR NUCLEAR PLANT UNIT 1 CONTAINMENT PENETRATIONS AND BARRIERS-TABLE 6.2.4-1

AMENDMENT 1

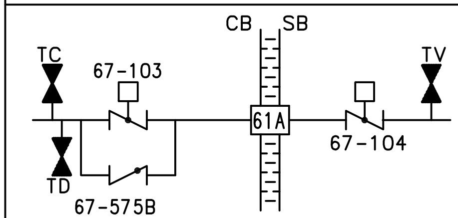
DETAIL	DWG NUMBER	GEN DES CRITERION	PROCESS FLUID	FLUID STATE	POSS LEAK PATHS	SYSTEM NUMBER AND PENETRATION DESCRIPTION	VALVE LOCATION	VALVE NUMBER	ESF POWER TRAIN	VALVE TYPE	ACTUATOR	PRI ACT MODE	SEC ACT MODE	ISOLATION SIGNAL	STROKE TIME	VALVE STATUS NORMAL	VALVE STATUS SHUTDOWN	VALVE STATUS POST-ACCIDENT	VALVE STATUS POWER FAILURE	ILRT	POS IND IN MCR	ESF	APP J TEST	ESSENTIAL/NON-ESS	SHIELD BLDG PENETRATION	NOTES
X-57B EL 720 AZ 354°15'	48N406	-	-	-	-	SPARE	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	A	-	-	
 EL 720 AZ 7°	47W845-3	56	W	C	AB DE	LWR CONT ERCW SUPPLY (67)	SB SB CB	1060A 83 89	- B A	RV BF BF	- MO MO	SA AT AT	- RM RM	- PB PB	- 66.0 66.0	C O O	C O O	V C C	- AI AI	- C C	N Y Y	N N N	AC AC AC	N	MK79	
X-58B EL 720 AZ 13°	47W600-75	54	W	C	BD	RCS PRESSURE SENSOR (68)	-	-	-	-	-	-	-	-	-	O	O	O	-	-	N	N	A	E	MK126	28

WATTS BAR NUCLEAR PLANT UNIT 1 CONTAINMENT PENETRATIONS AND BARRIERS-TABLE 6.2.4-1

AMENDMENT 1

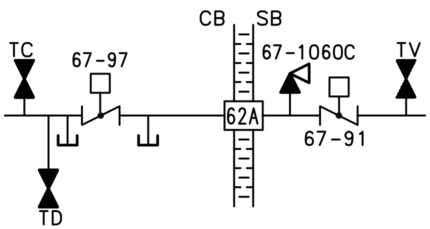
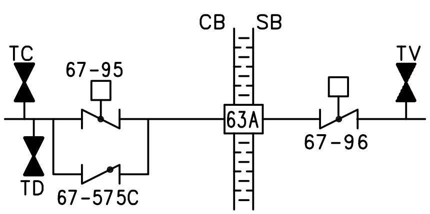
PENETRATION DATA							VALVE DATA																			
DETAIL	DWG NUMBER	GEN DES CRITERION	PROCESS FLUID	FLUID STATE	POSS LEAK PATHS	SYSTEM NUMBER AND PENETRATION DESCRIPTION	VALVE LOCATION	VALVE NUMBER	ESF POWER TRAIN	VALVE TYPE	ACTUATOR	PRI ACT MODE	SEC ACT MODE	ISOLATION SIGNAL	STROKE TIME	VALVE STATUS NORMAL	VALVE STATUS SHUTDOWN	VALVE STATUS POST-ACCIDENT	POWER FAILURE	ILRT	POS IND IN MCR	ESF	APP J TEST	ESSENTIAL/NON-ESS	SHIELD BLDG PENETRATION	NOTES
 EL 720 AZ 8°30'	47W845-3	56	W	C	AB DE	LWR CONT ERCW RETURN (67)	CB SB CB	87 88 575A	A B -	BF BF CK	MO MO -	AT AT SA	RM RM -	PB PB -	66.0 66.0 -	O O O	O O O	C C V	AI AI -	C C -	Y Y N	N N N	AC AC AC	N	MK78	8
X-59B EL 720 AZ 14°30'	48W406	-	-	-	-	SPARE	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	A	-	-	
 EL 720 AZ 173°	47W845-3	56	W	C	AB DE	LWR CONT ERCW SUPPLY (67)	SB SB CB	1060B 99 105	- A B	RV BF BF	- MO MO	SA AT AT	- RM RM	- PB PB	- 66.0 66.0	C O O	C O O	V C C	- AI AI	- C C	N Y Y	N N N	AC AC AC	N	MK77	

WATTS BAR NUCLEAR PLANT UNIT 1 CONTAINMENT PENETRATIONS AND BARRIERS-TABLE 6.2.4-1

PENETRATION DATA							VALVE DATA																		
															VALVE STATUS										
DETAIL	DWG NUMBER	GEN DES CRITERION	PROCESS FLUID	FLUID STATE	POSS LEAK PATHS	SYSTEM NUMBER AND PENETRATION DESCRIPTION	VALVE LOCATION	VALVE NUMBER	ESF POWER TRAIN	VALVE TYPE	ACTUATOR	PRI ACT MODE	SEC ACT MODE	ISOLATION SIGNAL	STROKE TIME	NORMAL	SHUTDOWN	POST-ACCIDENT	POWER FAILURE	ILRT	POS IND IN MCR	ESF	APP J TEST	ESSENTIAL/NON-ESS	SHIELD BLDG PENETRATION	NOTES
X-60B EL 720 AZ 174°15'	48W406	-	-	-	-	SPARE	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	A	-	-	
 EL 720 AZ 171°30'	47W845-3	56	W	C	AB DE	LWR CONT ERCW RETURN (67)	CB SB CB	103 104 575B	B A -	BF BF CK	MO MO -	AT AT SA	RM RM -	PB PB -	66.0 66.0 -	O O O	O O O	C C V	AI AI -	C C -	Y Y N	N N N	AC AC AC	N	MK76	8
X-61B EL 720 AZ 175°30'	48W406	-	-	-	-	SPARE	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	A	-	-	

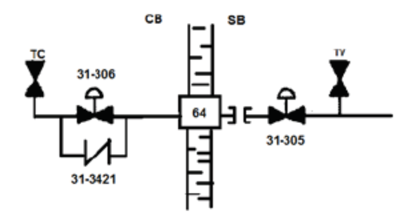
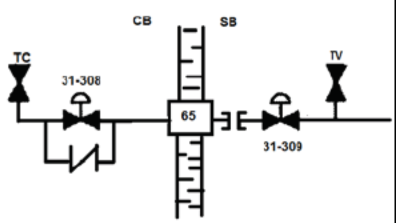
WATTS BAR NUCLEAR PLANT UNIT 1 CONTAINMENT PENETRATIONS AND BARRIERS-TABLE 6.2.4-1

AMENDMENT 1

PENETRATION DATA							VALVE DATA																			
DETAIL	DWG NUMBER	GEN DES CRITERION	PROCESS FLUID	FLUID STATE	POSS LEAK PATHS	SYSTEM NUMBER AND PENETRATION DESCRIPTION	VALVE LOCATION	VALVE NUMBER	ESF POWER TRAIN	VALVE TYPE	ACTUATOR	PRI ACT MODE	SEC ACT MODE	ISOLATION SIGNAL	STROKE TIME	VALVE STATUS NORMAL	VALVE STATUS SHUTDOWN	VALVE STATUS POST-ACCIDENT	POWER FAILURE	ILRT	POS IND IN MCR	ESF	APP J TEST	ESSENTIAL/NON-ESS	SHIELD BLDG PENETRATION	NOTES
 EL 720 AZ 187°	47W845-3	56	W	C	AB DE	LWR CONT ERCW SUPPLY (67)	SB SB CB	1060C 91 97	- B A	RV BF BF	- MO MO	SA AT AT	- RM RM	- PB PB	- 66.0 66.0	C O O	C O O	V C C	- AI AI	- C C	N Y Y	N N N	AC AC AC	N	MK74	
X-62B EL 720 AZ 193°	48W406	-	-	-	-	SPARE	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	A	-	-	
 EL 720 AZ 189°30'	47W845-3	56	W	C	AB DE	LWR CONT ERCW RETURN (67)	CB SB CB	95 96 575C	A B -	BF BF CK	MO MO -	AT AT SA	RM RM -	PB PB -	66.0 66.0 -	O O O	O O O	C C V	AI AI -	C C -	Y Y N	N N N	AC AC AC	N	MK75	8

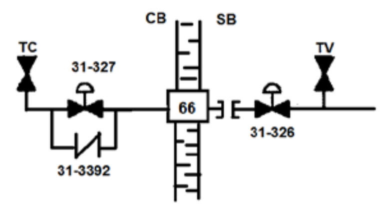
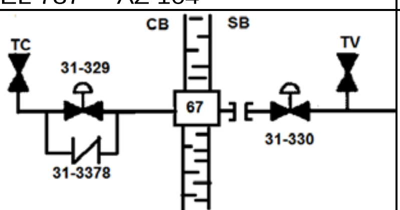
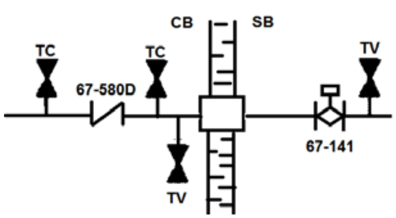
WATTS BAR NUCLAR PLANT UNIT 1 - CONTAINMENT PENETRATIONS AND BARRIERS

TABLE 6.2.4-1

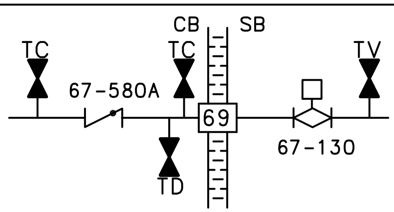
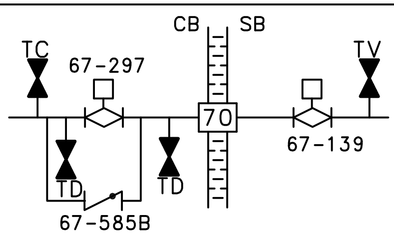
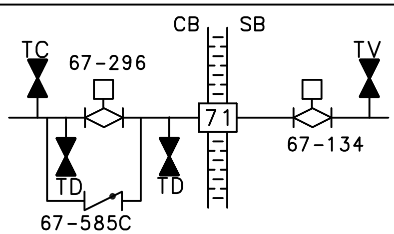
DETAIL	DWG NUMBER	GEN DES CRITERION	PROCESS FLUID	FLUID STATE	POSS LEAK PATHS	SYSTEM NUMBER AND PENETRATION DESCRIPTION	VALVE LOCATION	VALVE NUMBER	ESF POWER TRAIN	VALVE TYPE	ACTUATOR	PRI ACT MODE	SEC ACT MODE	ISOLATION SIGNAL	STROKE TIME	NORMAL	SHUTDOWN	POST-ACCIDENT	POWER FAILURE	ILRT	POS IND IN MCR	ESF	APP J TEST	ESSENTIAL/NON-ESS	SHIELD BLDG PENETRATION	NOTES
PENETRATION DATA							VALVE DATA									VALVE STATUS										
X-63B EL 720 AZ 194°30'	48W406	-	-	-	-	SPARE	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	A	-	-	
 EL 737 AZ 65°	47W865-5	56	W	C	AB D	INST RM CHILL H_2O RETURN (31)	CB SB CB	306 305 3421	A B -	GL GL CK	AO AO -	AT AT SA	RM RM -	PA PA -	10.0 10.0 -	C C C	C C C	C C V	C C -	C C -	Y Y N	N N N	A A A	N	MK92	24
 EL 738 AZ 65°	47W865-5	56	W	C	AB D	INST RM CHILL H_2O SUPPLY (31)	CB SB CB	308 309 3407	A B -	GL GL CK	AO AO -	AT AT SA	RM RM -	PA PA -	10.0 10.0 -	C C C	C C C	C C V	C C -	C C -	Y Y N	N N N	A A A	N	MK90	24

WATTS BAR NUCLAR PLANT UNIT 1 - CONTAINMENT PENETRATIONS AND BARRIERS

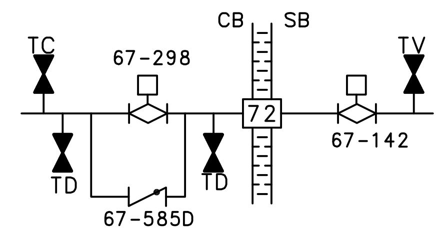
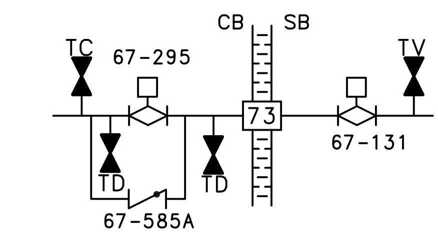
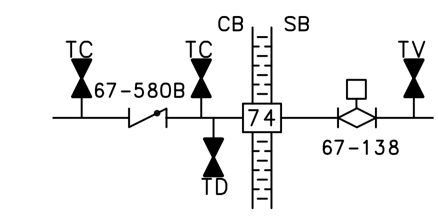
TABLE 6.2.4-1

DETAIL	DWG NUMBER	GEN DES CRITERION	PROCESS FLUID	FLUID STATE	POSS LEAK PATHS	SYSTEM NUMBER AND PENETRATION DESCRIPTION	VALVE LOCATION	VALVE NUMBER	ESF POWER TRAIN	VALVE TYPE	ACTUATOR	PRI ACT MODE	SEC ACT MODE	ISOLATION SIGNAL	STROKE TIME	VALVE STATUS NORMAL	VALVE STATUS SHUTDOWN	VALVE STATUS POST-ACCIDENT	VALVE STATUS POWER FAILURE	ILRT	POS IND IN MCR	ESF	APP J TEST	ESSENTIAL/NON-ESS	SHIELD BLDG PENETRATION	NOTES
PENETRATION DATA							VALVE DATA																			
 EL 737 AZ 104°	47W865-5	56	W	C	ABD	INST RM CHILL H_2O RETURN (31)	CB SB CB	327 326 3392	B A -	GL GL CK	AO AO -	AT AT SA	RM RM -	PA PA -	10.0 10.0 -	C C C	C C C	C C V	C C -	C C O	Y Y N	N N N	A A A	N	MK93	24
 EL 738 AZ 104°	47W865-5	56	W	C	ABD	INST RM CHILL H_2O SUPPLY (31)	CB SB CB	329 330 3378	A B -	GL GL CK	AO AO -	AT AT SA	RM RM -	PA PA -	10.0 10.0 -	C C C	C C C	C C V	C C -	C C -	Y Y N	N N N	A A A	N	MK91	24
 EL 794'6" AZ 301°15'	47W845-3	56	W	C	ABD	UPPER CONT ERCW SUPPLY (67)	CB SB	5800 141	- B	CK PG	- MO	SA AT	- RM	- PB	- 66.0	- O	- O	- C	- AI	- C	N Y	N N	AC AC	N	MK88	

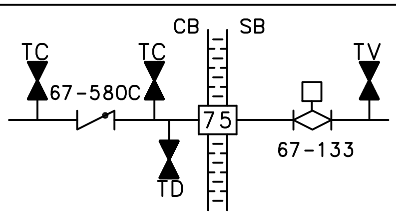
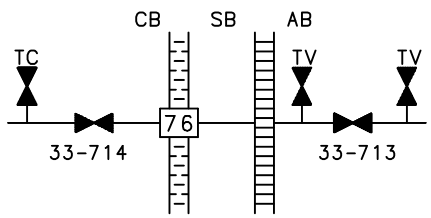
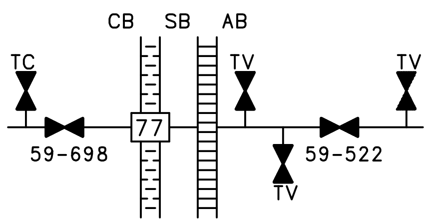
WATTS BAR NUCLEAR PLANT UNIT 1 CONTAINMENT PENETRATIONS AND BARRIERS-TABLE 6.2.4-1

PENETRATION DATA							VALVE DATA																			
DETAIL	DWG NUMBER	GEN DES CRITERION	PROCESS FLUID	FLUID STATE	POSS LEAK PATHS	SYSTEM NUMBER AND PENETRATION DESCRIPTION	VALVE LOCATION	VALVE NUMBER	ESF POWER TRAIN	VALVE TYPE	ACTUATOR	PRI ACT MODE	SEC ACT MODE	ISOLATION SIGNAL	STROKE TIME	VALVE STATUS NORMAL	VALVE STATUS SHUTDOWN	VALVE STATUS POST-ACCIDENT	VALVE STATUS POWER FAILURE	ILRT	POS IND IN MCR	ESF	APP J TEST	ESSENTIAL/NON-ESS	SHIELD BLDG PENETRATION	NOTES
 EL 796'6" AZ 301°15'	47W845-3	56	W	C	AB DE	UPPER CONT ERCW SUPPLY (67)	CB SB	580A 130	– A	CK PG	– MO	SA AT	– RM	– PB	– 66.0	– O	– O	– C	– AI	– C	N Y	N N	AC AC	N	MK86	
 EL 798'6" AZ 301°15'	47W845-3	56	W	C	AB DE	UPPER CONT ERCW RETURN (67)	CB SB CB	297 139 585B	B A –	PG PG CK	MO MO –	AT AT SA	RM RM –	PB PB –	66.0 66.0 –	O O O	O O O	C C V	AI AI –	C C –	Y Y N	N N N	AC AC AC	N	MK84	8
 EL 800'6" AZ 301°15'	47W845-3	56	W	C	AB DE	UPPER CONT ERCW RETURN (67)	CB SB CB	296 134 585C	A B –	PG PG CK	MO MO –	AT AT SA	RM RM –	PB PB –	66.0 66.0 –	O O O	O O O	C C V	AI AI –	C C –	Y Y N	N N N	AC AC AC	N	MK82	8

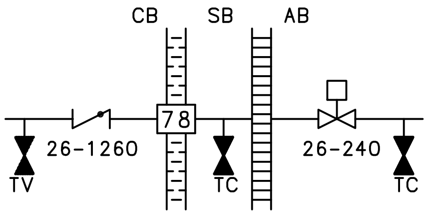
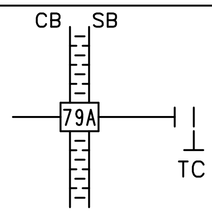
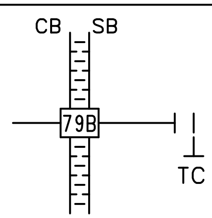
WATTS BAR NUCLEAR PLANT UNIT 1 CONTAINMENT PENETRATIONS AND BARRIERS-TABLE 6.2.4-1

PENETRATION DATA							VALVE DATA																			
																VALVE STATUS										
DETAIL	DWG NUMBER	GEN DES CRITERION	PROCESS FLUID	FLUID STATE	POSS LEAK PATHS	SYSTEM NUMBER AND PENETRATION DESCRIPTION	VALVE LOCATION	VALVE NUMBER	ESF POWER TRAIN	VALVE TYPE	ACTUATOR	PRI ACT MODE	SEC ACT MODE	ISOLATION SIGNAL	STROKE TIME	NORMAL	SHUTDOWN	POST-ACCIDENT	POWER FAILURE	ILRT	POS IND IN MCR	ESF	APP J TEST	ESSENTIAL/NON-ESS	SHIELD BLDG PENETRATION	NOTES
 EL 793'6" AZ 301°15'	47W845-3	56	W	C	AB DE	UPPER CONT ERCW RETURN (67)	CB SB CB	298 142 585D	B A -	PG PG CK	MO MO -	AT AT SA	RM RM -	PB PB -	66.0 66.0 -	O O O	O O O	C C V	AI AI -	C C -	Y Y N	N N N	AC AC AC	N	MK89	8
 EL 795'6" AZ 301°15'	47W845-3	56	W	C	AB DE	UPPER CONT ERCW RETURN (67)	CB SB CB	295 131 585A	A B -	PG PG CK	MO MO -	AT AT SA	RM RM -	PB PB -	66.0 66.0 -	O O O	O O O	C C V	AI AI -	C C -	Y Y N	N N N	AC AC AC	N	MK87	8
 EL 797'6" AZ 301°15'	47W845-3	56	W	C	AB DE	UPPER CONT ERCW SUPPLY (67)	CB SB	580B 138	- B	CK PG	- MO	SA AT	- RM	- PB	- 66.0	O O	O O	C C	- AI	- C	N Y	N N	AC AC	N	MK85	

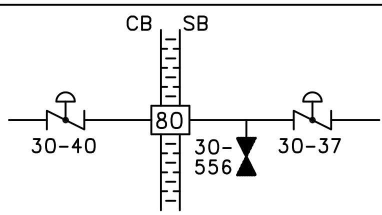
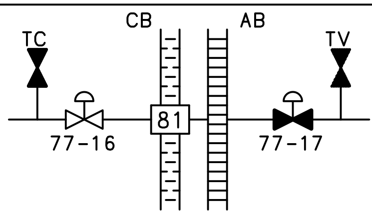
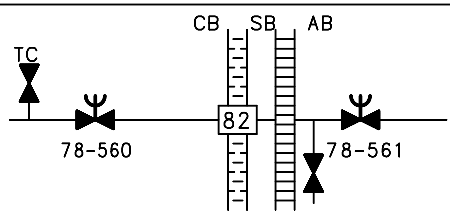
WATTS BAR NUCLEAR PLANT UNIT 1 CONTAINMENT PENETRATIONS AND BARRIERS-TABLE 6.2.4-1

PENETRATION DATA							VALVE DATA																		
DETAIL	DWG NUMBER	GEN DES CRITERION	PROCESS FLUID	FLUID STATE	POSS LEAK PATHS	SYSTEM NUMBER AND PENETRATION DESCRIPTION	VALVE LOCATION	VALVE NUMBER	ESF POWER TRAIN	VALVE TYPE	ACTUATOR	PRI ACT MODE	SEC ACT MODE	ISOLATION SIGNAL	STROKE TIME	VALVE STATUS NORMAL	VALVE STATUS SHUTDOWN	VALVE STATUS POST-ACCIDENT	POWER FAILURE	ILRT	POS IND IN MCR	ESF	APP J TEST	ESSENTIAL/NON-ESS	SHIELD BLDG PENETRATION	NOTES
 EL 799'6" AZ 301°15'	47W845-3	56	W	C	AB DE	UPPER CONT ERCW SUPPLY (67)	CB SB	580C 133	- A	CK PG	- MO	SA AT	- RM	- PB	- 66.0	O O	O O	C C	- AI	- C	N Y	N N	AC AC	N	MK83	
 EL 711 AZ 300°	47W846-2	56	A	C	AD	SERVICE AIR (33)	CB AB	714 713	- -	DI DI	M M	LM LM	- -	- -	- -	C C	O O	C C	- -	C C	N N	N N	AC AC	N	MK97	
 EL 710'6" AZ 299°	47W856-1	56	W	C	AD	DEMIN WATER (59)	CB AB	698 522	- -	DI DI	M M	LM LM	- -	- -	- -	C C	O O	C C	- -	C C	N N	N N	AC AC	N	MK96	

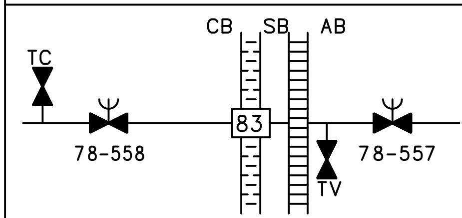
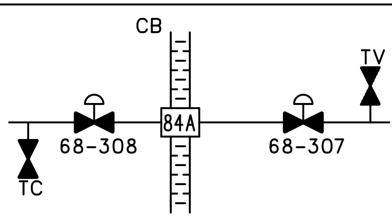
WATTS BAR NUCLEAR PLANT UNIT 1 CONTAINMENT PENETRATIONS AND BARRIERS-TABLE 6.2.4-1

PENETRATION DATA							VALVE DATA																			
																VALVE STATUS										
DETAIL	DWG NUMBER	GEN DES CRITERION	PROCESS FLUID	FLUID STATE	POSS LEAK PATHS	SYSTEM NUMBER AND PENETRATION DESCRIPTION	VALVE LOCATION	VALVE NUMBER	ESF POWER TRAIN	VALVE TYPE	ACTUATOR	PRI ACT MODE	SEC ACT MODE	ISOLATION SIGNAL	STROKE TIME	NORMAL	SHUTDOWN	POST-ACCIDENT	POWER FAILURE	ILRT	POS IND IN MCR	ESF	APP J TEST	ESSENTIAL/NON-ESS	SHIELD BLDG PENETRATION	NOTES
 EL 798'9" AZ 301°	47W850-9	56	A	C	AB D	FIRE PROTECTION (26)	CB AB	1260 240	- A	CK GA	- MO	SA AT	- RM	- PA	- 20.0	C O	C O	C C	- AI	- C	N Y	N N	AC AC	N	MK98	
 EL 808 AZ 289°	47W814-1 47W462-7	56	I	C	AB	ICE BLOWING (61)	SB			BL	M	LM	-	-	-	C	V	C	-	C	N	N	AB	N	MK23	
 EL 809 AZ 290°	47W814-1 47W462-7	56	I	C	AB	NEGATIVE RETURN (61)	SB			BL	M	LM	-	-	-	C	V	C	-	C	N	Y	AB	N	MK24	

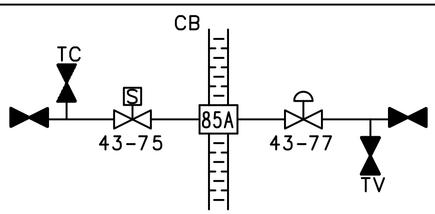
WATTS BAR NUCLEAR PLANT UNIT 1 CONTAINMENT PENETRATIONS AND BARRIERS–TABLE 6.2.4–1

PENETRATION DATA							VALVE DATA																			
DETAIL	DWG NUMBER	GEN DES CRITERION	PROCESS FLUID	FLUID STATE	POSS LEAK PATHS	SYSTEM NUMBER AND PENETRATION DESCRIPTION	VALVE LOCATION	VALVE NUMBER	ESF POWER TRAIN	VALVE TYPE	ACTUATOR	PRI ACT MODE	SEC ACT MODE	ISOLATION SIGNAL	STROKE TIME	VALVE STATUS: NORMAL	VALVE STATUS: SHUTDOWN	VALVE STATUS: POST-ACCIDENT	VALVE STATUS: POWER FAILURE	ILRT	POS IND IN MCR	ESF	APP J TEST	ESSENTIAL/NON-ESS	SHIELD BLDG PENETRATION	NOTES
 EL 720 AZ 286°30'	47W866-1	56	A	C	AB CE	LWR COMP PRESS RELIEF (30)	CB SB SB	40 37 556	A B –	BF BF TC	AO AO –	AT AT LM	RM RM –	CV CV –	4.0 4.0 –	V V C	C C C	C C C	C C C	C C C	Y Y N	N N N	AC AC AC	N	MK71	3,6,14
 EL 718 AZ 287°	47W830-1	56	A	C	AD	RC DR TK TO GAS ANALYZER (77)	CB AB	16 17	B A	DI DI	AO AO	AT AT	RM RM	PA PA	10.0 10.0	V V	O O	C C	C C	C C	Y Y	N N	AC AC	N	AS081	
 EL 718 AZ 282°30'	47W855-1	56	W	C	AB D	REFUEL CAV PRFCN PUMP SUCT (78)	CB AB	560 561	– –	DI DI	M M	LM LM	– –	– –	– –	C C	V V	C C	– –	C C	N N	N N	AC AC	N	MK94	

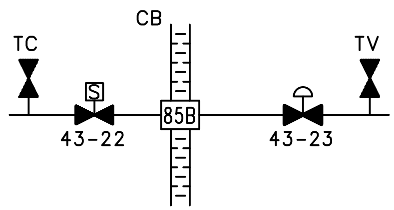
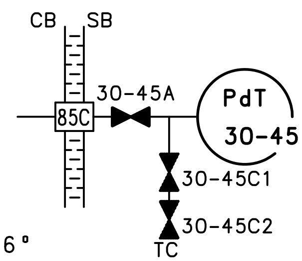
WATTS BAR NUCLEAR PLANT UNIT 1 CONTAINMENT PENETRATIONS AND BARRIERS-TABLE 6.2.4-1

PENETRATION DATA							VALVE DATA																		
															VALVE STATUS										
DETAIL	DWG NUMBER	GEN DES CRITERION	PROCESS FLUID	FLUID STATE	POSS LEAK PATHS	SYSTEM NUMBER AND PENETRATION DESCRIPTION	VALVE LOCATION	VALVE NUMBER	ESF POWER TRAIN	VALVE TYPE	ACTUATOR	PRI ACT MODE	SEC ACT MODE	ISOLATION SIGNAL	STROKE TIME	NORMAL	SHUTDOWN	POST-ACCIDENT	POWER FAILURE	ILRT	POS IND IN MCR	ESF	APP J TEST	ESSENTIAL/NON-ESS	SHIELD BLDG PENETRATION	NOTES
 EL 733 AZ 294°	47W855-1	56	W	C	AB D	REFUEL CAV PRFCN PUMP SUCTION (78)	CB AB	558 557	- -	DI DI	M M	LM LM	- -	- -	- -	C C	V V	C C	- -	C C	N N	N N	AC AC	N	MK95	
 EL 723 AZ 307°30'	47W625-8	56	N	C	AD	P.R.T. TO GAS ANALYZER (68)	CB SB	308 307	B A	GL GL	AO AO	AT AT	RM RM	PA PA	10.0 10.0	V V	O O	C C	C C	C C	Y Y	N N	AC AC	N	MK99M	
X-84B EL 723 AZ 307°30'	47W600-292	-	W	C	BD	RVLIS (68)	-	-	-	-	-	-	-	-	-	O	O	O	-	-	N	N	A	E	MK101	21

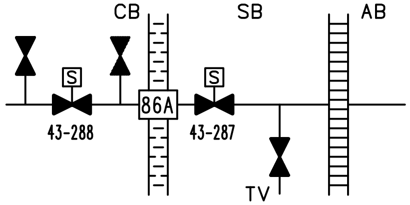
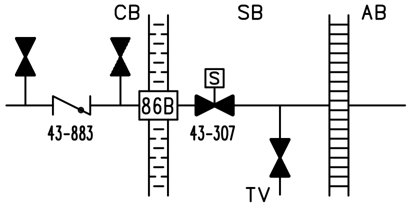
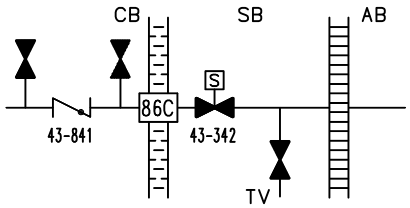
WATTS BAR NUCLEAR PLANT UNIT 1 CONTAINMENT PENETRATIONS AND BARRIERS-TABLE 6.2.4-1

PENETRATION DATA							VALVE DATA																			
DETAIL	DWG NUMBER	GEN DES CRITERION	PROCESS FLUID	FLUID STATE	POSS LEAK PATHS	SYSTEM NUMBER AND PENETRATION DESCRIPTION	VALVE LOCATION	VALVE NUMBER	ESF POWER TRAIN	VALVE TYPE	ACTUATOR	PRI ACT MODE	SEC ACT MODE	ISOLATION SIGNAL	STROKE TIME	VALVE STATUS NORMAL	VALVE STATUS SHUTDOWN	VALVE STATUS POST-ACCIDENT	POWER FAILURE	ILRT	POS IND IN MCR	ESF	APP J TEST	ESSENTIAL/NON-ESS	SHIELD BLDG PENETRATION	NOTES
X-84C EL 723 AZ 307°30'	47W600-292	–	W	C	BD	RVLIS (68)	–	–	–	–	–	–	–	–	–	O	O	O	–	–	N	N	A	E	MK101	21
X-84D EL 723 AZ 307°30'	47W600-292	–	W	C	BD	RVLIS (68)	–	–	–	–	–	–	–	–	–	O	O	O	–	–	N	N	A	E	MK101	21
 EL 723 AZ 306°	47W625-7	56	W	H	AB D	EX LT DN HX TO BORON ANALYZER (43) (NOT USED FOR UNIT 1 OPERATION)	CB SB	75 77	B A	GL GL	SO AO	AT AT	RM RM	PA PA	5.0 5.0	V V	C C	C C	C C	C C	Y Y	N N	AC AC	N	MK100M	15

WATTS BAR NUCLEAR PLANT UNIT 1 CONTAINMENT PENETRATIONS AND BARRIERS-TABLE 6.2.4-1

PENETRATION DATA							VALVE DATA																			
DETAIL	DWG NUMBER	GEN DES CRITERION	PROCESS FLUID	FLUID STATE	POSS LEAK PATHS	SYSTEM NUMBER AND PENETRATION DESCRIPTION	VALVE LOCATION	VALVE NUMBER	ESF POWER TRAIN	VALVE TYPE	ACTUATOR	PRI ACT MODE	SEC ACT MODE	ISOLATION SIGNAL	STROKE TIME	VALVE STATUS: NORMAL	VALVE STATUS: SHUTDOWN	VALVE STATUS: POST-ACCIDENT	VALVE STATUS: POWER FAILURE	ILRT	POS IND IN MCR	ESF	APP J TEST	ESSENTIAL/NON-ESS	SHIELD BLDG PENETRATION	NOTES
 EL 723 AZ 306°	47W625-1	55	W	H	AB D	HOT LEG SAMPLE LOOPS 1 & 3 (43)	CB SB	22 23	B A	GL GL	SO AO	AT AT	RM RM	PA PA	10.0 10.0	V V	V V	C C	C C	C C	Y Y	N N	AC AC	N	MK100M	15
 EL 723 AZ 306°	47W600-89	–	A	C	B	dP SENSOR (30)	SB SB SB SB	SNSR. 45C1 45C2 45A	– – – –	– GL GL GL	– M M M	– LM LM LM	– – – –	– – – –	– – – –	O C C O	O V V O	O C C O	– – – –	– C C O	– N N N	N N N N	A A – A	E –	–	12
X-85D EL 723 AZ 306°	47W331-2	–	–	–	–	SPARE	–	–	–	–	–	–	–	–	–	–	–	–	–	–	–	–	A	–	–	

WATTS BAR NUCLEAR PLANT UNIT 1 CONTAINMENT PENETRATIONS AND BARRIERS-TABLE 6.2.4-1

PENETRATION DATA							VALVE DATA																			
DETAIL	DWG NUMBER	GEN DES CRITERION	PROCESS FLUID	FLUID STATE	POSS LEAK PATHS	SYSTEM NUMBER AND PENETRATION DESCRIPTION	VALVE LOCATION	VALVE NUMBER	ESF POWER TRAIN	VALVE TYPE	ACTUATOR	PRI ACT MODE	SEC ACT MODE	ISOLATION SIGNAL	STROKE TIME	VALVE STATUS NORMAL	VALVE STATUS SHUTDOWN	VALVE STATUS POST-ACCIDENT	POWER FAILURE	ILRT	POS IND IN MCR	ESF	APP J TEST	ESSENTIAL/NON-ESS	SHIELD BLDG PENETRATION	NOTES
 EL 721'6" AZ 307°30'	47W625-15	56	A	C	AB D	PAS CONT AIR INTK (43)	CB SB	288 287	A A	GL GL	SO SO	RM RM	LM LM	- -	- -	C C	C C	V V	C C	C C	Y Y	N N	AC AC	E	MK61	19
 EL 721'6" AZ 307°30'	47W625-15	56	A	C	AB D	PAS CONT AIR RTRN (43)	CB SB	883 307	- A	CK GL	- SO	SA RM	- LM	- -	- -	C C	C C	V V	- C	- C	N Y	N N	AC AC	E	MK33	19
 EL 721'6" AZ 307°30'	47W625-15	56	W	C	AB D	PAS CONT SUMP RTRN (43)	CB SB	841 342	- A	CK GL	- SO	SA RM	- LM	- -	- -	C C	C C	V V	- C	- C	N Y	N N	AC AC	E	MK54	19

MIS285

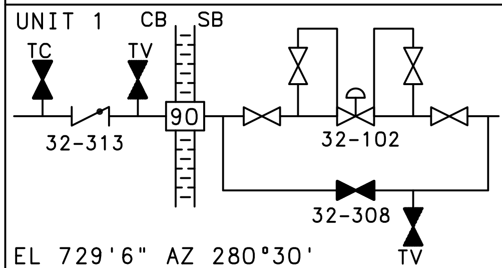
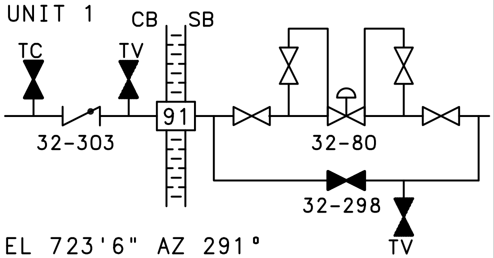
WATTS BAR NUCLEAR PLANT UNIT 1 CONTAINMENT PENETRATIONS AND BARRIERS-TABLE 6.2.4-1

PENETRATION DATA							VALVE DATA																			
																VALVE STATUS										
DETAIL	DWG NUMBER	GEN DES CRITERION	PROCESS FLUID	FLUID STATE	POSS LEAK PATHS	SYSTEM NUMBER AND PENETRATION DESCRIPTION	VALVE LOCATION	VALVE NUMBER	ESF POWER TRAIN	VALVE TYPE	ACTUATOR	PRI ACT MODE	SEC ACT MODE	ISOLATION SIGNAL	STROKE TIME	NORMAL	SHUTDOWN	POST-ACCIDENT	POWER FAILURE	ILRT	POS IND IN MCR	ESF	APP J TEST	ESSENTIAL/NON-ESS	SHIELD BLDG PENETRATION	NOTES
X-86D EL 721'6" AZ 307°30'	47W331-2	-	-	-	-	SPARE	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	A	-	-	
X-87A EL 721'6" AZ 306°	47W331-2	-	-	-	-	SPARE	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	A	-	-	
X-87B EL 721'6" AZ 306°	47W600-292 47W331-2	-	W	C	BD	RVLIS (68)	-	-	-	-	-	-	-	-	-	-	-	-	-	-	N	N	A	E	-	21

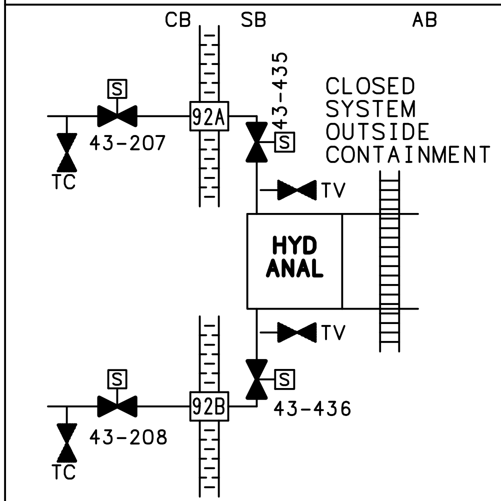
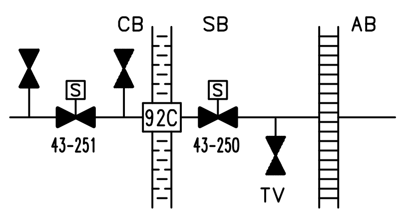
WATTS BAR NUCLEAR PLANT UNIT 1 CONTAINMENT PENETRATIONS AND BARRIERS-TABLE 6.2.4-1

PENETRATION DATA							VALVE DATA																			
																VALVE STATUS										
DETAIL	DWG NUMBER	GEN DES CRITERION	PROCESS FLUID	FLUID STATE	POSS LEAK PATHS	SYSTEM NUMBER AND PENETRATION DESCRIPTION	VALVE LOCATION	VALVE NUMBER	ESF POWER TRAIN	VALVE TYPE	ACTUATOR	PRI ACT MODE	SEC ACT MODE	ISOLATION SIGNAL	STROKE TIME	NORMAL	SHUTDOWN	POST-ACCIDENT	POWER FAILURE	ILRT	POS IND IN MCR	ESF	APP J TEST	ESSENTIAL/NON-ESS	SHIELD BLDG PENETRATION	NOTES
X-87C EL 721'6" AZ 306°	47W600-292 47W331-2	-	W	C	BD	RVLIS (68)	-	-	-	-	-	-	-	-	-	-	-	-	-	-	N	N	A	E	MK102	21
X-87D EL 721'6" AZ 306°	47W600-292 47W331-2	-	W	C	BD	RVLIS (68)	-	-	-	-	-	-	-	-	-	-	-	-	-	-	N	N	A	E	MK102	21
X-88 EL 733 AZ 277°30'	48W406	-	-	-	-	SPARE	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	A	-	MK103	29

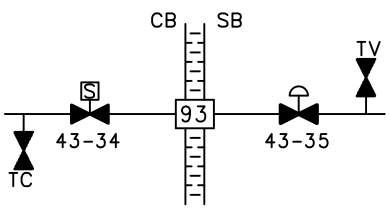
WATTS BAR NUCLEAR PLANT UNIT 1 CONTAINMENT PENETRATIONS AND BARRIERS-TABLE 6.2.4-1

PENETRATION DATA							VALVE DATA																			
DETAIL	DWG NUMBER	GEN DES CRITERION	PROCESS FLUID	FLUID STATE	POSS LEAK PATHS	SYSTEM NUMBER AND PENETRATION DESCRIPTION	VALVE LOCATION	VALVE NUMBER	ESF POWER TRAIN	VALVE TYPE	ACTUATOR	PRI ACT MODE	SEC ACT MODE	ISOLATION SIGNAL	STROKE TIME	VALVE STATUS NORMAL	VALVE STATUS SHUTDOWN	VALVE STATUS POST-ACCIDENT	VALVE STATUS POWER FAILURE	ILRT	POS IND IN MCR	ESF	APP J TEST	ESSENTIAL/NON-ESS	SHIELD BLDG PENETRATION	NOTES
X-89 EL 732 AZ 278°30'	48W406	-	-	-	-	SPARE	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	A	-	MK104	30
 EL 729'6" AZ 280°30'	47W848-1	56	A	C	AB D	CONTROL AIR TR-B (32)	CB SB SB	313 102 308	- B -	CK GL GA	- AO M	SA AT LM	- RM -	- PB -	- 10.0 -	O O C	O O C	C C C	- C AI	- C C	N Y N	N N N	AC AC AC	N	MK105	10
 EL 723'6" AZ 291°	47W848-1	56	A	C	AB D	CONTROL AIR TR-A (32)	CB SB SB	303 80 298	- A -	CK GL GA	- AO M	SA AT LM	- RM -	- PB -	- 10.0 -	O O C	O O C	C C C	- C AI	- C C	N Y N	N N N	AC AC AC	N	MK56	10

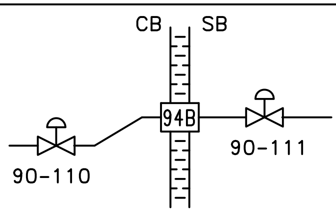
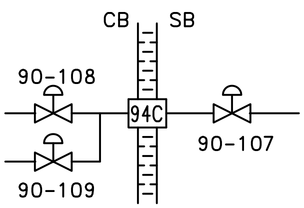
WATTS BAR NUCLEAR PLANT UNIT 1 CONTAINMENT PENETRATIONS AND BARRIERS-TABLE 6.2.4-1

PENETRATION DATA							VALVE DATA																			
DETAIL	DWG NUMBER	GEN DES CRITERION	PROCESS FLUID	FLUID STATE	POSS LEAK PATHS	SYSTEM NUMBER AND PENETRATION DESCRIPTION	VALVE LOCATION	VALVE NUMBER	ESF POWER TRAIN	VALVE TYPE	ACTUATOR	PRI ACT MODE	SEC ACT MODE	ISOLATION SIGNAL	STROKE TIME	VALVE STATUS NORMAL	VALVE STATUS SHUTDOWN	VALVE STATUS POST-ACCIDENT	POWER FAILURE	ILRT	POS IND IN MCR	ESF	APP J TEST	ESSENTIAL/NON-ESS	SHIELD BLDG PENETRATION	NOTES
	47W625-11	56	A	C	BD	H2 ANALYZER TR-B (43)	CB SB	207 435	B B	GL GL	SO SO	RM RM	- -	- -	- -	C C	C C	O O	C C	C C	Y Y	N N	AC AC	E E	MK-33 MK-61	20
EL 723'6" AZ 290°	47W625-11	56	A	C	BD	H2 ANALYZER TR-B (43)	CB SB	208 436	B B	GL GL	SO SO	RM RM	- -	- -	- -	C C	C C	O O	C C	C C	Y Y	N N	AC AC	E E	MK-33 MK-61	20
 EL 723'6" AZ 290°	47W625-15	55	W	H	AB D	PAS HOT LEG 1 TR-A (43)	CB SB	251 250	A A	GL GL	SO SO	RM RM	LM LM	- -	- -	C C	C C	V V	C C	C C	Y Y	N N	AC AC	E	MK34	19

WATTS BAR NUCLEAR PLANT UNIT 1 CONTAINMENT PENETRATIONS AND BARRIERS-TABLE 6.2.4-1

PENETRATION DATA							VALVE DATA																		
															VALVE STATUS										
DETAIL	DWG NUMBER	GEN DES CRITERION	PROCESS FLUID	FLUID STATE	POSS LEAK PATHS	SYSTEM NUMBER AND PENETRATION DESCRIPTION	VALVE LOCATION	VALVE NUMBER	ESF POWER TRAIN	VALVE TYPE	ACTUATOR	PRI ACT MODE	SEC ACT MODE	ISOLATION SIGNAL	STROKE TIME	NORMAL	SHUTDOWN	POST-ACCIDENT	POWER FAILURE	ILRT	POS IND IN MCR	ESF	APP J TEST	ESSENTIAL/NON-ESS	SHIELD BLDG PENETRATION	NOTES
X-92D EL 723'6" AZ 290°	47W331-3	-	-	-	-	SPARE	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	A	-	-	
 EL 723'6" AZ 289°	47W625-2	56	W	C	ABD	ACCUM SAMPLE (43)	CB SB	34 35	B A	GL GL	SO AO	AT AT	RM RM	PA PA	5.0 5.0	V V	C C	C C	C C	C C	Y Y	N N	AC AC	N	MK58	
X-94A EL 741 AZ 294°	47W600-105 48W406	-	A	C	-	SPARE	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	A	-	-	

WATTS BAR NUCLEAR PLANT UNIT 1 CONTAINMENT PENETRATIONS AND BARRIERS-TABLE 6.2.4-1

PENETRATION DATA							VALVE DATA																			
DETAIL	DWG NUMBER	GEN DES CRITERION	PROCESS FLUID	FLUID STATE	POSS LEAK PATHS	SYSTEM NUMBER AND PENETRATION DESCRIPTION	VALVE LOCATION	VALVE NUMBER	ESF POWER TRAIN	VALVE TYPE	ACTUATOR	PRI ACT MODE	SEC ACT MODE	ISOLATION SIGNAL	STROKE TIME	VALVE STATUS NORMAL	VALVE STATUS SHUTDOWN	VALVE STATUS POST-ACCIDENT	POWER FAILURE	ILRT	POS IND IN MCR	ESF	APP J TEST	ESSENTIAL/NON-ESS	SHIELD BLDG PENETRATION	NOTES
 EL 741 AZ 294°	47W600-105	56	A	C	AB D	LOWER COMP AIR MON INTAKE (90)	CB SB	110 111	B A	GL GL	AO AO	AT AT	RM RM	CV CV	5.0 5.0	O O	O O	C C	C C	C C	Y Y	N N	AC AC	N	MK59M	
 EL 741 AZ 294°	47W600-105	56	A	C	AB D	LOWER COMP AIR MON RETURN (90)	CB SB CB	108 107 109	B A B	GL GL GL	AO AO AO	AT AT AT	RM RM RM	CV CV CV	5.0 5.0 5.0	O O O	O O O	C C C	C C C	C C C	Y Y Y	N N N	AC AC AC	N	MK59M	
X-95A EL 741 AZ 293°	47W600-105	-	-	-	-	SPARE	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	A	-	-	

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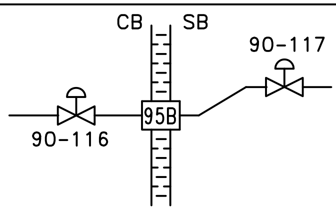
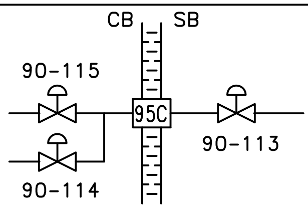
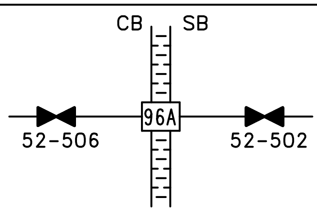
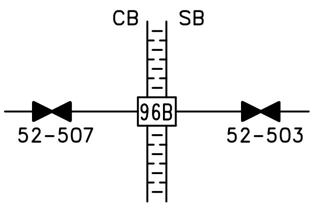
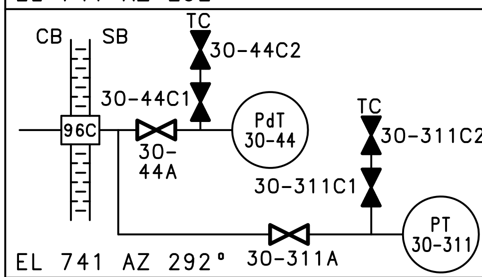
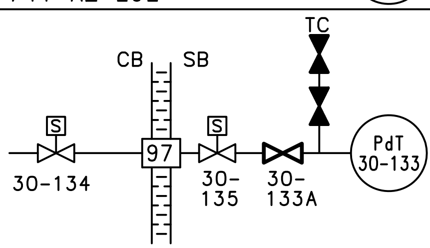
WATTS BAR NUCLEAR PLANT UNIT 1 CONTAINMENT PENETRATIONS AND BARRIERS-TABLE 6.2.4-1

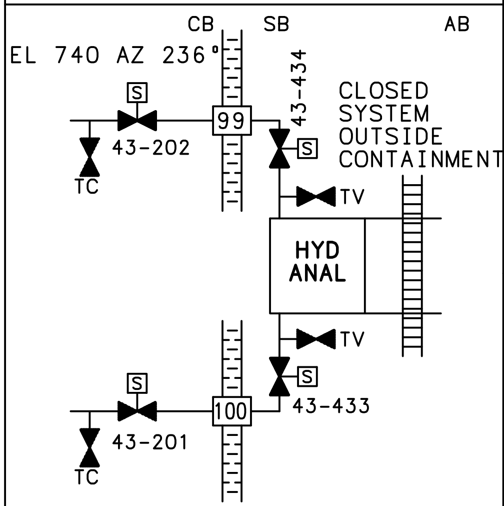
PENETRATION DATA							VALVE DATA																		
																VALVE STATUS									
DETAIL	DWG NUMBER	GEN DES CRITERION	PROCESS FLUID	FLUID STATE	POSS LEAK PATHS	SYSTEM NUMBER AND PENETRATION DESCRIPTION	VALVE LOCATION	VALVE NUMBER	ESF POWER TRAIN	VALVE TYPE	ACTUATOR	PRI ACT MODE	SEC ACT MODE	ISOLATION SIGNAL	STROKE TIME	NORMAL	SHUTDOWN	POST-ACCIDENT	POWER FAILURE	ILRT	POS IND IN MCR	ESF	APP J TEST	ESSENTIAL/NON-ESS	SHIELD BLDG PENETRATION	NOTES
 EL 741 AZ 293°	47W600-105	56	A	C	AB D	UPPER COMP AIR MON INTAKE (90)	CB SB	116 117	B A	GL GL	AO AO	AT AT	RM RM	CV CV	5.0 5.0	O O	O O	C C	C C	C C	Y Y	N N	AC AC	N	MK60M	
 EL 741 AZ 293°	47W600-105	56	A	C	AB D	UPPER COMP AIR MON RETURN (90)	CB SB CB	114 113 115	B A B	GL GL GL	AO AO AO	AT AT AT	RM RM RM	CV CV CV	5.0 5.0 5.0	O O O	O O O	C C C	C C C	O C O	Y Y Y	N N N	AC AC AC	N	MK60M	
 EL 741 AZ 292°	47W331-3	56	A	C	B	ILRT SENSOR LINE (52)	CB SB	506 502	- -	GL GL	M M	LM LM	- -	- -	- -	C C	V V	C C	- -	O O	N N	N N	C C	N	-	

WATTS BAR NUCLEAR PLANT UNIT 1 CONTAINMENT PENETRATIONS AND BARRIERS-TABLE 6.2.4-1

PENETRATION DATA							VALVE DATA																		
DETAIL	DWG NUMBER	GEN DES CRITERION	PROCESS FLUID	FLUID STATE	POSS LEAK PATHS	SYSTEM NUMBER AND PENETRATION DESCRIPTION	VALVE LOCATION	VALVE NUMBER	ESF POWER TRAIN	VALVE TYPE	ACTUATOR	PRI ACT MODE	SEC ACT MODE	ISOLATION SIGNAL	STROKE TIME	VALVE STATUS NORMAL	VALVE STATUS SHUTDOWN	VALVE STATUS POST-ACCIDENT	POWER FAILURE	ILRT	POS IND IN MCR	ESF	APP J TEST	ESSENTIAL/NON-ESS	SHIELD BLDG PENETRATION	NOTES
 EL 741 AZ 292°	47W331-3	56	A	C	B	ILRT SENSOR LINE (52)	CB SB	507 503	- -	GL GL	M M	LM LM	- -	- -	- -	C C	V V	C C	- -	O O	N N	N N	C C	N	-	
 EL 741 AZ 292°	47W600-89 47W331-3	-	A	C	B	dP SENSOR (30)	SB SB SB SB SB SB SB	SNSR. 44C1 44C2 311C1 311C2 49A 311A	- - - - - - -	- GL GL GL GL GL GL	- M M M M M M	- LM LM LM LM LM LM	- - - - - - -	- - - - - - -	- - - - - - -	O C C C C O O	O V V V V O O	O C C C C O O	- - - - - - -	- C C C C O O	- N N N N N N	N N N N N N N	A A - A - A A	E	-	12,13
 EL 741 AZ 291°	47W600-89	56	A	C	B	dP SENSOR (30)	CB SB SB	134 135 133A	B A -	GL GL GL	SO SO M	AT AT LM	RM RM -	PA PA -	10.0 10.0 -	O O O	O O O	C C O	C C -	C C O	Y Y N	N N N	AC AC AC	N	-	

WATTS BAR NUCLEAR PLANT UNIT 1 CONTAINMENT PENETRATIONS AND BARRIERS-TABLE 6.2.4-1

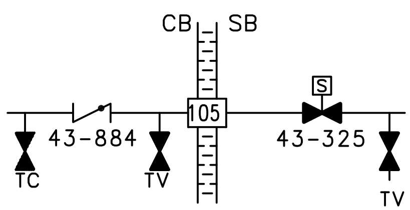
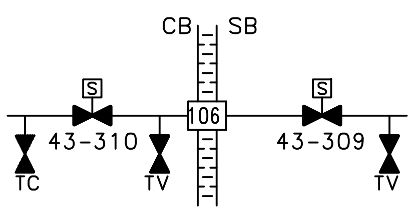
PENETRATION DATA							VALVE DATA																		
DETAIL	DWG NUMBER	GEN DES CRITERION	PROCESS FLUID	FLUID STATE	POSS LEAK PATHS	SYSTEM NUMBER AND PENETRATION DESCRIPTION	VALVE LOCATION	VALVE NUMBER	ESF POWER TRAIN	VALVE TYPE	ACTUATOR	PRI ACT MODE	SEC ACT MODE	ISOLATION SIGNAL	STROKE TIME	VALVE STATUS NORMAL	VALVE STATUS SHUTDOWN	VALVE STATUS POST-ACCIDENT	POWER FAILURE	ILRT	POS IND IN MCR	ESF	APP J TEST	ESSENTIAL/NON-ESS	SHIELD BLDG PENETRATION	NOTES
X-98 EL 741 AZ 290°	48W406	-	-	-	-	SPARE	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	A	-	-	
EL 740 AZ 236°	47W625-11	56	A	C	BD	HYDROGEN ANALYZER TR-A (43)	CB SB	202 434	A A	GL GL	SO SO	RM RM	- -	- -	- -	C C	C C	O O	C C	C C	Y Y	N N	AC AC	E E	MK-54 -	20
EL 740 AZ 237°	47W625-11	56	A	C	BD	HYDROGEN ANALYZER TR-A (43)	CB SB	201 433	A A	GL GL	SO SO	RM RM	- -	- -	- -	C C	C C	O O	C C	C C	Y Y	N N	AC AC	E E	MK-54 -	20



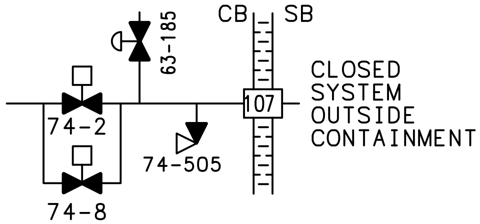
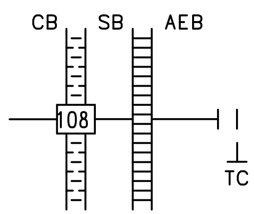
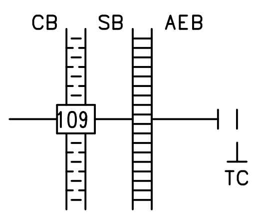
WATTS BAR NUCLEAR PLANT UNIT 1 CONTAINMENT PENETRATIONS AND BARRIERS-TABLE 6.2.4-1

PENETRATION DATA							VALVE DATA																			
																VALVE STATUS										
DETAIL	DWG NUMBER	GEN DES CRITERION	PROCESS FLUID	FLUID STATE	POSS LEAK PATHS	SYSTEM NUMBER AND PENETRATION DESCRIPTION	VALVE LOCATION	VALVE NUMBER	ESF POWER TRAIN	VALVE TYPE	ACTUATOR	PRI ACT MODE	SEC ACT MODE	ISOLATION SIGNAL	STROKE TIME	NORMAL	SHUTDOWN	POST-ACCIDENT	POWER FAILURE	ILRT	POS IND IN MCR	ESF	APP J TEST	ESSENTIAL/NON-ESS	SHIELD BLDG PENETRATION	NOTES
X-101 EL 723'6" AZ 288°	48W406	-	-	-	-	SPARE	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	A	-	-	
X-102 EL 728 AZ 236°	48W406	-	-	-	-	SPARE	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	A	-	-	
X-103 EL 723'6" AZ 287°	48W406	-	-	-	-	SPARE	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	A	-	-	

WATTS BAR NUCLEAR PLANT UNIT 1 CONTAINMENT PENETRATIONS AND BARRIERS–TABLE 6.2.4–1

PENETRATION DATA							VALVE DATA																			
DETAIL	DWG NUMBER	GEN DES CRITERION	PROCESS FLUID	FLUID STATE	POSS LEAK PATHS	SYSTEM NUMBER AND PENETRATION DESCRIPTION	VALVE LOCATION	VALVE NUMBER	ESF POWER TRAIN	VALVE TYPE	ACTUATOR	PRI ACT MODE	SEC ACT MODE	ISOLATION SIGNAL	STROKE TIME	VALVE STATUS NORMAL	VALVE STATUS SHUTDOWN	VALVE STATUS POST-ACCIDENT	VALVE STATUS POWER FAILURE	ILRT	POS IND IN MCR	ESF	APP J TEST	ESSENTIAL/NON-ESS	SHIELD BLDG PENETRATION	NOTES
X-104 EL 728 AZ 237°	48W406	-	-	-	-	SPARE	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	A	-	-	
 EL 722'6" AZ 288°	47W625-15	56	A	C	AB D	PAS CONT AIR RTRN TR-B (43)	CB SB	884 325	- B	CK GL	- SO	SA RM	- LM	- -	- -	C C	C C	V V	- C	C C	N Y	N N	AC AC	E	MK33	19
 EL 727 AZ 236°	47W625-15	56	W	H	AB D	PAS HOT LEG 3 TR-B (43)	CB SB	310 309	B B	GL GL	SO SO	RM RM	LM LM	- -	- -	C C	C C	V V	C C	C C	Y Y	N N	AC AC	E	MK48	19

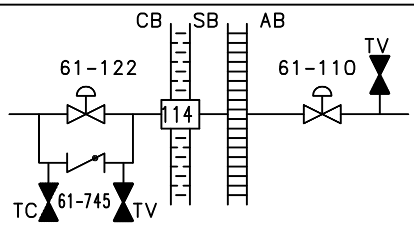
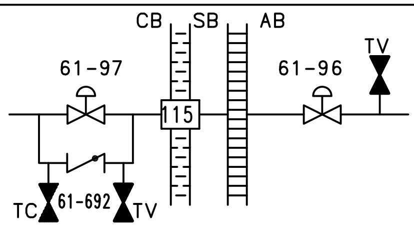
WATTS BAR NUCLEAR PLANT UNIT 1 CONTAINMENT PENETRATIONS AND BARRIERS-TABLE 6.2.4-1

PENETRATION DATA							VALVE DATA																			
																VALVE STATUS										
DETAIL	DWG NUMBER	GEN DES CRITERION	PROCESS FLUID	FLUID STATE	POSS LEAK PATHS	SYSTEM NUMBER AND PENETRATION DESCRIPTION	VALVE LOCATION	VALVE NUMBER	ESF POWER TRAIN	VALVE TYPE	ACTUATOR	PRI ACT MODE	SEC ACT MODE	ISOLATION SIGNAL	STROKE TIME	NORMAL	SHUTDOWN	POST-ACCIDENT	POWER FAILURE	ILRT	POS IND IN MCR	ESF	APP J TEST	ESSENTIAL/NON-ESS	SHIELD BLDG PENETRATION	NOTES
 EL 717'11.5" AZ 300°30'	47W810-1 47W432-2	55	W	H	BD	RHR SUPPLY TO PUMPS (74,63)	CB CB CB CB	2 8 505 185	B A - A	GA GA RV GL	MO MO - AO	RM RM - RM	LM LM - -	- - - PA	- - - 10.0	C C C V	V V C V	V V V C	AI AI - C	O C C C	Y Y N Y	N N N N	- - - -	E	MK38	
 EL 711'6" AZ 218°	47W435-18 -22	-	A	C	AB E	MAINT. PORT	AB	-	-	BL	M	LM	-	-	-	C	V	C	-	C	N	N	AB	N	AS108	
 EL 711'6" AZ 222°	47W435-18 -22	-	A	C	AB E	MAINT. PORT	AB	-	-	BL	M	LM	-	-	-	C	V	C	-	C	N	N	AB	N	AS109	

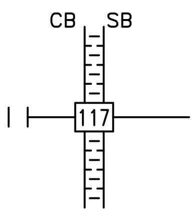
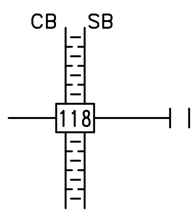
WATTS BAR NUCLEAR PLANT UNIT 1 CONTAINMENT PENETRATIONS AND BARRIERS-TABLE 6.2.4-1

PENETRATION DATA							VALVE DATA																			
DETAIL	DWG NUMBER	GEN DES CRITERION	PROCESS FLUID	FLUID STATE	POSS LEAK PATHS	SYSTEM NUMBER AND PENETRATION DESCRIPTION	VALVE LOCATION	VALVE NUMBER	ESF POWER TRAIN	VALVE TYPE	ACTUATOR	PRI ACT MODE	SEC ACT MODE	ISOLATION SIGNAL	STROKE TIME	VALVE STATUS NORMAL	VALVE STATUS SHUTDOWN	VALVE STATUS POST-ACCIDENT	VALVE STATUS POWER FAILURE	ILRT	POS IND IN MCR	ESF	APP J TEST	ESSENTIAL/NON-ESS	SHIELD BLDG PENETRATION	NOTES
X-110 EL 711'6" AZ 209°	48N406 47W435-22	-	-	-	B	SPARE	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	A	-	MK32	
X-111 EL 845'9" AZ 90°	48W406	-	-	-	-	SPARE	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	A	-	-	
X-112 EL 845'9" AZ 270°	48W406	-	-	-	-	SPARE	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	A	-	-	

WATTS BAR NUCLEAR PLANT UNIT 1 CONTAINMENT PENETRATIONS AND BARRIERS–TABLE 6.2.4-1

PENETRATION DATA							VALVE DATA																			
																VALVE STATUS										
DETAIL	DWG NUMBER	GEN DES CRITERION	PROCESS FLUID	FLUID STATE	POSS LEAK PATHS	SYSTEM NUMBER AND PENETRATION DESCRIPTION	VALVE LOCATION	VALVE NUMBER	ESF POWER TRAIN	VALVE TYPE	ACTUATOR	PRI ACT MODE	SEC ACT MODE	ISOLATION SIGNAL	STROKE TIME	NORMAL	SHUTDOWN	POST-ACCIDENT	POWER FAILURE	ILRT	POS IND IN MCR	ESF	APP J TEST	ESSENTIAL/NON-ESS	SHIELD BLDG PENETRATION	NOTES
X-113 EL 740 AZ 216°	48W406	–	–	–	–	SPARE	–	–	–	–	–	–	–	–	–	–	–	–	–	–	–	–	A	–	–	
 EL 775 AZ 300°	47W814-2	56	G	C	AB D	GLYCOL FLOOR COOLING (61)	CB AB CB	122 110 745	B A –	DI DI CK	AO AO –	AT AT SA	RM RM –	PA PA –	30.0 30.0 –	O O O	O O O	C C V	C C –	O O O	Y Y N	Y Y Y	C C C	N	MK110	8
 EL 771'6" AZ 300°	47W814-2	56	G	C	AB D	GLYCOL FLOOR COOLING (61)	CB AB CB	97 96 692	B A –	DI DI CK	AO AO –	AT AT SA	RM RM –	PA PA –	30.0 30.0 –	O O O	O O O	C C V	C C –	O O O	Y Y N	Y Y Y	C C C	N	MK111	8

WATTS BAR NUCLEAR PLANT UNIT 1 CONTAINMENT PENETRATIONS AND BARRIERS-TABLE 6.2.4-1

PENETRATION DATA							VALVE DATA																			
																VALVE STATUS										
DETAIL	DWG NUMBER	GEN DES CRITERION	PROCESS FLUID	FLUID STATE	POSS LEAK PATHS	SYSTEM NUMBER AND PENETRATION DESCRIPTION	VALVE LOCATION	VALVE NUMBER	ESF POWER TRAIN	VALVE TYPE	ACTUATOR	PRI ACT MODE	SEC ACT MODE	ISOLATION SIGNAL	STROKE TIME	NORMAL	SHUTDOWN	POST-ACCIDENT	POWER FAILURE	ILRT	POS IND IN MCR	ESF	APP J TEST	ESSENTIAL/NON-ESS	SHIELD BLDG PENETRATION	NOTES
X-116 EL 760 AZ 300°	48W406	–	–	–	–	SPARE	–	–	–	–	–	–	–	–	–	–	–	–	–	–	–	–	A	–	MK7	
 EL 758 AZ 300°	47W301-1	–	A	C	AB	MAINT. PORT	CB	–	–	BL	M	LM	–	–	–	C	V	C	–	C	N	N	AB	N	MK8	
 EL 708'6" AZ 209°	72-4333-318 47W862-1	–	W	C	AB C	LAYUP WATER TREATMENT (41)	SB	–	–	BL	M	LM	–	–	–	C	O	C	–	O	N	N	AB	N	MK18	27

WATTS BAR NUCLEAR PLANT UNIT 1 CONTAINMENT PENETRATIONS AND BARRIERS-TABLE 6.2.4-1

PENETRATION DATA							VALVE DATA																			
DETAIL	DWG NUMBER	GEN DES CRITERION	PROCESS FLUID	FLUID STATE	POSS LEAK PATHS	SYSTEM NUMBER AND PENETRATION DESCRIPTION	VALVE LOCATION	VALVE NUMBER	ESF POWER TRAIN	VALVE TYPE	ACTUATOR	PRI ACT MODE	SEC ACT MODE	ISOLATION SIGNAL	STROKE TIME	VALVE STATUS NORMAL	VALVE STATUS SHUTDOWN	VALVE STATUS POST-ACCIDENT	POWER FAILURE	ILRT	POS IND IN MCR	ESF	APP J TEST	ESSENTIAL/NON-ESS	SHIELD BLDG PENETRATION	NOTES
X-119 EL 844'5" AZ 90°	48W406	-	-	-	-	SPARE	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	A	-	-	
X-120 EL 844'5" AZ 270°	48W406	-	-	-	-	SPARE	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	A	-	-	

WATTS BAR NUCLAR PLANT UNIT 1 - CONTAINMENT PENETRATIONS AND BARRIERS

TABLE 6.2.4-1

ABBREVIATIONS FOR TABLE

VALVE TYPE

GL = GLOBE

GA = GATE

BF = BUTTERFLY

CK = CHECK

A = ANGLE

RV = RELIEF

DI = DIAPHRAGM

BL = BLIND FLANGE

TC = TEST CONNECTION

TD = TEST DRAIN

TV = TEST VENT

WV = WAFER VALVE

D = DOOR

BA = BALL

PG = PLUG

ACTUATORS

AO = AIR OPERATED

MO = MOTOR OPERATED

SO = SOLENOID OPERATED

ILRT

C = CLOSED

O = OPEN

ISOLATION SIGNAL

CV = CONTAINMENT VENT ISOLATION

PA = PHASE A

PB = PHASE B

PR = PUMP RUN

CP = CONTAINMENT PRESSURE

SI = SAFETY INJECTION

AFIS = AUXILIARY FEEDWATER ISOLATION SIGNAL

MS = MAIN STEAM ISOLATION SIGNAL

FW = FEEDWATER ISOLATION SIGNAL

ABBREVIATIONS FOR TABLE

ACTUATION MODE

RM = REMOTE MANUAL

LM = LOCAL MANUAL

SA = SELF ACTUATOR

AT = AUTO TRIP

AO = AUTO OPEN

VALVE POSITIONS

O = OPEN

C = CLOSED

V = VARIES

AI = AS IS

LC = LOCKED CLOSED

PROCESS FLUID STATE

C = COLD

H = HOT

POSSIBLE LEAKAGE PATHS

SEE FSAR SECTION 6.2.4.3.1

FLUID

A = AIR

W = WATER

S = STEAM

N = NITROGEN

I = ICE

G = GLYCOL

BUILDING STRUCTURES

CB = CONTAINMENT BUILDING

SB = SHIELD BUILDING

AB = AUXILIARY BUILDING

AEB = ADDITIONAL EQUIPMENT BUILDING

1. A dash is printed in data spaces that do not apply to the penetration being described.

2. An m at the end of a containment or shield building penetration number indicates the penetration has more than one line going through it.

3. Piping systems isolated by a Containment Ventilation (CV) signal are also isolated by a Phase A signal.

4. Piping inside the containment vessel has five branches. Identical isolation valves are installed in each branch. One valve is normally open. Three other valves are normally closed and also receive Phase A signal. The fifth valve is a relief valve which relieves into containment.

5. The piping is joined by a line leading to the nitrogen supply outside of the shield building. This line is isolated by an air operated gate valve with Phase A isolation.

6. Ducting outside the shield building has two branches. Identical secondary containment isolation valves are installed in each branch.

7. Piping outside of the shield building has nine branches. Each line is isolated by a relief valve.

8. Inside the containment vessel a line is provided for pressure relief of the volume between the isolation valves. This line is isolated by a check valves.

9. The piping is joined by a line used for S.I.S. testing outside of the shield building. This line is isolated by an air operated globe valve with remote manual actuation from the main control room.

10. The piping has a bypass line attached to it inside the annulus. This bypass line is isolated by a locked closed gate valve with local manual operation.

11. Isolation inside containment is accomplished by a closed system.

12. The DP sensors are closed systems outside of containment that are attached directly to the containment. No in line isolation valves are employed for these sensors refer to SECTION 6.2.4.3.

13. Post Accident Monitoring System (PAMS) wide range pressure sensors also monitor pressure through this penetration.

14. The secondary containment isolation valve(s), which represent an additional boundary outside containment in HVAC systems, are used to prevent bypass leakage from the annulus.

15. Isolation valves can be manually actuated electrically or pneumatically.

16. Two blind flanges are employed in the annulus, one acts as a primary containment isolation barrier and other as a secondary containment isolation barrier. The primary and secondary containment penetrations are not connected except for the type A test pressurization.

17. Outside containment, relief valves release into this line in the event of SIS overpressure, thus the system is almost always closed, but could open for pressure relief anytime SIS is on.

18. Outside containment, a one inch line ties in between the outer isolation valve and primary containment. This line is isolated by a manually locked closed containment isolation valve.

19. The containment isolation valves in this line require simultaneous open signals from two separate switches in order to open. One switch is located in the Main Control Room and the other is located in the Remote Sampling Station.

20. This penetration contains both the intake and return lines for a hydrogen analyzer. These lines form a closed system outside containment and each has a containment isolation valve inside and outside containment.

21. This line transmits pressure form the primary coolant system to pressure instrumentation for determining the reactor vessel fluid level. The line is fluid filled and double diaphragmed to prevent communication between the primary system fluid and the Auxiliary Building. No primary system fluid travels through the penetration since the inner diaphragm is located near the Reactor Vessel. Since double diaphragms are employed for containment isolation, no containment isolation valves are used.

22. This line joins the secondary side of the steam generator(s) inside containment and is therefore considered a closed system inside containment. The isolation valves which exist outboard of containment are not leakrate tested due to flooding of the steam generators post LOCA.

WATTS BAR NUCLAR PLANT UNIT 1 - CONTAINMENT PENERTRATIONS AND BARRIERS

TABLE 6.2.4-1

23. Between primary containment and the outer isolation valve in this line, an auxiliary feedwater line adjoins. The auxiliary feedwater line has isolation valves which are isolated in the same manner as the auxiliary feedwater isolation valves listed for X-40A and X-40B.

24. Piping for X-64,X-65, X-66, and X-67 has been capped in the Annulus. Therefore, the containment isolation functions associated are no longer required. Additionally, the components associated with these penetrations are only required to be type A tested in accordance with the Appendix J program.

25. Azimuth information for these lines can be found in Chicago Bridge & Iron Co. Steel Containment Vessel Data Package (RIMS # B48850425301).

26. No type C leakage rate test is performed on this line, but a water inventory leakage rate test is performed on the MOV located outside containment. This water leakage test assures that the water leg, located immediately inboard of the MOV, will maintain a sufficient inventory to maintain a pressure of at least 1.1 times the maximum calculated containment accident pressure on the MOV for 30 days after an accident.

27. Penetration X-118 is opened by removing the blind flange in the annulus to allow the water which has collected in the annulus from leakage through the penetrations or seepage through the concrete to drain the Reactor Building floor and equipment drain sump.

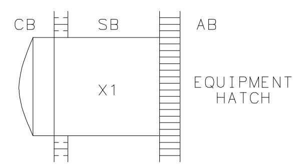
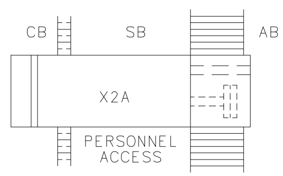
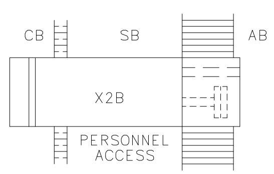
28. This line transmits pressure from the primary coolant system to pressure instrumentation. The line is fluid filled and double diaphragmed to prevent communication between the primary system fluid and the Auxiliary Building. No primary system fluid travels through the penetration since the inner diaphragm is located near the reactor vessel. Since double diaphragms are employed for containment isolation, no containment isolation valves are used.

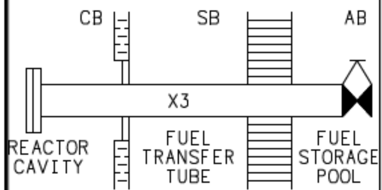
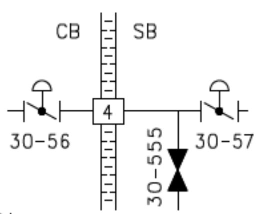
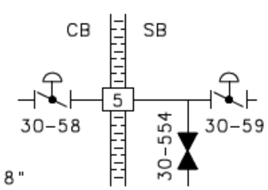
29. System 26 HPFP line supplies the cable tray interaction (RB Annulus) through Shield Building Pen. MK 103 located at the same EL and AZ as containment Pen. X-88 (Ref. DWG. 47W850-9).

30. System 26 HPFP line supplies the stand pipe (RB Annulus) through Shield Building Pen. MK 104 located at the same EL and AZ as containment Pen X-89 (Ref. Dwg. 47W850-9).

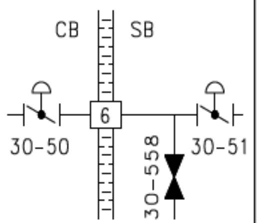
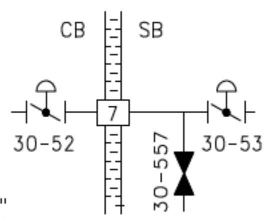
31. The testing program defines position of the inner and outer doors associated with X-2A and X-2B during the performance of 1-SI-0-703, containment integrated leak rate test, for Appendix J requirements of 10 CFR 50.

WATTS BAR NUCLEAR PLANT CONTAINMENT PENETRATIONS AND BARRIERS-TABLE 6.2.4-1

PENETRATION DATA							VALVE DATA																		
															VALVE STATUS										
DETAILS	DWG NUMBER	GEN DES CRITERION	PROCESS FLUID	FLUID STATE	POSS LEAK PATHS	SYSTEM NUMBER AND PENETRATION DESCRIPTION	VALVE LOCATION	VALVE NUMBER	ESF POWER TRAIN	VALVE TYPE	ACTUATOR	PRI ACT MODE	SEC ACT MODE	ISOLATION SIGNAL	STROKE TIME	NORMAL	SHUTDOWN	POST-ACCIDENT	POWER FAILURE	ILRT	POS IND IN MCR	ESF	APP J TEST	ESSENTIAL/NON-ESS	SHIELD BLDG PENETRATION	NOTES
 EL 764'7.5" AZ 285°	44N250	–	A	C	AB	EQUIPMENT HATCH	–	–	–	D	M	LM	–	–	–	C	V	C	–	C	–	N	AB	N	–	
 EL 719'3.63" AZ 62°	44N240	–	A	C	AD	PERSONNEL AIR LOCK	IN	–	–	D	M	LM	–	–	–	C	O	C	–	O	Y	N	AB	N	MK40	
							OUT	–	–	D	M	LM	–	–	–	C	O	C	–	C	Y	N	AB	N		
 EL 760'3.63" AZ 255°	44N240	–	A	C	AD	PERSONNEL AIR LOCK	IN	–	–	D	M	LM	–	–	–	C	O	C	–	O	Y	N	AB	N	MK40	
							OUT	–	–	D	M	LM	–	–	–	C	O	C	–	C	Y	N	AB	N		

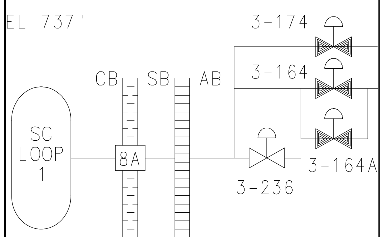
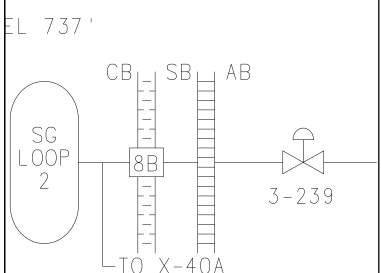
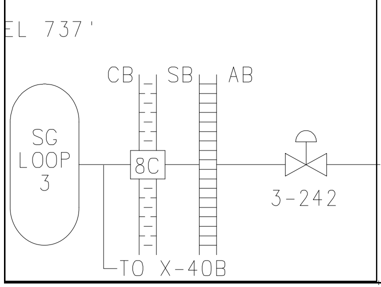
WATTS BAR NUCLEAR PLANT CONTAINMENT PENETRATIONS AND BARRIERS-TABLE 6.2.4-1																										
PENETRATION DATA							VALVE DATA																			
															VALVE STATUS											
DETAILS	DWG NUMBER	GEN DES CRITERION	PROCESS FLUID	FLUID STATE	POSS LEAK PATHS	SYSTEM NUMBER AND PENETRATION DESCRIPTION	VALVE LOCATION	VALVE NUMBER	ESF POWER TRAIN	VALVE TYPE	ACTUATOR	PRI ACT MODE	SEC ACT MODE	ISOLATION SIGNAL	STROKE TIME	NORMAL	SHUTDOWN	POST-ACCIDENT	POWER FAILURE	ILRT	POS IND IN MCR	ESF	APP J TEST	ESSENTIAL/NON-ESS	SHIELD BLDG PENETRATION	NOTES
 EL 711'3" AZ 261° 50'	47W455-1 72-4333 -PS1 (CB & 1 CONTRACT #75320)	56	W	C	AB D	FUEL TRANSFER TUBE	AB	–	–	BL	M	LM	–	–	–	C	V	C	–	C	N	N	AB	N	MK62	
 EL 735 AZ 36°30'	47W866-1	56	A	C	AB CE	LOWER COMPARTMENT PURGE AIR EXHAUST (30)	CB SB SB	56 57 555	A B -	BF BF GL	AO AO M	AT AT LM	RM RM -	CV CV -	40 40 -	V V LC	O O LC	C C LC	C C -	C C C	Y Y N	N N N	AC AC AC	N	F-005	3, 6, 14
 EL 739'8" AZ 116°	47W866-1	56	A	C	AB CE	INST RM PURGE AIR EXHAUST (30)	CB SB SB	58 59 554	B A -	BF BF GL	AO AO M	AT AT LM	RM RM -	CV CV -	40 40 -	V V LC	O O LC	C C LC	C C -	C C C	Y Y N	N N N	AC AC AC	N	F-005	3, 6, 14

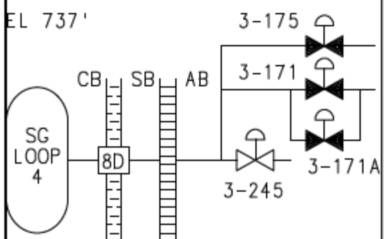
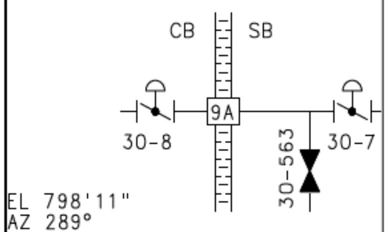
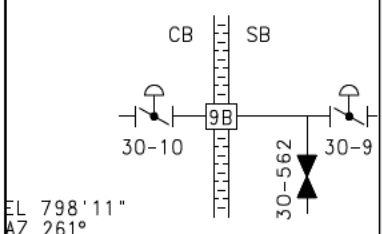
WATTS BAR NUCLEAR PLANT CONTAINMENT PENETRATIONS AND BARRIERS-TABLE 6.2.4-1

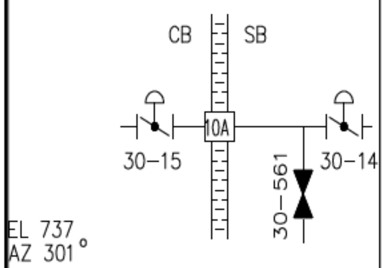
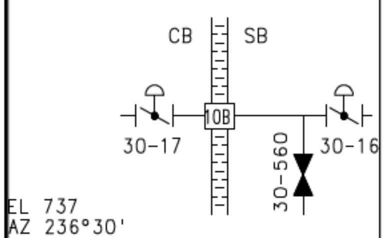
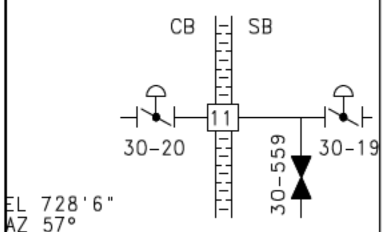
PENETRATION DATA							VALVE DATA																			
															VALVE STATUS											
DETAILS	DWG NUMBER	GEN DES CRITERION	PROCESS FLUID	FLUID STATE	POSS LEAK PATHS	SYSTEM NUMBER AND PENETRATION DESCRIPTION	VALVE LOCATION	VALVE NUMBER	ESF POWER TRAIN	VALVE TYPE	ACTUATOR	PRI ACT MODE	SEC ACT MODE	ISOLATION SIGNAL	STROKE TIME	NORMAL	SHUTDOWN	POST-ACCIDENT	POWER FAILURE	ILRT	POS IND IN MCR	ESF	APP J TEST	ESSENTIAL/NON-ESS	SHIELD BLDG PENETRATION	NOTES
 EL 749'3" AZ 293°	47W866-1	56	A	C	AB CE	UPPER COMPARTMENT PURGE AIR EXHAUST (30)	CB SB SB	50 51 558	B A -	BF BF GL	AO AO M	AT AT LM	RM RM -	CV CV -	4 0 4 0 -	V V LC	O O LC	C C LC	C C -	C C C	Y Y N	N N N	AC AC AC	N	F-005	3, 6, 14
 EL 751'4" AZ 249°30'	47W866-1	56	A	C	AB CE	UPPER COMPARTMENT PURGE AIR EXHAUST (30)	CB SB SB	52 53 557	A B -	BF BF GL	AO AO M	AT AT LM	RM RM -	CV CV -	4 0 4 0 -	V V LC	O O LC	C C LC	C C -	C C C	Y Y N	N N N	AC AC AC	N	F-005	3,6, 14
X-8 EL 790'0" AZ 266°	48W406	–	–	–	–	SPARE	–	–	–	–	–	–	–	–	–	–	–	–	–	–	–	–	A	–	–	

WATTS BAR NUCLEAR PLANT CONTAINMENT PENETRATIONS AND BARRIERS-TABLE 6.2.4-1

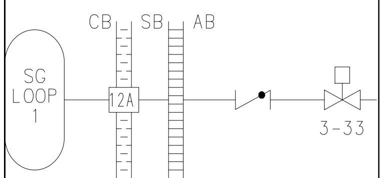
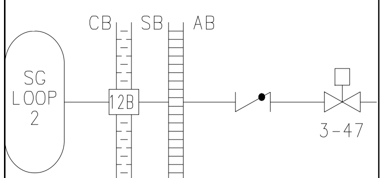
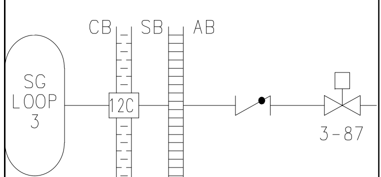
PENETRATION DATA | VALVE DATA (VALVE STATUS: NORMAL, SHUTDOWN, POST-ACCIDENT, POWER FAILURE)

DETAILS	DWG NUMBER	GEN DES CRITERION	PROCESS FLUID	FLUID STATE	POSS LEAK PATHS	SYSTEM NUMBER AND PENETRATION DESCRIPTION	VALVE LOCATION	VALVE NUMBER	ESF POWER TRAIN	VALVE TYPE	ACTUATOR	PRI ACT MODE	SEC ACT MODE	ISOLATION SIGNAL	STROKE TIME	NORMAL	SHUTDOWN	POST-ACCIDENT	POWER FAILURE	ILRT	POS IND IN MCR	ESF	APP J TEST	ESSENTIAL/NON-ESS	SHIELD BLDG PENETRATION	NOTES
	47W803 -1,2	57	W	H	BCE	FEEDWATER BYPASS (3)	AB	236 174 164 164A	A,B B A A	GL GL GL A	AO AO AO AO	AT RM RM RM	LM LM LM LM	FW – – –	– – – –	O C C C	C C C C	C V V V	C C O C	C C C C	Y Y Y Y	N N N N	A A A A	N	MK112	22,23,25
	47W803 -1,2	57	W	H	BCE	FEEDWATER BYPASS (3)	AB	239	A,B	GL	AO	AT	LM	FW	–	O	C	C	C	C	Y	N	A	N	MK113	22 25
	47W803 -1,2	57	W	H	BCE	FEEDWATER BYPASS (3)	AB	242	A,B	GL	AO	AT	LM	FW	–	O	C	C	C	C	Y	N	A	N	MK114	22,25

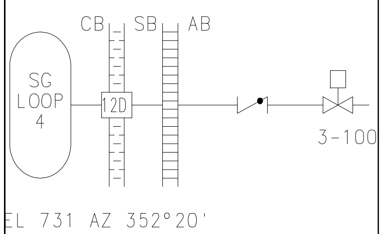
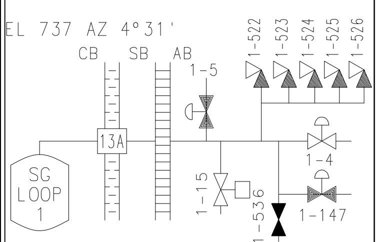
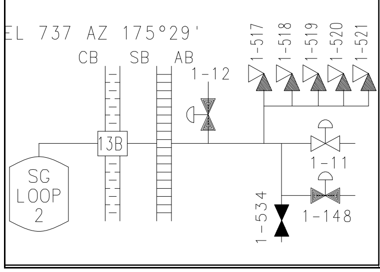
WATTS BAR NUCLEAR PLANT CONTAINMENT PENETRATIONS AND BARRIERS-TABLE 6.2.4-1																										
PENETRATION DATA								VALVE DATA								VALVE STATUS										
DETAILS	DWG NUMBER	GEN DES CRITERION	PROCESS FLUID	FLUID STATE	POSS LEAK PATHS	SYSTEM NUMBER AND PENETRATION DESCRIPTION	VALVE LOCATION	VALVE NUMBER	ESF POWER TRAIN	VALVE TYPE	ACTUATOR	PRI ACT MODE	SEC ACT MODE	ISOLATION SIGNAL	STROKE TIME	NORMAL	SHUTDOWN	POST-ACCIDENT	POWER FAILURE	ILRT	POS IND IN MCR	ESF	APP J TEST	ESSENTIAL/NON-ESS	SHIELD BLDG PENETRATION	NOTES
	47W803 -1,2	57	W	H	BC E	FEEDWATER BYPASS (3)	AB	245	A,B	GL	AO	AT	LM	FW	–	O	C	C	C	C	Y	N	A	N	MK115	22, 23, 25
								175	A	GL	AO	RM	LM	–	–	C	C	V	C	C	Y	N	A			
								171	B	GL	AO	RM	LM	–	–	C	C	V	O	C	Y	N	A			
								171A	B	A	AO	RM	LM	–	–	C	C	V	C	C	Y	N	A			
	47W866-1	56	A	C	AB CE	UPPER COMPARTMENT PURGE AIR SUPPLY (30)	CB	8	B	BF	AO	AT	RM	CV	4 0	V	O	C	C	C	Y	N	AC	N	F-002	3,6,14
							SB	7	A	BF	AO	AT	RM	CV	4 0	V	O	C	C	C	Y	N	AC			
							SB	563	-	GL	M	LM	-	-	-	LC	LC	LC	-	C	N	N	AC			
	47W866-1	56	A	C	AB CE	UPPER COMPARTMENT PURGE AIR SUPPLY (30)	CB	10	A	BF	AO	AT	RM	CV	4 0	V	O	C	C	C	Y	N	AC	N	F-002	3,6,14
							SB	9	B	BF	AO	AT	RM	CV	4 0	V	O	C	C	C	Y	N	AC			
							SB	562	-	GL	M	LM	-	-	-	LC	LC	LC	-	C	N	N	AC			

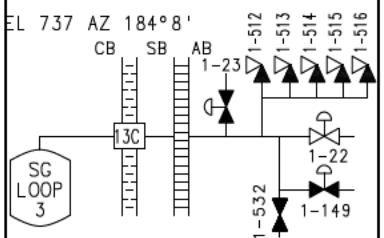
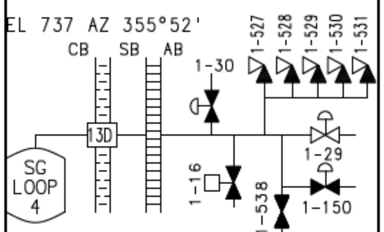
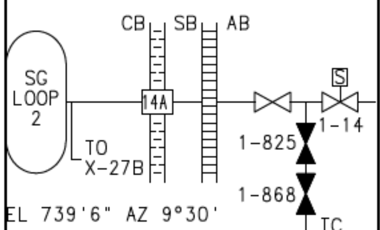
WATTS BAR NUCLEAR PLANT CONTAINMENT PENETRATIONS AND BARRIERS-TABLE 6.2.4-1																										
PENETRATION DATA							VALVE DATA																			
															VALVE STATUS											
DETAILS	DWG NUMBER	GEN DES CRITERION	PROCESS FLUID	FLUID STATE	POSS LEAK PATHS	SYSTEM NUMBER AND PENETRATION DESCRIPTION	VALVE LOCATION	VALVE NUMBER	ESF POWER TRAIN	VALVE TYPE	ACTUATOR	PRI ACT MODE	SEC ACT MODE	ISOLATION SIGNAL	STROKE TIME	NORMAL	SHUTDOWN	POST-ACCIDENT	POWER FAILURE	ILRT	POS IND IN MCR	ESF	APP J TEST	ESSENTIAL/NON-ESS	SHIELD BLDG PENETRATION	NOTES
	47W866-1	56	A	C	AB CE	LOWER COMPARTMENT PURGE AIR SUPPLY (30)	CB SB SB	15 14 561	B A -	BF BF GL	AO AO M	AT AT LM	RM RM -	CV CV -	4 0 4 0 -	V V LC	O O LC	C C LC	C C -	C C C	Y Y N	N N N	AC AC AC	N	F-002	3,6,14
	47W866-1	56	A	C	AB CE	LOWER COMPARTRMENT PURGE AIR SUPPLY (30)	CB SB SB	17 16 560	A B -	BF BF GL	AO AO M	AT AT LM	RM RM -	CV CV -	4 0 4 0 -	V V LC	C C LC	C C LC	C C -	C C C	Y Y N	N N N	AC AC AC	N	F-002	3,6,14
	47W866-1	56	A	C	AB CE	INST RM PURGE AIR SUPPLY (30)	CB SB SB	20 19 559	A B -	BF BF GL	AO AO M	AT AT LM	RM RM -	CV CV -	4 0 4 0 -	V V LC	O O LC	C C LC	C C -	C C C	Y Y N	N N N	AC AC AC	N	F-002	3,6,14

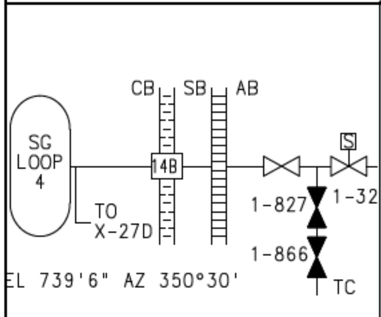
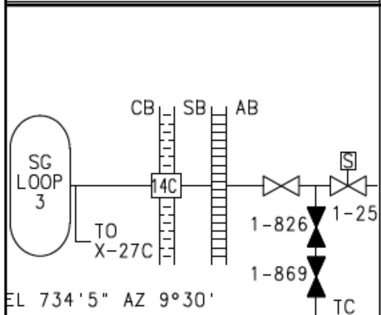
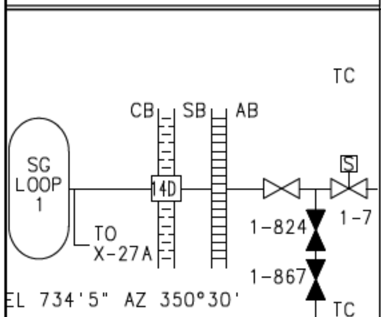
WATTS BAR NUCLEAR PLANT CONTAINMENT PENETRATIONS AND BARRIERS-TABLE 6.2.4-1

PENETRATION DATA								VALVE DATA																		
																VALVE STATUS										
DETAILS	DWG NUMBER	GEN DES CRITERION	PROCESS FLUID	FLUID STATE	POSS LEAK PATHS	SYSTEM NUMBER AND PENETRATION DESCRIPTION	VALVE LOCATION	VALVE NUMBER	ESF POWER TRAIN	VALVE TYPE	ACTUATOR	PRI ACT MODE	SEC ACT MODE	ISOLATION SIGNAL	STROKE TIME	NORMAL	SHUTDOWN	POST-ACCIDENT	POWER FAILURE	ILRT	POS IND IN MCR	ESF	APP J TEST	ESSENTIAL/NON-ESS	SHIELD BLDG PENETRATION	NOTES
 EL 731 AZ 7°40'	47W803-1	57	W	H	BCE	FEEDWATER (3)	AB	33	A	GA	MO	AT	LM	FW	–	O	C	C	AI	C	Y	N	A	N	MK70	22
 EL 731 AZ 172°20'	47W803-1	57	W	H	BCE	FEEDWATER (3)	AB	47	B	GA	MO	AT	LM	FW	–	O	C	C	AI	C	Y	N	A	N	MK69	22
 EL 731 AZ 187°40'	47W803-1	57	W	H	BCE	FEEDWATER (3)	AB	87	A	GA	MO	AT	LM	FW	–	O	C	C	AI	C	Y	N	A	N	MK68	22

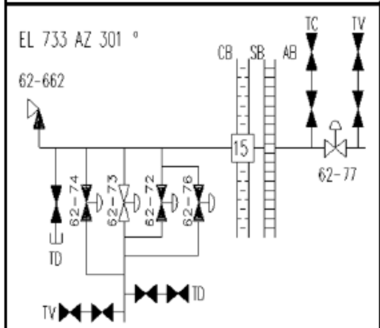
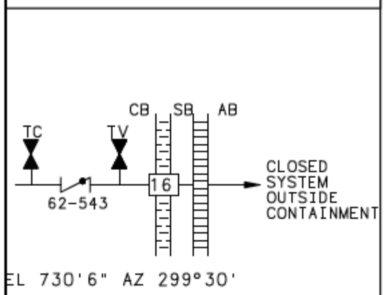
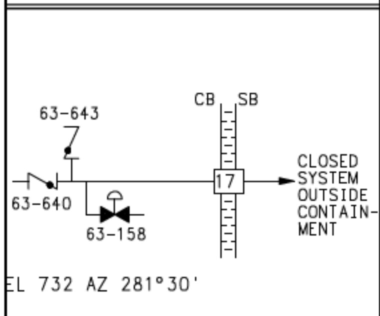
WATTS BAR NUCLEAR PLANT CONTAINMENT PENETRATIONS AND BARRIERS-TABLE 6.2.4-1

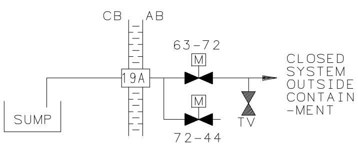
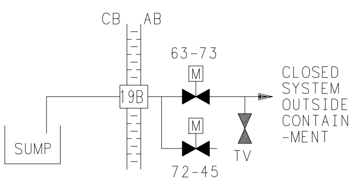
PENETRATION DATA							VALVE DATA								VALVE STATUS											
DETAILS	DWG NUMBER	GEN DES CRITERION	PROCESS FLUID	FLUID STATE	POSS LEAK PATHS	SYSTEM NUMBER AND PENETRATION DESCRIPTION	VALVE LOCATION	VALVE NUMBER	ESF POWER TRAIN	VALVE TYPE	ACTUATOR	PRI ACT MODE	SEC ACT MODE	ISOLATION SIGNAL	STROKE TIME	NORMAL	SHUTDOWN	POST-ACCIDENT	POWER FAILURE	ILRT	POS IND IN MCR	ESF	APP J TEST	ESSENTIAL/NON-ESS	SHIELD BLDG PENETRATION	NOTES
	47W803-1	57	W	H	BCE	FEEDWATER (3)	AB	100	B	GA	MO	AT	LM	FW	–	O	C	C	AI	C	Y	N	A	N	MK67	22
	47W801-1 47W803-2	57	S	H	BCE	MAIN STEAM (1)	AB	4	A, B	GL	AO	AT	LM	MS	–	O	C	C	C	C	Y	N	A	N	MK66	22
							AB	5	A, B	RV	AO	AT	RM	–	–	C	C	C	C	C	Y	N	A			
							AB	15	A	GA	MO	AT	RM	–	–	O	C	V	AT	C	Y	N	A			
							AB	147	A	GA	AO	AT	RM	MS	–	C	V	C	C	C	Y	N	A			
							AB	522	–	RV	–	SA	–	–	–	C	C	C	–	C	N	N	A			
							AB	523	–	RV	–	SA	–	–	–	C	C	C	–	C	N	N	A			
							AB	524	–	RV	–	SA	–	–	–	C	C	C	–	C	N	N	A			
							AB	525	–	RV	–	SA	–	–	–	C	C	C	–	C	N	N	A			
							AB	526	–	RV	–	SA	–	–	–	C	C	C	–	C	N	N	A			
							AB	536	–	GL	M	LM	–	–	–	C	V	C	–	C	N	N	A			
	47W801-1	57	S	H	BCE	MAIN STEAM (1)	AB	11	A, B	GL	AO	AT	LM	MS	–	O	C	C	C	C	Y	N	A	N	MK65	22
							AB	12	A, B	RV	AO	AT	RM	–	–	C	C	C	C	C	Y	N	A			
							AB	148	B	GA	AO	AT	RM	MS	–	C	V	C	C	C	Y	N	A			
							AB	517	–	RV	–	SA	–	–	–	C	C	C	–	C	N	N	A			
							AB	518	–	RV	–	SA	–	–	–	C	C	C	–	C	N	N	A			
							AB	519	–	RV	–	SA	–	–	–	C	C	C	–	C	N	N	A			
							AB	520	–	RV	–	SA	–	–	–	C	C	C	–	C	N	N	A			
							AB	521	–	RV	–	SA	–	–	–	C	C	C	–	C	N	N	A			
							AB	534	–	GL	M	LM	–	–	–	C	V	C	–	C	N	N	A			

WATTS BAR NUCLEAR PLANT CONTAINMENT PENETRATIONS AND BARRIERS-TABLE 6.2.4-1																										
PENETRATION DATA							VALVE DATA																			
															VALVE STATUS											
DETAILS	DWG NUMBER	GEN DES CRITERION	PROCESS FLUID	FLUID STATE	POSS LEAK PATHS	SYSTEM NUMBER AND PENETRATION DESCRIPTION	VALVE LOCATION	VALVE NUMBER	ESF POWER TRAIN	VALVE TYPE	ACTUATOR	PRI ACT MODE	SEC ACT MODE	ISOLATION SIGNAL	STROKE TIME	NORMAL	SHUTDOWN	POST-ACCIDENT	POWER FAILURE	ILRT	POS IND IN MCR	ESF	APP J TEST	ESSENTIAL/NON-ESS	SHIELD BLDG PENETRATION	NOTES
	47W801-1	57	S	H	BC	MAIN STEAM	AB	22	A, B	GL	AO	AT	LM	MS	–	O	C	C	C	C	Y	N	A	N	MK64	22
					E	(1)	AB	23	A, B	RV	AO	AT	RM	–	–	C	C	C	C	C	Y	N	A			
							AB	149	A	GA	AO	AT	RM	MS	–	C	V	C	C	C	Y	N	A			
							AB	512	–	RV	–	SA	–	–	–	C	C	C	–	C	N	N	A			
							AB	513	–	RV	–	SA	–	–	–	C	C	C	–	C	N	N	A			
							AB	514	–	RV	–	SA	–	–	–	C	C	C	–	C	N	N	A			
							AB	515	–	RV	–	SA	–	–	–	C	C	C	–	C	N	N	A			
							AB	516	–	RV	–	SA	–	–	–	C	C	C	–	C	N	N	A			
							AB	532	–	GL	M	LM	–	–	–	C	V	C	–	C	N	N	A			
	47W801-1	57	S	H	BC	MAIN STEAM	AB	29	A, B	GL	AO	AT	LM	MS	–	O	C	C	C	C	Y	N	A	N	MK63	22
	47W803-2				E	(1)	AB	16	A	GA	MO	AT	RM	_	–	C	C	V	AI	C	Y	N	A			
							AB	30	A, B	RV	AO	AT	RM	_	–	C	_	C	C	C	Y	N	A			
							AB	150	B	GL	AO	AT	RM	MS	–	C	V	C	C	C	Y	N	A			
							AB	527	–	RV	–	SA	–	–	–	C	C	C	–	C	N	N	A			
							AB	528	–	RV	–	SA	–	–	–	C	C	C	–	C	N	N	A			
							AB	529	–	RV	–	SA	–	–	–	C	C	C	–	C	N	N	A			
							AB	530	–	RV	–	SA	–	–	–	C	C	C	–	C	N	N	A			
							AB	531	–	RV	–	SA	–	–	–	C	C	C	–	C	N	N	A			
							AB	538	–	GL	M	LM	–	–	–	C	V	C	–	C	N	N	A			
	47W801-2	57	W	H	BE	STEAM GENERATOR BLOWDOWN (1)	AB	14	A	GL	SO	AT	RM	PA	15.0	O	C	C	C	C	Y	N	A	N	AS14A	22

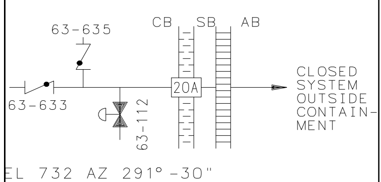
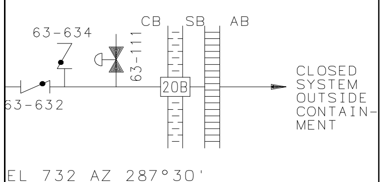
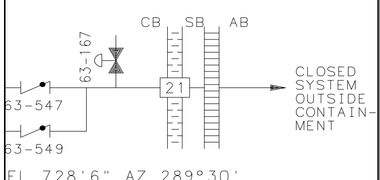
WATTS BAR NUCLEAR PLANT CONTAINMENT PENETRATIONS AND BARRIERS-TABLE 6.2.4-1																										
PENETRATION DATA							VALVE DATA																			
															VALVE STATUS											
DETAILS	DWG NUMBER	GEN DES CRITERION	PROCESS FLUID	FLUID STATE	POSS LEAK PATHS	SYSTEM NUMBER AND PENETRATION DESCRIPTION	VALVE LOCATION	VALVE NUMBER	ESF POWER TRAIN	VALVE TYPE	ACTUATOR	PRI ACT MODE	SEC ACT MODE	ISOLATION SIGNAL	STROKE TIME	NORMAL	SHUTDOWN	POST-ACCIDENT	POWER FAILURE	ILRT	POS IND IN MCR	ESF	APP J TEST	ESSENTIAL/NON-ESS	SHIELD BLDG PENETRATION	NOTES
	47W801-2	57	W	H	BE	STEAM GENERATOR BLOWDOWN (1)	AB	32	A	GL	SO	AT	RM	PA	15.0	O	C	C	C	C	Y	N	A	N	AS14B	22
	47W801-2	57	W	H	BE	STEAM GENERATOR BLOWDOWN (1)	AB	25	B	GL	SO	AT	RM	PA	15.0	O	C	C	C	C	Y	N	A	N	AS14C	22
	47W801-2	57	W	H	BE	STEAM GENERATOR BLOWDOWN (1)	AB	7	B	GL	SO	AT	RM	PA	15.0	O	C	C	C	C	Y	N	A	N	AS14D	22

WATTS BAR NUCLEAR PLANT CONTAINMENT PENETRATIONS AND BARRIERS-TABLE 6.2.4-1

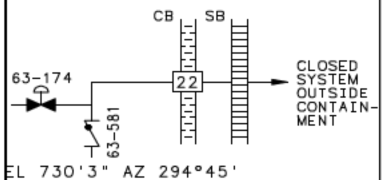
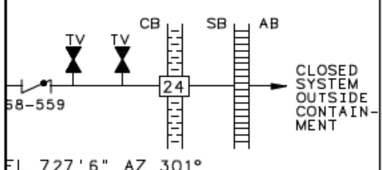
PENETRATION DATA							VALVE DATA																			
DETAILS	DWG NUMBER	GEN DES CRITERION	PROCESS FLUID	FLUID STATE	POSS LEAK PATHS	SYSTEM NUMBER AND PENETRATION DESCRIPTION	VALVE LOCATION	VALVE NUMBER	ESF POWER TRAIN	VALVE TYPE	ACTUATOR	PRI ACT MODE	SEC ACT MODE	ISOLATION SIGNAL	STROKE TIME	VALVE STATUS: NORMAL	VALVE STATUS: SHUTDOWN	VALVE STATUS: POST-ACCIDENT	VALVE STATUS: POWER FAILURE	ILRT	POS IND IN MCR	ESF	APP J TEST	ESSENTIAL/NON-ESS	SHIELD BLDG PENETRATION	NOTES
	47W809-1	55	W	H	BD	CVCS LETDOWN (62)	CB CB CB CB CB AB	72 73 74 662 76 77	A A A – A B	GL GL GL RV GL GL	AO AO AO – AO AO	AT AT AT SA AT AT	RM RM RM – RM RM	PA PA PA – PA PA	10.0 10.0 10.0 – 10.0 10.0	C O C C C O	V V V C C V	C C C C C C	C C C C C C	C C C C C C	Y Y Y N Y Y	N N N N N N	AC AC AC AC AC AC	N	AS15	4
	47W809-1	55	W	C	AD	CVCS NORMAL CHARGING (62)	CB	543	–	CK	–	SA	RM	–	–	O	V	C	–	–	N	N	A	N	MK19	
	47W811-1 47W810-1	55	W	C	BD	RHR HOT LEG INJECTION (63)	CB CB CB	640 643 158	– – –	CK CK GL	– – AO	SA SA RM	– – –	– – -	– – –	C C V	V V V	C C C	– – C	– – C	N N Y	Y Y Y	A A A	E	MK27	

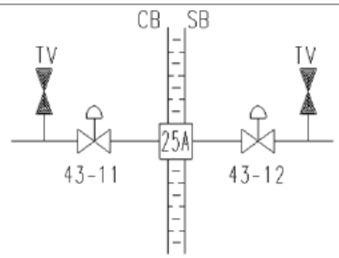
WATTS BAR NUCLEAR PLANT CONTAINMENT PENETRATIONS AND BARRIERS-TABLE 6.2.4-1																									
PENETRATION DATA							VALVE DATA																		
															VALVE STATUS										
DETAILS	DWG NUMBER	GEN DES CRITERION	PROCESS FLUID	FLUID STATE	POSS LEAK PATHS	SYSTEM NUMBER AND PENETRATION DESCRIPTION	VALVE LOCATION	VALVE NUMBER	ESF POWER TRAIN	VALVE TYPE	ACTUATOR	PRI ACT MODE	SEC ACT MODE	ISOLATION SIGNAL	STROKE TIME	NORMAL	SHUTDOWN	POST-ACCIDENT	POWER FAILURE	ILRT	POS IND IN MCR	ESF	APP J TEST	ESSENTIAL/NON-ESS	SHIELD BLDG PENETRATION	NOTES
X-18 EL 740' AZ 145	48W406	–	–	–	–	SPARE	–	–	–	–	–	–	–	–	–	–	–	–	–	–	–	–	A	–	–	
EL 689 AZ 318°  	47W811-1 47W812-1	–	W	H	D	SUMP SUCTION TO RHR PUMP 2A-A (63,72)	AB AB	72 44	A A	GA GA	MO MO	RM RM	LM LM	– –	– –	C C	C C	V V	AI AI	C C	Y Y	Y Y	A A	E	–	
EL 689 AZ 325°  	47W811-1 47W812-1	–	W	H	D	SUMP SUCTION TO RHR PUMP 2B-B (63,72)	AB AB	73 45	B B	GA GA	MO MO	RM RM	LM LM	– –	– –	C C	C C	V V	AI AI	C C	Y Y	Y Y	A A	E	–	

WATTS BAR NUCLEAR PLANT CONTAINMENT PENETRATIONS AND BARRIERS-TABLE 6.2.4-1

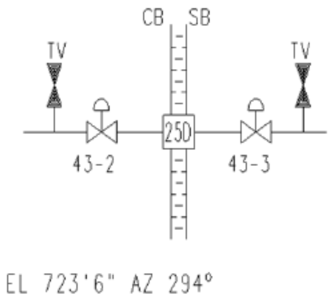
PENETRATION DATA								VALVE DATA								VALVE STATUS										
DETAILS	DWG NUMBER	GEN DES CRITERION	PROCESS FLUID	FLUID STATE	POSS LEAK PATHS	SYSTEM NUMBER AND PENETRATION DESCRIPTION	VALVE LOCATION	VALVE NUMBER	ESF POWER TRAIN	VALVE TYPE	ACTUATOR	PRI ACT MODE	SEC ACT MODE	ISOLATION SIGNAL	STROKE TIME	NORMAL	SHUTDOWN	POST-ACCIDENT	POWER FAILURE	ILRT	POS IND IN MCR	ESF	APP J TEST	ESSENTIAL/NON-ESS	SHIELD BLDG PENETRATION	NOTES
	47W811-1	55	W	C	B D	RHR PUMP DISCHARGE TR B (63)	CB CB CB	112 635 633	– – –	GL CK CK	AO – –	RM SA SA	– – –	– – –	– – –	V C C	V C C	C V V	C – –	C – –	Y N N	N Y Y	A A A	E	AS20A	
	47W811-1	55	W	C	BD	RHR PUMP DISCHARGE TR A (63)	CB CB CB	111 632 634	– – –	GL CK CK	AO – –	RM SA SA	– – –	– – –	– – –	V C C	V C C	C V V	C – –	C – –	Y N N	N Y Y	A A A	E	AS20B	
	47W811-1	55	W	C	BD	SIS PUMP DISCHARGE TO HOT LEGS TR B (63)	CB CB CB	167 547 549	– – –	GL CK CK	AO – –	RM SA SA	– – –	– – –	– – –	C – –	C – –	V – –	C – –	C – –	Y N N	N Y Y	A A A	E	AS21	

WATTS BAR NUCLEAR PLANT CONTAINMENT PENETRATIONS AND BARRIERS-TABLE 6.2.4-1

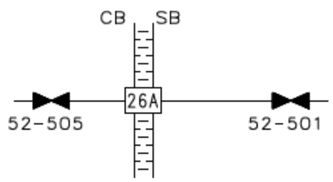
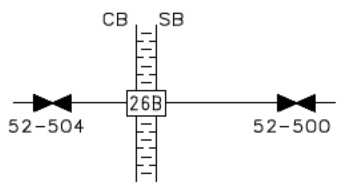
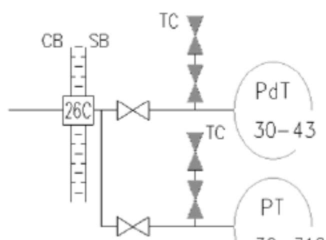
DETAILS	DWG NUMBER	GEN DES CRITERION	PROCESS FLUID	FLUID STATE	POSS LEAK PA HS	SYSTEM NUMBER AND PENETRATION DESCRIPTION	VALVE LOCATION	VALVE NUMBER	ESF POWER TRAIN	VALVE TYPE	ACTUATOR	PRI ACT MODE	SEC ACT MODE	ISOLATION SIGNAL	STROKE TIME	NORMAL	SHUTDOWN	POST ACCIDENT	POWER FAILURE	ILRT	POS IND IN MCR	ESF	APP J TEST	ESSENTIAL/NON-ESS	SHIELD BLDG PENETRATION	NOTES
PENETRATION DATA							VALVE DATA									VALVE STATUS										
	47W811-1	55	W	C	BD	SIS CHARGING PUMP DISCHARGE (63)	CB CB	174 581	– –	GL CK	AO –	RM SA	– –	– –	– –	V –	V –	C –	C –	C –	Y N	N Y	A A	E	AS22	
X-23 EL 729' AZ 283°	47W625-15	–	–	–	–	SPARE	–	–	–	–	–	–	–	–	–	–	–	–	–	–	–	–	A	–	–	
	47W813-1 47W811-1 47W812-1	56	W	H	BD	RELIEF VALVE DISCHARGE (68)	CB	559	–	CK	–	SA	–	–	–	C	C	C	–	–	N	Y	-	E	AS24	25

WATTS BAR NUCLEAR PLANT CONTAINMENT PENETRATIONS AND BARRIERS-TABLE 6.2.4-1																										
PENETRATION DATA							VALVE DATA																			
															VALVE STATUS											
DETAILS	DWG NUMBER	GEN DES CRITERION	PROCESS FLUID	FLUID STATE	POSS LEAK PATHS	SYSTEM NUMBER AND PENETRATION DESCRIPTION	VALVE LOCATION	VALVE NUMBER	ESF POWER TRAIN	VALVE TYPE	ACTUATOR	PRI ACT MODE	SEC ACT MODE	ISOLATION SIGNAL	STROKE TIME	NORMAL	SHUTDOWN	POST-ACCIDENT	POWER FAILURE	ILRT	POS IND IN MCR	ESF	APP J TEST	ESSENTIAL/NON-ESS	SHIELD BLDG PENETRATION	NOTES
 EL 723'6" AZ 294°	47W625-1	55	W	H	AD	PRESSURIZER LIQUID SAMPLE (43)	CB SB	11 12	B A	GL GL	AO AO	AT AT	RM RM	PA PA	5 0 5 0	V V	V V	C C	C C	C C	Y Y	N N	AC AC	N	MK44H	15
X-25B EL 723'6" AZ 294°	47W331-2	–	–	–	–	SPARE	–	–	–	–	–	–	–	–	–	–	–	–	–	–	–	–	A	–	–	
X-25C EL 723'6" AZ 294°	47W331-2	–	–	–	–	SPARE	–	–	–	–	–	–	–	–	–	–	–	–	–	–	–	–	A	–	–	

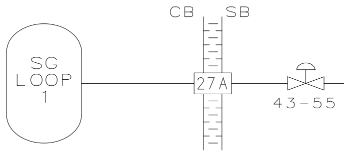
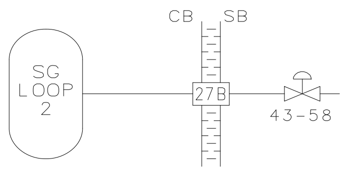
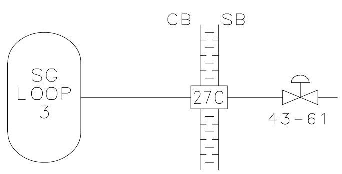
WATTS BAR NUCLEAR PLANT CONTAINMENT PENETRATIONS AND BARRIERS-TABLE 6.2.4-1

DETAILS	DWG NUMBER	GEN DES CRITERION	PROCESS FLUID	FLUID STATE	POSS LEAK PATHS	SYSTEM NUMBER AND PENETRATION DESCRIPTION	VALVE LOCATION	VALVE NUMBER	ESF POWER TRAIN	VALVE TYPE	ACTUATOR	PRI ACT MODE	SEC ACT MODE	ISOLATION SIGNAL	STROKE TIME	VALVE STATUS: NORMAL	VALVE STATUS: SHUTDOWN	VALVE STATUS: POST-ACCIDENT	VALVE STATUS: POWER FAILURE	ILRT	POS IND IN MCR	ESF	APP J TEST	ESSENTIAL/NON-ESS	SHIELD BLDG PENETRATION	NOTES
PENETRATION DATA							VALVE DATA																			
	47W625-1	55	S	H	AD	PRESSURIZER STEAM SAMPLE (43)	CB SB	2 3	B A	GL GL	AO AO	AT AT	RM RM	PA PA	5 0 5 0	V V	V V	C C	C C	C C	Y Y	N N	AC AC	N	MK44D	15

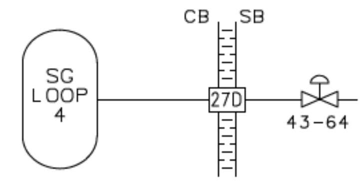
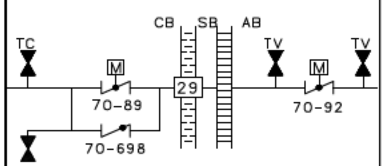
WATTS BAR NUCLEAR PLANT CONTAINMENT PENETRATIONS AND BARRIERS-TABLE 6.2.4-1

DETAILS	PENETRATION DATA: DWG NUMBER	GEN DES CRITERION	PROCESS FLUID	FLUID STATE	POSS LEAK PATHS	SYSTEM NUMBER AND PENETRATION DESCRIPTION	VALVE DATA: VALVE LOCATION	VALVE NUMBER	ESF POWER TRAIN	VALVE TYPE	ACTUATOR	PRI ACT MODE	SEC ACT MODE	ISOLATION SIGNAL	STROKE TIME	VALVE STATUS: NORMAL	SHUTDOWN	POST ACCIDENT	POWER FAILURE	ILRT	POS IND IN MCR	ESF	APP J TEST	ESSENTIAL/NON-ESS	SHIELD BLDG PENETRATION	NOTES
 EL 723'6" AZ 293°	47W331-3	56	A	C	B	ILRT SENSOR LINE (52) TEST INSTRUMENTATION	CB SB	505 501	– –	GL GL	M M	LM LM	– –	– –	– –	C C	V V	C C	– –	O O	N N	N N	AC AC	N	–	
 EL 723'6" AZ 293°	47W331-3	56	A	C	B	ILRT SENSOR LINE (52) TEST INSTRUMENTATION	CB SB	504 500	– –	GL GL	M M	LM LM	– –	– –	– –	C C	V V	C C	– –	O O	N N	N N	AC AC	N	–	
 EL 723'6" AZ 293°	47W600-89 47W331-3	–	A	C	B	dP SENSOR (30)	SB	SNSR	–	–	–	–	–	–	–	O	–	O	–	C	N	N	A	E	–	12, 13

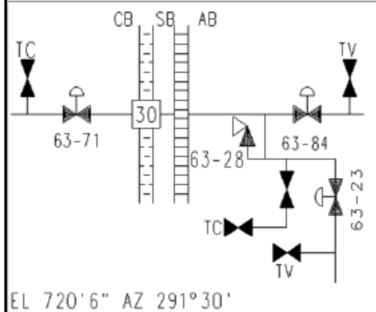
WATTS BAR NUCLEAR PLANT CONTAINMENT PENETRATIONS AND BARRIERS-TABLE 6.2.4-1

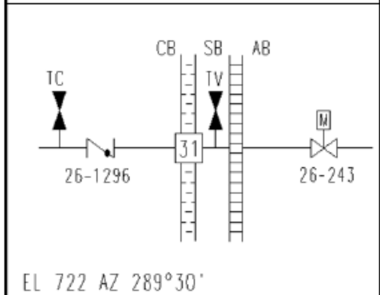
PENETRATION DATA							VALVE DATA																		
															VALVE STATUS										
DETAILS	DWG NUMBER	GEN DES CRITERION	PROCESS FLUID	FLUID STATE	POSS LEAK PATHS	SYSTEM NUMBER AND PENETRATION DESCRIPTION	VALVE LOCATION	VALVE NUMBER	ESF POWER TRAIN	VALVE TYPE	ACTUATOR	PRI ACT MODE	SEC ACT MODE	ISOLATION SIGNAL	STROKE TIME	NORMAL	SHUTDOWN	POST-ACCIDENT	POWER FAILURE	ILRT	POS IND IN MCR	ESF	APP J TEST	ESSENTIAL/NON-ESS	SHIELD BLDG PENETRATION	NOTES
 EL 723'6" AZ 292°	47W625-2	56	W	H	A B D	STM GEN NO. 1 SAMPLE (43) SAMPLE AND WATER QUALITY	SB	55	A	GL	AO	AT	RM	PA	10.0	V	V	C	C	C	Y	N	A	N	MK046M	22
 EL 723'6" AZ 292°	47W625-2	56	W	H	AB D	STM GEN NO. 2 SAMPLE (43)	SB	58	A	GL	AO	AT	RM	PA	10.0	V	V	C	C	C	Y	N	A	N	MK46M	22
 EL 723'6" AZ 292°	47W625-2	56	W	H	AB D	STM GEN NO. 3 SAMPLE (43)	SB	61	A	GL	AO	AT	RM	PA	10.0	V	V	C	C	C	Y	N	A	N	MK46M	22

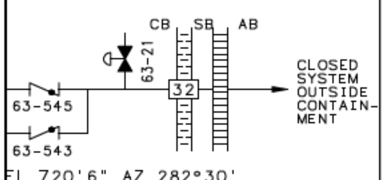
WATTS BAR NUCLEAR PLANT CONTAINMENT PENETRATIONS AND BARRIERS-TABLE 6.2.4-1

	PENETRATION DATA						VALVE DATA																			
															VALVE STATUS											
DETAILS	DWG NUMBER	GEN DES CRITERION	PROCESS FLUID	FLUID STATE	POSS LEAK PATHS	SYSTEM NUMBER AND PENETRATION DESCRIPTION	VALVE LOCATION	VALVE NUMBER	ESF POWER TRAIN	VALVE TYPE	ACTUATOR	PRI ACT MODE	SEC ACT MODE	ISOLATION SIGNAL	STROKE TIME	NORMAL	SHUTDOWN	POST ACCIDENT	POWER FAILURE	ILRT	POS IND IN MCR	ESF	APP J TEST	ESSENTIAL/NON-ESS	SHIELD BLDG PENETRATION	NOTES
 EL 723'6" AZ 292°	47W625-2	56	W	H	AB D	STM GEN NO. 4 SAMPLE (43)	SB	64	A	GL	AO	AT	RM	PA	10.0	V	V	C	C	C	Y	N	A	N	MK46M	22
X-28 EL 721' AZ 287°30'	47W625-15	–	–	–	–	SPARE	–	–	–	–	–	–	–	–	–	–	–	–	–	–	–	–	A	–	–	
 EL 720 AZ 288°30'	47W859-2 47W859-3	56	W	C	AD	CCS FROM RC PUMP COOLERS (70)	CB AB CB	89 92 698	B A –	BF BF CK	MO MO –	AT AT SA	– – –	PB PB –	66.0 66.0 –	O O O	O O O	C C V	AI AI –	C C –	Y Y N	N N N	AC AC AC	N	MK50	8

WATTS BAR NUCLEAR PLANT CONTAINMENT PENETRATIONS AND BARRIERS-TABLE 6.2.4-1

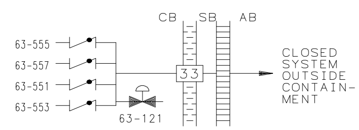
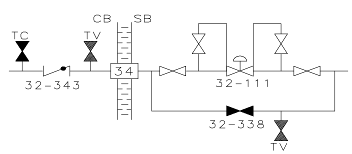
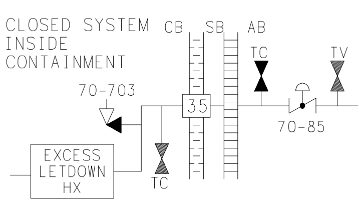






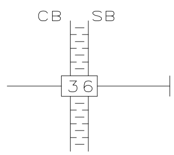
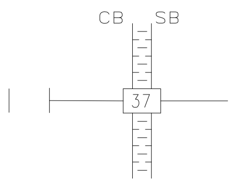
PENETRATION DATA							VALVE DATA									VALVE STATUS										
DETAILS	DWG NUMBER	GEN DES CRITERION	PROCESS FLUID	FLUID STATE	POSS LEAK PA HS	SYSTEM NUMBER AND PENETRATION DESCRIPTION	VALVE LOCATION	VALVE NUMBER	ESF POWER TRAIN	VALVE TYPE	ACTUATOR	PRI ACT MODE	SEC ACT MODE	ISOLATION SIGNAL	STROKE TIME	NORMAL	SHUTDOWN	POST ACCIDENT	POWER FAILURE	ILRT	POS IND IN MCR	ESF	APP J TEST	ESSENTIAL/NON-ESS	SHIELD BLDG PENETRATION	NOTES
	47W811-1	56	W	C	BD	ACCUM TO HOLDUP TANKS (63)	CB	71	A	GL	AO	AT	RM	PA	10.0	V	V	C	C	C	Y	N	AC	N	AS30	9
							AB	84	B	GL	AO	AT	–	PA	10.0	V	V	C	C	C	Y	N	AC			
							AB	23	B	GL	AO	AT	–	PA	10.0	V	V	C	C	C	Y	N	AC			
							AB	28	–	RV	–	SA	–	–	–	C	C	V	–	C	N	N	AC			
	47W850-9	56	A	C	AB D	FIRE PROECTION (26)	CB	1296	–	CK	–	SA	–	–	–	C	C	C	–	C	N	N	AC	N	MK49	
							AB	243	A	GA	MO	AT	RM	PA	20.0	O	O	C	C	C	Y	N	AC			
	47W811-1	55	W	C	BD	SI TO HOT LEGS (63)	CB	545	–	CK	–	SA	–	–	–	C	C	V	–	–	N	Y	A	E	AS32	
							CB	543	–	CK	–	SA	–	–	–	C	C	V	–	–	N	Y	A			
							CB	21	–	GL	AO	RM	–	–	–	V	V	C	C	C	Y	N	A			

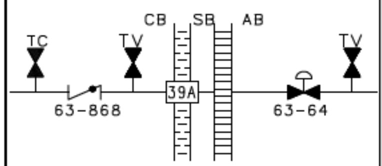
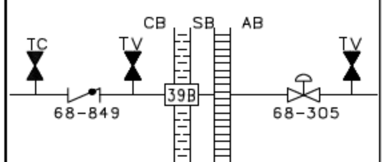
WATTS BAR NUCLEAR PLANT CONTAINMENT PENETRATIONS AND BARRIERS-TABLE 6.2.4-1

PENETRATION DATA							VALVE DATA								VALVE STATUS										
DETAILS	DWG NUMBER	GEN DES CRITERION	PROCESS FLUID	FLUID STATE	POSS LEAK PATHS	SYSTEM NUMBER AND PENETRATION DESCRIPTION	VALVE LOCATION	VALVE NUMBER	ESF POWER TRAIN	VALVE TYPE	ACTUATOR	PRI ACT MODE	SEC ACT MODE	ISOLATION SIGNAL	STROKE TIME	NORMAL	SHUTDOWN	POST-ACCIDENT	POWER FAILURE	ILRT	POS IND IN MCR	ESF	APP J TEST	ESSENTIAL/NON-ESS	SHIELD BLDG PENETRATION	NOTES
 EL 720'6" AZ 277°30'	47W811-1	55	W	C	BD	SI TO LOW HEAD SI (63)	CB	553	–	CK	–	SA	–	–	–	C	C	V	–	–	N	Y	A	E	AS33	
							CB	551	–	CK	–	SA	–	–	–	C	C	V	–	–	N	Y	A			
							CB	557	–	CK	–	SA	–	–	–	C	C	V	–	–	N	Y	A			
							CB	555	–	CK	–	SA	–	–	–	C	C	V	–	–	N	Y	A			
							CB	121	–	GL	AO	RM	–	–	–	V	V	C	C	C	Y	N	A			
 EL 720'6" AZ 299°30'	47W848-1	56	A	C	AB D	CONTROL AIR I & C (32)	CB	343	–	CK	–	SA	–	–	–	O	O	C	–	–	N	N	AC	N	MK43	10
							SB	111	A	GL	AO	AT	RM	PB	10.0	O	O	C	C	C	Y	N	AC			
							SB	338	–	GL	M	LM	–	–	–	C	C	C	AI	C	N	N	AC			
 EL 720'6" AZ 301°30'	47W859-2, 3	57	W	C	AD	CCS FROM EXCESS LETDN HX (70)	CB	703	–	RV	–	SA	–	–	–	V	V	V	C	C	N	N	AC	N	MK42	11
							AB	85	B	BF	AO	AT	RM	PA	10.0	V	C	C	C	C	Y	N	AC			

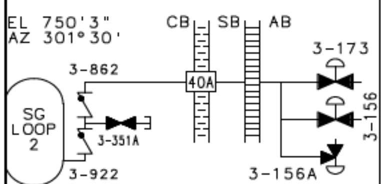
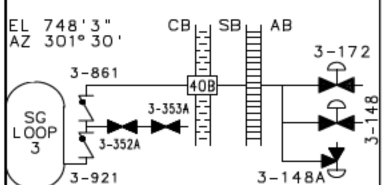
Unit 2

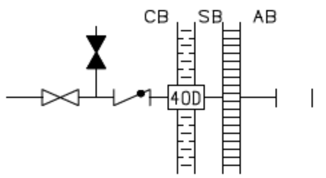
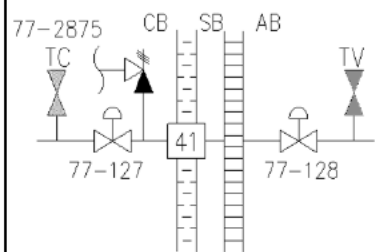
WATTS BAR NUCLEAR PLANT CONTAINMENT PENETRATIONS AND BARRIERS-TABLE 6.2.4-1

PENETRATION DATA							VALVE DATA																			
																VALVE STATUS										
DETAILS	DWG NUMBER	GEN DES CRITERION	PROCESS FLUID	FLUID STATE	POSS LEAK PATHS	SYSTEM NUMBER AND PENETRATION DESCRIPTION	VALVE LOCATION	VALVE NUMBER	ESF POWER TRAIN	VALVE TYPE	ACTUATOR	PRI ACT MODE	SEC ACT MODE	ISOLATION SIGNAL	STROKE TIME	NORMAL	SHUTDOWN	POST-ACCIDENT	POWER FAILURE	ILRT	POS IND IN MCR	ESF	APP J TEST	ESSENTIAL/NON-ESS	SHIELD BLDG PENETRATION	NOTES
 EL 742'6" AZ 280°	72-4334-310	–	A	C	AB	SG. CHEM. CLEANING	SB	–	–	BL	M	LM	–	–	–	C	V	C	–	C	N	N	AB	N	MK20	
 EL 771'6" AZ 265°	47W301-1	–	A	C	AB	MAINT. PORT	CB	–	–	BL	M	LM	–	–	–	C	V	C	–	C	N	N	AB	N	MK14	
X-38 EL 771'6" AZ 268°	48N406	–	–	–	–	SPARE	–	–	–	–	–	–	–	–	–	–	–	–	–	–	–	–	A	–	–	

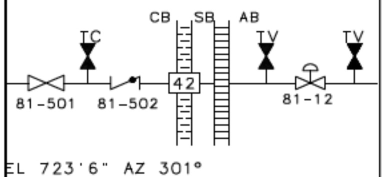
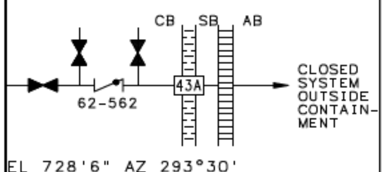
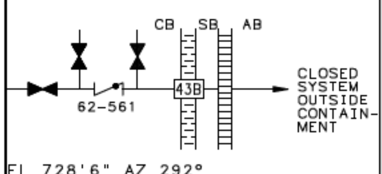
WATTS BAR NUCLEAR PLANT CONTAINMENT PENETRATIONS AND BARRIERS-TABLE 6.2.4-1																										
PENETRATION DATA							VALVE DATA																			
															VALVE STATUS											
DETAILS	DWG NUMBER	GEN DES CRITERION	PROCESS FLUID	FLUID STATE	POSS LEAK PATHS	SYSTEM NUMBER AND PENETRATION DESCRIPTION	VALVE LOCATION	VALVE NUMBER	ESF POWER TRAIN	VALVE TYPE	ACTUATOR	PRI ACT MODE	SEC ACT MODE	ISOLATION SIGNAL	STROKE TIME	NORMAL	SHUTDOWN	POST-ACCIDENT	POWER FAILURE	ILRT	POS IND IN MCR	ESF	APP J TEST	ESSENTIAL/NON-ESS	SHIELD BLDG PENETRATION	NOTES
 EL 720'6" AZ 280°	47W830-6	56	N2	C	AD	N2 TO ACCUM (63)	CB AB	868 64	– A	CK GL	– AO	SA AT	– RM	– PA	– 10.0	O V	O O	C C	– C	– C	N Y	N N	AC AC	N	MK55M	
 EL 720'6" AZ 280°	47W830-6	56	N2	C	AD	N2 TO PRESS RELIEF TK (68)	CB AB	849 305	– A	CK DI	– AO	SA AT	– RM	– PA	– 10.0	O O	O O	C C	– C	– C	N Y	N N	AC AC	N	MK55M	
X-39C EL 720'6" AZ 280°	47W331-2	–	–	–	–	SPARE	–	–	–	–	–	–	–	–	–	–	–	–	–	–	–	–	A	–	–	

WATTS BAR NUCLEAR PLANT CONTAINMENT PENETRATIONS AND BARRIERS-TABLE 6.2.4-1

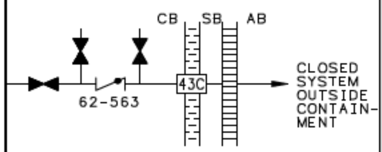
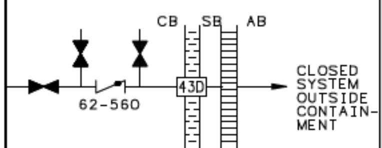
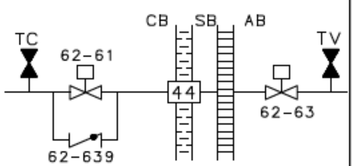
PENETRATION DATA							VALVE DATA																			
DETAILS	DWG NUMBER	GEN DES CRITERION	PROCESS FLUID	FLUID STATE	POSS LEAK PATHS	SYSTEM NUMBER AND PENETRATION DESCRIPTION	VALVE LOCATION	VALVE NUMBER	ESF POWER TRAIN	VALVE TYPE	ACTUATOR	PRI ACT MODE	SEC ACT MODE	ISOLATION SIGNAL	STROKE TIME	VALVE STATUS: NORMAL	VALVE STATUS: SHUTDOWN	VALVE STATUS: POST-ACCIDENT	VALVE STATUS: POWER FAILURE	ILRT	POS IND IN MCR	ESF	APP J TEST	ESSENTIAL/NON-ESS	SHIELD BLDG PENETRATION	NOTES
X-39D EL 720'6" AZ 280°	47W331-2	–	–	–	–	SPARE	–	–	–	–	–	–	–	–	–	–	–	–	–	–	–	–	A	–	–	
	47W803-2	57	W	C	BC E	AUX FW (3)	AB AB AB	156 156A 173	A A B	GL A GL	AO AO AO	RM RM RM	LM LM LM	– – –	– – –	C C C	C C C	V V V	O C C	C C C	Y Y Y	N N N	A A A	E	MK9	22
	47W803-2	57	W	C	BC E	AUX FW (3)	AB AB AB	148 148A 172	B B A	GL A GL	AO AO AO	RM RM RM	LM LM LM	– – –	– – –	C C C	C C C	V V V	O C C	C C C	Y Y Y	N N N	A A A	E	MK11	22

WATTS BAR NUCLEAR PLANT CONTAINMENT PENETRATIONS AND BARRIERS-TABLE 6.2.4-1																										
PENETRATION DATA							VALVE DATA																			
															VALVE STATUS											
DETAILS	DWG NUMBER	GEN DES CRITERION	PROCESS FLUID	FLUID STATE	POSS LEAK PATHS	SYSTEM NUMBER AND PENETRATION DESCRIPTION	VALVE LOCATION	VALVE NUMBER	ESF POWER TRAIN	VALVE TYPE	ACTUATOR	PRI ACT MODE	SEC ACT MODE	ISOLATION SIGNAL	STROKE TIME	NORMAL	SHUTDOWN	POST-ACCIDENT	POWER FAILURE	ILRT	POS IND IN MCR	ESF	APP J TEST	ESSENTIAL/NON-ESS	SHIELD BLDG PENETRATION	NOTES
X-40C EL 750'3" AZ 299°30'	48N406	–	–	–	–	SPARE	–	–	–	–	–	–	–	–	–	–	–	–	–	–	–	–	A	–	–	
 EL 748'3" AZ 299°30'	47W846-2 47W492-2	–	A	C	AD	SERVICE AIR (33)	AB	–	–	BL	M	LM	–	–	–	C	C	C	–	C	–	N	AB	N	MK12	
 EL 719'6" AZ 294°	47W851-1	56	W	C	AD	FL SUMP PUMP DISCH (77)	CB AB CB	127 128 2875	B A –	BA BA RV	AO AO –	AT AT –	RM RM –	PA PA –	10.0 10.0 –	O O C	O O C	C C V	C C C	C C C	Y Y N	N N N	AC AC AC	N	MK47	

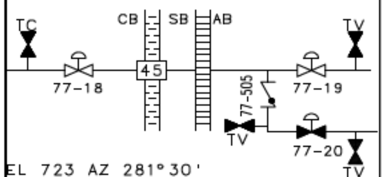
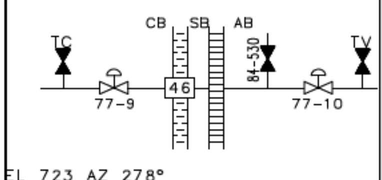
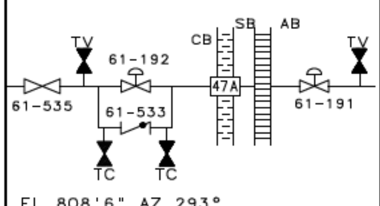
WATTS BAR NUCLEAR PLANT CONTAINMENT PENETRATIONS AND BARRIERS-TABLE 6.2.4-1

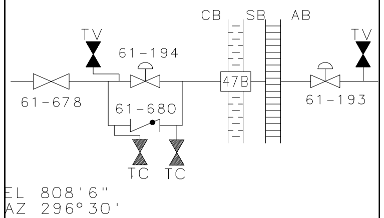
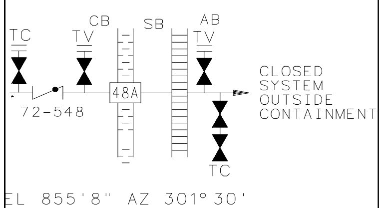
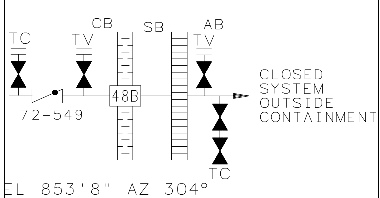
DETAILS	PENETRATION DATA: DWG NUMB R	GEN DES CRITERION	PROCESS FLUID	FLUID STATE	POSS LEAK PATHS	SYSTEM NUMBER AND PENETRATION DESCRIPTION	VALVE DATA: VALVE LOCATION	VALVE NUMBER	ESF POWER TRAIN	VALVE TYPE	ACTUATOR	PRI ACT MODE	SEC ACT MODE	ISOLATION SIGNAL	STROKE TIME	VALVE STATUS: NORMAL	SHUTDOWN	POST-ACCIDENT	POWER FAILURE	ILRT	POS IND IN MCR	ESF	APP J TEST	ESSENTIAL/NON-ESS	SHIELD BLDG PENETRATION	NOTES
	47W819-1	56	W	C	ABD	PRESS RLF TK MAKE-UP (81)	CB AB	502 12	– A	CK DI	– AO	SA AT	– RM	– PA	– 10.0	V V	C C	C C	– C	– C	N Y	N N	AC AC	N	MK39	
	47W809-1	55	W	C	AD	TO RCP SEALS (62)	CB	562	–	CK	–	SA	–	–	–	O	C	C	–	–	N	N	A	N	MK31	
	47W809-1	55	W	C	AD	TO RCP SEALS (62)	CB	561	–	CK	–	SA	–	–	–	O	C	C	–	–	N	N	A	N	MK29	

WATTS BAR NUCLEAR PLANT CONTAINMENT PENETRATIONS AND BARRIERS-TABLE 6.2.4-1

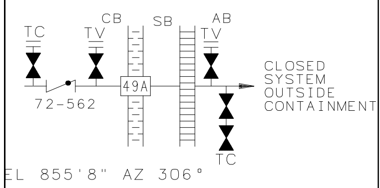
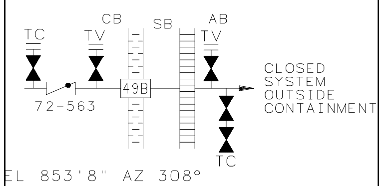
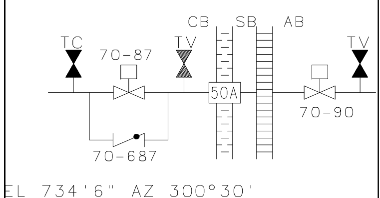
DETAILS	PENETRATION DATA: DWG NUMBER	GEN DES CRITERION	PROCESS FLUID	FLUID STATE	POSS LEAK PATHS	SYSTEM NUMBER AND PENETRATION DESCRIPTION	VALVE DATA: VALVE LOCATION	VALVE NUMBER	ESF POWER TRAIN	VALVE TYPE	ACTUATOR	PRI ACT MODE	SEC ACT MODE	ISOLATION SIGNAL	STROKE TIME	VALVE STATUS: NORMAL	SHUTDOWN	POST-ACCIDENT	POWER FAILURE	ILRT	POS IND IN MCR	ESF	APP J TEST	ESSENTIAL/NON-ESS	SHIELD BLDG PENETRATION	NOTES
 EL 727 AZ 293°30'	47W809-1	55	W	C	AD	TO RCP SEALS (62)	CB	563	–	CK	–	SA	–	–	–	O	C	C	–	–	N	N	A	N	MK28	
 EL 727 AZ 292°	47W809-1	55	W	C	AD	TO RCP SEALS (62)	CB	560	–	CK	–	SA	–	–	–	O	C	C	–	–	N	N	A	N	MK30	
 EL 727'6" AZ 281°30'	47W809-1	55	W	C	AD	FROM RCP SEALS (62)	CB AB CB	61 63 639	B A –	GA GA CK	MO MO –	AT AT SA	RM RM –	PA PA –	10.0 10.0 –	O O O	V V O	C C V	AI AI –	C C –	Y Y N	N N N	AC AC AC	N	MK37	

WATTS BAR NUCLEAR PLANT CONTAINMENT PENETRATIONS AND BARRIERS-TABLE 6.2.4-1

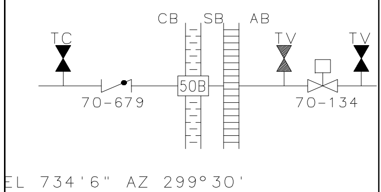
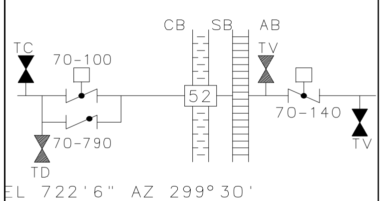
DETAILS	PENETRATION DATA						VALVE DATA																			
	DWG NUMBER	GEN DES CRITERION	PROCESS FLUID	FLUID STATE	POSS LEAK PATHS	SYSTEM NUMBER AND PENETRATION DESCRIPTION	VALVE LOCATION	VALVE NUMBER	ESF POWER TRAIN	VALVE TYPE	ACTUATOR	PRI ACT MODE	SEC ACT MODE	ISOLATION SIGNAL	STROKE TIME	VALVE STATUS: NORMAL	SHUTDOWN	POST-ACCIDENT	POWER FAILURE	ILRT	POS IND IN MCR	ESF	APP J TEST	ESSENTIAL/NON-ESS	SHIELD BLDG PENETRATION	NOTES
	47W830-1	56	N2	C	BD	RC DRAIN TK AND PRT TO VH (77)	CB AB AB	18 19 20	B A A	DI DI DI	AO AO AO	AT AT AT	RM RM RM	PA PA PA	10.0 10.0 10.0	O O V	O O O	C C C	C C C	C C C	Y Y Y	N N N	AC AC AC	N	AS45	
	47W830-1 47W809-7	56	W	C	BD	RC DRAIN TK PUMP DISCHARGE (77/84)	CB AB AB	9 10 530	B A –	DI DI GL	AO AO M	AT AT LM	RM RM –	PA PA –	10.0 10.0 –	O O C	O O C	C C C	C C –	C C C	Y Y N	N N N	AC AC AC	N	AS46	18
	47W814-2	56	G	C	BD	GLYCOL SUPPLY (61)	CB AB CB	192 191 533	B A –	DI DI CK	AO AO –	AT AT SA	RM RM –	PA PA –	30.0 30.0 –	O O O	O O O	C C V	C C –	O O O	Y Y N	N N N	C C C	N	MK25	8

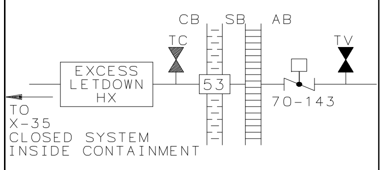
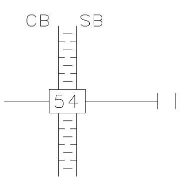
WATTS BAR NUCLEAR PLANT CONTAINMENT PENETRATIONS AND BARRIERS-TABLE 6.2.4-1																										
PENETRATION DATA							VALVE DATA																			
															VALVE STATUS											
DETAILS	DWG NUMBER	GEN DES CRITERION	PROCESS FLUID	FLUID STATE	POSS LEAK PATHS	SYSTEM NUMBER AND PENETRATION DESCRIPTION	VALVE LOCATION	VALVE NUMBER	ESF POWER TRAIN	VALVE TYPE	ACTUATOR	PRI ACT MODE	SEC ACT MODE	ISOLATION SIGNAL	STROKE TIME	NORMAL	SHUTDOWN	POST-ACCIDENT	POWER FAILURE	ILRT	POS IND IN MCR	ESF	APP J TEST	ESSENTIAL/NON-ESS	SHIELD BLDG PENETRATION	NOTES
	47W814-2	56	G	C	BD	GLYCOL RETURN (61)	CB AB CB	194 193 680	B A –	DI DI CK	AO AO –	AT AT SA	RM RM –	PA PA –	30.0 30.0 –	O O O	O O O	C C V	C C –	O O O	Y Y N	N N N	C C C	N	MK26	8
	47W812-1	56	W	C	AD	CONT. SPRAY (72)	CB	548	–	CK	–	SA	–	–	–	C	C	V	–	–	N	Y	A	E	MK4	26
	47W812-1	56	W	C	AD	CONT. SPRAY (72)	CB	549	–	CK	–	SA	–	–	–	C	C	V	–	–	N	Y	A	E	MK3	26

WATTS BAR NUCLEAR PLANT CONTAINMENT PENETRATIONS AND BARRIERS-TABLE 6.2.4-1

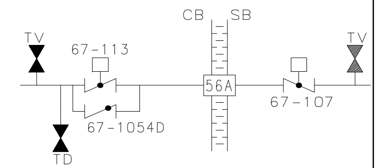
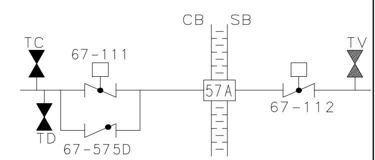
DETAILS	PENETRATION DATA: DWG NUMBER	GEN DES CRITERION	PROCESS FLUID	FLUID STATE	POSS LEAK PATHS	SYSTEM NUMBER AND PENETRATION DESCRIPTION	VALVE DATA: VALVE LOCATION	VALVE NUMBER	ESF POWER TRAIN	VALVE TYPE	ACTUATOR	PRI ACT MODE	SEC ACT MODE	ISOLATION SIGNAL	STROKE TIME	VALVE STATUS: NORMAL	SHUTDOWN	POST-ACCIDENT	POWER FAILURE	ILRT	POS IND IN MCR	ESF	APP J TEST	ESSENTIAL/NON-ESS	SHIELD BLDG PENETRATION	NOTES
	47W812-1	56	W	C	AD	RHR SPRAY (72)	CB	562	–	CK	–	SA	–	–	–	C	C	V	–	–	N	Y	A	E	MK2	26
	47W812-1	56	W	C	AD	RHR SPRAY (72)	CB	563	–	CK	–	SA	–	–	–	C	C	V	–	–	N	Y	A	E	MK1	26
	47W859-3 47W859-2	56	W	C	AD	RCP THERM BARRIER RETURN (70)	CB AB CB	87 90 687	B A –	GA GA CK	MO MO –	AT AT –	RM RM –	PB PB –	66.0 66.0 –	O O O	O O O	C C V	AI AI –	C C –	Y Y N	N N N	AC AC AC	N	MK16	8

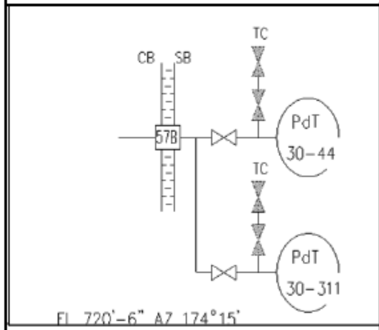
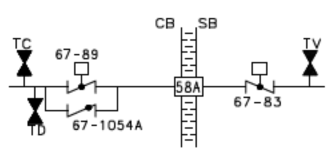
WATTS BAR NUCLEAR PLANT CONTAINMENT PENETRATIONS AND BARRIERS-TABLE 6.2.4-1

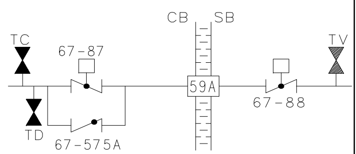
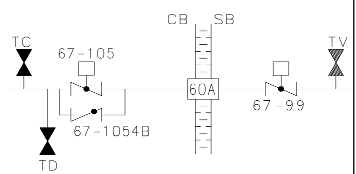
DETAILS	DWG NUMBER	GEN DES CRITERION	PROCESS FLUID	FLUID STATE	POSS LEAK PATHS	SYSTEM NUMBER AND PENETRATION DESCRIPTION	VALVE LOCATION	VALVE NUMBER	ESF POWER TRAIN	VALVE TYPE	ACTUATOR	PRI ACT MODE	SEC ACT MODE	ISOLATION SIGNAL	STROKE TIME	NORMAL	SHUTDOWN	POST-ACCIDENT	POWER FAILURE	ILRT	POS IND IN MCR	ESF	APP J TEST	ESSENTIAL/NON-ESS	SHIELD BLDG PENETRATION	NOTES
	47W859-3 47W859-2	56	W	C	AD	RCP THERM BARRIER SUPPLY (70)	CB AB	679 134	– B	CK GA	– MO	SA AT	– RM	– PB	– 66.0	O O	O O	C C	– AI	– C	N Y	N N	AC AC	N	MK17	
X-51 EL 728'6" AZ 286°30'	48W406	–	–	–	–	SPARE	–	–	–	–	–	–	–	–	–	–	–	–	–	–	–	–	A	–	–	
	47W859-2 47W859-3	56	W	C	AD	CCS TO RCP COOLERS (70)	AB CB CB	140 100 790	B A –	BF BF CK	MO MO –	AT AT SA	RM RM –	PB PB –	66.0 66.0 –	O O O	O O O	C C V	AI AI –	C C –	Y Y N	N N N	AC AC AC	N	MK40	8

WATTS BAR NUCLEAR PLANT CONTAINMENT PENETRATIONS AND BARRIERS-TABLE 6.2.4-1																										
PENETRATION DATA							VALVE DATA																			
															VALVE STATUS											
DETAILS	DWG NUMBER	GEN DES CRITERION	PROCESS FLUID	FLUID STATE	POSS LEAK PATHS	SYSTEM NUMBER AND PENETRATION DESCRIPTION	VALVE LOCATION	VALVE NUMBER	ESF POWER TRAIN	VALVE TYPE	ACTUATOR	PRI ACT MODE	SEC ACT MODE	ISOLATION SIGNAL	STROKE TIME	NORMAL	SHUTDOWN	POST-ACCIDENT	POWER FAILURE	ILRT	POS IND IN MCR	ESF	APP J TEST	ESSENTIAL/NON-ESS	SHIELD BLDG PENETRATION	NOTES
 EL 721'6" AZ 301°	47W859-2 47W859-3	57	W	C	AD	CCS TO EXCESS COOLERS LETDN HX (70)	AB	143	A	BF	MO	AT	RM	PA	66.0	V	C	C	AI	C	Y	N	AC	N	MK41	11
 EL 740 AZ 90°	72-4334-315	–	A	C	BC	IITA RENEWAL	SB	–	–	BL	M	LM	–	–	–	C	V	C	–	O	N	N	AB	N	MK72	16
X-55 EL 771'6" AZ 262°	48W406	–	–	–	–	SPARE	–	–	–	–	–	–	–	–	–	–	–	–	–	–	–	–	A	–	–	

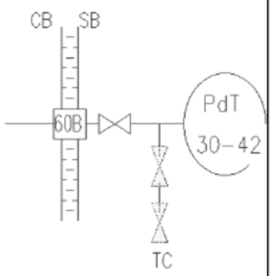
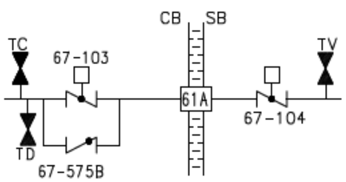
WATTS BAR NUCLEAR PLANT CONTAINMENT PENETRATIONS AND BARRIERS-TABLE 6.2.4-1

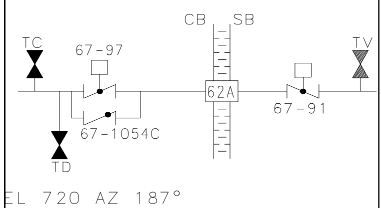
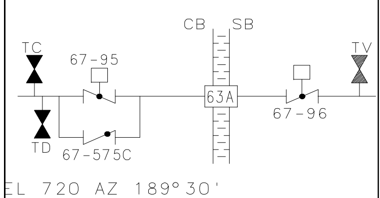
PENETRATION DATA							VALVE DATA																		
															VALVE STATUS										
DETAILS	DWG NUMBER	GEN DES CRITERION	PROCESS FLUID	FLUID STATE	POSS LEAK PATHS	SYSTEM NUMBER AND PENETRATION DESCRIPTION	VALVE LOCATION	VALVE NUMBER	ESF POWER TRAIN	VALVE TYPE	ACTUATOR	PRI ACT MODE	SEC ACT MODE	ISOLATION SIGNAL	STROKE TIME	NORMAL	SHUTDOWN	POST-ACCIDENT	POWER FAILURE	ILRT	POS IND IN MCR	ESF	APP J TEST	ESSENTIAL/NON-ESS	SHIELD BLDG PENETRATION	NOTES
 EL 720 AZ 353°	47W845-3	56	W	C	AB DE	LWR CONT ERCW SUPPLY (67)	SB CB CB	107 113 1054D	A B –	BF BF CK	MO MO –	AT AT SA	RM RM –	PB PB –	66.0 66.0 –	O O O	O O O	C C V	AI AI –	C C –	Y Y N	N N N	AC AC AC	N	MK80	8
X-56B EL 720 AZ 355°30'	48W406	–	–	–	–	SPARE	–	–	–	–	–	–	–	–	–	–	–	–	–	–	–	–	A	–	–	
 EL 720 AZ 351°30'	47W845-3	56	W	C	AB DE	LWR CONT ERCW RETURN (67)	CB SB CB	111 112 575D	B A –	BF BF CK	MO MO –	AT AT SA	RM RM –	PB PB –	66.0 66.0 –	O O O	O O O	C C V	AI AI –	C C –	Y Y N	N N N	AC AC AC	N	MK81	8

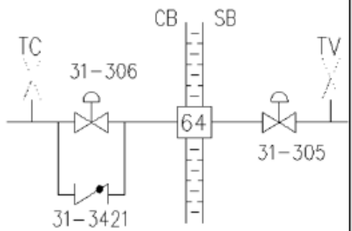
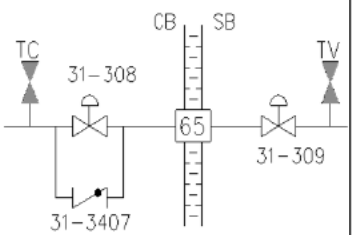
WATTS BAR NUCLEAR PLANT CONTAINMENT PENETRATIONS AND BARRIERS-TABLE 6.2.4-1																										
PENETRATION DATA							VALVE DATA																			
															VALVE STATUS											
DETAILS	DWG NUMBER	GEN DES CRITERION	PROCESS FLUID	FLUID STATE	POSS LEAK PATHS	SYSTEM NUMBER AND PENETRATION DESCRIPTION	VALVE LOCATION	VALVE NUMBER	ESF POWER TRAIN	VALVE TYPE	ACTUATOR	PRI ACT MODE	SEC ACT MODE	ISOLATION SIGNAL	STROKE TIME	NORMAL	SHUTDOWN	POST ACCIDENT	POWER FAILURE	ILRT	POS IND IN MCR	ESF	APP J TEST	ESSENTIAL/NON-ESS	SHIELD BLDG PENETRATION	NOTES
	47W600-89	–	A	C	B	dP SENSOR (30)	SB	SNSR	–	–	–	–	–	–	–	O	O	O	–	–	–	N	A	E	–	12
 EL 720' AZ 7°	47W845-3	56	W	C	AB DE	LWR CONT ERCW SUPPLY (67)	CB SB CB	1054A 83 89	– B A	CK BF BF	– MO MO	SA AT AT	– RM RM	– PB PB	– 66.0 66.0	O O O	O O O	V C C	– AI AI	– C C	N Y Y	N N N	AC AC AC	N	MK79	
X-58B EL 720 AZ 13°	47W600-75	54	W	C	BD	RCS PRESSURE SENSOR (68)	–	–	–	–	–	–	–	–	–	–	O	O	O	–	N	N	A	E	MK126	28

WATTS BAR NUCLEAR PLANT CONTAINMENT PENETRATIONS AND BARRIERS-TABLE 6.2.4-1																										
PENETRATION DATA							VALVE DATA																			
															VALVE STATUS											
DETAILS	DWG NUMBER	GEN DES CRITERION	PROCESS FLUID	FLUID STATE	POSS LEAK PATHS	SYSTEM NUMBER AND PENETRATION DESCRIPTION	VALVE LOCATION	VALVE NUMBER	ESF POWER TRAIN	VALVE TYPE	ACTUATOR	PRI ACT MODE	SEC ACT MODE	ISOLATION SIGNAL	STROKE TIME	NORMAL	SHUTDOWN	POST-ACCIDENT	POWER FAILURE	ILRT	POS IND IN MCR	ESF	APP J TEST	ESSENTIAL/NON-ESS	SHIELD BLDG PENETRATION	NOTES
 EL 720 AZ 8°30'	47W845-3	56	W	C	AB DE	LWR CONT ERCW RETURN (67)	CB SB CB	87 88 575A	A B –	BF BF CK	MO MO –	AT AT SA	RM RM –	PB PB –	66.0 66.0 –	O O O	O O O	C C V	AI AI –	C C –	Y Y N	N N N	AC AC AC	N	MK78	8
X-59B EL 720 AZ 14°30'	48W406	–	–	–	–	SPARE	–	–	–	–	–	–	–	–	–	–	–	–	–	–	–	–	A	–	–	
 EL 720 AZ 173°	47W845-3	56	W	C	AB DE	LWR CONT ERCW SUPPLY (67)	CB SB CB	1054B 99 105	– A B	CK BF BF	– MO MO	SA AT AT	– RM RM	– PB PB	– 66.0 66.0	O O O	O O O	V C C	– AI AI	– C C	N Y Y	N N N	AC AC AC	N	MK77	8

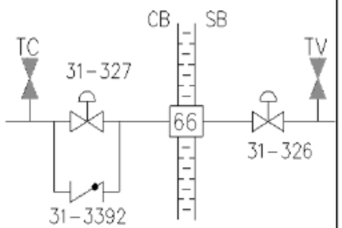
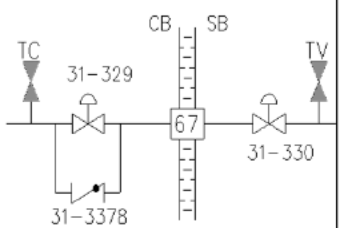
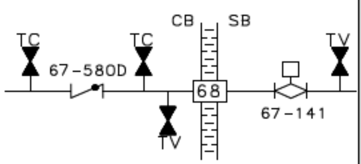
WATTS BAR NUCLEAR PLANT CONTAINMENT PENETRATIONS AND BARRIERS-TABLE 6.2.4-1

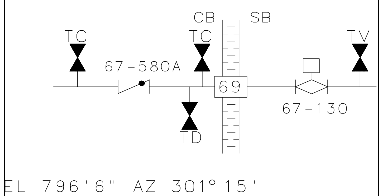
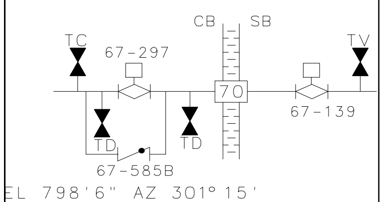
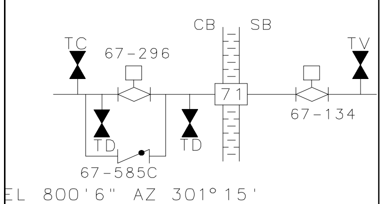
PENETRATION DATA							VALVE DATA																			
																VALVE STATUS										
DETAILS	DWG NUMBER	GEN DES CRITERION	PROCESS FLUID	FLUID STATE	POSS LEAK PATHS	SYSTEM NUMBER AND PENETRATION DESCRIPTION	VALVE LOCATION	VALVE NUMBER	ESF POWER TRAIN	VALVE TYPE	ACTUATOR	PRI ACT MODE	SEC ACT MODE	ISOLATION SIGNAL	STROKE TIME	NORMAL	SHUTDOWN	POST-ACCIDENT	POWER FAILURE	ILRT	POS IND IN MCR	ESF	APP J TEST	ESSENTIAL/NON-ESS	SHIELD BLDG PENETRATION	NOTES
 EL 720'-6" AZ 174°15'	47W600	–	A	C	B	dP SENSOR (30)	SB	SNSR	–	–	–	–	–	–	–	O	O	O	–	–	–	N	A	E	–	12
 EL 720' AZ 171° 30'	47W845-3	56	W	C	AB DE	LWR CONT ERCW RETURN (67)	CB SB CB	103 104 575B	B A –	BF BF CK	MO MO –	AT AT SA	RM RM –	PB PB –	66.0 66.0 –	O O O	O O O	C C V	AI AI –	C C –	Y Y N	N N N	AC AC AC	N	MK76	8
X-61B EL 720 AZ 175° 30'	48W406	–	–	–	–	SPARE	–	–	–	–	–	–	–	–	–	–	–	–	–	–	–	–	A	–	–	

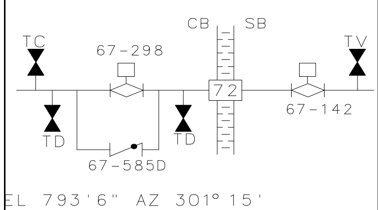
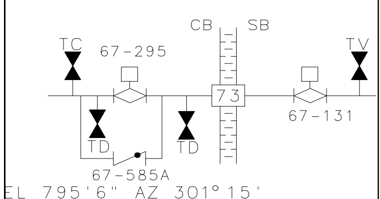
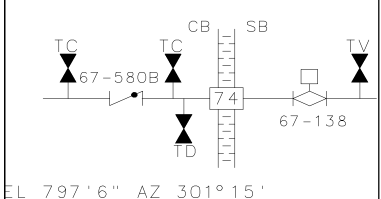
WATTS BAR NUCLEAR PLANT CONTAINMENT PENETRATIONS AND BARRIERS-TABLE 6.2.4-1																										
PENETRATION DATA							VALVE DATA																			
															VALVE STATUS											
DETAILS	DWG NUMBER	GEN DES CRITERION	PROCESS FLUID	FLUID STATE	POSS LEAK PATHS	SYSTEM NUMBER AND PENETRATION DESCRIPTION	VALVE LOCATION	VALVE NUMBER	ESF POWER TRAIN	VALVE TYPE	ACTUATOR	PRI ACT MODE	SEC ACT MODE	ISOLATION SIGNAL	STROKE TIME	NORMAL	SHUTDOWN	POST-ACCIDENT	POWER FAILURE	ILRT	POS IND IN MCR	ESF	APP J TEST	ESSENTIAL/NON-ESS	SHIELD BLDG PENETRATION	NOTES
	47W845-3	56	W	C	AB DE	LWR CONT ERCW SUPPLY (67)	CB SB CB	1054C 91 97	– B A	CK BF BF	– MO MO	SA AT AT	– RM RM	– PB PB	– 66.0 66.0	O O O	O O O	V C C	– AI AI	– C C	N Y Y	N N N	AC AC AC	N	MK74	8
X-62B EL 720 AZ 193°	48W406	–	–	–	–	SPARE	–	–	–	–	–	–	–	–	–	–	–	–	–	–	–	–	A	–	–	
	47W845-3	56	W	C	AB DE	LWR CONT ERCW RETURN (67)	CB SB CB	95 96 575C	A B –	BF BF CK	MO MO –	AT AT SA	RM RM –	PB PB –	66.0 66.0 –	O O O	O O O	C C V	AI AI –	C C –	Y Y N	N N N	AC AC AC	N	MK75	8

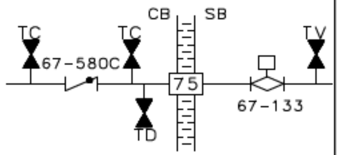
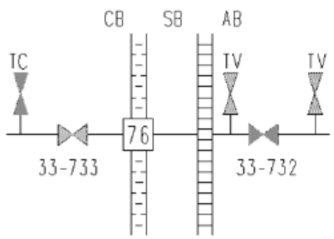
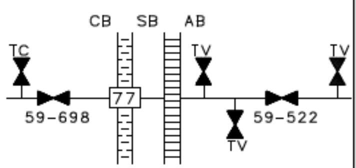
WATTS BAR NUCLEAR PLANT CONTAINMENT PENETRATIONS AND BARRIERS-TABLE 6.2.4-1																										
PENETRATION DATA							VALVE DATA																			
															VALVE STATUS											
DETAILS	DWG NUMBER	GEN DES CRITERION	PROCESS FLUID	FLUID STATE	POSS LEAK PATHS	SYSTEM NUMBER AND PENETRATION DESCRIPTION	VALVE LOCATION	VALVE NUMBER	ESF POWER TRAIN	VALVE TYPE	ACTUATOR	PRI ACT MODE	SEC ACT MODE	ISOLATION SIGNAL	STROKE TIME	NORMAL	SHUTDOWN	POST-ACCIDENT	POWER FAILURE	ILRT	POS IND IN MCR	ESF	APP J TEST	ESSENTIAL/NON-ESS	SHIELD BLDG PENETRATION	NOTES
X-63B EL 720 AZ 194°30'	48W406	–	–	–	–	SPARE	–	–	–	–	–	–	–	–	–	–	–	–	–	–	–	–	A	–	–	
 EL 737 AZ 65°	47W865-5	56	W	C	AB D	INST RM CHILL H2O RETURN (31)	CB SB CB	306 305 3421	A B –	GL GL CK	AO AO –	AT AT SA	RM RM –	PA PA –	10.0 10.0 –	O O O	O O O	C C V	C C –	C C –	Y Y N	N N N	AC AC AC	N	MK92	8
 EL 738 AZ 65°	47W865-5	56	W	C	AB D	INST RM CHILL H2O SUPPLY (31)	CB SB CB	308 309 3407	A B –	GL GL CK	AO AO –	AT AT SA	RM RM –	PA PA –	10.0 10.0 –	O O O	O O O	C C V	C C –	C C –	Y Y N	N N N	AC AC AC	N	MK90	8

WATTS BAR NUCLEAR PLANT CONTAINMENT PENETRATIONS AND BARRIERS-TABLE 6.2.4-1

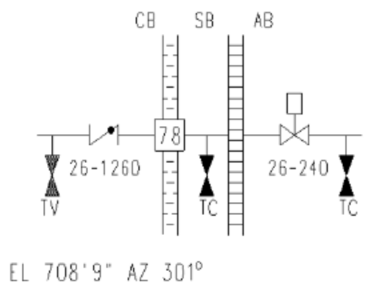
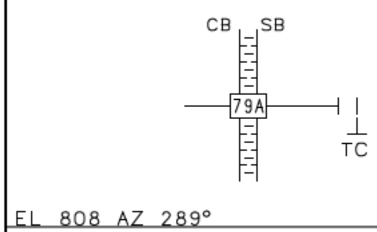
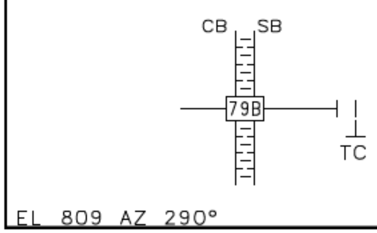
	PENETRATION DATA						VALVE DATA																			
																VALVE STATUS										
DETAILS	DWG NUMBER	GEN DES CRITERION	PROCESS FLUID	FLUID STATE	POSS LEAK PATHS	SYSTEM NUMBER AND PENETRATION DESCRIPTION	VALVE LOCATION	VALVE NUMBER	ESF POWER TRAIN	VALVE TYPE	ACTUATOR	PRI ACT MODE	SEC ACT MODE	ISOLATION SIGNAL	STROKE TIME	NORMAL	SHUTDOWN	POST-ACCIDENT	POWER FAILURE	ILRT	POS IND IN MCR	ESF	APP J TEST	ESSENTIAL/NON-ESS	SHIELD BLDG PENETRATION	NOTES
 EL 737 AZ 104°	47W865-5	56	W	C	AB D	INST RM CHILL H2O RETURN (31)	CB SB CB	327 326 3392	B A –	GL GL CK	AO AO –	AT AT SA	RM RM –	PA PA –	10.0 10.0 –	O O O	O O O	C C V	C C –	C C O	Y Y N	N N N	AC AC AC	N	MK93	8
 EL 738 AZ 104°	47W865-5	56	W	C	AB D	INST RM CHILL H2O SUPPLY (31)	CB SB CB	329 330 3378	B A –	GL GL CK	AO AO –	AT AT SA	RM RM –	PA PA –	10.0 10.0 –	O O O	O O O	C C V	C C –	C C O	Y Y N	N N N	AC AC AC	N	MK91	8
 EL 794'6" AZ 301°15'	47W845-3	56	W	C	AB DE	UPPER CONT ERCW SUPPLY (67)	CB SB	580D 141	– B	CK PG	– MO	SA AT	– RM	– PB	– 66.0	– O	– O	– C	– AI	– C	N Y	N N	AC AC	N	MK88	

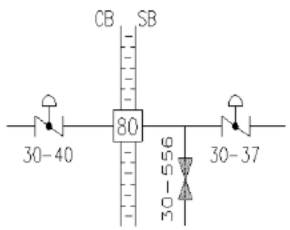
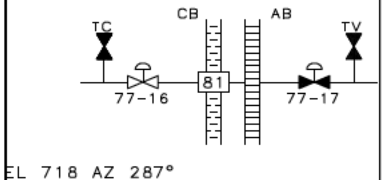
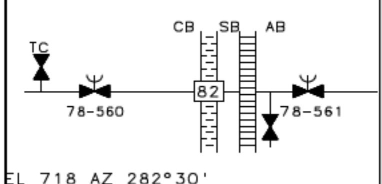
WATTS BAR NUCLEAR PLANT CONTAINMENT PENETRATIONS AND BARRIERS-TABLE 6.2.4-1																										
PENETRATION DATA							VALVE DATA																			
																VALVE STATUS										
DETAILS	DWG NUMBER	GEN DES CRITERION	PROCESS FLUID	FLUID STATE	POSS LEAK PATHS	SYSTEM NUMBER AND PENETRATION DESCRIPTION	VALVE LOCATION	VALVE NUMBER	ESF POWER TRAIN	VALVE TYPE	ACTUATOR	PRI ACT MODE	SEC ACT MODE	ISOLATION SIGNAL	STROKE TIME	NORMAL	SHUTDOWN	POST-ACCIDENT	POWER FAILURE	ILRT	POS IND IN MCR	ESF	APP J TEST	ESSENTIAL/NON-ESS	SHIELD BLDG PENETRATION	NOTES
	47W845-3	56	W	C	AB DE	UPPER CONT ERCW SUPPLY (67)	CB SB	580A 130	– A	CK PG	– MO	SA AT	– RM	– PB	– 66.0	– O	– O	– C	– AI	– C	N Y	N N	AC AC	N	MK86	
	47W845-3	56	W	C	AB DE	UPPER CONT ERCW RETURN (67)	CB SB CB	297 139 585B	B A –	PG PG CK	MO MO –	AT AT SA	RM RM –	PB PB –	66.0 66.0 –	O O O	O O O	C C V	AI AI –	C C –	Y Y N	N N N	AC AC AC	N	MK84	8
	47W845-3	56	W	C	AB DE	UPPER CONT ERCW RETURN (67)	CB SB CB	296 134 585C	A B –	PG PG CK	MO MO –	AT AT SA	RM RM –	PB PB –	66.0 66.0 –	O O O	O O O	C C V	AI AI –	C C –	Y Y N	N N N	AC AC AC	N	MK82	8

WATTS BAR NUCLEAR PLANT CONTAINMENT PENETRATIONS AND BARRIERS-TABLE 6.2.4-1																										
PENETRATION DATA							VALVE DATA																			
															VALVE STATUS											
DETAILS	DWG NUMBER	GEN DES CRITERION	PROCESS FLUID	FLUID STATE	POSS LEAK PATHS	SYSTEM NUMBER AND PENETRATION DESCRIPTION	VALVE LOCATION	VALVE NUMBER	ESF POWER TRAIN	VALVE TYPE	ACTUATOR	PRI ACT MODE	SEC ACT MODE	ISOLATION SIGNAL	STROKE TIME	NORMAL	SHUTDOWN	POST-ACCIDENT	POWER FAILURE	ILRT	POS IND IN MCR	ESF	APP J TEST	ESSENTIAL/NON-ESS	SHIELD BLDG PENETRATION	NOTES
	47W845-3	56	W	C	AB DE	UPPER CONT ERCW RETURN (67)	CB SB CB	298 142 585D	B A –	PG PG CK	MO MO –	AT AT SA	RM RM –	PB PB –	66.0 66.0 –	O O O	O O O	C C V	AI AI –	C C –	Y Y N	N N N	AC AC AC	N	MK89	8
	47W845-3	56	W	C	AB DE	UPPER CONT ERCW RETURN (67)	CB SB CB	295 131 585A	A B –	PG PG CK	MO MO –	AT AT SA	RM RM –	PB PB –	66.0 66.0 –	O O O	O O O	C C V	AI AI –	C C –	Y Y N	N N N	AC AC AC	N	MK87	8
	47W845-3	56	W	C	AB DE	UPPER CONT ERCW SUPPLY (67)	CB SB	580B 138	– B	CK PG	– MO	SA AT	– RM	– PB	– 66.0	O O	O O	C C	– AI	– C	N Y	N N	AC AC	N	MK85	

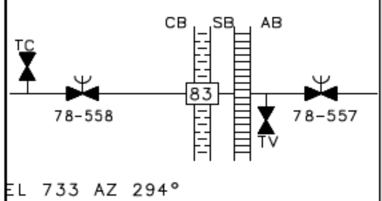
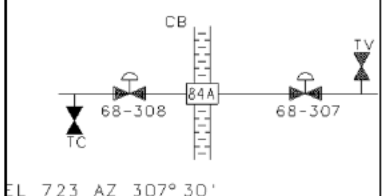
WATTS BAR NUCLEAR PLANT CONTAINMENT PENETRATIONS AND BARRIERS-TABLE 6.2.4-1																									
PENETRATION DATA							VALVE DATA								VALVE STATUS										
DETAILS	DWG NUMBER	GEN DES CRITERION	PROCESS FLUID	FLUID STATE	POSS LEAK PATHS	SYSTEM NUMBER AND PENETRATION DESCRIPTION	VALVE LOCATION	VALVE NUMBER	ESF POWER TRAIN	VALVE TYPE	ACTUATOR	PRI ACT MODE	SEC ACT MODE	ISOLATION SIGNAL	STROKE TIME	NORMAL	SHUTDOWN	POST-ACCIDENT	POWER FAILURE	ILRT	POS IND IN MCR	ESF	APP J TEST	ESSENTIAL/NON-ESS	SHIELD BLDG PENETRATION	NOTES
 EL 799'6" AZ 301°15'	47W845-3	56	W	C	AB DE	UPPER CONT ERCW SUPPLY (67)	CB SB	580C 133	– A	CK PG	– MO	SA AT	– RM	– PB	– 66.0	O O	O O	C C	– AI	– C	N Y	N N	AC AC	N	MK83	
 EL 711 AZ 300°	47W846-2	56	A	C	AD	SERVICE AIR (33)	CB AB	733 732	– –	DI DI	M M	LM LM	– –	– –	– –	C C	O O	C C	– –	C C	N N	N N	AC AC	N	MK97	
 EL 710'6" AZ 299°	47W856-1	56	W	C	AD	DEMIN WATER (59)	CB AB	698 522	– –	DI DI	M M	LM LM	– –	– –	– –	C C	O O	C C	– –	C C	N N	N N	AC AC	N	MK96	

WATTS BAR NUCLEAR PLANT CONTAINMENT PENETRATIONS AND BARRIERS-TABLE 6.2.4-1

PENETRATION DATA							VALVE DATA																		
DETAILS	DWG NUMBER	GEN DES CRITERION	PROCESS FLUID	FLUID STATE	POSS LEAK PATHS	SYSTEM NUMBER AND PENETRATION DESCRIPTION	VALVE LOCATION	VALVE NUMBER	ESF POWER TRAIN	VALVE TYPE	ACTUATOR	PRI ACT MODE	SEC ACT MODE	ISOLATION SIGNAL	STROKE TIME	VALVE STATUS: NORMAL	VALVE STATUS: SHUTDOWN	VALVE STATUS: POST-ACCIDENT	VALVE STATUS: POWER FAILURE	ILRT	POS IND IN MCR	ESF	APP J TEST	ESSENTIAL/NON-ESS	SHIELD BLDG PENETRATION	NOTES
	47W850-9	56	A	C	AB D	FIRE PROTECTION (26)	CB AB	1260 240	– A	CK GA	– MO	SA AT	– RM	– PA	– 20.0	C O	C O	C C	– AI	– C	N Y	N N	AC AC	N	MK98	
	47W814-1 47W462-7	56	I	C	AB	ICE BLOWING (61)	SB			BL	M	LM	–	–	–	C	V	C	–	C	N	N	AB	N	MK23	
	47W814-1 47W462-7	56	I	C	AB	NEGATIVE RETURN (61)	SB			BL	M	LM	–	–	–	C	V	C	–	C	N	Y	AB	N	MK24	

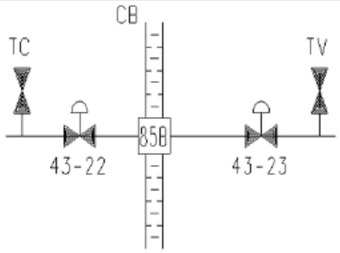
WATTS BAR NUCLEAR PLANT CONTAINMENT PENETRATIONS AND BARRIERS-TABLE 6.2.4-1																										
PENETRATION DATA							VALVE DATA																			
															VALVE STATUS											
DETAILS	DWG NUMBER	GEN DES CRITERION	PROCESS FLUID	FLUID STATE	POSS LEAK PATHS	SYSTEM NUMBER AND PENETRATION DESCRIPTION	VALVE LOCATION	VALVE NUMBER	ESF POWER TRAIN	VALVE TYPE	ACTUATOR	PRI ACT MODE	SEC ACT MODE	ISOLATION SIGNAL	STROKE TIME	NORMAL	SHUTDOWN	POST-ACCIDENT	POWER FAILURE	ILRT	POS IND IN MCR	ESF	APP J TEST	ESSENTIAL/NON-ESS	SHIELD BLDG PENETRATION	NOTES
	47W866-1	56	A	C	AB CE	LWR COMP PRESS RELIEF (30)	CB SB SB	40 37 556	A B -	BF BF GL	AO AO M	AT AT LM	RM RM -	CV CV -	40 40 -	V V LC	C C LC	C C LC	C C -	C C C	Y Y N	N N N	AC AC AC	N		3,6,14
	47W830-1	56	A	C	AD	RC DR TK TO GAS ANALYZER (77)	CB AB	16 17	B A	DI DI	AO AO	AT AT	RM RM	PA PA	10.0 10.0	V V	O O	C C	C C	C C	Y Y	N N	AC AC	N	ASO81	
	47W855-1	56	W	C	AB D	REFUEL CAV PRFCN PUMP SUCT (78)	CB AB	560 561	– –	DI DI	M M	LM LM	– –	– –	– –	C C	V V	C C	– –	C C	N N	N N	AC AC	N	MK94	

WATTS BAR NUCLEAR PLANT CONTAINMENT PENETRATIONS AND BARRIERS-TABLE 6.2.4-1

DETAILS	DWG NUMBER	GEN DES CRITERION	PROCESS FLUID	FLUID STATE	POSS LEAK PATHS	SYSTEM NUMBER AND PENETRATION DESCRIPTION	VALVE LOCATION	VALVE NUMBER	ESF POWER TRAIN	VALVE TYPE	ACTUATOR	PRI ACT MODE	SEC ACT MODE	ISOLATION SIGNAL	STROKE TIME	NORMAL	SHUTDOWN	POST ACCIDENT	POWER FAILURE	ILRT	POS IND IN MCR	ESF	APP J TEST	ESSENTIAL/NON-ESS	SHIELD BLDG PENETRATION	NOTES
PENETRATION DATA							VALVE DATA									VALVE STATUS										
	47W855-1	56	W	C	AB D	REFUEL CAV PRFCN PUMP SUCT (78)	CB AB	558 557	– –	DI DI	M M	LM LM	– –	– –	– –	C C	V V	C C	– –	C C	N N	N N	AC AC	N	MK95	
	47W625-8	56	N	C	AD	P.R.T. TO GAS ANALYZER (68)	CB SB	308 307	B A	GL GL	AO AO	AT AT	RM RM	PA PA	10.0 10.0	V V	O O	C C	C C	C C	Y Y	N N	AC AC	N	MK99M	
X-84B EL 723 AZ 307° 30'	47W600-292	–	W	C	BD	RVLIS (68)	–	–	–	–	–	–	–	–	–	O	O	O	–	–	N	N	A	E	MK101	21

WATTS BAR NUCLEAR PLANT CONTAINMENT PENETRATIONS AND BARRIERS-TABLE 6.2.4-1

DETAILS	DWG NUMBER	GEN DES CRITERION	PROCESS FLUID	FLUID STATE	POSS LEAK PATHS	SYSTEM NUMBER AND PENETRATION DESCRIPTION	VALVE LOCATION	VALVE NUMBER	ESF POWER TRAIN	VALVE TYPE	ACTUATOR	PRI ACT MODE	SEC ACT MODE	ISOLATION SIGNAL	STROKE TIME	NORMAL	SHUTDOWN	POST ACCIDENT	POWER FAILURE	ILRT	POS IND IN MCR	ESF	APP J TEST	ESSENTIAL/NON-ESS	SHIELD BLDG PENETRATION	NOTES
PENETRATION DATA							VALVE DATA									VALVE STATUS										
X-84C EL 723 AZ 307°30'	47W600-292	–	W	C	BD	RVLIS (68)	–	–	–	–	–	–	–	–	–	O	O	O	–	–	N	N	A	E	MK101	21
X-84D EL 723 AZ 307°30'	47W600-292	–	W	C	BD	RVLIS (68)	–	–	–	–	–	–	–	–	–	O	O	O	–	–	N	N	A	E	MK101	21
X-85A EL 723' AZ 306°	47W331-2	–	–	–	–	SPARE	–	–	–	–	–	–	–	–	–	–	–	–	–	–	–	–	A	–	–	

WATTS BAR NUCLEAR PLANT CONTAINMENT PENETRATIONS AND BARRIERS-TABLE 6.2.4-1																										
PENETRATION DATA							VALVE DATA									VALVE STATUS										
DETAILS	DWG NUMBER	GEN DES CRITERION	PROCESS FLUID	FLUID STATE	POSS LEAK PATHS	SYSTEM NUMBER AND PENETRATION DESCRIPTION	VALVE LOCATION	VALVE NUMBER	ESF POWER TRAIN	VALVE TYPE	ACTUATOR	PRI ACT MODE	SEC ACT MODE	ISOLATION SIGNAL	STROKE TIME	NORMAL	SHUTDOWN	POST ACCIDENT	POWER FAILURE	ILRT	POS IND IN MCR	ESF	APP J TEST	ESSENTIAL/NON-ESS	SHIELD BLDG PENETRATION	NOTES
 EL 723' AZ 306°	47W625-1	55	W	H	AB D	HOT LEG SAMPLE LOOPS 1 & 3 (43)	CB SB	22 23	B A	GL GL	AO AO	AT AT	RM RM	PA PA	10.0 10.0	V V	V V	C C	C C	C C	Y Y	N N	AC AC	N	MK100M	15
X-85C EL 723' AZ 306°	47W331-2	–	–	–	–	SPARE	–	–	–	–	–	–	–	–	–	–	–	–	–	–	–	–	A	–	–	
X-85D EL 723 AZ 306°	47W331-2	–	–	–	–	SPARE	–	–	–	–	–	–	–	–	–	–	–	–	–	–	–	–	A	–	–	

WATTS BAR NUCLEAR PLANT CONTAINMENT PENETRATIONS AND BARRIERS-TABLE 6.2.4-1

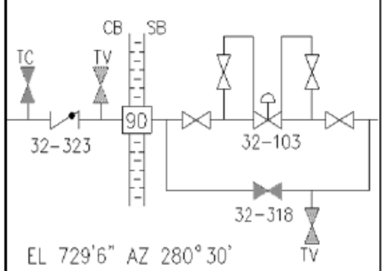
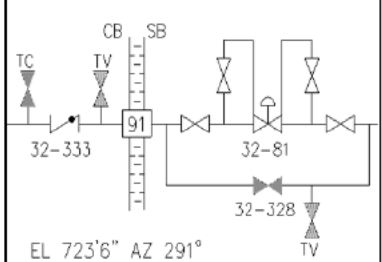
	PENETRATION DATA						VALVE DATA								VALVE STATUS										
DETAILS	DWG NUMBER	GEN DES CRITERION	PROCESS FLUID	FLUID STATE	POSS LEAK PATHS	SYSTEM NUMBER AND PENETRATION DESCRIPTION	VALVE LOCATION	VALVE NUMBER	ESF POWER TRAIN	VALVE TYPE	ACTUATOR	PRI ACT MODE	SEC ACT MODE	ISOLATION SIGNAL	STROKE TIME	NORMAL	SHUTDOWN	POST-ACCIDENT	POWER FAILURE	ILRT	POS IND IN MCR	ESF	APP J TEST	ESSENTIAL/NON-ESS	SHIELD BLDG PENETRATION	NOTES
X-86A EL 721'-6" AZ 307°30'	47W625-15	–	–	–	–	SPARE	–	–	–	–	–	–	–	–	–	–	–	–	–	–	–	–	A	–	–	
X-86B EL 721'-6" AZ 307°30'	47W625-15	–	–	–	–	SPARE	–	–	–	–	–	–	–	–	–	–	–	–	–	–	–	–	A	–	–	
X-86C EL 721'-6" AZ 307°30'	47W625-15	–	–	–	–	SPARE	–	–	–	–	–	–	–	–	–	–	–	–	–	–	–	–	A	–	–	

WATTS BAR NUCLEAR PLANT CONTAINMENT PENETRATIONS AND BARRIERS-TABLE 6.2.4-1

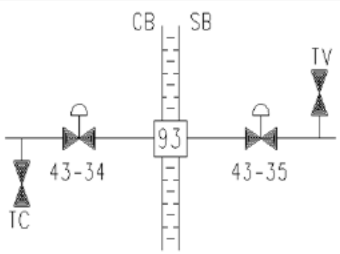
DETAILS	DWG NUMBER	GEN DES CRITERION	PROCESS FLUID	FLUID STATE	POSS LEAK PATHS	SYSTEM NUMBER AND PENETRATION DESCRIPTION	VALVE LOCATION	VALVE NUMBER	ESF POWER TRAIN	VALVE TYPE	ACTUATOR	PRI ACT MODE	SEC ACT MODE	ISOLATION SIGNAL	STROKE TIME	VALVE STATUS: NORMAL	VALVE STATUS: SHUTDOWN	VALVE STATUS: POST-ACCIDENT	VALVE STATUS: POWER FAILURE	ILRT	POS IND IN MCR	ESF	APP J TEST	ESSENTIAL/NON-ESS	SHIELD BLDG PENETRATION	NOTES
X-86D EL 721'6" AZ 307°30'	47W331-2	–	–	–	–	SPARE	–	–	–	–	–	–	–	–	–	–	–	–	–	–	–	–	A	–	–	
X-87A EL 721'6" AZ 306°	47W331-2	–	–	–	–	SPARE	–	–	–	–	–	–	–	–	–	–	–	–	–	–	–	–	A	–	–	
X-87B EL 721'6" AZ 306°	47W600-292 47W331-2	–	W	C	BD	RVLIS (68)	–	–	–	–	–	–	–	–	–	–	–	–	–	–	N	N	A	E	–	21

WATTS BAR NUCLEAR PLANT CONTAINMENT PENETRATIONS AND BARRIERS-TABLE 6.2.4-1

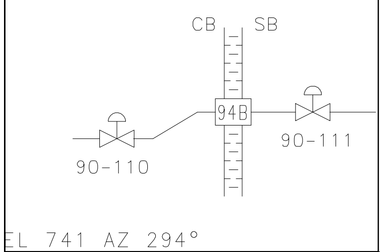
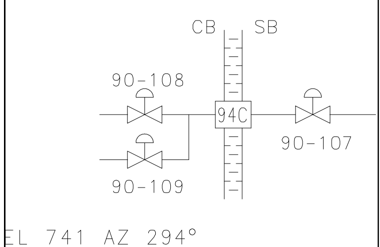
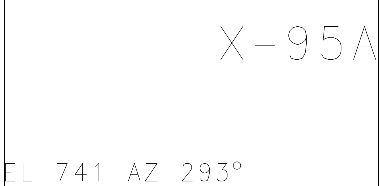
PENETRATION DATA							VALVE DATA										VALVE STATUS									
DETAILS	DWG NUMBER	GEN DES CRITERION	PROCESS FLUID	FLUID STATE	POSS LEAK PATHS	SYSTEM NUMBER AND PENETRATION DESCRIPTION	VALVE LOCATION	VALVE NUMBER	ESF POWER TRAIN	VALVE TYPE	ACTUATOR	PRI ACT MODE	SEC ACT MODE	ISOLATION SIGNAL	STROKE TIME	NORMAL	SHUTDOWN	POST-ACCIDENT	POWER FAILURE	ILRT	POS IND IN MCR	ESF	APP J TEST	ESSENTIAL/NON-ESS	SHIELD BLDG PENETRATION	NOTES
X-87C EL 721'6" AZ 306°	47W600-292 47W331-2	–	W	C	BD	RVLIS (68)	–	–	–	–	–	–	–	–	–	–	–	–	–	–	N	N	A	E	MK102	21
X-87D EL 721'6" AZ 306°	47W600-292 47W331-2	–	W	C	BD	RVLIS (68)	–	–	–	–	–	–	–	–	–	–	–	–	–	–	N	N	A	E	MK102	21
X-88 EL 733 AZ 277°30'	48W406 47W850-9	–	–	–	–	SPARE	–	–	–	–	–	–	–	–	–	–	–	–	–	–	–	–	A	–	MK-103	29

WATTS BAR NUCLEAR PLANT CONTAINMENT PENETRATIONS AND BARRIERS-TABLE 6.2.4-1																										
PENETRATION DATA							VALVE DATA																			
															VALVE STATUS											
DETAILS	DWG NUMBER	GEN DES CRITERION	PROCESS FLUID	FLUID STATE	POSS LEAK PATHS	SYSTEM NUMBER AND PENETRATION DESCRIPTION	VALVE LOCATION	VALVE NUMBER	ESF POWER TRAIN	VALVE TYPE	ACTUATOR	PRI ACT MODE	SEC ACT MODE	ISOLATION SIGNAL	STROKE TIME	NORMAL	SHUTDOWN	POST-ACCIDENT	POWER FAILURE	ILRT	POS IND IN MCR	ESF	APP J TEST	ESSENTIAL/NON-ESS	SHIELD BLDG PENETRATION	NOTES
X-89 EL 732 AZ 278° 30'	48W406 47W850-9	–	–	–	–	SPARE	–	–	–	–	–	–	–	–	–	–	–	–	–	–	–	–	A	–	MK-104	30
	47W848-1	56	A	C	AB D	CONTROL AIR TR-B (32)	CB SB SB	323 103 318	– B –	CK GL GA	– AO M	SA AT LM	– RM –	– PB –	– 10.0 –	O O C	O O C	C C C	– C AI	– C C	N Y N	N N N	AC AC AC	N	MK105	10
	47W848-1	56	A	C	AB D	CONTROL AIR TR-A (32)	CB SB SB	333 81 328	– A –	CK GL GA	– AO M	SA AT LM	– RM –	– PB –	– 10.0 –	O O C	O O C	C C C	– C AI	– C C	N Y N	N N N	AC AC AC	N	MK56	10

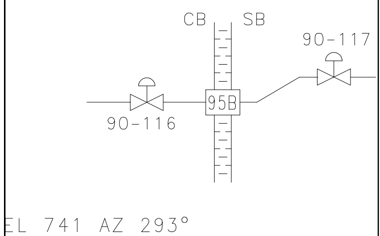
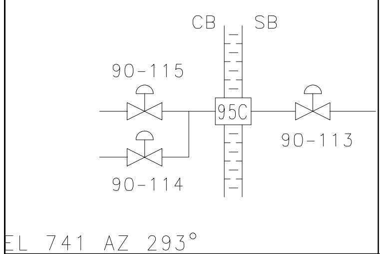
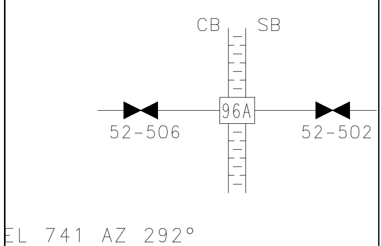
DETAILS	DWG NUMBER	GEN DES CRITERION	PROCESS FLUID	FLUID STATE	POSS LEAK PATHS	SYSTEM NUMBER AND PENETRATION DESCRIPTION	VALVE LOCATION	VALVE NUMBER	ESF POWER TRAIN	VALVE TYPE	ACTUATOR	PRI ACT MODE	SEC ACT MODE	ISOLATION SIGNAL	STROKE TIME	NORMAL	SHUTDOWN	POST-ACCIDENT	POWER FAILURE	ILRT	POS IND IN MCR	ESF	APP J TEST	ESSENTIAL/NON-ESS	SHIELD BLDG PENETRATION	NOTES
	WATTS BAR NUCLEAR PLANT CONTAINMENT PENETRATIONS AND BARRIERS-TABLE 6.2.4-1																									
	PENETRATION DATA						**VALVE DATA**									VALVE STATUS										
X-92A EL 723 AZ 290°	47W625-11	–	–	–	–	SPARE	–	–	–	–	–	–	–	–	–	–	–	–	–	–	–	–	A	–	–	
X-92B EL 723 AZ 290°	47W625-11	–	–	–	–	SPARE	–	–	–	–	–	–	–	–	–	–	–	–	–	–	–	–	A	–	–	
X-92C EL 723'-6" AZ 290°	47W625-15	–	–	–	–	SPARE	–	–	–	–	–	–	–	–	–	–	–	–	–	–	–	–	A	–	–	

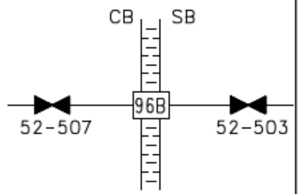
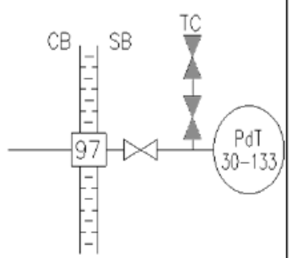
WATTS BAR NUCLEAR PLANT CONTAINMENT PENETRATIONS AND BARRIERS-TABLE 6.2.4-1																										
PENETRATION DATA							VALVE DATA																			
																VALVE STATUS										
DETAILS	DWG NUMBER	GEN DES CRITERION	PROCESS FLUID	FLUID STATE	POSS LEAK PATHS	SYSTEM NUMBER AND PENETRATION DESCRIPTION	VALVE LOCATION	VALVE NUMBER	ESF POWER TRAIN	VALVE TYPE	ACTUATOR	PRI ACT MODE	SEC ACT MODE	ISOLATION SIGNAL	STROKE TIME	NORMAL	SHUTDOWN	POST-ACCIDENT	POWER FAILURE	ILRT	POS IND IN MCR	ESF	APP J TEST	ESSENTIAL/NON-ESS	SHIELD BLDG PENETRATION	NOTES
X-92D EL 723'6" AZ 290°	47W331-3	–	–	–	–	SPARE	–	–	–	–	–	–	–	–	–	–	–	–	–	–	–	–	A	–	–	
 EL 723'6" AZ 289°	47W625-2	56	W	C	AB D	ACCUM SAMPLE (43)	CB SB	34 35	B A	GL GL	AO AO	AT AT	RM RM	PA PA	5 0 5 0	V V	C C	C C	C C	C C	Y Y	N N	AC AC	N	MK58	
X-94A EL 741 AZ 294°	47W600-105 48W406	–	A	C	–	SPARE	–	–	–	–	–	–	–	–	–	–	–	–	–	–	–	–	A	–	–	

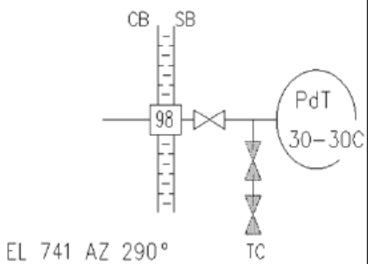
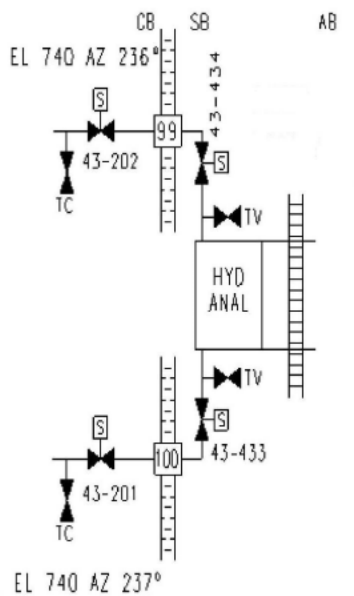
Unit 2

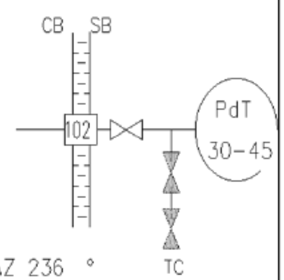
WATTS BAR NUCLEAR PLANT CONTAINMENT PENETRATIONS AND BARRIERS-TABLE 6.2.4-1																										
PENETRATION DATA							VALVE DATA																			
															VALVE STATUS											
DETAILS	DWG NUMBER	GEN DES CRITERION	PROCESS FLUID	FLUID STATE	POSS LEAK PATHS	SYSTEM NUMBER AND PENETRATION DESCRIPTION	VALVE LOCATION	VALVE NUMBER	ESF POWER TRAIN	VALVE TYPE	ACTUATOR	PRI ACT MODE	SEC ACT MODE	ISOLATION SIGNAL	STROKE TIME	NORMAL	SHUTDOWN	POST-ACCIDENT	POWER FAILURE	ILRT	POS IND IN MCR	ESF	APP J TEST	ESSENTIAL/NON-ESS	SHIELD BLDG PENETRATION	NOTES
	47W600-105	56	A	C	AB D	LOWER COMP AIR MON INTAKE (90)	CB SB	110 111	B A	GL GL	AO AO	AT AT	RM RM	CV CV	5 0 5 0	O O	O O	C C	C C	C C	Y Y	N N	AC AC	N	MK59M	
	47W600-105	56	A	C	AB D	LOWER COMP AIR MON RETURN (90)	CB SB CB	108 107 109	B A B	GL GL GL	AO AO AO	AT AT AT	RM RM RM	CV CV CV	5 0 5 0 5 0	O O O	O O O	C C C	C C C	C C C	Y Y Y	N N N	AC AC AC	N	MK59M	
	47W600-105	–	–	–	–	SPARE	–	–	–	–	–	–	–	–	–	–	–	–	–	–	–	–	A	–	–	

WATTS BAR NUCLEAR PLANT CONTAINMENT PENETRATIONS AND BARRIERS-TABLE 6.2.4-1

DETAILS	DWG NUMBER	GEN DES CRITERION	PROCESS FLUID	FLUID STATE	POSS LEAK PATHS	SYSTEM NUMBER AND PENETRATION DESCRIPTION	VALVE LOCATION	VALVE NUMBER	ESF POWER TRAIN	VALVE TYPE	ACTUATOR	PRI ACT MODE	SEC ACT MODE	ISOLATION SIGNAL	STROKE TIME	VALVE STATUS: NORMAL	VALVE STATUS: SHUTDOWN	VALVE STATUS: POST-ACCIDENT	VALVE STATUS: POWER FAILURE	ILRT	POS IND IN MCR	ESF	APP J TEST	ESSENTIAL/NON-ESS	SHIELD BLDG PENETRATION	NOTES
	47W600-105	56	A	C	AB D	UPPER COMP AIR MON INTAKE (90)	CB SB	116 117	B A	GL GL	AO AO	AT AT	RM RM	CV CV	5 0 5 0	O O	O O	C C	C C	C C	Y Y	N N	AC AC	N	MK60M	
	47W600-105	56	A	C	AB D	UPPER COMP AIR MON RETURN (90)	CB SB CB	114 113 115	B A B	GL GL GL	AO AO AO	AT AT AT	RM RM RM	CV CV CV	5 0 5 0 5 0	O O O	O O O	C C C	C C C	C C C	Y Y Y	N N N	AC AC AC	N	MK60M	
	47W331-3	56	A	C	B	ILRT SENSOR LINE (52)	CB SB	506 502	– –	GL GL	M M	LM LM	– –	– –	– –	C C	V V	C C	– –	O O	N N	N N	AC AC	N	–	

WATTS BAR NUCLEAR PLANT CONTAINMENT PENETRATIONS AND BARRIERS-TABLE 6.2.4-1																									
PENETRATION DATA							VALVE DATA																		
															VALVE STATUS										
DETAILS	DWG NUMBER	GEN DES CRITERION	PROCESS FLUID	FLUID STATE	POSS LEAK PATHS	SYSTEM NUMBER AND PENETRATION DESCRIPTION	VALVE LOCATION	VALVE NUMBER	ESF POWER TRAIN	VALVE TYPE	ACTUATOR	PRI ACT MODE	SEC ACT MODE	ISOLATION SIGNAL	STROKE TIME	NORMAL	SHUTDOWN	POST ACCIDENT	POWER FAILURE	ILRT	POS IND IN MCR	ESF	APP J TEST	ESSENTIAL/NON-ESS	SHIELD BLDG PENETRATION	NOTES
 EL 741 AZ 292°	47W331-3	56	A	C	B	ILRT SENSOR LINE (52)	CB SB	507 503	– –	GL GL	M M	LM LM	– –	– –	– –	C C	V V	C C	– –	O O	N N	N N	AC AC	N	–	
X-96C 741 AZ 292°	47W600-89	–	–	–	–	SPARE	–	–	–	–	–	–	–	–	–	–	–	–	–	–	–	–	A	–	–	
 EL 741 AZ 291°	47W866-1 2-47W600-89	–	A	C	B	dP SENSOR (30)	SB	SNSR	–	–	–	–	–	–	–	O	O	O	–	–	–	N	A	N	–	12

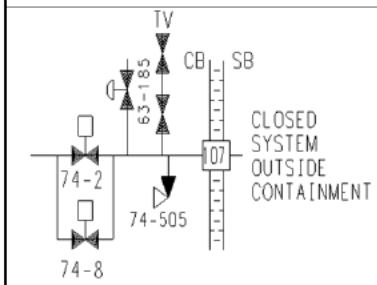
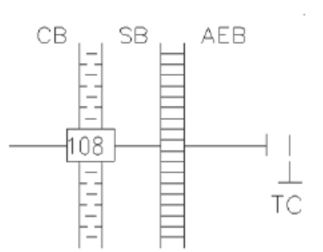
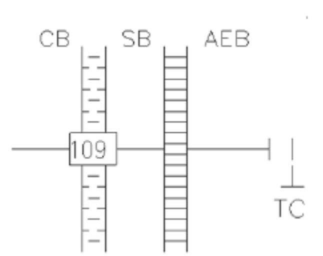
WATTS BAR NUCLEAR PLANT CONTAINMENT PENETRATIONS AND BARRIERS-TABLE 6.2.4-1																										
PENETRATION DATA											VALVE DATA															
																VALVE STATUS										
DETAILS	DWG NUMBER	GEN DES CRITERION	PROCESS FLUID	FLUID STATE	POSS LEAK PATHS	SYSTEM NUMBER AND PENETRATION DESCRIPTION	VALVE LOCATION	VALVE NUMBER	ESF POWER TRAIN	VALVE TYPE	ACTUATOR	PRI ACT MODE	SEC ACT MODE	ISOLATION SIGNAL	STROKE TIME	NORMAL	SHUTDOWN	POST-ACCIDENT	POWER FAILURE	ILRT	POS IND IN MCR	ESF	APP J TEST	ESSENTIAL/NON-ESS	SHIELD BLDG PENETRATION	NOTES
	2-47W600-89	–	A	C	B	dP SENSOR (30)	SB	SNSR	–	–	–	–	–	–	–	O	O	O	–	–	–	N	A	N	–	12
	47W625-11	56	A	C	BD	HYDROGEN ANALYZER (43)	CB	202	A	GL	SO	RM	–	–	–	C	C	O	C	C	Y	N	AC	E	MK-34	
							SB	434	A	GL	SO	RM	–	–	–	C	C	O	C	C	Y	N	AC			
	47W625-11	56	A	C	BD	HYDROGEN ANALYZER (43)	CB	201	A	GL	SO	RM	–	–	–	C	C	O	C	C	Y	N	AC	E	MK-34	
							SB	433	A	GL	SO	RM	–	–	–	C	C	O	C	C	Y	N	AC			

WATTS BAR NUCLEAR PLANT CONTAINMENT PENETRATIONS AND BARRIERS-TABLE 6.2.4-1																										
PENETRATION DATA							VALVE DATA																			
															VALVE STATUS											
DETAILS	DWG NUMBER	GEN DES CRITERION	PROCESS FLUID	FLUID STATE	POSS LEAK PATHS	SYSTEM NUMBER AND PENETRATION DESCRIPTION	VALVE LOCATION	VALVE NUMBER	ESF POWER TRAIN	VALVE TYPE	ACTUATOR	PRI ACT MODE	SEC ACT MODE	ISOLATION SIGNAL	STROKE TIME	NORMAL	SHUTDOWN	POST-ACCIDENT	POWER FAILURE	ILRT	POS IND IN MCR	ESF	APP J TEST	ESSENTIAL/NON-ESS	SHIELD BLDG PENETRATION	NOTES
X-101 EL 723'6" AZ 288°	48W406	–	–	–	–	SPARE	–	–	–	–	–	–	–	–	–	–	–	–	–	–	–	–	A	–	–	
 EL 728' AZ 236 °	2-47W600-89	–	A	C	B	dP SENSOR (30)	SB	SNSR	–	–	–	–	–	–	–	O	O	O	–	–	–	N	A	E	–	12
X-103 EL 723'6" AZ 287°	48W406	–	–	–	–	SPARE	–	–	–	–	–	–	–	–	–	–	–	–	–	–	–	–	A	–	–	

WATTS BAR NUCLEAR PLANT CONTAINMENT PENETRATIONS AND BARRIERS-TABLE 6.2.4-1

PENETRATION DATA							VALVE DATA																			
																VALVE STATUS										
DETAILS	DWG NUMBER	GEN DES CRITERION	PROCESS FLUID	FLUID STATE	POSS LEAK PATHS	SYSTEM NUMBER AND PENETRATION DESCRIPTION	VALVE LOCATION	VALVE NUMBER	ESF POWER TRAIN	VALVE TYPE	ACTUATOR	PRI ACT MODE	SEC ACT MODE	ISOLATION SIGNAL	STROKE TIME	NORMAL	SHUTDOWN	POST ACCIDENT	POWER FAILURE	ILRT	POS IND IN MCR	ESF	APP J TEST	ESSENTIAL/NON-ESS	SHIELD BLDG PENETRATION	NOTES
X-104 EL 728 AZ 237°	48W406	–	–	–	–	SPARE	–	–	–	–	–	–	–	–	–	–	–	–	–	–	–	–	A	–	–	
X-105 EL 722'-6" AZ 288°	47W625-15	–	–	–	–	SPARE	–	–	–	–	–	–	–	–	–	–	–	–	–	–	–	–	A	–	–	
X-106 EL 727' AZ 236°	47W625-15	–	–	–	–	SPARE	–	–	–	–	–	–	–	–	–	–	–	–	–	–	–	–	A	–	–	

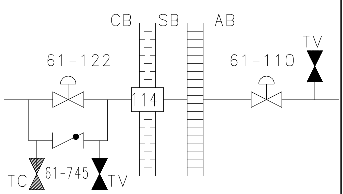
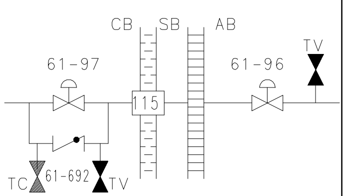
WATTS BAR NUCLEAR PLANT CONTAINMENT PENETRATIONS AND BARRIERS-TABLE 6.2.4-1

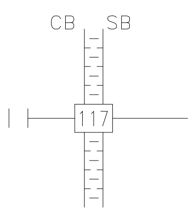
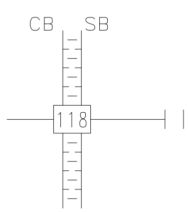
DETAILS	DWG NUMBER	GEN DES CRITERION	PROCESS FLUID	FLUID STATE	POSS LEAK PATHS	SYSTEM NUMBER AND PENETRATION DESCRIPTION	VALVE LOCATION	VALVE NUMBER	ESF POWER TRAIN	VALVE TYPE	ACTUATOR	PRI ACT MODE	SEC ACT MODE	ISOLATION SIGNAL	STROKE TIME	NORMAL	SHUTDOWN	POST ACCIDENT	POWER FAILURE	ILRT	POS IND IN MCR	ESF	APP J TEST	ESSENTIAL/NON-ESS	SHIELD BLDG PENETRATION	NOTES
PENETRATION DATA							VALVE DATA									VALVE STATUS										
 EL 717'11.5" AZ 300°30'	47W810-1 47W432-2	55	W	H	BD	RHR SUPPLY TO PUMPS (74, 63)	CB CB CB CB	2 8 505 185	B A – A	GA GA RV GL	MO MO – AO	RM RM – RM	LM LM – –	– – – PA	– – – 10.0	C C C V	V V C V	V V V C	AI AI – C	O C C C	Y Y N Y	N N N N	– – – –	E	MK38	
 EL 711'6" AZ 218°	47W435-18 -22	–	A	C	AB E	MAINT PORT	AB	–	–	BL	M	LM	–	–	–	C	V	C	–	C	N	N	AB	N	AS108	
 EL 711'6" AZ 222°	47W435-18 -22	–	A	C	AB E	MAINT PORT	AB	–	–	BL	M	LM	–	–	–	C	V	C	–	C	N	N	AB	N	AS109	

WATTS BAR NUCLEAR PLANT CONTAINMENT PENETRATIONS AND BARRIERS-TABLE 6.2.4-1

PENETRATION DATA							VALVE DATA																			
															VALVE STATUS											
DETAILS	DWG NUMBER	GEN DES CRITERION	PROCESS FLUID	FLUID STATE	POSS LEAK PATHS	SYSTEM NUMBER AND PENETRATION DESCRIPTION	VALVE LOCATION	VALVE NUMBER	ESF POWER TRAIN	VALVE TYPE	ACTUATOR	PRI ACT MODE	SEC ACT MODE	ISOLATION SIGNAL	STROKE TIME	NORMAL	SHUTDOWN	POST-ACCIDENT	POWER FAILURE	ILRT	POS IND IN MCR	ESF	APP J TEST	ESSENTIAL/NON-ESS	SHIELD BLDG PENETRATION	NOTES
X-110 EL 711'6" AZ 209°	48N406 47W435-22	–	–	–	B	SPARE	–	–	–	–	–	–	–	–	–	–	–	–	–	–	–	–	A	–	MK32	
X-111 EL 845'9" AZ 90°	48W406	–	–	–	–	SPARE	–	–	–	–	–	–	–	–	–	–	–	–	–	–	–	–	A	–	–	
X-112 EL 845'9" AZ 270°	48W406	–	–	–	–	SPARE	–	–	–	–	–	–	–	–	–	–	–	–	–	–	–	–	A	–	–	

WATTS BAR NUCLEAR PLANT CONTAINMENT PENETRATIONS AND BARRIERS-TABLE 6.2.4-1

PENETRATION DATA							VALVE DATA									VALVE STATUS										
DETAILS	DWG NUMBER	GEN DES CRITERION	PROCESS FLUID	FLUID STATE	POSS LEAK PATHS	SYSTEM NUMBER AND PENETRATION DESCRIPTION	VALVE LOCATION	VALVE NUMBER	ESF POWER TRAIN	VALVE TYPE	ACTUATOR	PRI ACT MODE	SEC ACT MODE	ISOLATION SIGNAL	STROKE TIME	NORMAL	SHUTDOWN	POST-ACCIDENT	POWER FAILURE	ILRT	POS IND IN MCR	ESF	APP J TEST	ESSENTIAL/NON-ESS	SHIELD BLDG PENETRATION	NOTES
X-113 EL 740 AZ 216°	48W406	–	–	–	–	SPARE	–	–	–	–	–	–	–	–	–	–	–	–	–	–	–	–	A	–	–	
 EL 775 AZ 300°	47W814-2	56	G	C	AB D	GLYCOL FLOOR COOLING (61)	CB AB CB	122 110 745	B A –	DI DI CK	AO AO –	AT AT SA	RM RM –	PA PA –	30.0 30.0 –	O O O	O O O	C C V	C C –	O O O	Y Y N	Y Y Y	C C C	N	MK110	8
 EL 771'6" AZ 300°	47W814-2	56	G	C	AB D	GLYCOL FLOOR COOLING (61)	CB AB CB	97 96 692	B A –	DI DI CK	AO AO –	AT AT SA	RM RM –	PA PA –	30.0 30.0 –	O O O	O O O	C C V	C C –	O O O	Y Y N	Y Y Y	C C C	N	MK111	8

WATTS BAR NUCLEAR PLANT CONTAINMENT PENETRATIONS AND BARRIERS-TABLE 6.2.4-1																										
PENETRATION DATA							VALVE DATA																			
															VALVE STATUS											
DETAILS	DWG NUMBER	GEN DES CRITERION	PROCESS FLUID	FLUID STATE	POSS LEAK PATHS	SYSTEM NUMBER AND PENETRATION DESCRIPTION	VALVE LOCATION	VALVE NUMBER	ESF POWER TRAIN	VALVE TYPE	ACTUATOR	PRI ACT MODE	SEC ACT MODE	ISOLATION SIGNAL	STROKE TIME	NORMAL	SHUTDOWN	POST-ACCIDENT	POWER FAILURE	ILRT	POS IND IN MCR	ESF	APP J TEST	ESSENTIAL/NON-ESS	SHIELD BLDG PENETRATION	NOTES
X-116 EL 760 AZ 300°	48W406	–	–	–	–	SPARE	–	–	–	–	–	–	–	–	–	–	–	–	–	–	–	–	A	–	MK7	
 EL 758 AZ 300°	47W301-1	–	A	C	AB	MAINT PORT	CB	–	–	BL	M	LM	–	–	–	C	V	C	–	C	N	N	AB	N	MK8	
 EL 708'6" AZ 209°	72-4334-318	–	W	C	AB C	LAYUP WATER TREATMENT (41) AND ANNULUS FLOOD MODE DRAIN	SB	–	–	BL	M	LM	–	–	–	C	O	C	–	O	N	N	AB	N	MK18	27

WATTS BAR NUCLEAR PLANT CONTAINMENT PENETRATIONS AND BARRIERS-TABLE 6.2.4-1																										
PENETRATION DATA							VALVE DATA																			
															VALVE STATUS											
DETAILS	DWG NUMBER	GEN DES CRITERION	PROCESS FLUID	FLUID STATE	POSS LEAK PATHS	SYSTEM NUMBER AND PENETRATION DESCRIPTION	VALVE LOCATION	VALVE NUMBER	ESF POWER TRAIN	VALVE TYPE	ACTUATOR	PRI ACT MODE	SEC ACT MODE	ISOLATION SIGNAL	STROKE TIME	NORMAL	SHUTDOWN	POST-ACCIDENT	POWER FAILURE	ILRT	POS IND IN MCR	ESF	APP J TEST	ESSENTIAL/NON-ESS	SHIELD BLDG PENETRATION	NOTES
X-119 EL 844'5" AZ 90°	48W406	–	–	–	–	SPARE	–	–	–	–	–	–	–	–	–	–	–	–	–	–	–	–	A	–	–	
X-120 EL 844'5" AZ 270°	48W406	–	–	–	–	SPARE	–	–	–	–	–	–	–	–	–	–	–	–	–	–	–	–	A	–	–	

TABLE 6.2.4-2 (Sheet 1 of 6)

POSSIBLE BYPASS LEAKAGE PATHS TO THE AUXILIARY BUILDING

Penetration Number	Penetrating Line Name	Description
X-2 A, B	Personnel Access Hatch	Any leakage would be treated by the ABGTS
X-3	Fuel Transfer Tube	Any leakage would be treated by the ABGTS
X-15	Chemical and Volume Letdown Line	Any leakage would be treated by the ABGTS
X-16*	Normal Charging Line	High water pressure maintained on outboard valve, even in the event of a single failure.
X-17	RHR Hot Leg Injection	System is in operation after an accident. Cross ties between pumps maintain flows in the event of a single failure.
X-19 A&B*	RHR Sump Suction line	Line is always filled with water. No atmospheric bypass leakage to the Auxiliary Building can occur after a LOCA
X-20 A&B*	SIS RHR Pump Discharge	System is in operation after a LOCA. Cross ties between pumps maintain flows and pressure in the event of a single failure.
X-21*	Safety Injection Pump Discharge	Line in use during a LOCA which will be pressurized even in the event of a single failure due to cross ties between pumps.
X-22*	Charging Pump Discharge	Same as for Penetration No. 21.
X-23	PAS Containment Air Sample	Any leakage would be treated by the ABGTS.
X-24*	SIS Relief Valve Discharge	Any leakage through the relief valve is prevented as the valves are pressurized outside containment by the SIS system. Any leakage would be into containment.
X-25A	Pressurizer Liquid Sample	Any leakage would be treated by the ABGTS.
X-25D	Pressurizer Steam Sample	Any leakage would be treated by the ABGTS.
X-27 A, B, C, D	Steam Generator Sample Lines	Any leakage would be treated by the ABGTS.
X-28	PAS Containment Sump Return Sample (Unit 1 Only)	Any leakage would be treated by the ABGTS.
X-29	CCS from RC Pump Coolers	Any leakage would be treated by the ABGTS.
X-30	Accumulator to Holdup Tank	Any leakage would be treated by the ABGTS.

TABLE 6.2.4-2 (Sheet 2 of 6)

POSSIBLE BYPASS LEAKAGE PATHS TO THE AUXILIARY BUILDING

Penetration Number	Penetrating Line Name	Description
X-31	Fire Protection	Any leakage would be treated by the ABGTS.
X-32*	Safety Injection Pump Discharge	Same as for Penetration No. 21.
X-33*	Safety Injection Pump Discharge	Same as for Penetration No. 21.
X-34	Control Air I&C	Any leakage would be treated by the ABGTS.
X-35	CCS from Excess Letdown Heat Exchanger	Same as for Penetration No. 29.
X-39A	N_2 to Accumulators	Any leakage would be treated by the ABGTS.
X-39B	N_2 to Pressurizer Relief Tank	Any leakage would be treated by the ABGTS.
X-40D	Service Air	Any leakage would be treated by the ABGTS.
X-41	Floor Sump Pump Discharge	Any leakage would be treated by the ABGTS.
X-42	Pressurizer Relief Tank Makeup	Any leakage would be treated by the ABGTS.
X-43 A*,B*,C*,D*	To RC Pump Seals	The line is pressurized during a LOCA even in the event of a single failure. If there was any leakage it would be treated by the ABGTS.
X-44	From RC Pump Seals	Any leakage would be treated by the ABGTS.
X-45	RC Drain Tank and PRT to Vent Header	Any leakage would be treated by the ABGTS.
X-46	RC Drain Tank Pump Discharge	Any leakage would be treated by the ABGTS.
X-47A	Glycol Line to Ice Condenser	Any leakage would be treated by the ABGTS.
X-47B	Glycol Line from Ice Condenser	Any leakage would be treated by the ABGTS.
X-48 A&B*	Containment Spray	System in operation after a LOCA. A 30-day water leg seal is maintained in this line.
X-49 A&B	RHR Spray	System in operation after a LOCA. System pressure maintained even in the event of a single failure due to pump cross ties.
X-50A	RCP Thermal Barrier Return	Same as for Penetration No. 29.

TABLE 6.2.4-2 (Sheet 3 of 6)

POSSIBLE BYPASS LEAKAGE PATHS TO THE AUXILIARY BUILDING

Penetration Number	Penetrating Line Name	Description
X-50B	RCP Thermal Barrier Supply	Any leakage would be treated by the ABGTS
X-52*	CCS to RC Pump Coolers	Same as for Penetration No. 43.
X-53*	CCS to Excess Letdown Heat Exchanger	Same as for Penetration No. 43.
X-56A*	Lower containment ERCW Supply	High water pressure maintained on outboard valve, even in the event of a single failure.
X-57A*	Lower containment ERCW Return	High water pressure maintained on outboard valve, even in the event of a single failure.
X-58A*	Lower containment ERCW Supply	High water pressure maintained on outboard valve, even in the event of a single failure.
X-58B	RCS Pressure Sensor	Any leakage would be treated by the ABGTS.
X-59A*	Lower containment ERCW Return	High water pressure maintained on outboard valve, even in the event of a single failure.
X-60A*	Lower containment ERCW Supply	High water pressure maintained on outboard valve, even in the event of a single failure.
X-61A*	Lower containment ERCW Return	High water pressure maintained on outboard valve, even in the event of a single failure.
X-62A*	Lower containment ERCW Supply	High water pressure maintained on outboard valve, even in the event of a single failure.
X-63A*	Lower containment ERCW Return	High water pressure maintained on outboard valve, even in the event of a single failure.
X-64	Instrument Room AC Chilled Water Return	Any leakage would be treated by the EGTS.
X-65	Instrument Room AC Chilled Water Supply	Any leakage would be treated by the EGTS.
X-71*	Upper Containment ERCW Return	High water pressure maintained on outboard valve, even in the event of

TABLE 6.2.4-2 (Sheet 4 of 6)

POSSIBLE BYPASS LEAKAGE PATHS TO THE AUXILIARY BUILDING

Penetration Number	Penetrating Line Name	Description
		of a single failure.
X-72*	Upper Containment ERCW Return	High water pressure maintained on outboard valve, even in the event of a single failure.
X-73*	Upper Containment ERCW Return	High water pressure maintained on outboard valve, even in the event of a single failure.
X-74*	Upper Containment ERCW Supply	High water pressure maintained on outboard valve, even in the event of a single failure.
X-75*	Upper Containment ERCW Supply	High water pressure maintained on outboard valve, even in the event of a single failure.
X-76	Service Air	Any leakage would be treated by the ABGTS.
X-77	Demineralized Water	Any leakage would be treated by the ABGTS.
X-78	Fire Protection	Any leakage would be treated by the ABGTS.
X-81	RC Drain Tank to Gas Analyzer	Any leakage would be treated by the ABGTS.
X-82	Refueling Cavity C-U Pump Suction	Any leakage would be treated by the ABGTS.
X-83	Refueling Cavity C-U Pump Suction	Any leakage would be treated by the ABGTS.
X-84A	Pressurizer Relief Tank to Gas Analyzer	Any leakage would be treated by the ABGTS.
X-84B	Reactor Vessel Level Indicating System	Any leakage would be treated by the ABGTS.
X-84C	Reactor Vessel Level Indicating System	Any leakage would be treated by the ABGTS.
X-84D	Reactor Vessel Level Indicating System	Any leakage would be treated by the ABGTS.
X-85A	Excess Letdown Heat Exchanger to Boron Analyze (Not used for Unit 1 Operation).	Any leakage would be treated by the ABGTS.

TABLE 6.2.4-2 (Sheet 5 of 6)

POSSIBLE BYPASS LEAKAGE PATHS TO THE AUXILIARY BUILDING

Penetration Number	Penetrating Line Name	Description
X-85B	Hot Leg Sample	Any leakage would be treated by the ABGTS.
X-86A	PAS Containment Air Sample (Unit 1 Only)	Any leakage would be treated by the ABGTS.
X-86B	PAS Containment Air Sample (Unit 1 Only)	Any leakage would be treated by the ABGTS.
X-86C	PAS Containment Sump Sample (Unit 1 Only)	Any leakage would be treated by the ABGTS.
X-87B	Reactor Vessel Level Indicating System	Any leakage would be treated by the ABGTS.
X-87C	Reactor Vessel Level Indicating System	Any leakage would be treated by the ABGTS.
X-87D	Reactor Vessel Level Indicating System	Any leakage would be treated by the ABGTS.
X-90	Control Air	Any leakage would be treated by the ABGTS.
X-91	Control Air	Any leakage would be treated by the ABGTS.
X-92 A, B	H_2 Analyzers (Unit 1 Only)	Any leakage would be treated by the ABGTS.
X-92C	PAS Hot Leg Sample (Unit 1 Only)	Any leakage would be treated by the ABGTS.
X-93	Accumulator Sample	Any leakage would be treated by the ABGTS.
X-94 B, C	Containment Atmosphere Radiation Monitor	Any leakage would be treated by the ABGTS.
X-95 B, C	Containment Atmosphere Radiation Monitor	Any leakage would be treated by the ABGTS.
X-99	H_2 Analyzers	Any leakage would be treated by the ABGTS.
X-100	H_2 Analyzers	Any leakage would be treated by the ABGTS.
X-105	PAS Containment Air Sample (Unit 1 Only)	Any leakage would be treated by the ABGTS.

TABLE 6.2.4-2 (Sheet 6 of 6)

POSSIBLE BYPASS LEAKAGE PATHS TO THE AUXILIARY BUILDING

Penetration Number	Penetrating Line Name	Description
X-106	PAS Hot Leg Sample (Unit 1 Only)	Any leakage would be treated by the ABGTS.
X-107	RHR Supply	Any leakage would be treated by the ABGTS.
X-114	Ice Condenser (from Glycol Floor Cooling Coils)	Any leakage would be treated by the ABGTS.
X-115	Ice Condenser (to Glycol Floor Cooling Coils)	Any leakage would be treated by the ABGTS.

* Not a bypass leakage path to the Auxiliary Building

TABLE 6.2.4-3 (Sheet 1 of 3)

PREVENTION OF BYPASS LEAKAGE TO THE ATMOSPHERE

Penetration Number	Penetration Line Name	Description
X-4	Lower Compartment Purge Air Exhaust	Leakoff lines to the annulus
X-5	Instrument Room Purge Air Exhaust	Leakoff lines to the annulus
X-6	Upper Compartment Purge Air Exhaust	Leakoff lines to the annulus
X-7	Upper Compartment Purge Air Exhaust	Leakoff lines to the annulus
X-8A	Feedwater Bypass	Secondary side of the steam generator is pressurized above containment pressure
X-8B	Feedwater Bypass	Same as for Penetration X-8A
X-8C	Feedwater Bypass	Same as for Penetration X-8A
X-8D	Feedwater Bypass	Same as for Penetration X-8A
X-9A	Upper Compartment Purge Air Supply	Leakoff lines to the annulus
X-9B	Upper Compartment Purge Air Supply	Leakoff lines to the annulus
X-10A	Lower Compartment Purge Air Supply	Leakoff lines to the annulus
X-10B	Lower Compartment Purge Air Supply	Leakoff lines to the annulus
X-11	Instrument Room Purge Air Supply	Leakoff lines to the annulus
X-12A	Feedwater	Same as for Penetration X-8A
X-12B	Feedwater	Same as for Penetration X-8A
X-12C	Feedwater	Same as for Penetration X-8A
X-12-D	Feedwater	Same as for Penetration X-8A
X-13A	Main Steam Line	Same as for Penetration X-8A

TABLE 6.2.4-3 (Sheet 2 of 3)

PREVENTION OF BYPASS LEAKAGE TO THE ATMOSPHERE

Penetration Number	Penetration Line Name	Description
X-13B	Main Steam Line	Same as for Penetration X-8A
X-13C	Main Steam Line	Same as for Penetration X-8A
X-13D	Main Steam Line	Same as for Penetration X-8A
X-14A, B, C, D	Steam Generator Blowdown Lines	Same as for Penetration X-8A
X-40A	Auxiliary Feedwater	Same as for Penetration X-8A
X-40B	Auxiliary Feedwater	Same as for Penetration X-8A
X-56A	Lower Compartment ERCW Supply	Leakage prevented by combination of water seal and piping traps in conjunction with acceptance limits for Appendix J Testing as defined in design output.
X-57A	Lower Compartment ERCW Return	Same as for Penetration X-56A
X-58A	Lower Compartment ERCW Supply	Same as for Penetration X-56A
X-59A	Lower Compartment ERCW Return	Same as for Penetration X-56A
X-60A	Lower Compartment ERCW Supply	Same as for Penetration X-56A
X-61A	Lower Compartment ERCW Return	Same as for Penetration X-56A
X-62A	Lower Compartment ERCW Supply	Same as for Penetration X-56A
X-63A	Lower Compartment ERCW Return	Same as for Penetration X-56A
X-68	Upper Compartment ERCW Supply	Same as for Penetration X-56A
X-69	Upper Compartment ERCW Supply	Same as for Penetration X-56A
X-70	Upper Compartment ERCW Return	Same as for Penetration X-56A
X-71	Upper Compartment ERCW Return	Same as for Penetration X-56A
X-72	Upper Compartment ERCW Return	Same as for Penetration X-56A
X-73	Upper Compartment ERCW Return	Same as for Penetration X-56A
X-74	Upper Compartment ERCW Supply	Same as for Penetration X-56A

TABLE 6.2.4-3 (Sheet 3 of 3)

PREVENTION OF BYPASS LEAKAGE TO THE ATMOSPHERE

Penetration Number	Penetration Line Name	Description
X-75	Upper Compartment ERCW Supply	Same as for Penetration X-56A
X-80	Lower Compartment Pressure Relief	Leakoff lines to the annulus.
X-108	Maintenance Port	Blind flange, double O-ring design with a zero leakage criteria; not open during power operation
X-109	Maintenance Port	Blind flange, double O-ring design with a zero leakage criteria; not open during power operation.

TABLE 6.2.4-4 (Sheet 1 of 4)
UNIT 1

INSTRUMENT LINES PENETRATING PRIMARY CONTAINMENT

Line Identification	No.	Penetration Number	Line Size Inches	Orifice	Inner Isolation Valve Number	Valve Type	Valve Location	Outer Isolation Valve Number	Valve Type	Valve Location
PAS Containment Air INTK LC Train B	1	X-23	3/8	No	43-319	Globe	Inside Prim Containment	43-318	Globe	Annulus
Pressurizer Liquid Sample	2	X-25A	3/8	No	43-11	Globe	Inside Prim Containment	43-12	Globe	Annulus
Containment Annulus ΔP Sensor[1]	3	X-25B	1/2	No	-	-	-	-	-	-
Containment Annulus ΔP Sensor[1]	4	X-25C	1/2	No	-	-	-	-	-	-
Pressurizer Steam Sample	5	X-25D	3/8	No	43-2	Globe	Inside Prim Containment	43-3	Globe	Annulus
Containment Annulus ΔP Sensor[1]	6	X-26C	1/2	No	-	-	-	-	-	-
Steam Generator No. 1 Sample	7	X-27A	3/8	No	43-54D	Globe	Inside Prim Containment	43-55	Globe	Annulus
Steam Generator No. 2 Sample	8	X-27B	3/8	No	43-56D	Globe	Inside Prim Containment	43-58	Globe	Annulus
Steam Generator No. 3 Sample	9	X-27C	3/8	No	43-59D	Globe	Inside Prim Containment	43-61	Globe	Annulus
Steam Generator No. 4 Sample	10	X-27D	3/8	No	43-63D	Globe	Inside Prim Containment	43-64	Globe	Annulus
PAS Containment Return Train B	11	X-28	3/8	No	43-834	Check	Inside Prim Containment	43-341	Globe	Annulus

TABLE 6.2.4-4 (Sheet 2 of 4)
UNIT 1

INSTRUMENT LINES PENETRATING PRIMARY CONTAINMENT

Line Identification	No.	Penetration Number	Line Size Inches	Orifice	Inner Isolation Valve Number	Valve Type	Valve Location	Outer Isolation Valve Number	Valve Type	Valve Location
Accum. to Holdup Tank	12	X-30	3/4	No	63-071	Globe	Inside Prim Containment	63-084	Globe	Outside Shield Building
Pressurizer Relief Tank to Gas Analyzer	14	X-84A	3/8	No	68-308	Globe	Inside Prim Containment	68-307	Globe	Annulus
Reactor Vessel Level Ind Sys	15	X-84B	3/16	No	-	-	-	-	-	-
Reactor Vessel Level Ind Sys	16	X-84C	3/16	No	-	-	-	-	-	-
Reactor Vessel Level Ind Sys	17	X-84D	3/16	No	-	-	-	-	-	-
Excess Letdown Heat Exchanger to Boron Analyzer	18	X-85A	3/8	No	43-75	Globe	Inside Prim Containment	43-77	Globe	Annulus
Hot Leg Sample - Loops 1 and 3	19	X-85B	3/8	No	43-22	Globe	Inside Prim Containment	43-23	Globe	Annulus
Containment Annulus ΔP Sensor[1]	20	X-85C	1/2	No	-	-	-	-	-	-
PAS Containment Air INTK UC Train A	21	X-86A	3/8	No	43-288	Globe	Inside Prim Containment	43-287	Globe	Annulus
PAS Containment Air RTRN Train A	22	X-86B	3/8	No	43-883	Check	Inside Prim Containment	43-307	Globe	Annulus

WBN

TABLE 6.2.4-4 (Sheet 3 of 4)
UNIT 1

INSTRUMENT LINES PENETRATING PRIMARY CONTAINMENT

Line Identification	No.	Penetration Number	Line Size Inches	Orifice	Inner Isolation Valve Number	Valve Type	Valve Location	Outer Isolation Valve Number	Valve Type	Valve Location
PAS Containment Sump	23	X-86C	3/8	No	43-841	Check	Inside Prim Containment	43-342	Globe	Annulus
RTRN Train A Reactor Vessel Level Ind	25	X-87B	3/16	No	-	-	-	-	-	-
Reactor Vessel Level Ind Sys	25	X-87B	3/16	No	_	_	_	_	_	_
Reactor Vessel Level Ind Sys	26	X-87C	3/16	No	-	-	-	-	-	-
Reactor Vessel Level Ind Sys	27	X-87D	3/16	No	-	-	-	-	-	-
Hydrogen Analyzer Train B	30	X-92A	3/8	No	43-207	Globe	Inside Prim Containment	43-435	Globe	Annulus
Hydrogen Analyzer Train B	31	X-92B	3/8	No	43-208	Globe	Inside Prim Containment	43-436	Globe	Annulus
PAS Hot Leg 1 - Train A	32	X-92C	3/8	No	43-251	Globe	Inside Prim Containment	43-250	Globe	Annulus
Accumulator Sample	33	X-93	3/8	No	43-34	Globe	Inside Prim Containment	43-35	Globe	Annulus
Upper Compartment Air Monitor	35	X-95C	1-1/2	No	90-114 90-115	Globe	Inside Prim Containment	90-113	Globe	Annulus
Upper Compartment Air Monitor	36	X-95B	1-1/2	No	90-116	Globe	Inside Prim Containment	90-117	Globe	Annulus

TABLE 6.2.4-4 (Sheet 4 of 4)
UNIT 1

INSTRUMENT LINES PENETRATING PRIMARY CONTAINMENT

Line Identification	No.	Penetration Number	Line Size Inches	Orifice	Inner Isolation Valve Number	Valve Type	Valve Location	Outer Isolation Valve Number	Valve Type	Valve Location
Lower Compartment Air Monitor	38	X-94C	1-1/2	No	90-108 90-109	Globe	Inside Prim Containment	90-107	Globe	Annulus
Lower Compartment Air Monitor	39	X-94B	1-1/2	No	90-110	Globe	Inside Prim Containment	90-111	Globe	Annulus
Containment Annulus ΔP Sensor[1]	40	X-96C	1/2	No	-	-	-	-	-	-
Containment Annulus ΔP Sensor	41	X-97	1/2	No	30-134	Globe	Inside Prim Containment	30-135	Globe	Annulus
Hydrogen Analyzer - Train A	42	X-99	3/8	No	43-202	Globe	Inside Prim Containment	43-434	Globe	Annulus
Hydrogen Analyzer - Train A	43	X-100	3/8	No	43-201	Globe	Inside Prim Containment	43-433	Globe	Annulus
PAS Containment Air RTRN Train B	45	X-105	3/8	No	43-884	Check	Inside Prim Containment	43-325	Globe	Annulus
PAS Hot Leg 3 - Train B	46	X-106	3/8	No	43-310	Globe	Inside Prim Containment	43-309	Globe	Annulus

[1]These have no in-line containment isolation valves - see Section 6.2.4.3 and Table 6.2.4-1.

TABLE 6.2.4-4 (Sheet 1 of 3)
UNIT 2

INSTRUMENT LINES PENETRATING PRIMARY CONTAINMENT

Line Identification	No.	Penetration Number	Line Size Inches	Orifice	Inner Isolation Valve Number	Valve Type	Valve Location	Outer Isolation Valve Number	Valve Type	Valve Location
Pressurizer Liquid Sample	1	X-25A	3/8	No	43-11	Globe	Inside Prim Containment	43-12	Globe	Annulus
Pressurizer Steam Sample	2	X-25D	3/8	No	43-2	Globe	Inside Prim Containment	43-3	Globe	Annulus
ΔP Sensor[1]	3	X-26C	1/2	No	-	-	-	-	-	-
Steam Generator No. 1 Sample and Water Quality	4	X-27A	3/8	No	43-54D	Globe	Inside Prim Containment	43-55	Globe	Annulus
Steam Generator No. 2 Sample	5	X-27B	3/8	No	43-56D	Globe	Inside Prim Containment	43-58	Globe	Annulus
Steam Generator No. 3 Sample	6	X-27C	3/8	No	43-59D	Globe	Inside Prim Containment	43-61	Globe	Annulus
Steam Generator No. 4 Sample	7	X-27D	3/8	No	43-63D	Globe	Inside Prim Containment	43-64	Globe	Annulus
ΔP Sensor[1]		X-57B	1/2	No	-	-	-	-	-	-
ΔP Sensor[1]		X-60B	1/2	No	-	-	-	-	-	-
ΔP Sensor[1]		X-102	1/2	No	-	-	-	-	-	-
RCS Pressure Sensor[2]		X-58B	1/4	No						
Accum. to Holdup Tank	8	X-30	3/4	No	63-071	Globe	Inside Prim Containment	63-084	Globe	Outside Shield Building

TABLE 6.2.4-4 (Sheet 2 of 3)
UNIT 2

INSTRUMENT LINES PENETRATING PRIMARY CONTAINMENT

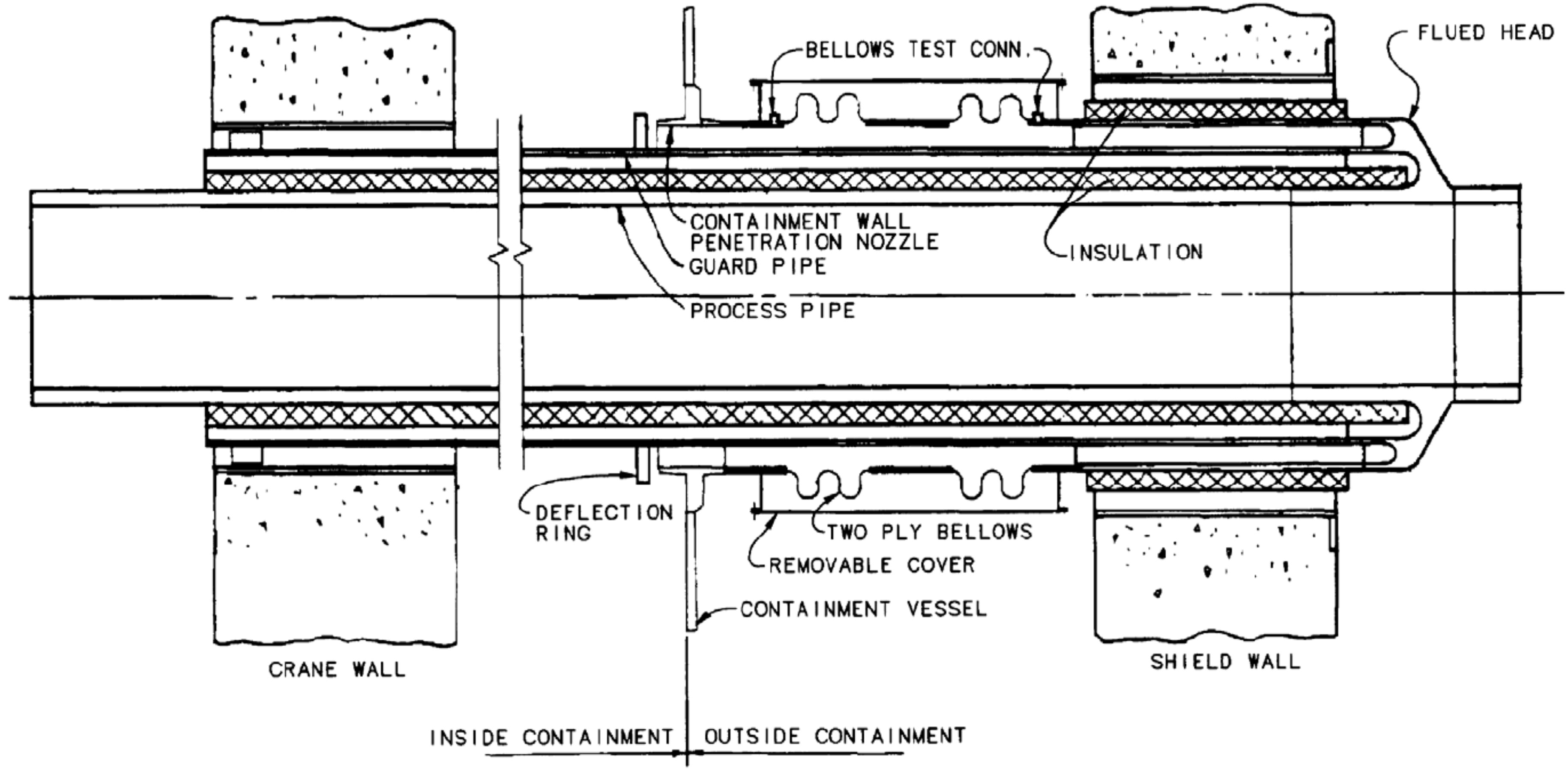
Line Identification	No.	Penetration Number	Line Size Inches	Orifice	Inner Isolation Valve Number	Valve Type	Valve Location	Outer Isolation Valve Number	Valve Type	Valve Location
Pressurizer Relief Tank to Gas Analyzer	9	X-84A	3/8	No	68-308	Globe	Inside Prim Containment	68-307	Globe	Annulus
Reactor Vessel Level Ind Sys[2]	10	X-84B	1/4	No	-	-	-	-	-	-
Reactor Vessel Level Ind Sys[2]	11	X-84C	1/4	No	-	-	-	-	-	-
Reactor Vessel Level Ind Sys[2]	12	X-84D	1/4	No	-	-	-	-	-	-
Hot Leg Sample - Loops 1 and 3	13	X-85B	3/8	No	43-22	Globe	Inside Prim Containment	43-23	Globe	-
Reactor Vessel Level Ind Sys[2]	16	X-87B	1/4	No	-	-	-	-	-	-
Reactor Vessel Level Ind Sys[2]	17	X-87C	1/4	No	-	-	-	-	-	-
Reactor Vessel Level Ind Sys[2]	18	X-87D	1/4	No	-	-	-	-	-	-
Accumulator Sample	19	X-93	3/8	No	43-34	Globe	Inside Prim Containment	43-35	Globe	Annulus
Lower Compartment Air Monitor Intake	20	X-94B	1-1/2	No	90-110	Globe	Inside Prim Containment	90-111	Globe	Annulus
Lower Compartment Air Monitor Return	21	X-94C	1-1/2	No	90-108 90-109	Globe	Inside Prim Containment	90-107	Globe	Annulus

TABLE 6.2.4-4 (Sheet 3 of 3)
UNIT 2

INSTRUMENT LINES PENETRATING PRIMARY CONTAINMENT

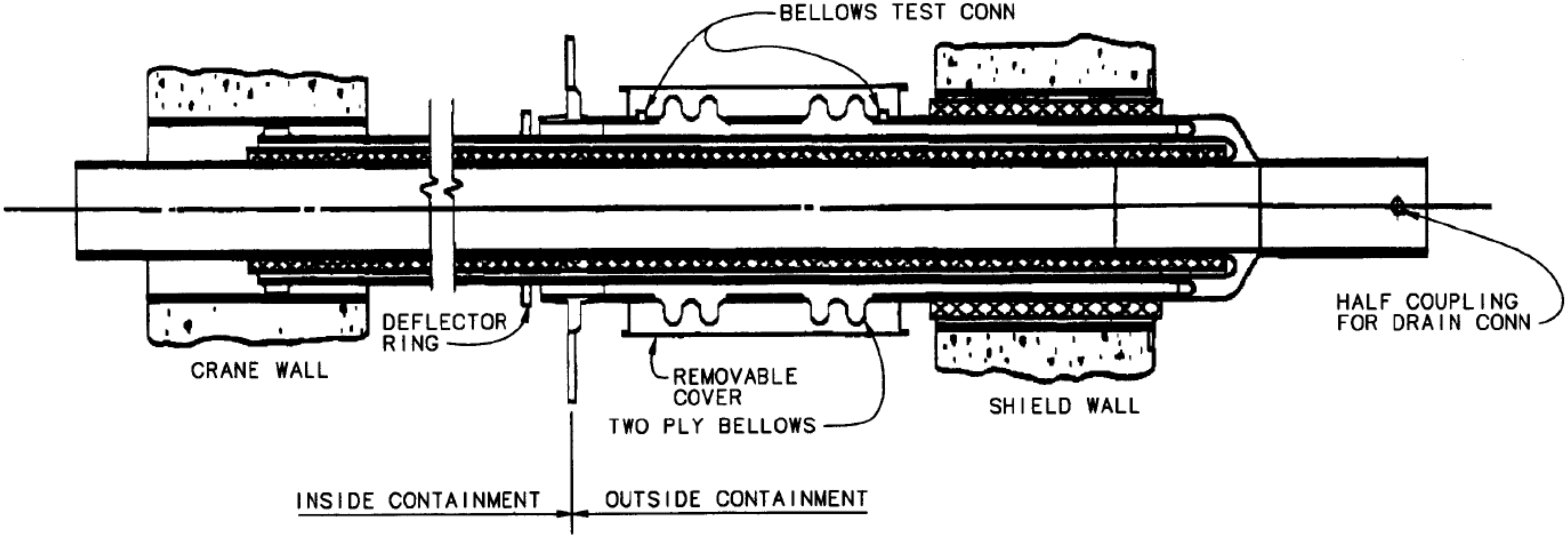
Line Identification	No.	Penetration Number	Line Size Inches	Orifice	Inner Isolation Valve Number	Valve Type	Valve Location	Outer Isolation Valve Number	Valve Type	Valve Location
Upper Compartment Air Monitor Intake	22	X-95B	1-1/2	No	90-116	Globe	Inside Prim Containment	90-117	Globe	Annulus
Upper Compartment Air Monitor Return	23	X-95C	1-1/2	No	90-114 90-115	Globe	Inside Prim Containment	90-113	Globe	Annulus
Containment ΔP Sensor[1] (PΔT-30-133)	24	X-97	1/2	No	-	-	-	-	-	-
Containment ΔP Sensor[1] (PΔT-30-30C)	25	X-98	1/2	No	-	-	-	-	-	-
Hydrogen Analyzer	26	X-99	3/8	No	43-202	Globe	Inside Prim Containment	43-434	Globe	Annulus
Hydrogen Analyzer	27	X-100	3/8	No	43-201	Globe	Inside Prim Containment	43-433	Globe	Annulus

[1] These have no in-line containment isolation valves - see Section 6.2.4.3 and Table 6.2.4-1.
[2] Capillary lines are 1/4" through the containment penetration. They transit to 3/16" on both sides of the containment penetration.



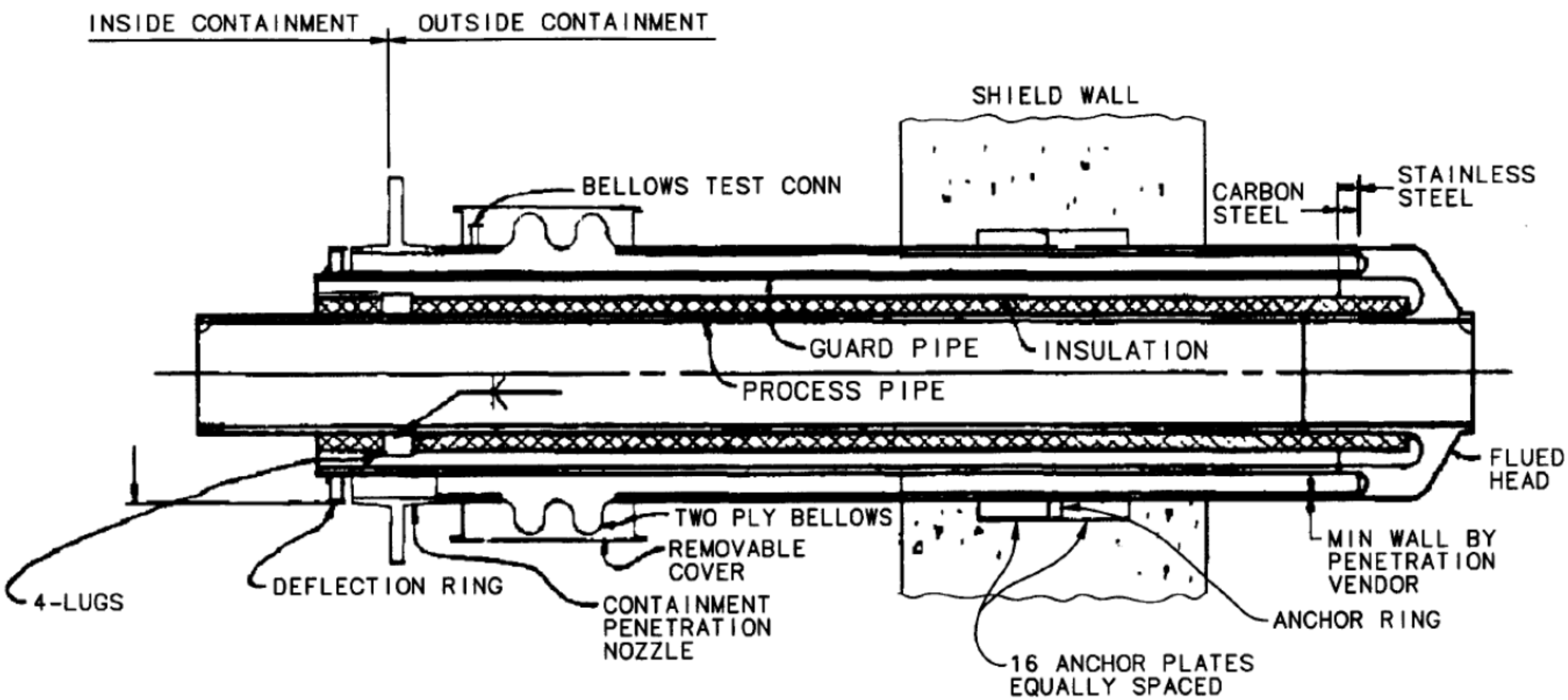
WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

Type I
Main Steam
X-13A, X-13B, X-13C, X-13D
FIGURE 6.2.4-1



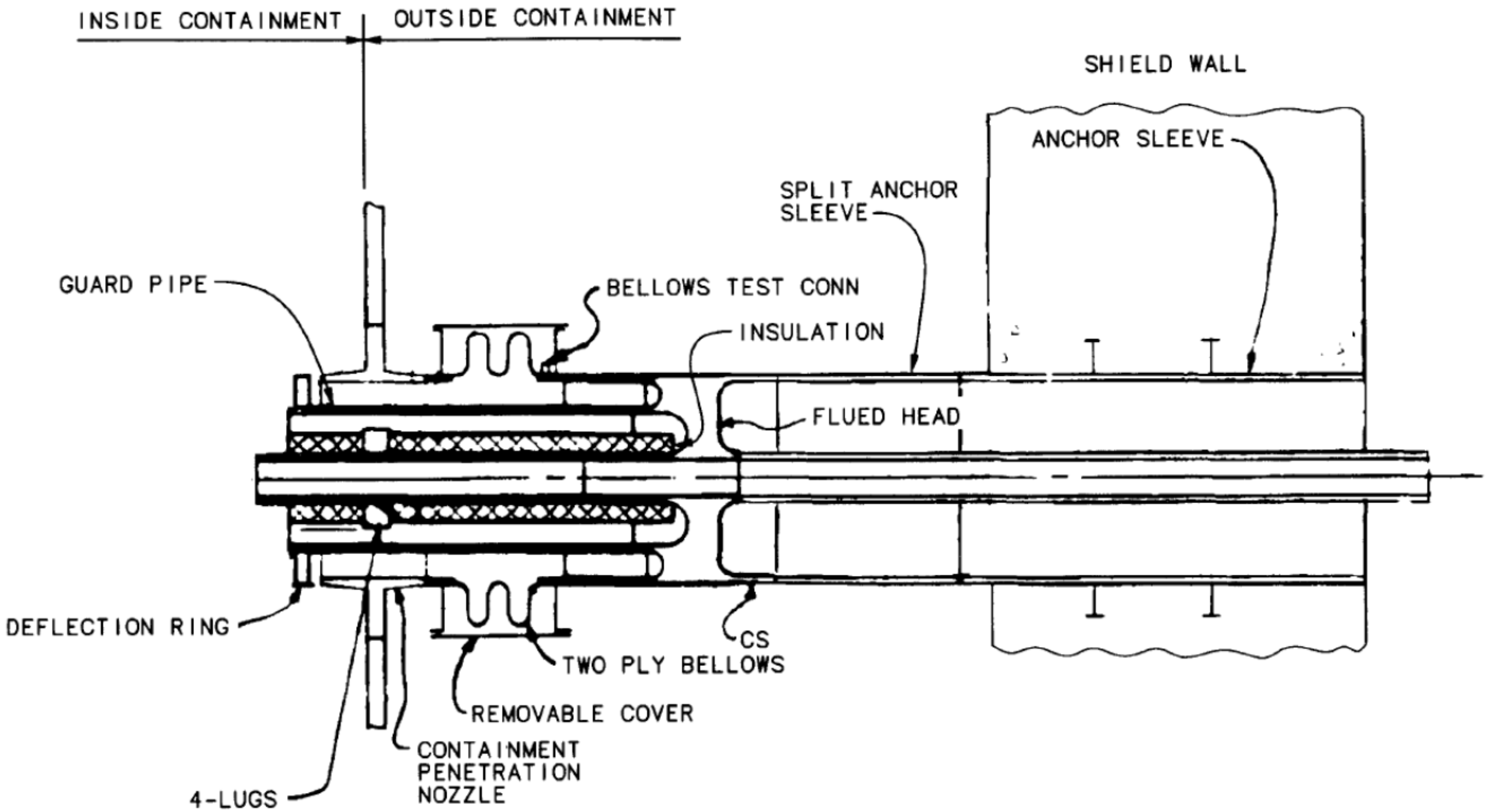
WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

Type II
Feedwater
X-12A, X-12B, X-12C, X-12D
FIGURE 6.2.4-2



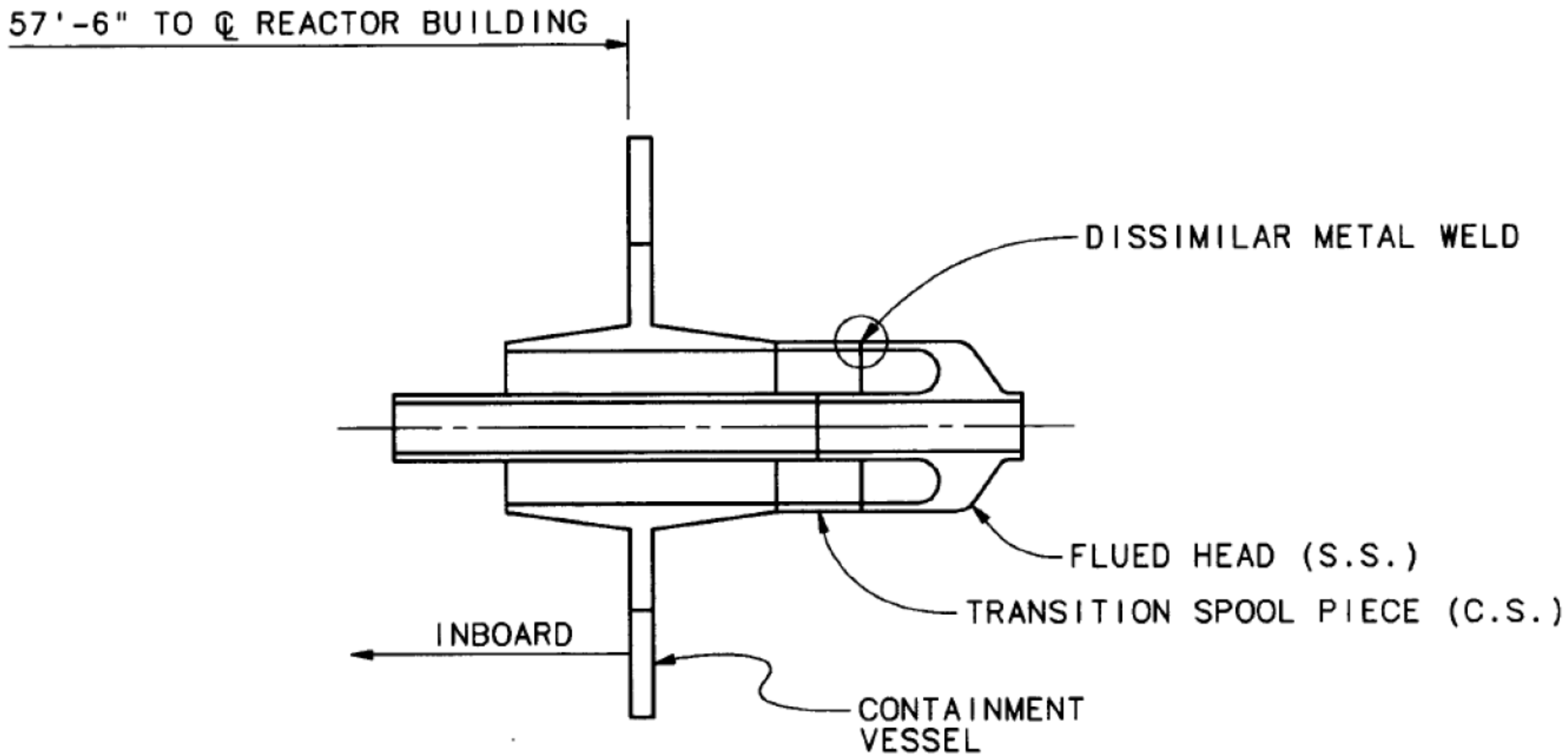
WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

Type III
Residual Heat Removal
Pump Return X-17
Pump Supply X-107
FIGURE 6.2.4-3



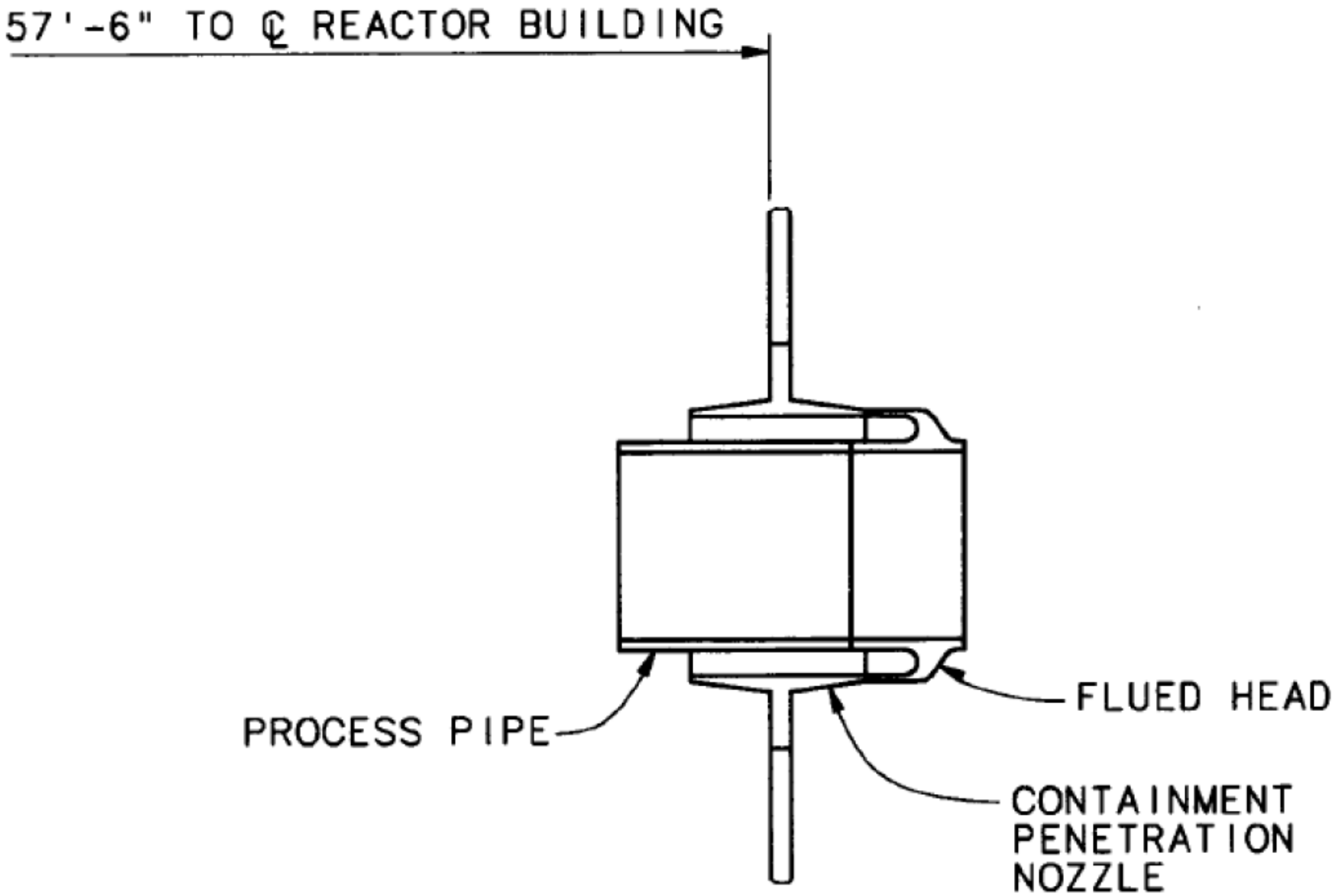
WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

Type IV & V
Type IV Socket Weld Ends
Type V Butt Weld Ends
FIGURE 6.2.4-4



WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

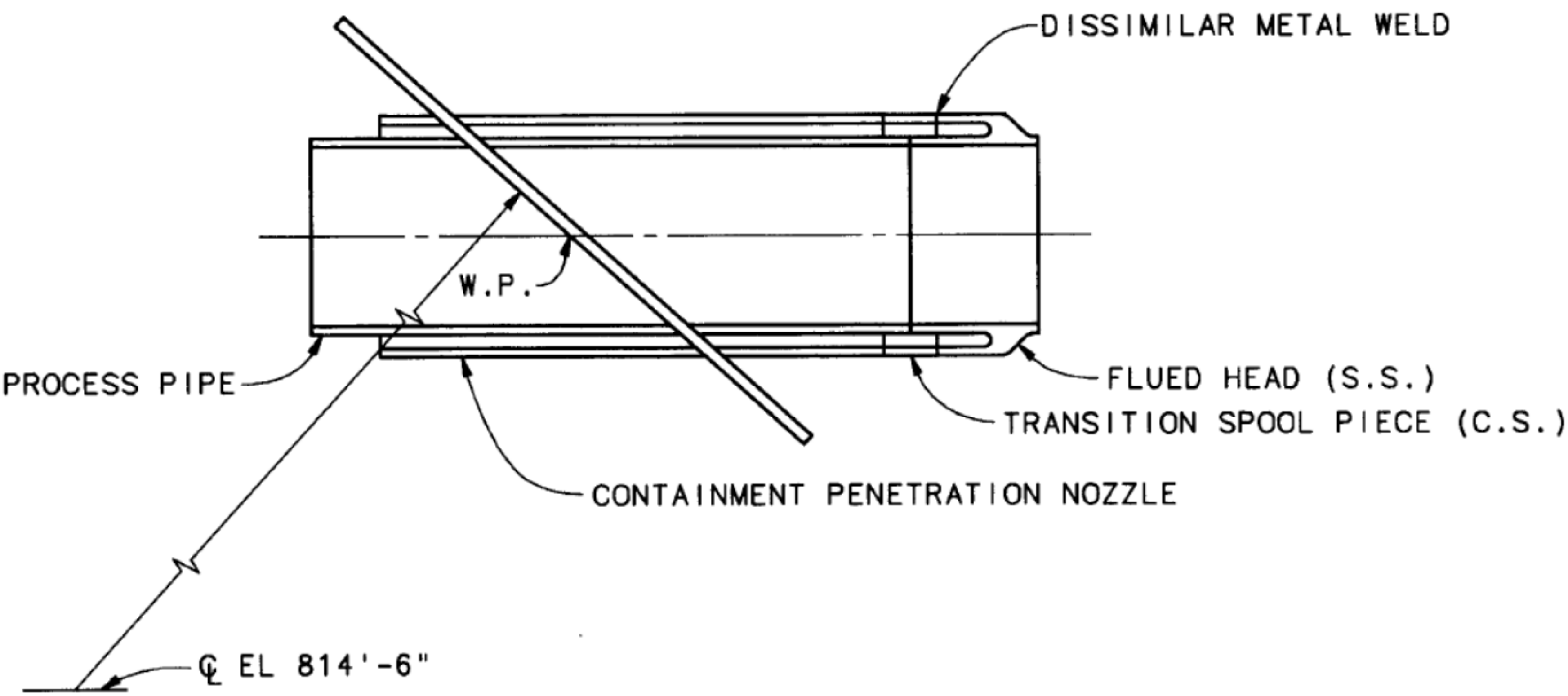
Type VI and VII
Type VI for Socket Weld
SS Process Lines,
Type VII for butt Weld
SS Process Lines
FIGURE 6.2.4-5



WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

Type VIII, for Butt Weld
C.S. Process Lines

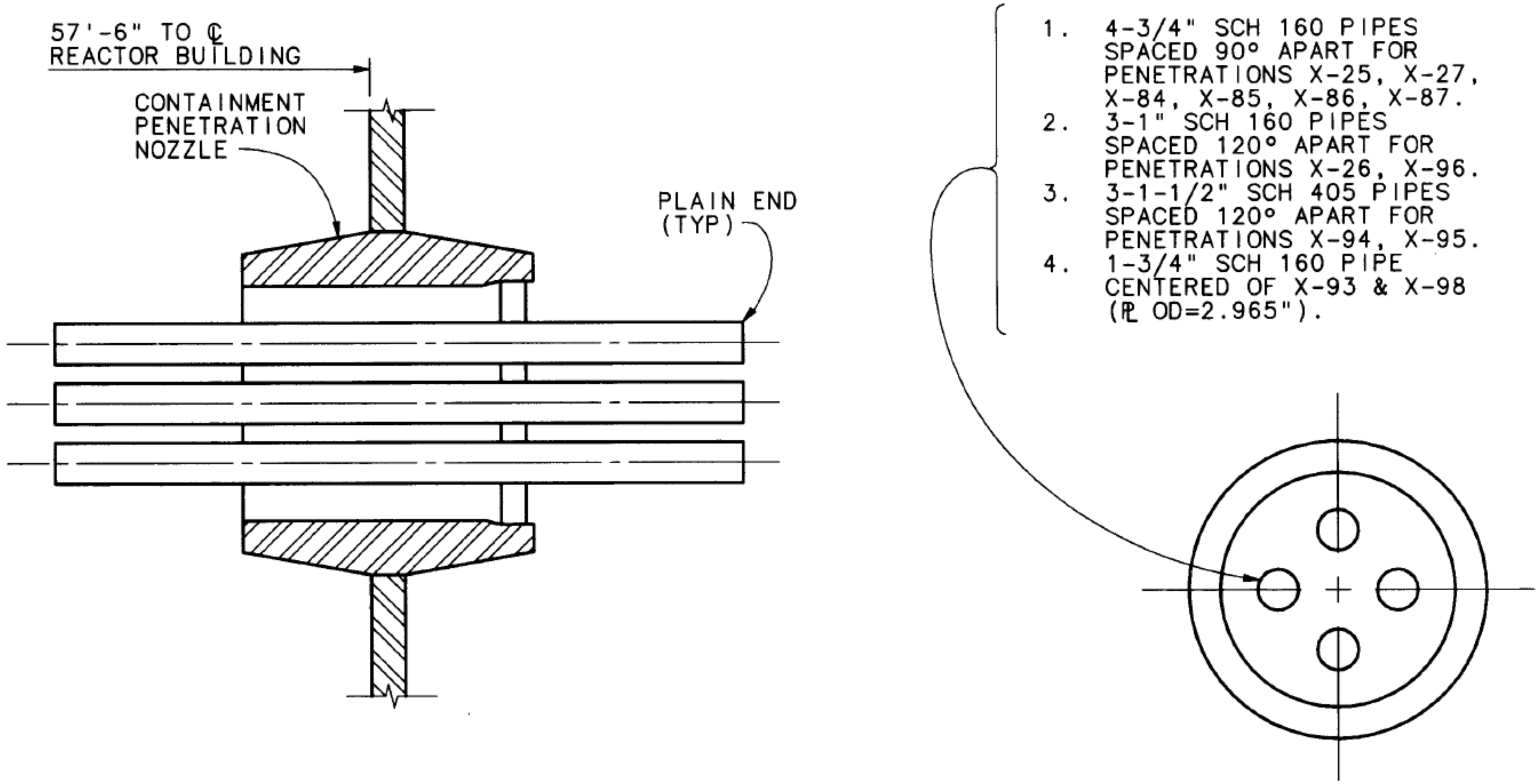
FIGURE 6.2.4-6



WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

Type IX
For SS Process Lines

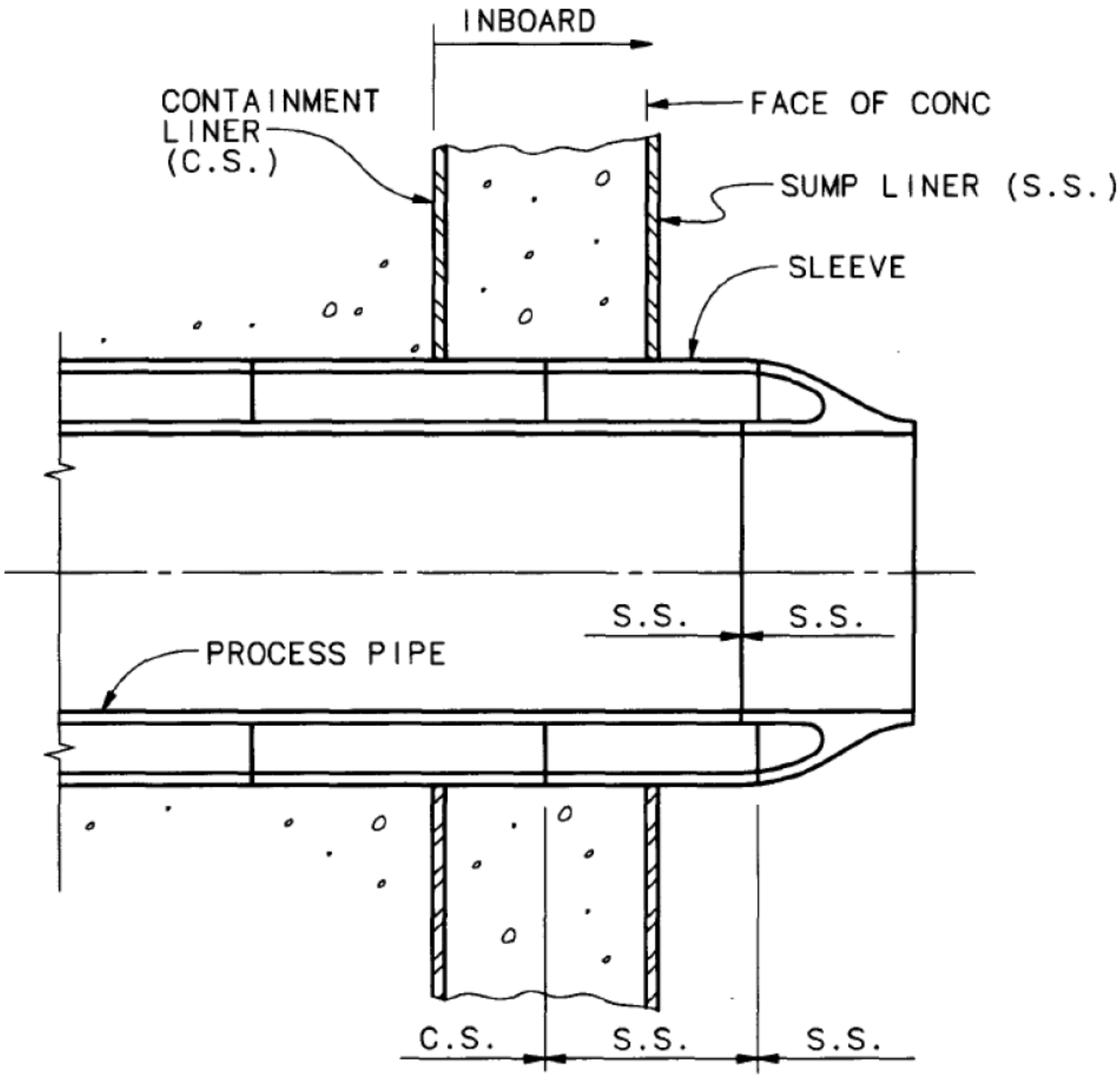
FIGURE 6.2.4-7



WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

Type X
Instrument Penetrations

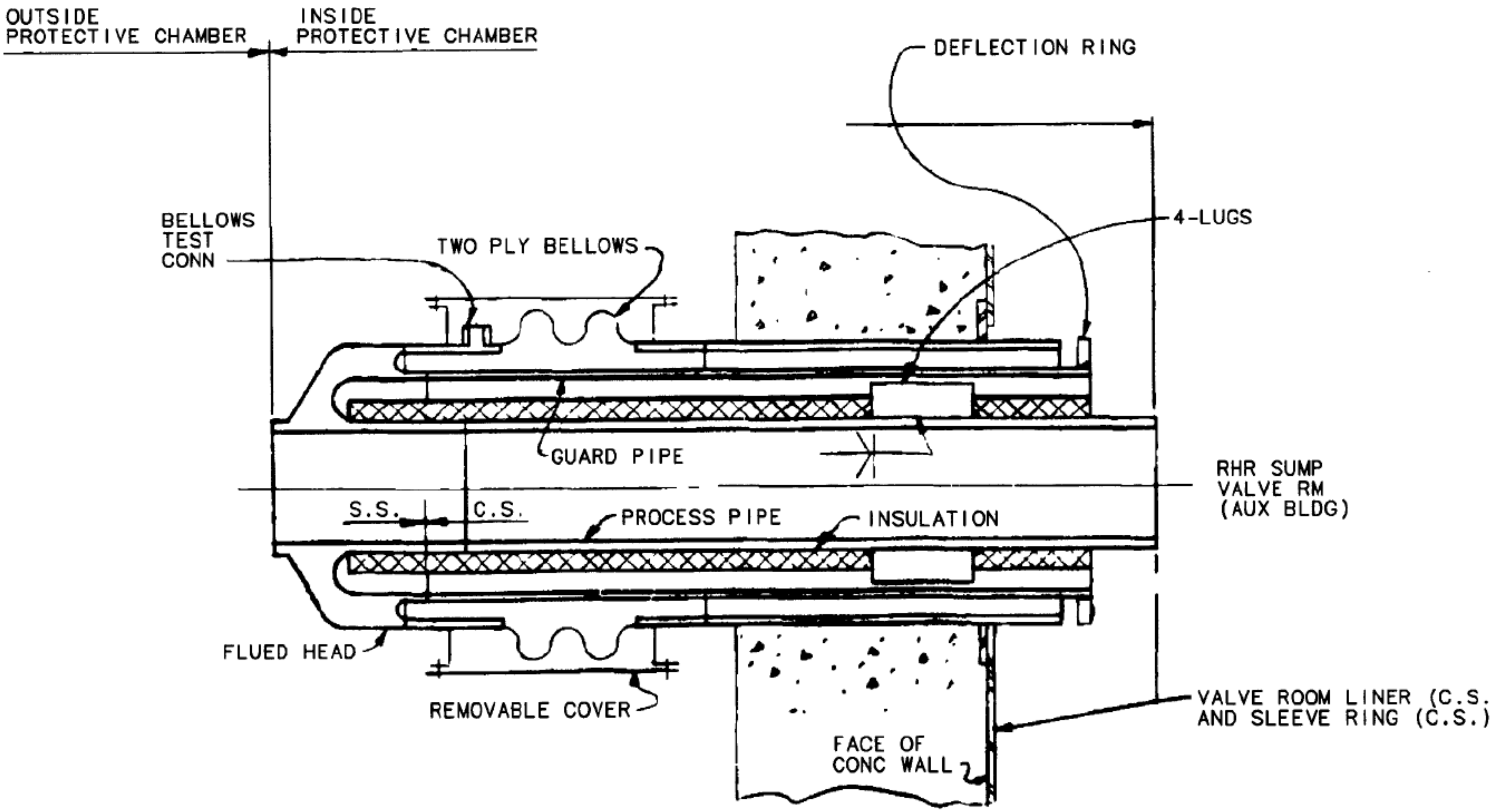
FIGURE 6.2.4-8



WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

Type XII
Emergency Sump

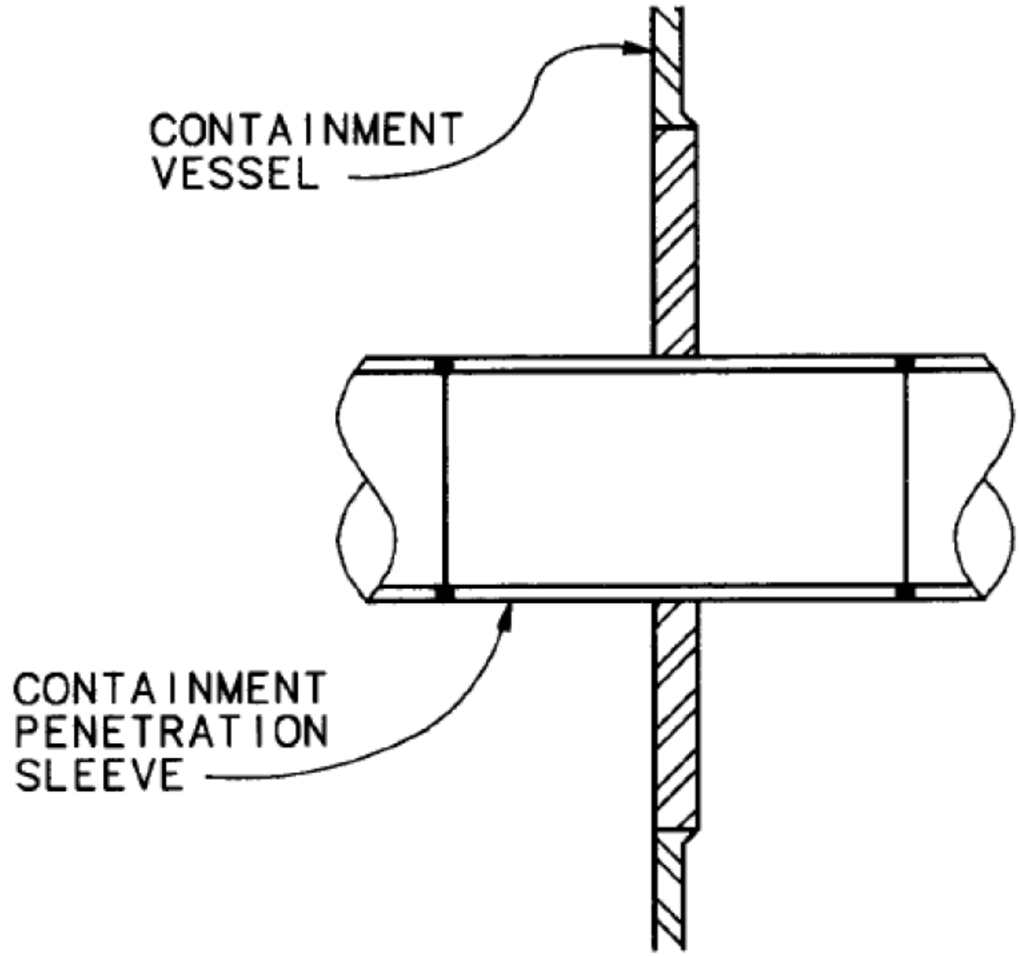
FIGURE 6.2.4-9



WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

Type XI
Emergency Sump

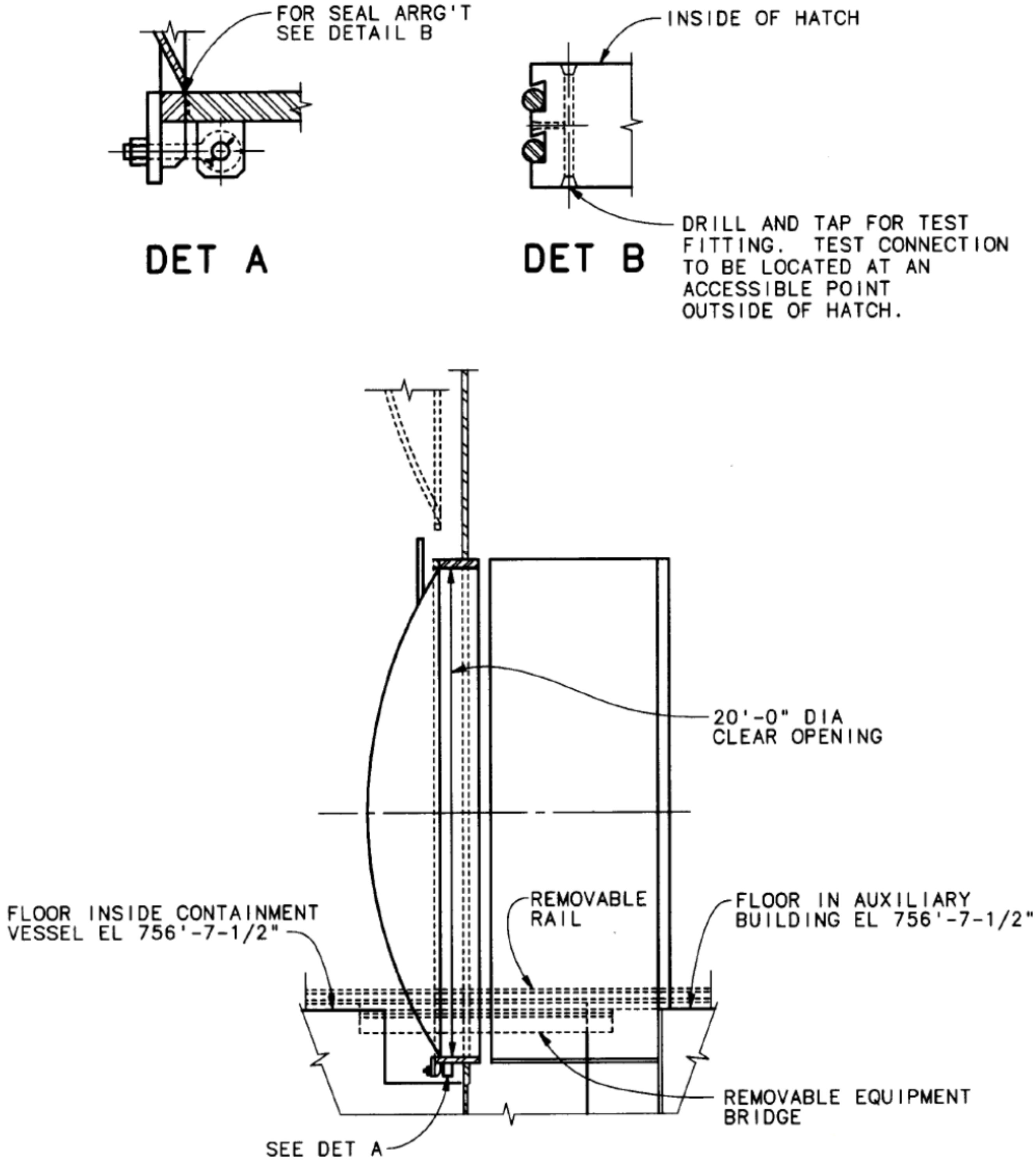
FIGURE 6.2.4-10



WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

Type XIII
Ventilation Duct Penetration

FIGURE 6.2.4-11

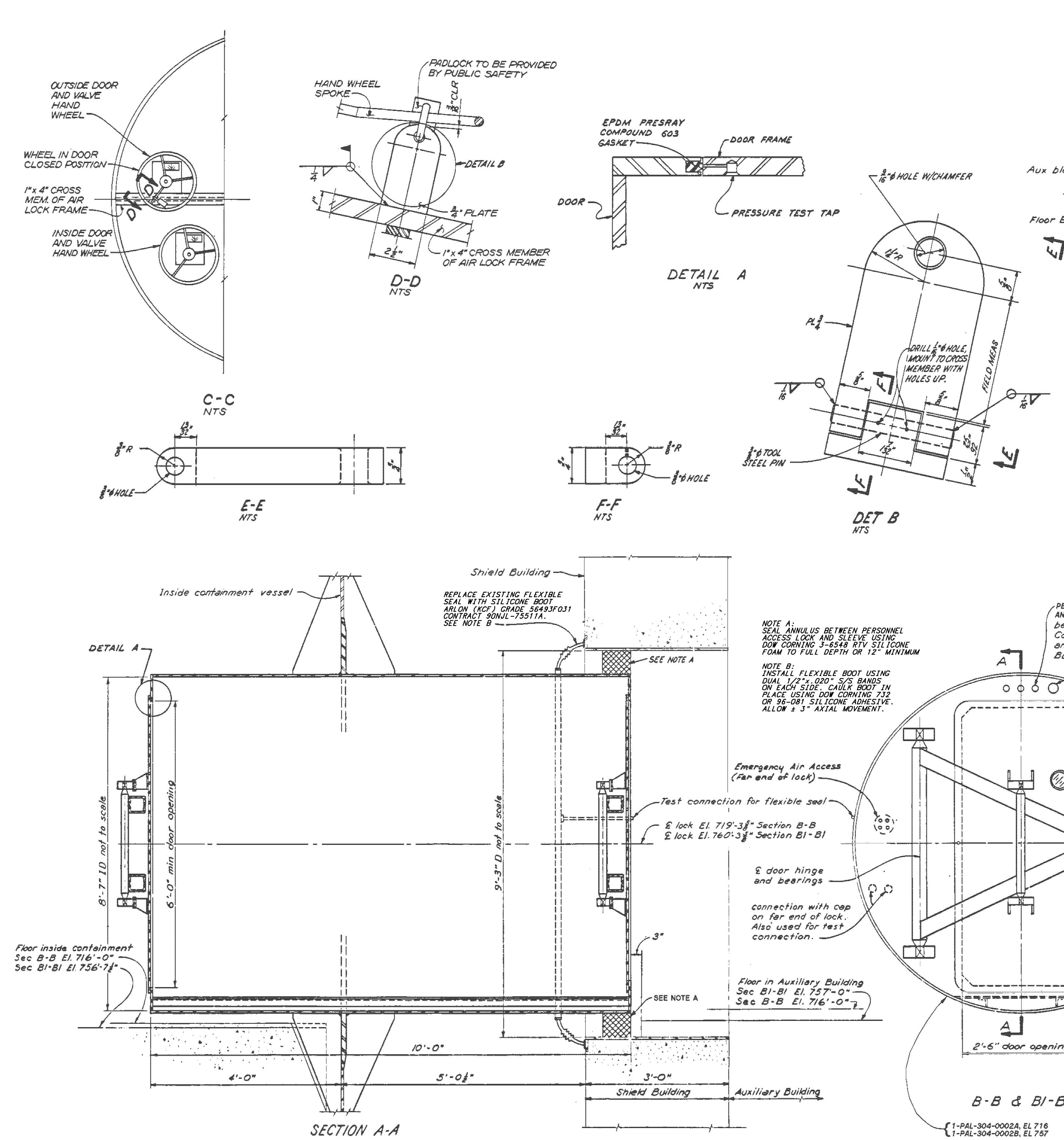


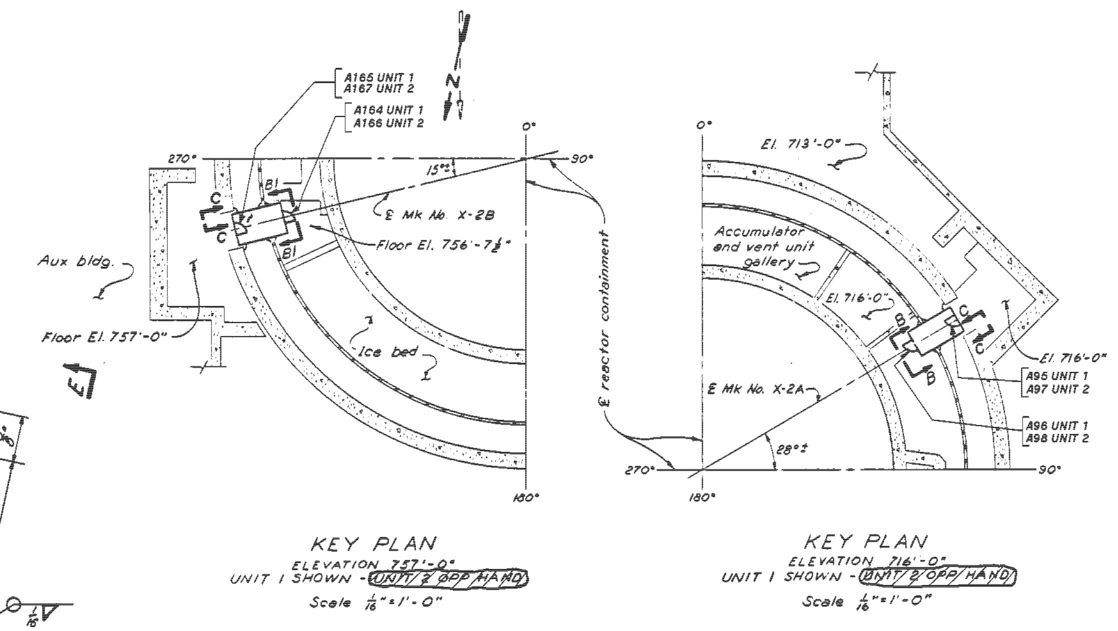
WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

Type XIV
Equipment Hatch

Figure 6.2.4-12

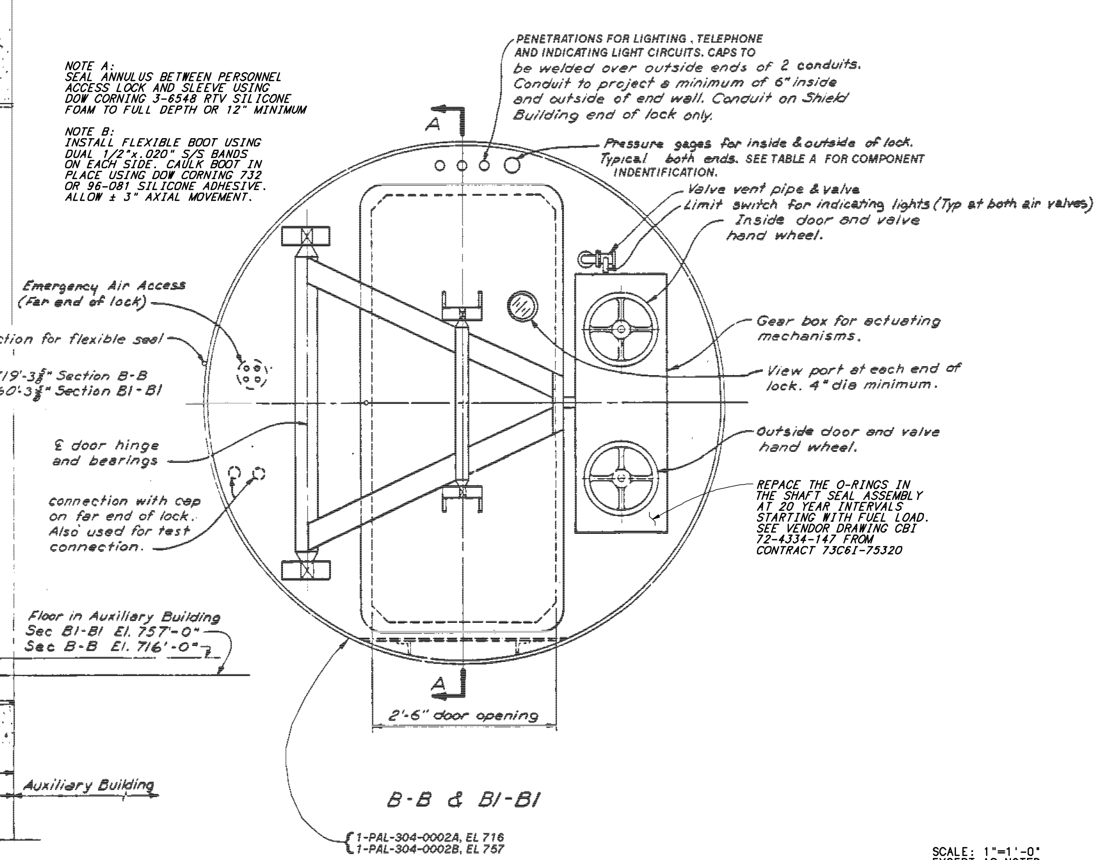
CAD MAINTAINED DRAWING





COMPONENT ID	DESCRIPTION	LOCATION
1-PI-304-0002A/1	PRESS IND OUTSIDE PAL AB END EL 716	EL 716, AZ 62
1-PI-304-0002A/2	PRESS IND INSIDE PAL AB END EL 716	EL 716, AZ 62
1-PI-304-0002A/3	PRESS IND INSIDE PAL RB END EL 716	EL 716, AZ 62
1-PI-304-0002A/4	PRESS IND OUTSIDE PAL RB END EL 716	EL 716, AZ 62
1-PI-304-0002B/1	PRESS IND OUTSIDE PAL AB END EL 757	EL 757, AZ 255
1-PI-304-0002B/2	PRESS IND INSIDE PAL AB END EL 757	EL 757, AZ 255
1-PI-304-0002B/3	PRESS IND INSIDE PAL RB END EL 757	EL 757, AZ 255
1-PI-304-0002B/4	PRESS IND OUTSIDE PAL RB END EL 757	EL 757, AZ 255

TABLE A



NOTES:
1. TWO LOCKS REQUIRED PER UNIT. FOUR LOCKS TOTAL REQUIRED.
2. LOCK SIZE - 6'-7" ID X ABOUT 10'-0" LONG.
3. DOOR OPENING SIZE - 2'-6" X 6'-0".
4. LOCKS DESIGNED FOR LOADS, PRESSURES AND TEMPERATURES, LOAD COMBINATIONS, AND STRESSES OUTLINED IN SPECIFICATION OR SHOWN ON THIS DRAWING.
5. LOCKS SEALED AT BOTH DOORS AND AT SHIELD BUILDING AS SHOWN.
6. DESIGNED FOR INTERNAL PRESSURE OF 13.5 PSIG AND TEST PRESSURE OF 16.9 PSIG.
7. EACH DOOR SHALL BE DESIGNED TO WITHSTAND DESIGN AND TEST PRESSURE INSIDE CONTAINMENT VESSEL WITH OTHER DOOR OPEN.
8. EACH DOOR SHALL BE CAPABLE (WITH SECURING DEVICES ON INSIDE DOOR IF REQUIRED) TO WITHSTAND DESIGN AND TEST PRESSURE INSIDE ACCESS LOCK WITHOUT LEAKING.
9. EACH DOOR SHALL REMAIN LEAKTIGHT WITH THE OTHER DOOR OPEN AND A NEGATIVE PRESSURE OF 0.5 PSIG INSIDE CONTAINMENT VESSEL.
10. LIVE LOAD OF 100 PSF OR 1000 LB MOVING CONCENTRATED LOAD AT ANY POINT.
11. LOCK OPERATION COMPLETELY MECHANICAL.
12. BOTH DOORS AND BOTH PRESSURE EQUALIZING VALVES ON EACH LOCK SHALL BE OPERABLE FROM OUTSIDE OF BOTH ENDS OF LOCK AND INSIDE LOCK.
13. OPERATION OF DOORS TO REQUIRE A MAXIMUM FORCE OF 40 LB.
14. DOORS AND VALVES INTERLOCKED SUCH THAT A DOOR AND/OR VALVE WILL NOT OPEN UNLESS THE DOOR AND VALVE AT THE OPPOSITE END OF THE LOCK ARE CLOSED. INTERLOCK TO HAVE MANUAL OVERRIDE BY SPECIAL TOOLS AND PROCEDURES.
15. DOOR HINGES TO ALLOW ADJUSTMENT IN ALL DIRECTIONS WITH EQUAL PRESSURE ON ALL SEALS.
16. INDICATORS PROVIDED INSIDE LOCK AND OUTSIDE OF BOTH ENDS OF LOCK TO SHOW THE POSITION OF BOTH DOORS AND BOTH VALVES.
17. LIMIT SWITCHES TO BE PROVIDED FOR REMOTE INDICATION ON THE UNIT CONTROL BOARD AS TO THE POSITION OF EACH DOOR AND EQUALIZING VALVE.
18. LOCKS DESIGNED TO ALLOW FOR 1-1/2" MINIMUM VERTICAL AND 1" MINIMUM HORIZONTAL MOVEMENT OF LOCKS WITH RESPECT TO SHIELD BUILDING DUE TO PRESSURE AND THERMAL EXPANSION.

B. ALL MATERIALS AND WORK BY TVA FIELD SHALL BE IN ACCORDANCE WITH GENERAL CONSTRUCTION SPECIFICATION N3G-881, QUALITY LEVEL II.

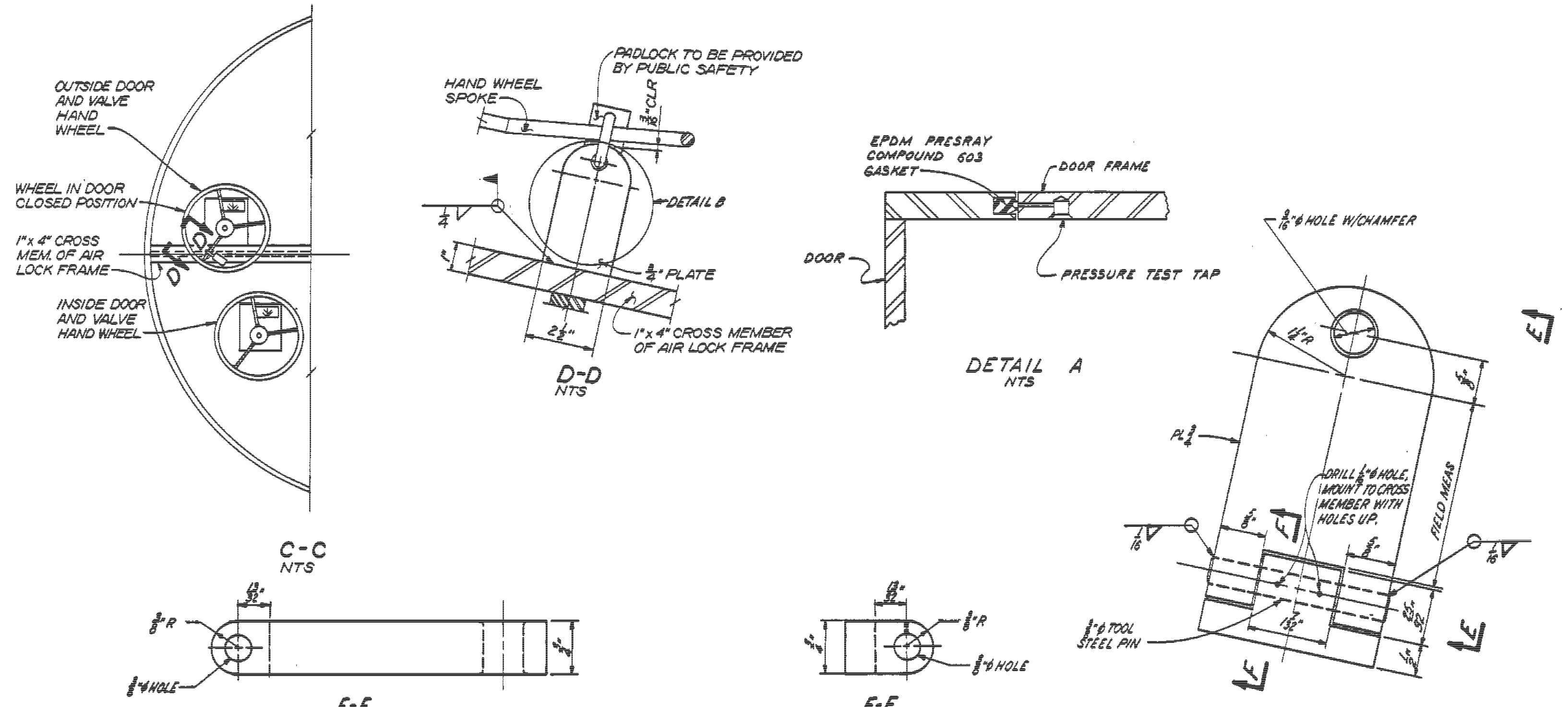
SCALE: 1"=1'-0"
EXCEPT AS NOTED

WATTS BAR
FINAL SAFETY
ANALYSIS REPORT

REACTOR BUILDING
UNIT 1
PERSONNEL ACCESS LOCK
ARRGT & DETAILS

TVA DWG NO. 44N240 RJ
FIGURE 6.2.4-13

CAD MAINTAINED DRAWING



C-C
NTS

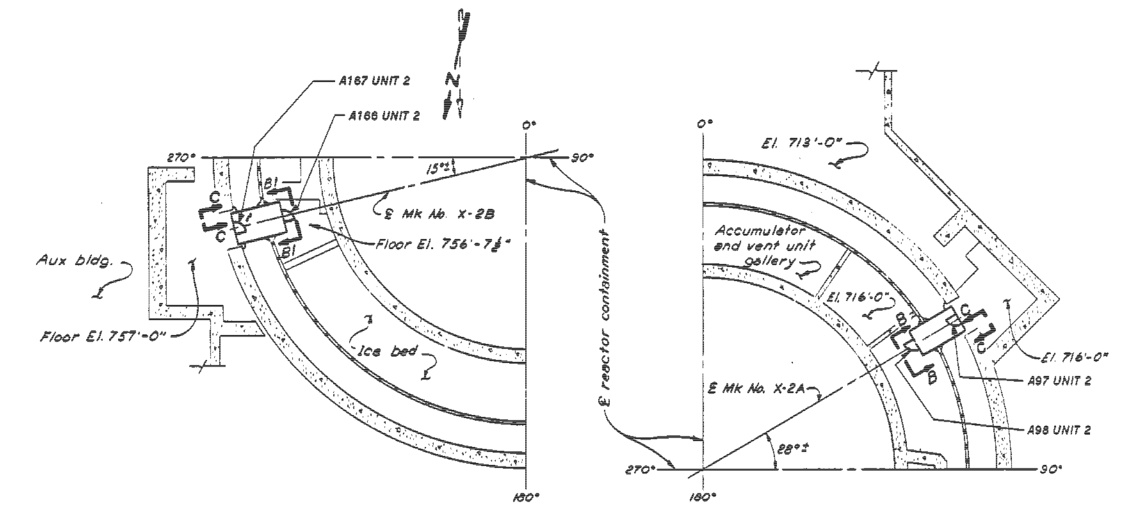
D-D
NTS

DETAIL A
NTS

E-E
NTS

F-F
NTS

DET B
NTS

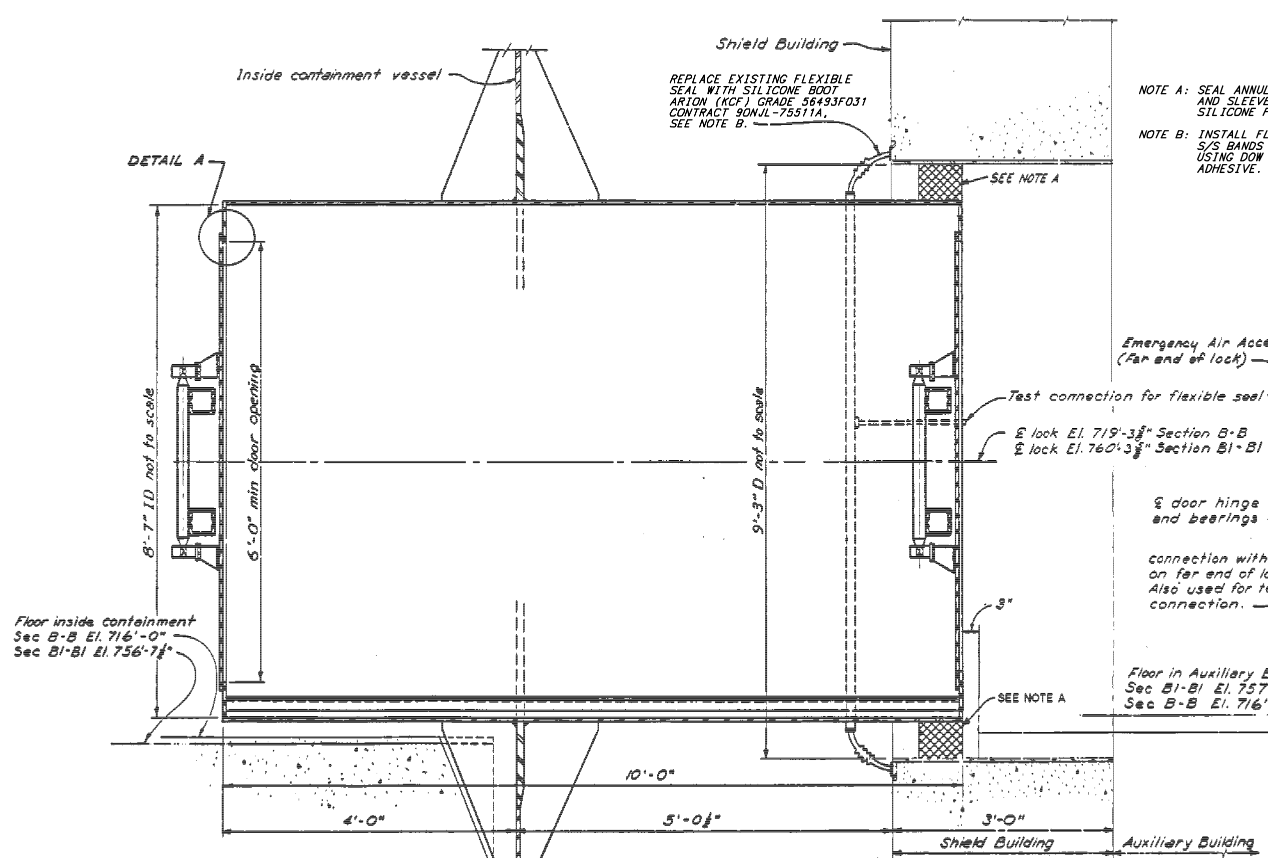


KEY PLAN
ELEVATION 757'-0"
UNIT 2 OPP HAND
Scale 1/16"=1'-0"

KEY PLAN
ELEVATION 716'-0"
UNIT 2 OPP HAND
Scale 1/16"=1'-0"

NOTES:

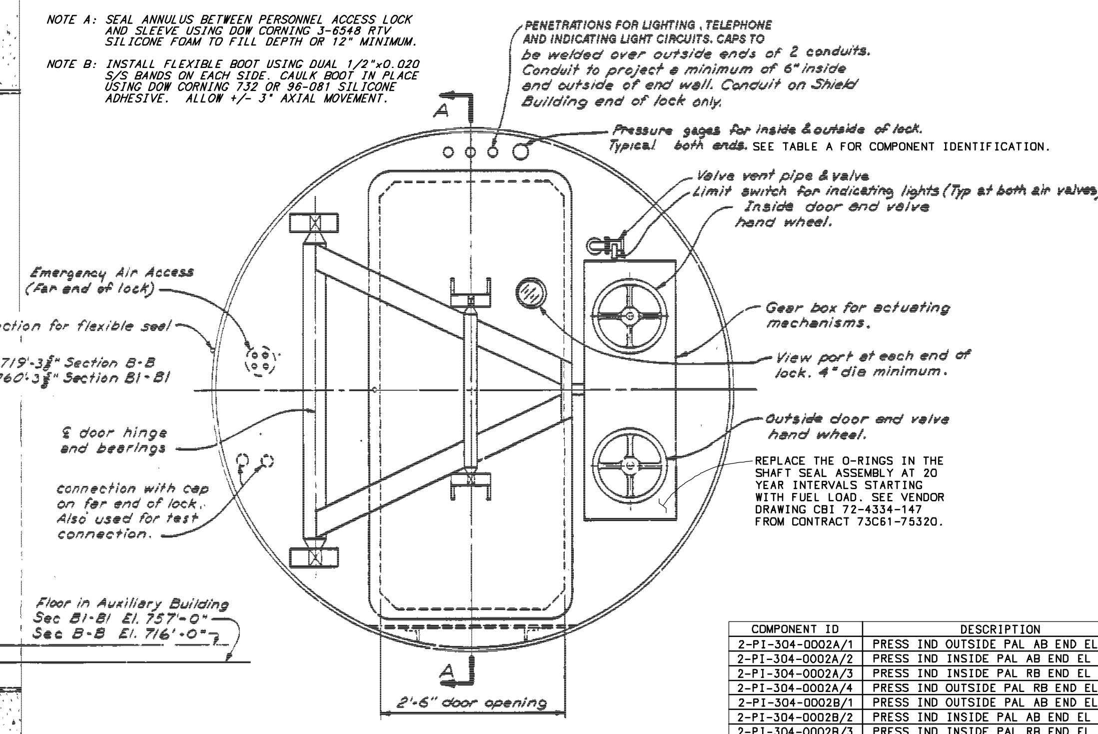
1. TWO LOCKS REQUIRED PER UNIT. FOUR LOCKS TOTAL REQUIRED.
2. LOCK SIZE - 8'-7" ID X ABOUT 10'-0" LONG.
3. DOOR OPENING SIZE - 2'-6" X 6'-0".
4. LOCKS DESIGNED FOR LOADS, PRESSURES AND TEMPERATURES, LOAD COMBINATIONS, AND STRESSES OUTLINED IN SPECIFICATION OR SHOWN ON THIS DRAWING.
5. LOCKS SEALED AT BOTH DOORS AND AT SHIELD BUILDING AS SHOWN.
6. DESIGNED FOR INTERNAL PRESSURE OF 13.5 PSIG AND TEST PRESSURE OF 16.9 PSIG.
7. EACH DOOR SHALL BE DESIGNED TO WITHSTAND DESIGN AND TEST PRESSURE INSIDE CONTAINMENT VESSEL WITH OTHER DOOR OPEN.
8. EACH DOOR SHALL BE CAPABLE (WITH SECURING DEVICES ON INSIDE DOOR IF REQUIRED) TO WITHSTAND DESIGN AND TEST PRESSURE INSIDE ACCESS LOCK WITHOUT LEAKING.
9. EACH DOOR SHALL REMAIN LEAKTIGHT WITH THE OTHER DOOR OPEN AND A NEGATIVE PRESSURE OF 0.5 PSIG INSIDE CONTAINMENT VESSEL.
10. LIVE LOAD OF 100 PSF OR 1000 LB MOVING CONCENTRATED LOAD AT ANY POINT.
11. LOCK OPERATION COMPLETELY MECHANICAL.
12. BOTH DOORS AND BOTH PRESSURE EQUALIZING VALVES ON EACH LOCK SHALL BE OPERABLE FROM OUTSIDE OF BOTH ENDS OF LOCK AND INSIDE LOCK.
13. OPERATION OF DOORS TO REQUIRE A MAXIMUM FORCE OF 40 LB.
14. DOORS AND VALVES INTERLOCKED SUCH THAT A DOOR AND/OR VALVE WILL NOT OPEN UNLESS THE DOOR AND VALVE AT THE OPPOSITE END OF THE LOCK ARE CLOSED. INTERLOCK TO HAVE MANUAL OVERRIDE BY SPECIAL TOOLS AND PROCEDURES.
15. DOOR HINGES TO ALLOW ADJUSTMENT IN ALL DIRECTIONS WITH EQUAL PRESSURE ON ALL SEALS.
16. INDICATORS PROVIDED INSIDE LOCK AND OUTSIDE OF BOTH ENDS OF LOCK TO SHOW THE POSITION OF BOTH DOORS AND BOTH VALVES.
17. LIMIT SWITCHES TO BE PROVIDED FOR REMOTE INDICATION ON THE UNIT CONTROL BOARD AS TO THE POSITION OF EACH DOOR AND EQUALIZING VALVE.
18. LOCKS DESIGNED TO ALLOW FOR 1-1/2" MINIMUM VERTICAL AND 1" MINIMUM HORIZONTAL MOVEMENT OF LOCKS WITH RESPECT TO SHIELD BUILDING DUE TO PRESSURE AND THERMAL EXPANSION.
19. ALL MATERIALS AND WORK BY TVA FIELD SHALL BE IN ACCORDANCE WITH GENERAL CONSTRUCTION SPECIFICATION N3G-881, QUALITY LEVEL II.



SECTION A-A

NOTE A: SEAL ANNULUS BETWEEN PERSONNEL ACCESS LOCK AND SLEEVE USING DOW CORNING 3-6548 RTV SILICONE FOAM TO FILL DEPTH OR 12" MINIMUM.

NOTE B: INSTALL FLEXIBLE BOOT USING DUAL 1/2"x0.020 S/S BANDS ON EACH SIDE. CAULK BOOT IN PLACE USING DOW CORNING 732 OR 96-081 SILICONE ADHESIVE. ALLOW +/- 3" AXIAL MOVEMENT.



B-B & B1-B1

2-PAL-304-0002A - EL. 716'
2-PAL-304-0002B - EL. 757'

COMPONENT ID	DESCRIPTION	LOCATION
2-PI-304-0002A/1	PRESS IND OUTSIDE PAL AB END EL 716	EL 716', AZ 62°
2-PI-304-0002A/2	PRESS IND INSIDE PAL AB END EL 716	EL 716', AZ 62°
2-PI-304-0002A/3	PRESS IND INSIDE PAL RB END EL 716	EL 716', AZ 62°
2-PI-304-0002A/4	PRESS IND OUTSIDE PAL RB END EL 716	EL 716', AZ 62°
2-PI-304-0002B/1	PRESS IND OUTSIDE PAL AB END EL 757	EL 757', AZ 255°
2-PI-304-0002B/2	PRESS IND INSIDE PAL AB END EL 757	EL 757', AZ 255°
2-PI-304-0002B/3	PRESS IND INSIDE PAL RB END EL 757	EL 757', AZ 255°
2-PI-304-0002B/4	PRESS IND OUTSIDE PAL RB END EL 757	EL 757', AZ 255°

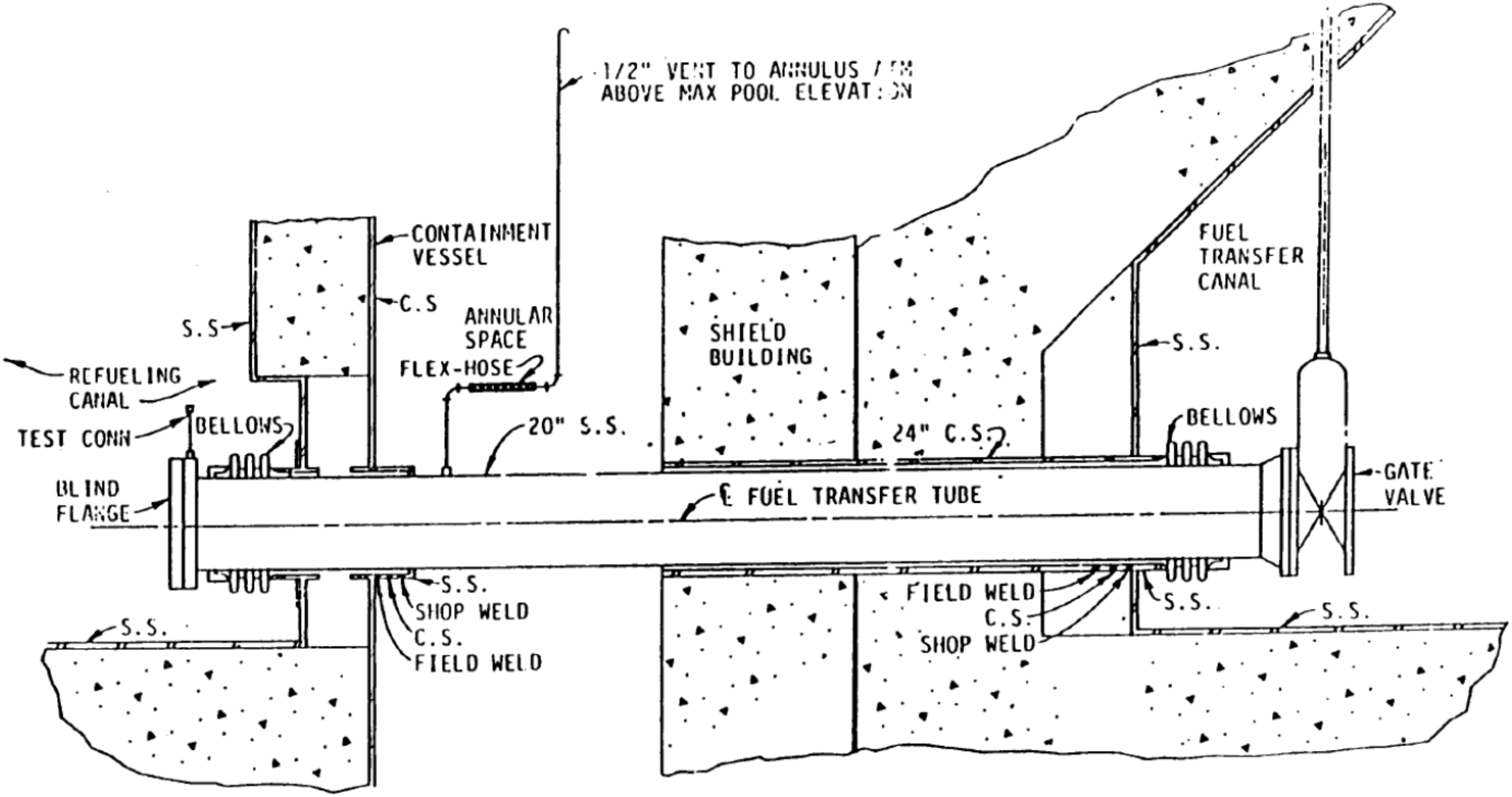
TABLE A

UFSAR AMENDMENT 1

WATTS BAR
FINAL SAFETY
ANALYSIS REPORT

REACTOR BUILDING
UNIT 2
PERSONNEL ACCESS LOCK
ARRGT & DETAILS

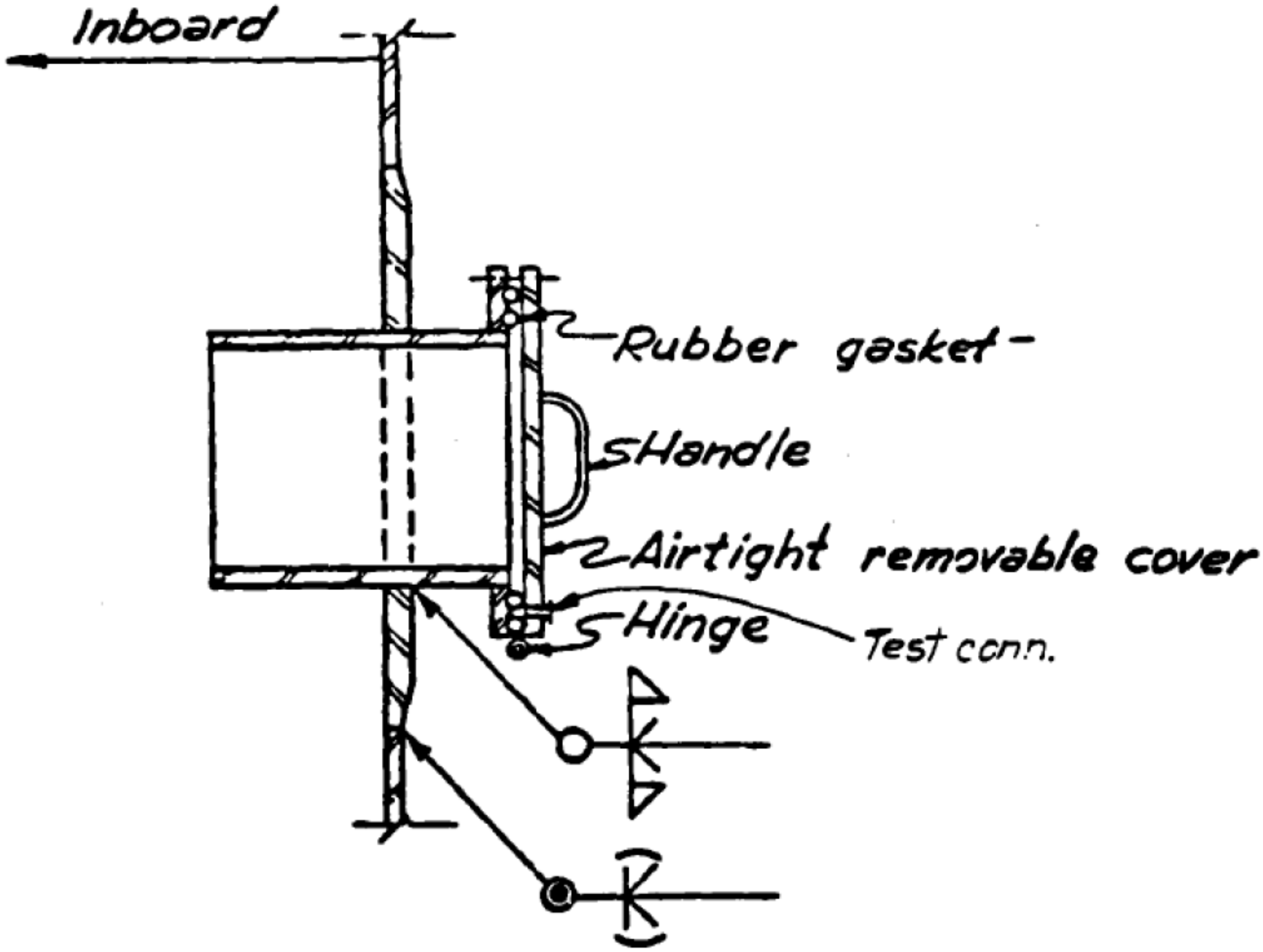
TVA DWG NO. 2-44N240 R4
FIGURE 6.2.4-13(U2)



WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

Type XVI
Transfer Tube

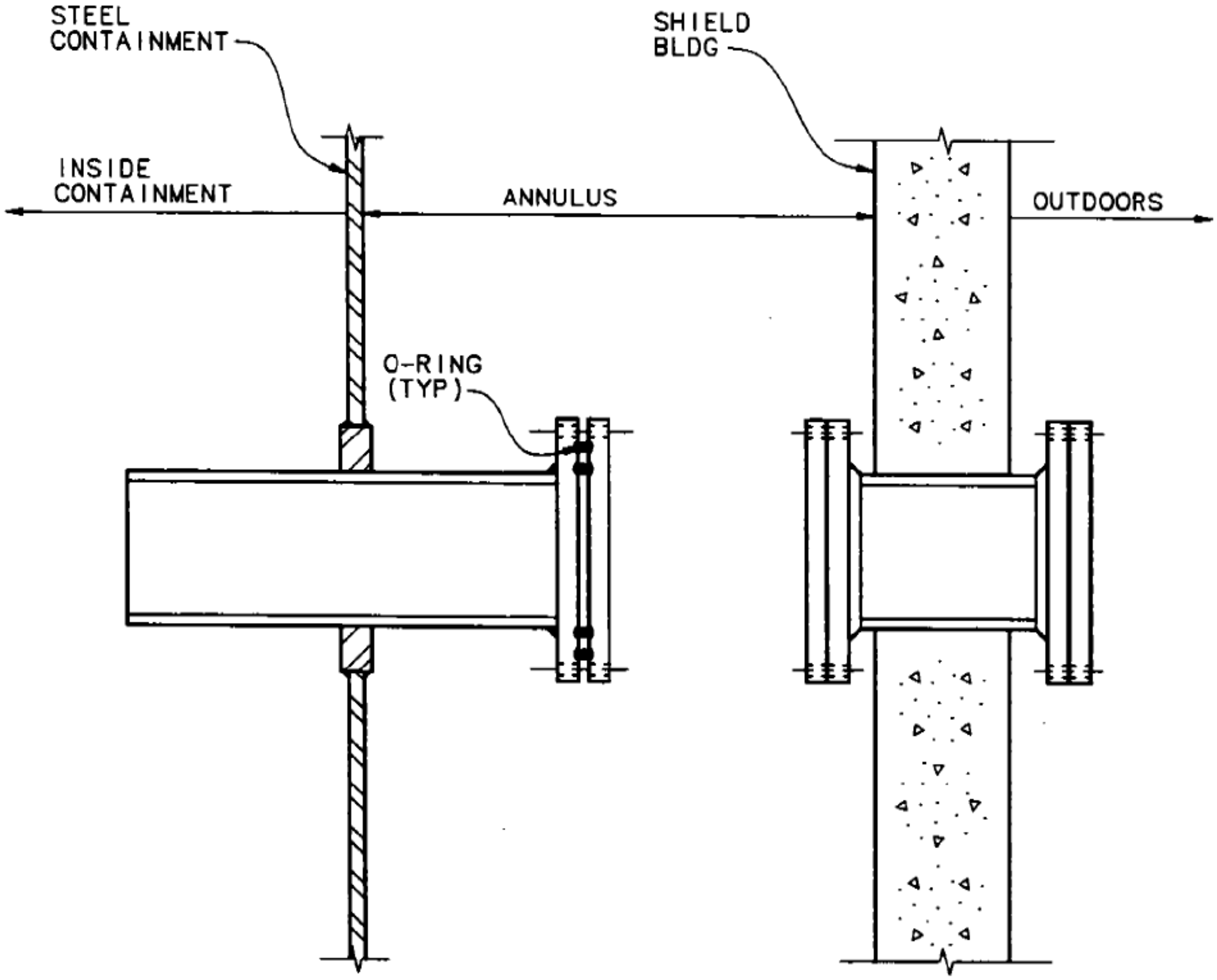
FIGURE 6.2.4-14



WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

Type XVII
Thimble Renewal Line

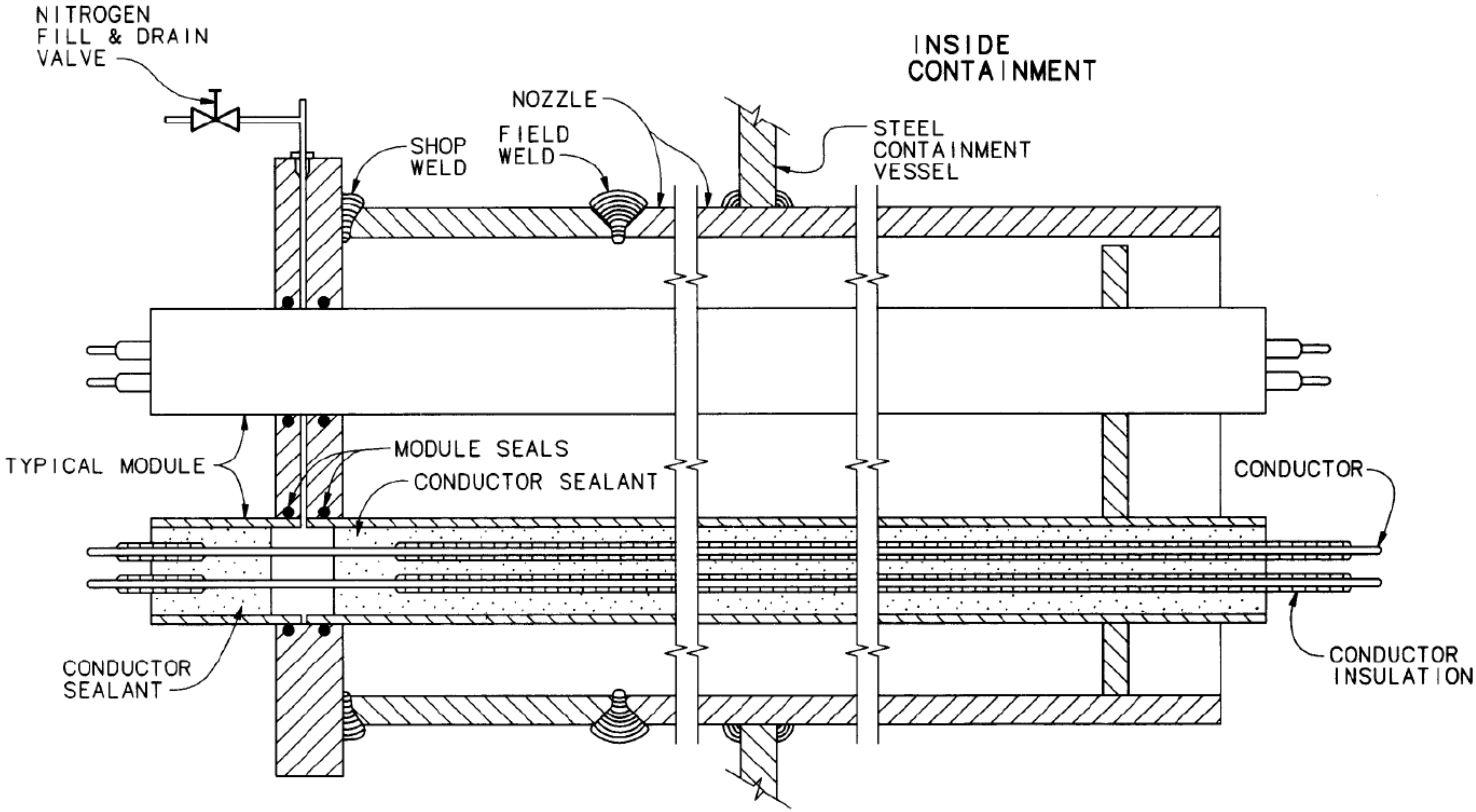
Figure 6.2.4-15



WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

Type XVIII
Ice Blowing Line

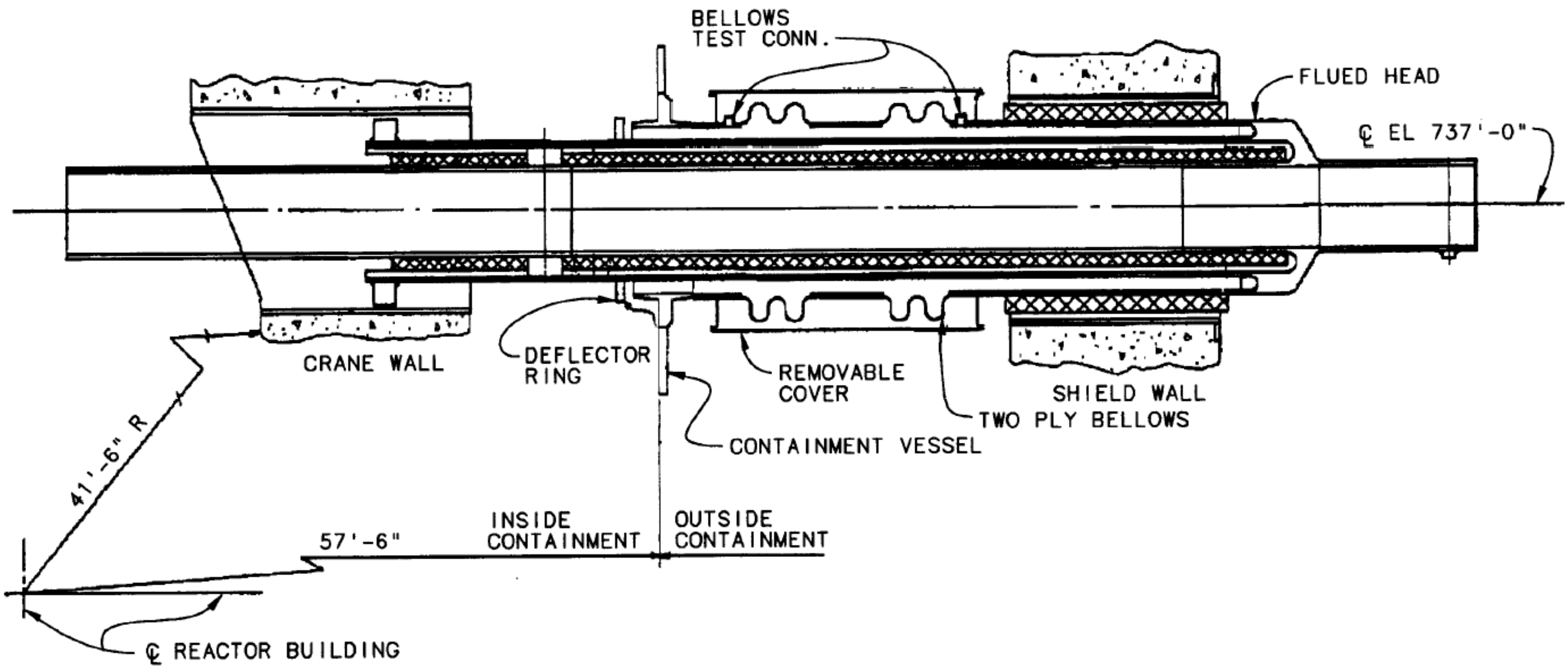
Figure 6.2.4-16



WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

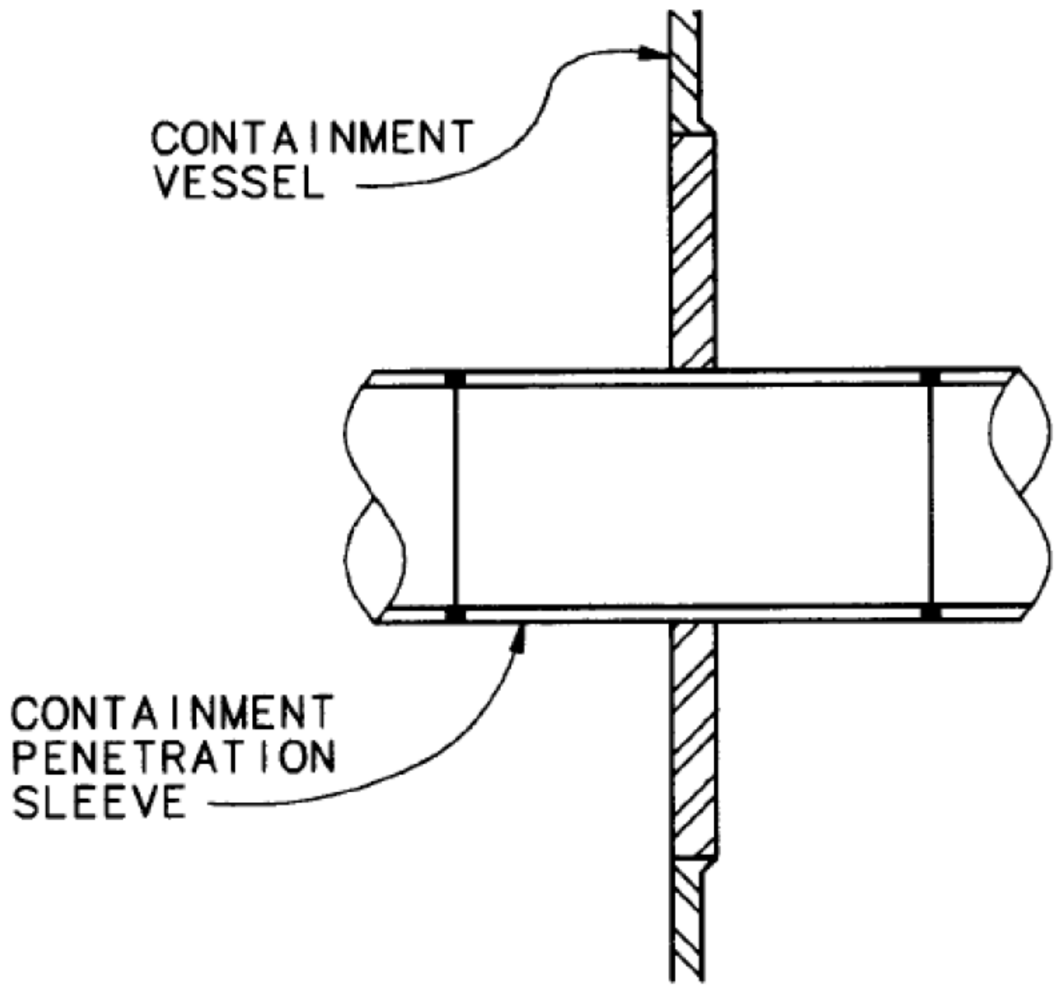
Type XIX
Electrical Penetration

FIGURE 6.2.4-17



WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

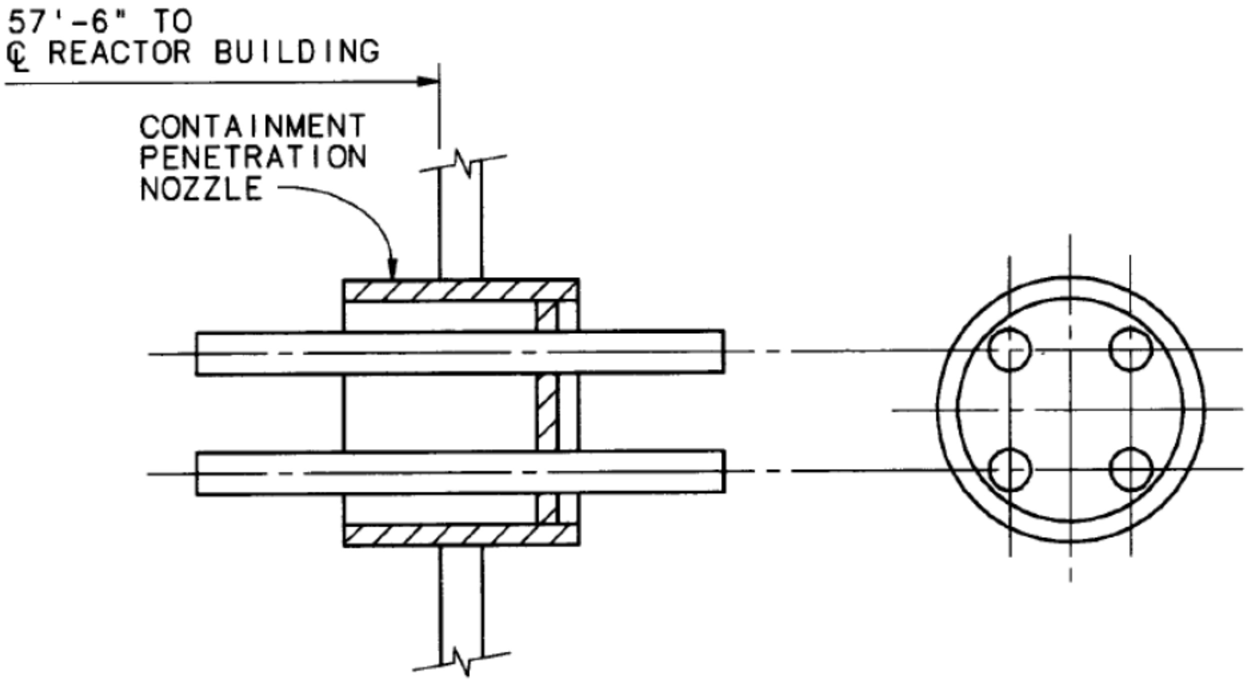
Type XX
Feedwater Bypass Penetrations
X-8A, X-8B, X-8C, X-8D
FIGURE 6.2.4-17A



WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

Type XXI
Upper and Lower Containment
ERCW Supply and Return
CCW from Excess Letdown
Heat Exchanger & from RC
Pump Coolers

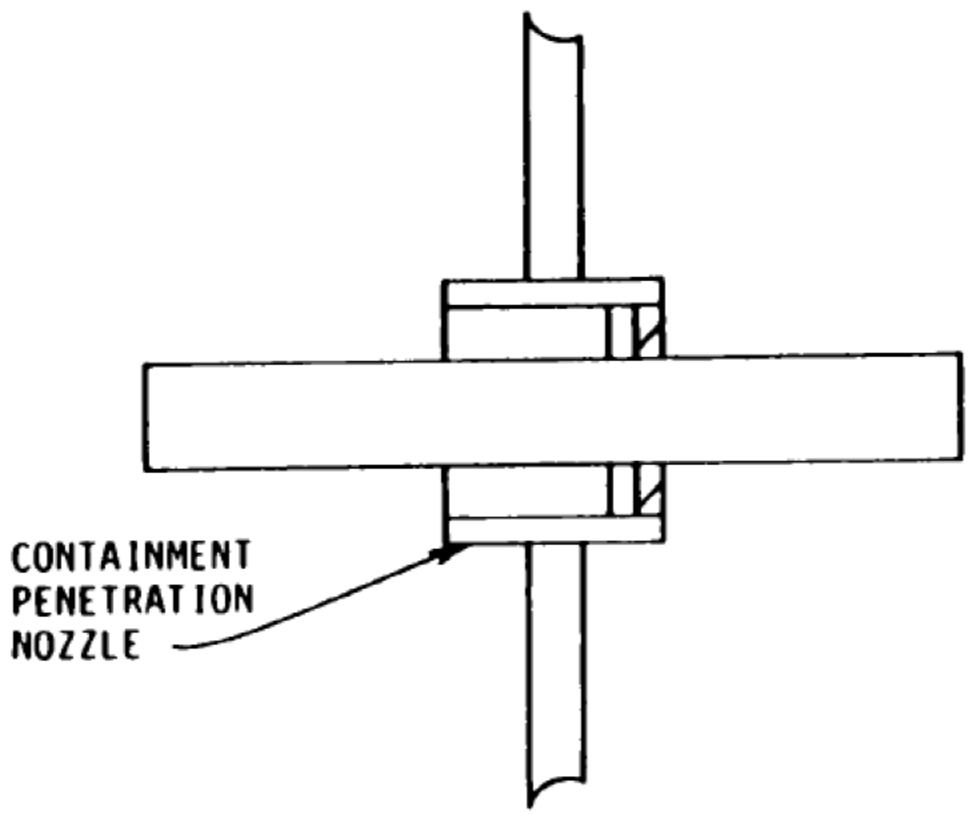
Figure 6.2.4-17B



WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

Type XXII
Multi Line Penetration
X-39

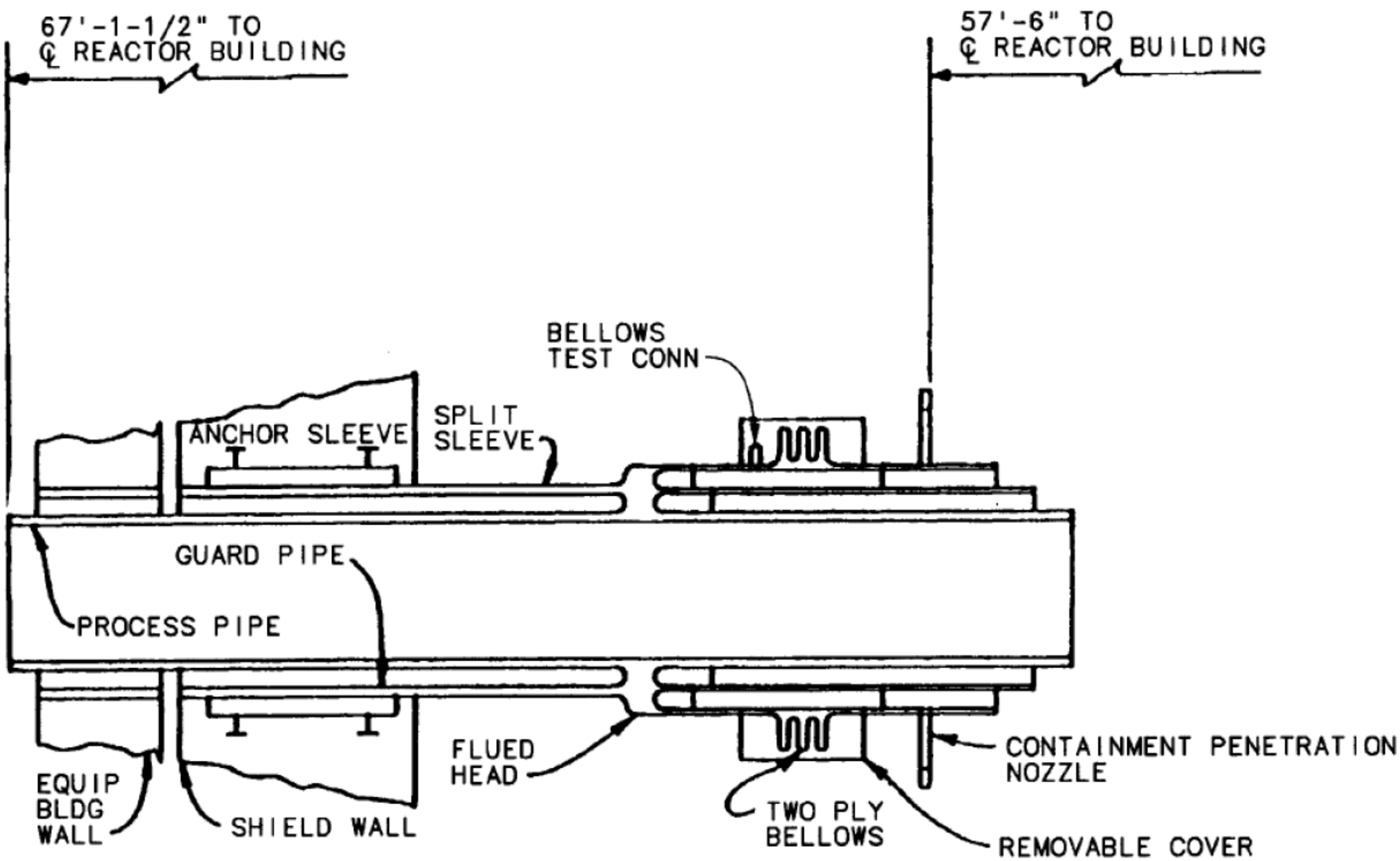
Figure 6.2.4-17C



WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

Type XXIII
Instrument Room Chilled
H20 Supply and Return

Figure 6.2.4-17D

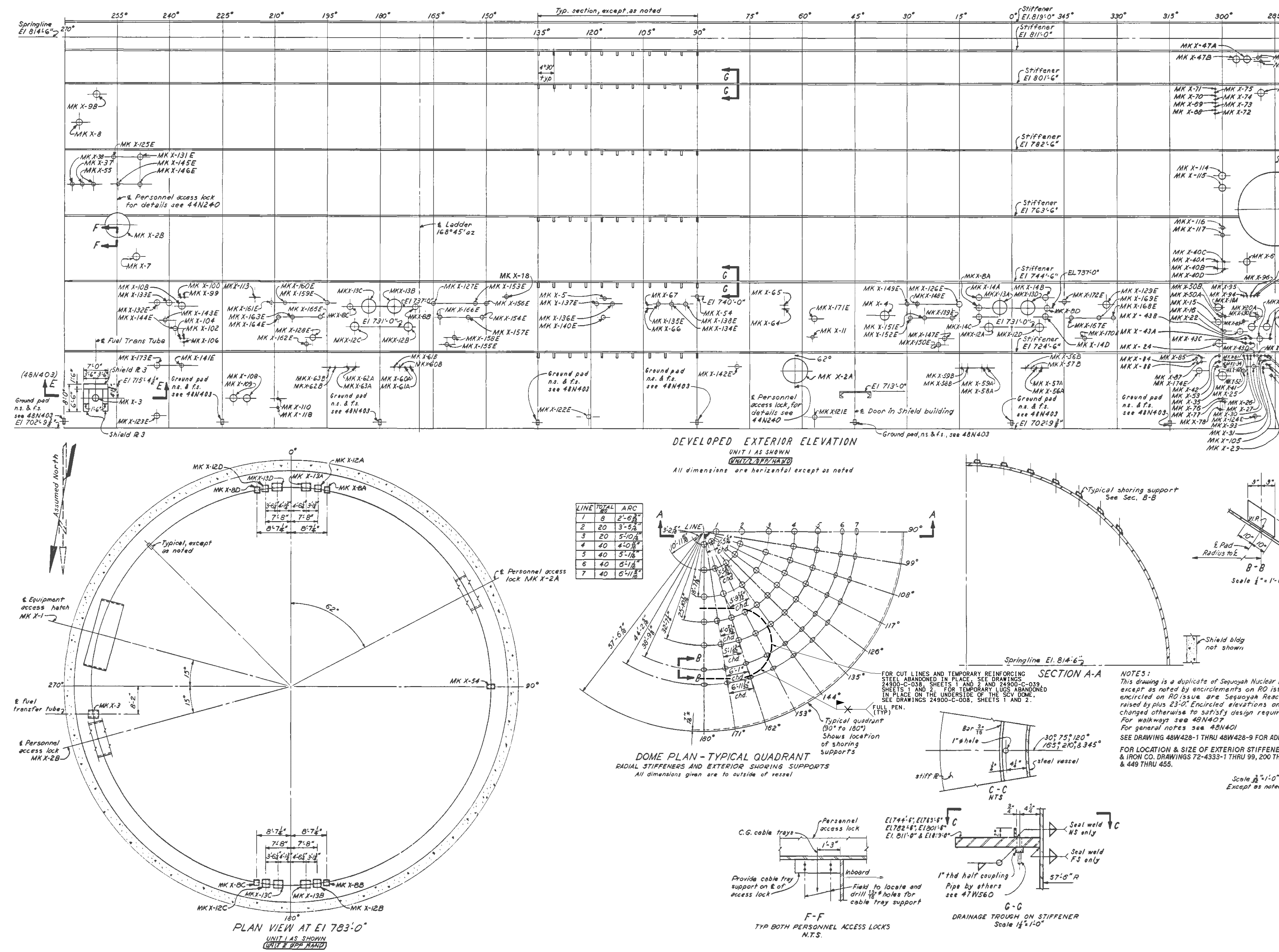


WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

Type XXIV
UHI X-108, X-109

FIGURE 6.2.4-17E

CAD MAINTAINED DRAWING



LINE	TOTAL ℄S	ARC
1	8	2'-6 3/32"
2	20	3'-5 7/16"
3	20	5'-10 1/16"
4	40	4'-0 11/16"
5	40	5'-1 7/16"
6	40	6'-1 1/16"
7	40	6'-11 5/16"

NOTES:

This drawing is a duplicate of Sequoyah Nuclear Plant drawing 48N405 R6 except as noted by encirclements on R0 issue. Elevations not encircled on R0 issue are Sequoyah Reactor Building elevations raised by plus 23'-0." Encircled elevations on R0 issue have been changed otherwise to satisfy design requirements.

For walkways see 48N407

For general notes see 48N401

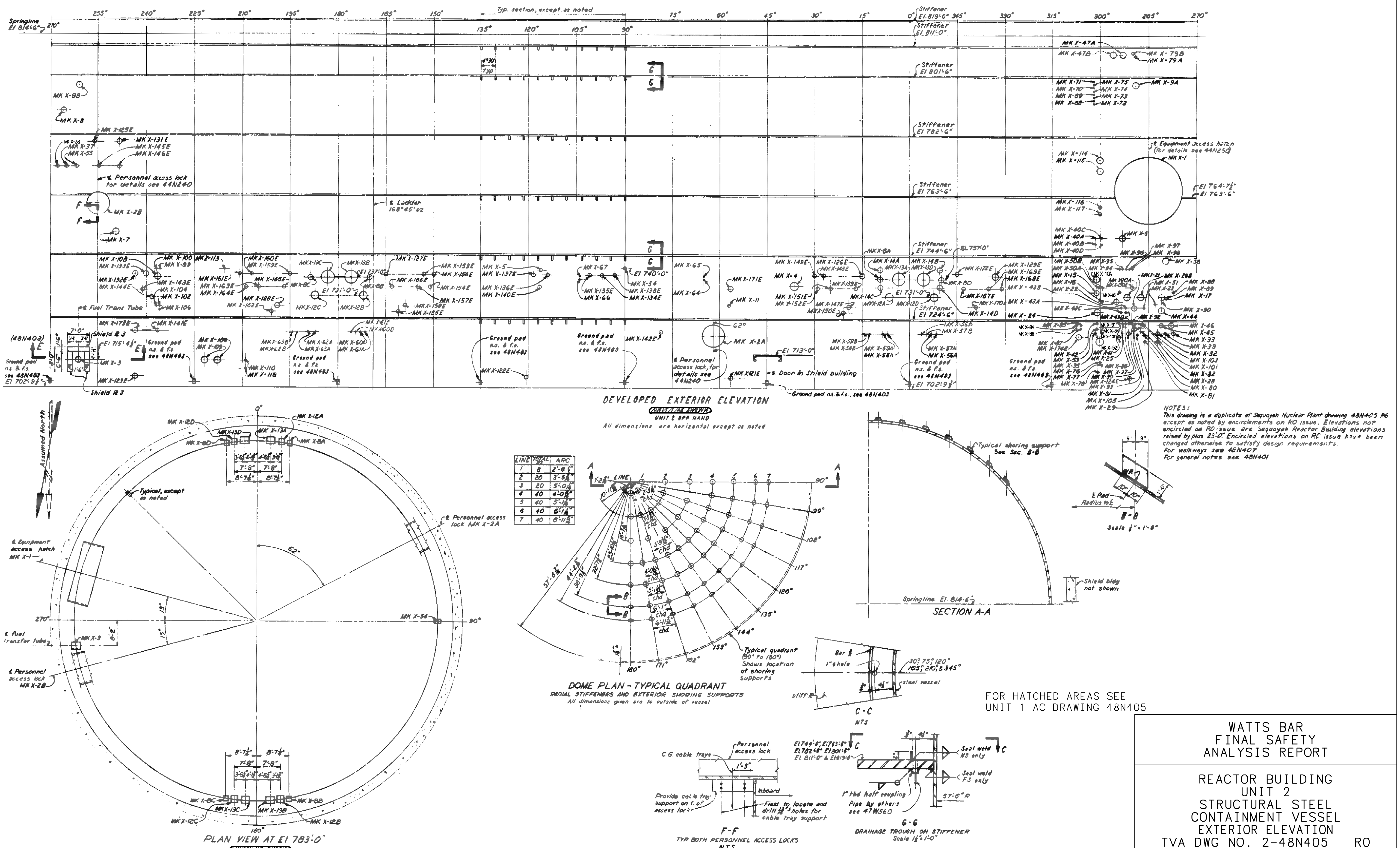
SEE DRAWING 48W428-1 THRU 48W428-9 FOR ADDITIONAL INFORMATION.

FOR LOCATION & SIZE OF EXTERIOR STIFFENERS SEE CHICAGO BRIDGE & IRON CO. DRAWINGS 72-4333-1 THRU 99, 200 THRU 213, 411 THRU 420 & 449 THRU 455.

Scale 3/32" = 1'-0"
Except as noted

WATTS BAR
FINAL SAFETY
ANALYSIS REPORT

REACTOR BUILDING
UNIT 1
STRUCTURAL STEEL
CONTAINMENT VESSEL
EXTERIOR ELEVATION
TVA DWG NO. 48N405 RF
FIGURE 6.2.4-18

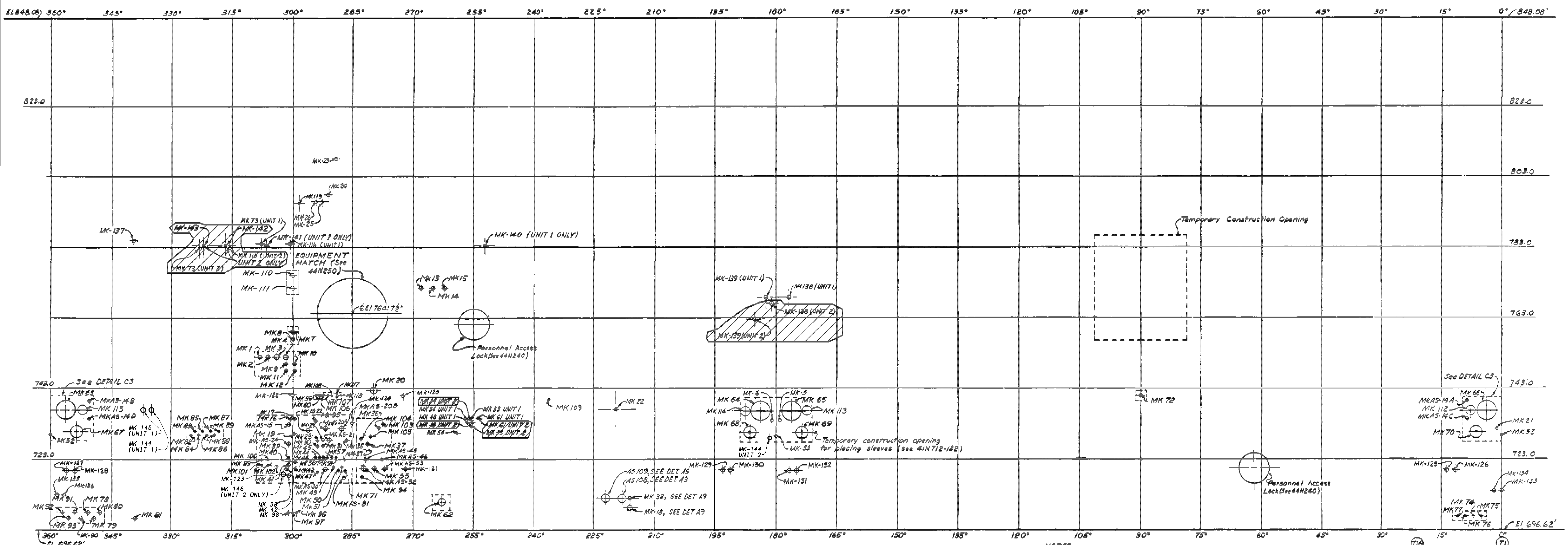


NOTES:
This drawing is a duplicate of Sequoyah Nuclear Plant drawing 48N405 R6 except as noted by encirclements on RO issue. Elevations not encircled on RO issue are Sequoyah Reactor Building elevations raised by plus 25'-0". Encircled elevations on RO issue have been changed otherwise to satisfy design requirements.
For walkways see 48N407
For general notes see 48N401

FOR HATCHED AREAS SEE
UNIT 1 AC DRAWING 48N405

WATTS BAR
FINAL SAFETY
ANALYSIS REPORT

REACTOR BUILDING
UNIT 2
STRUCTURAL STEEL
CONTAINMENT VESSEL
EXTERIOR ELEVATION
TVA DWG NO. 2-48N405 RO
FIGURE 6.2.4-18(U2)



DEVELOPED EXTERIOR ELEVATION
UNIT 1 AS SHOWN
UNIT 2 OPP HAND
(About ℄ between units)

NOTES:
1. FOR NOTES AND COMPANION DWGS, SEE SHT 1.
2. FOR COLLAR DETAILS, SEE 48N946-3.
3. FOR SHIELD WALL SLEEVE SEALS, SEE 47W470-10.
** MULTI-PROCESS LINE.

SLEEVE DATA

SLEEVE MK. NO.	PENETRATION NO.	PIPE SIZE	SLEEVE SIZE	LENGTH	AZIMUTH	℄ ELEVATION	QTY. UNIT 1	SYSTEM	B.M. MK. NO.
1	X49B	8"	12"	3'-0"	308°30'	752'-0"	1	RHR SPRAY	55
2	X49A	8"	12"	3'-0"	306°0'	752'-0"	1	RHR SPRAY	65
3	X48B	10"	16"	3'-0"	304°0'	752'-0"	1	CONTAINMENT SPRAY	45
4	X48A	10"	16"	3'-0"	301°30'	752'-0"	1	CONTAINMENT SPRAY	45
5		2"	4"	3'-0"	179°0'	731'-6"	1	CONTROL AIR	75
6		2"	4"	3'-0"	181°-0'	731'-6"	1	CONTROL AIR	75
7	X116		16"	3'-0"	300°0'	760'-0"	1	SPARE	45
8	X117		16"	3'-4"	300°0'	758'-0"	1	MAINTENANCE PORT (DET C9)	45
9	X40A	4"	10"	3'-0"	301°-30'	750'-3"	1	AUX. FEEDWATER	60
10			8"	3'-0"	299°-30'	750'-3"	1	SPARE	65
11	X40B	4"	10"	3'-0"	301°30'	748'-3"	1	AUX FEEDWATER	60
12	X40D	3"	8"	3'-0"	299°30'	748'-3"	1	SERVICE AIR	65
13	X38		12"	3'-0"	268°-0'	771'-6"	1	SPARE	65
14	X37		12"	3'-0"	265°0'	771'-6"	1	MAINTENANCE PORT (DET C9)	55
15	X55		12"	3'-0"	262°0'	771'-6"	1	SPARE	55
16	X50A	3"	6"	3'-0"	300°30'	734'-6"	1	COMPONENT COOLING SYS	70
17	X50B	3"	6"	3'-0"	299°30'	734'-6"	1	COMPONENT COOLING SYS	70
18	X118	4"	8"	3'-0"	216°0'	708'-6"	1	WET LAYUP H_2O TREATMENT	DET J31 A8
19	X16	3"	6"	3'-0"	299°30'	730'-6"	1	NOR. CHRG.	70
20	X36		24"	3'-0"	280°0'	742'-6"	1	SPARE	40
21			4"	3'-0"	1°-0'	731'-6"	1	SPARE (DET L3)	75
22	NONE	**	8"	3'-0"	220°-0'	737'-0"	1	RADIATION MONITOR	65
23	X79A	4"	10"	4'-0"	289°-0'	808'-0"	1	ICE BLOWING (DET F3)	60
24	X79B	6"	10"	4'-0"	291°-0'	798'-0"	1	NEG. RETURN (DET F3)	60
25	X47A	6"	12"	3'-0"	293°-0'	796'-0"	1	GLYCOL IN	55
26	X47B	6"	12"	3'-0"	295°0'	796'-0"	1	GLYCOL OUT	55
27	X17	SEE 47W331-1			281°30'	792'-0"		RHR RETURN	
28	X43C	2"	6"	3'-0"	293°30'	728'-6"	1	TO RC PUMP SEALS	70
29	X43B	2"	6"	3'-0"	292°0'	728'-6"	1	TO RC PUMP SEALS	70
30	X43D	2"	6"	3'-0"	293°30'	727'-0"	1	TO RC PUMP SEALS	70

SLEEVE DATA

SLEEVE MK. NO.	PENETRATION NO.	PIPE SIZE	SLEEVE SIZE	LENGTH	AZIMUTH	℄ ELEVATION	QTY. UNIT 1	SYSTEM	B.M. MK. NO.
31	X43A	2"	6"	3'-0"	292°0'	727'-0"	1	TO RC PUMP SEALS	70
32	X110		6"	3'-0"	216°-0'	711'-6"	1	SPARE	-
33		**	3"	3'-0"	256°-30' / 255°-21'	734'-9" / 732'-3"		I & C (47W625-16)	80
34	X-92C	**	3"	3'-0"	256°-30' / 255°-21'	734'-3"	1	I & C (47W625-16)	80
35	X51		10"	3'-0"	286°-30'	728'-6"	1	SPARE	60
36	X23	3/8"	3"	3'-0"	283°-0'	729'-0"	1	POST ACCIDENT SAMPLING	80
37	X44	4"	12"	3'-0"	281°-30'	727'-6"	1	SEAL WATER OUT	55
38	X107	SEE 47W331-1			300°-30'	717'-11½"		RHR SUPPLY	
39	X42	3"	6"	3'-0"	301°-0'	723'-6"	1	PRIM WTR TO PRT	70
40	X52	6"	10"	3'-0"	299°-30'	722'-6"	1	COMPONENT COOLING SYS	60
41	X53	6"	10"	3'-0"	301°-0'	721'-6"	1	COMPONENT COOLING SYS	60
42	X35	6"	10"	3'-0"	301°-30'	720'-6"	1	COMPONENT COOLING SYS	60
43	X34	2"	8"	3'-0"	299°30'	720'-6"	1	CONTROL AIR	65
44	X25	**	8"	3'-0"	294°-0'	723'-6"	1	I & C	65
45	X26		8"	3'-0"	293°-0'	723'-6"	1	SPARE	65
46	X27	**	8"	3'-0"	292°-0'	723'-6"	1	I & C	65
47	X41	2"	6"	3'-0"	294°-0'	719'-6"	1	FLOOR SUMP PMP DISCH	70
48		**	3"	3'-0"	(256°-30') / 255°-21'	733'-3"		I & C (47W625-16)	80
49	X31	4"	10"	3'-0"	289°-30'	722'-0"	1	FIRE PROTECTION	60
50	X29	6"	10"	3'-0"	288°-30'	720'-0"	1	COMPONENT COOLING SYS	60
51			3"	3'-0"	287°30'	721'-0"	1	SPARE DET L3	80
52	NONE		8"	3'-0"	0°-0'	729'-6"	1	SPARE	65
53	NONE		8"	3'-0"	180°-0'	729'-6"	1	SPARE	65
54	X28	**	4"	3'-0"	259°-0'	730'-3"	1	I & C (47W625-16)	75
55	X39	**	10"	3'-0"	280°-0'	720'-6"	1	I & C	60
56	X91	2"	6"	3'-0"	291°-0'	723'-6"	1	CONTROL AIR	70
57	X92		10"	3'-0"	290°-0'	723'-6"	1	SPARE	60
58	X93	3/8"	3"	3'-0"	289°-0'	723'-6"	1	I & C	80
59	X94	**	6"	3'-0"	294°-0'	741'-0"	1	CONT SAMPLE	70
60	X95	**	6"	3'-0"	293°-0'	741'-0"	1	CONT SAMPLE RETURN	70

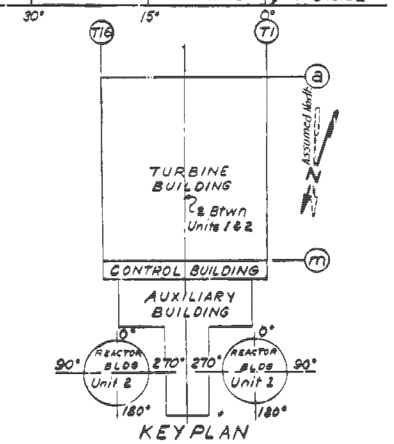
SLEEVE DATA

SLEEVE MK. NO.	PENETRATION NO.	PIPE SIZE	SLEEVE SIZE	LENGTH	AZIMUTH	℄ ELEVATION	QTY. UNIT 1	SYSTEM	B.M. MK. NO.
61		**	3"	3'-0"	[illegible]	[illegible]	1	I & C (47W625-16)	80
62	X3	20"	24"			711'-2⅜"	1	FUEL TRANSFER - DET A3	85
63	X13D	32"	66"	BY OTHERS		737'-0"	1	MAIN STEAM - DETAIL C3	
64	X13C	32"	66"	BY OTHERS		737'-0"	1	MAIN STEAM - DETAIL C3	
65	X13B	32"	66"	BY OTHERS		737'-0"	1	MAIN STEAM - DETAIL C3	
66	X13A	32"	66"	BY OTHERS		737'-0"	1	MAIN STEAM - DETAIL C3	
67	X12D	16"	40"	BY OTHERS		731'-0"	1	FEEDWATER - DETAIL C3	
68	X12C	16"	40"	BY OTHERS		731'-0"	1	FEEDWATER - DETAIL C3	
69	X12B	16"	40"	BY OTHERS		731'-0"	1	FEEDWATER - DETAIL C3	
70	X12A	16"	40"	BY OTHERS		731'-0"	1	FEEDWATER - DETAIL C3	
71	X80		8"	3'-0"	286°30'	720'-0"	1	SPARE	65
72	X54		8"	3'-6"	90°-0'	740'-0"	1	THIMBLE RENEWAL SEE DETAIL B3	88
73	NONE	[illegible]	4"	3'-0"	[illegible]	[illegible]	1	I & C - EGTS SENSOR	75
74	X62A	8"	10"	3'-0"	7°-0'	706'-6"	1	LOWER CONT ERCW SUPPLY	60
75	X63A	8"	10"	3'-0"	5°-0'	706'-6"	1	LOWER CONT ERCW RETURN	60
76	X61A	8"	10"	3'-0"	9°-0'	706'-6"	1	LOWER CONT ERCW RETURN	60
77	X60A	8"	10"	3'-0"	11°-0'	706'-6"	1	LOWER CONT ERCW SUPPLY	60
78	X59A	6"	10"	3'-0"	351°-0'	708'-0"	1	LOWER CONT ERCW RETURN	60
79	X58A	6"	10"	3'-0"	349°30'	706'-6"	1	LOWER CONT ERCW SUPPLY	60
80	X56A	6"	10"	3'-0"	345°-0'	708'-0"	1	LOWER CONT ERCW SUPPLY	60
81	X57A	6"	10"	3'-0"	339°-0'	706'-6"	1	LOWER CONT ERCW RETURN	60
82	X71	2"	4"	3'-0"	325°-0'	730'-0"	1	UPPER CONT ERCW RETURN	75
83	X75	2"	4"	3'-0"	326°-0'	731'-0"	1	UPPER CONT ERCW SUPPLY	75
84	X70	2"	4"	3'-0"	323°-0'	730'-0"	1	UPPER CONT ERCW RETURN	75
85	X74	2"	4"	3'-0"	322°-0'	731'-0"	1	UPPER CONT ERCW SUPPLY	75
86	X69	2"	4"	3'-0"	321°-0'	730'-0"	1	UPPER CONT ERCW SUPPLY	75
87	X73	2"	4"	3'-0"	320°-0'	731'-0"	1	UPPER CONT ERCW RETURN	75
88	X68	2"	4"	3'-0"	319°-0'	730'-0"	1	UPPER CONT ERCW SUPPLY	75
89	X72	2"	4"	3'-0"	318°-0'	731'-0"	1	UPPER CONT ERCW RETURN	75
90	X65	2"	6"	3'-0"	352°30'	706'-6"	1	INST RM CH WATER RETURN	70

SLEEVE DATA

SLEEVE MK. NO.	PENETRATION NO.	PIPE SIZE	SLEEVE SIZE	LENGTH	AZIMUTH	℄ ELEVATION	QTY. UNIT 1	SYSTEM	B.M. MK. NO.
91	X67	2"	6"	3'-0"	354°-0'	708'-0"	1	INST RM CH WATER SUPPLY	70
92	X64	2"&3/4" COND	6"	3'-0"	357°-0'	708'-0"	1	INST RM CH WATER RETURN & ELECTRICAL CONDUIT	70
93	X66	2"	6"	3'-0"	355°30'	706'-6"	1	INST RM CH WATER SUPPLY	70
94	X82	6"	10"	3'-0"	282°30'	718'-0"	1	REFUELING CAVITY CLEANUP PUMP SUCTION	60
95	X83	4"	8"	3'-0"	294°-0'	733'-0"	1	REFUELING CAVITY CLEANUP PUMP DISCHARGE	65
96	X77	2"	6"	3'-0"	299°0'	708'-9"	1	DEMIN WATER	70
97	X76	2"	6"	3'-0"	300°-0'	708'-9"	1	SERVICE AIR	70
98	X78	4"	6"	3'-0"	301°-0'	708'-9"	1	FIRE PROTECTION	70
99	X84	**	6"	3'-0"	307°-30'	723'-0"	1	GAS ANALYZER	70
100	X85	**	6"	3'-0"	306°-0'	723'-0"	1	I & C SAMPLE	70
101	X86	**	6"	3'-0"	307°30'	721'-6"	1	I & C SAMPLE	70
102	X87	**	6"	3'-0"	306°-0'	721'-6"	1	I & C SAMPLE	70
103		4"	8"	3'-0"	277°-30'	733'-0"	1	FIRE PROTECTION	65
104		4"	8"	3'-0"	278°30'	732'-0"	1	FIRE PROTECTION	65
105	X90	2"	4"	3'-0"	280°-30'	729'-6"	1	CONTROL AIR	75
106	X96		6"	3'-0"	292°-0'	741'-0"	1	SPARE	70
107	X97		3"	3'-0"	291°-0'	741'-0"	1	SPARE	80
108	X98	1" / 3/4"	3"	3'-0"	290°-0'	741'-0"	1	ELECTRICAL CONDUIT (U1) / I & C	80
109	X99	NONE / 1"	3"	3'-0"	236°-0'	740'-0"	1	SPARE / I & C	80
110	X114	2"	6"	3'-0"	300°-0'	775'-0"	1	GLYCOL FLOOR COOLING	70
111	X115	2"	6"	3'-0"	300°-0'	771'-6"	1	GLYCOL FLOOR COOLING	70
112	X8A	6"	30"	3'-0"	—	737'-0"	1	FEEDWATER - DETAIL C3	38
113	X8B	6"	30"	3'-0"	—	737'-0"	1	FEEDWATER - DETAIL C3	38
114	X8C	6"	30"	3'-0"	—	737'-0"	1	FEEDWATER - DETAIL C3	38
115	X8D	6"	30"	3'-0"	—	737'-0"	1	FEEDWATER - DETAIL C3	38
116	NONE	[illegible]	4"	3'-0"	[illegible]	[illegible]	1	I & C - EGTS SENSOR	75
				(CONT ON SHEET 7)					

UNIT 1 ONLY



KEYPLAN
SCALE: NTS
SCALE: NTS (865'-6½" Radius)

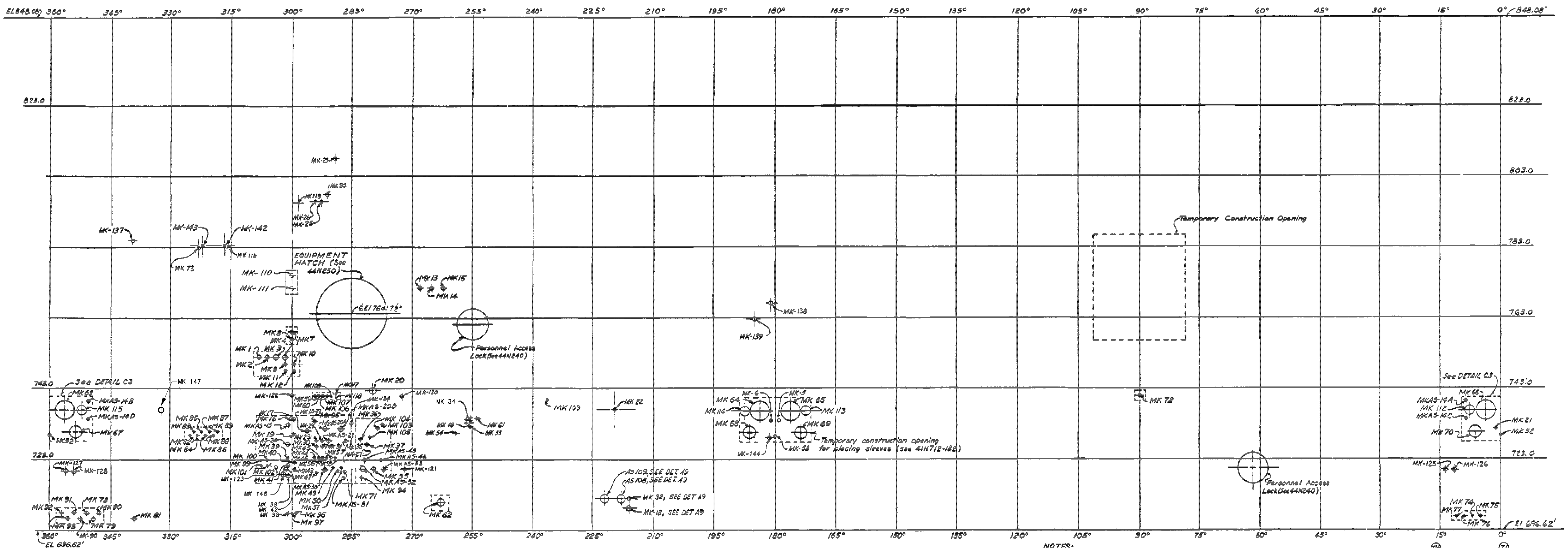
UFSAR AMENDMENT 1

WATTS BAR
FINAL SAFETY
ANALYSIS REPORT

POWERHOUSE
REACTOR BUILDING-UNIT 1
MECHANICAL
SLEEVES - SHIELD
BUILDING
TVA DWG NO. 47W470-2 RL
FIGURE 6.2.4-19

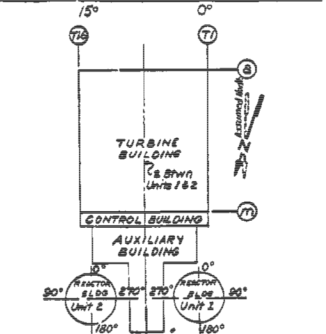
CAD MAINTAINED DRAWING

CAD MAINTAINED DRAWING



DEVELOPED EXTERIOR ELEVATION
UNIT 2 OPP HAND
(About ℄ between Units)

NOTES:
1. FOR NOTES AND COMPANION DWGS, SEE SHT 1.
2. FOR COLLAR DETAILS, SEE 48N946-3.
3. FOR SHIELD WALL SLEEVE SEALS, SEE 47W470-10.
** MULTI-PROCESS LINE.



KEYPLAN
SCALE: NTS

SLEEVE DATA

SLEEVE MK. NO.	PENETRATION NO.	PIPE SIZE	SLEEVE SIZE	LENGTH	AZIMUTH	& ELEVATION	QTY. UNIT 2	SYSTEM	B.M. MK. NO.
1	X49B	8"	12"	3'-0"	308°30'	752'-0"	1	RHR SPRAY	55
2	X49A	8"	12"	3'-0"	306°0'	752'-0"	1	RHR SPRAY	55
3	X48B	10"	16"	3'-0"	304°0'	752'-0"	1	CONTAINMENT SPRAY	45
4	X48A	10"	16"	3'-0"	301°30'	752'-0"	1	CONTAINMENT SPRAY	45
5		2"	4"	3'-0"	179°0'	731'-6"	1	CONTROL AIR	75
6		2"	4"	3'-0"	181°-0'	731'-6"	1	CONTROL AIR	75
7			16"	3'-0"	300°-0'	760'-0"	1	SPARE	45
8	X117		16"	3'-4"	300°-0'	758'-0"	1	MAINTENANCE PORT DET C3	45
9	X40A	4"	10"	3'-0"	301°-30'	750'-3"	1	AUX. FEEDWATER	60
10			8"	3'-0"	299°-30'	750'-3"	1	SPARE	65
11	X40B	4"	10"	3'-0"	301°30'	748'-3'	1	AUX FEEDWATER	60
12	X40D	3"	8"	3'-0"	299°30'	748'-5"	1	SERVICE AIR	65
13			12"	3'-0"	268°-0'	771'-6"	1	SPARE	65
14	X37		12"	3'-0"	265°0'	771'-6"	1	MAINTENANCE PORT DET C3	55
15			12"	3'-0"	262°0'	771'-6"	1	SPARE	55
16	X50A	3"	6"	3'-0"	300°30'	734'-6"	1	COMPONENT COOLING SYS	70
17	X50B	3"	6"	3'-0"	299°30'	734'-6"	1	COMPONENT COOLING SYS	70
18	X118		8"	3'-0"	216°0'	708'-6"	1	SPARE	DET G1
19	X16	3"	6"	3'-0"	299°30'	730'-6"	1	NOR. CHRG.	70
20			24"	3'-0"	280°0'	742'-6"	1	SPARE	40
21			4"	3'-0"	1°-0'	731'-6"	1	SPARE	75
22	NONE	**	8"	2'-11¼"	220°-4'	738'-1½"	1	N2 SG PORV (I&C)	65
23	X79A	4"	10"	4'-0"	289°-0'	808'-0"	1	ICE BLOWING (DET F3)	60
24	X79B	6"	10"	4'-0"	291°-0'	798'-0"	1	NEG. RETURN (DET F3)	60
25	X47A	6"	12"	3'-0"	293°-0'	796'-0"	1	GLYCOL IN	55
26	X47B	6"	12"	3'-0"	295°0'	796'-0"	1	GLYCOL OUT	55
27	X17	SEE 47W331-1			281°-30'	732'-0"		RHR RETURN	X
28	X43C	2"	6"	3'-0"	293°30'	728'-6"	1	TO RC PUMP SEALS	70
29	X43B	2"	6"	3'-0"	292°0'	728'-6"	1	TO RC PUMP SEALS	70
30	X43D	2"	6"	3'-0"	293°30'	727'-0"	1	TO RC PUMP SEALS	70

SLEEVE DATA

SLEEVE MK. NO.	PENETRATION NO.	PIPE SIZE	SLEEVE SIZE	LENGTH	AZIMUTH	& ELEVATION	QTY. UNIT 2	SYSTEM	B.M. MK. NO.
31	X43A	2"	6"	3'-0"	292°0'	727'-0"	1	TO RC PUMP SEALS	70
32			6"	3'-0"	216°0'	711'-6"	1	SPARE	-
33			3"	3'-0"	255°-21'	722'-8"	1	SPARE	80
34	X-92C		3"	3'-0"	255°-21'	734'-3"	1	SPARE	80
35			10"	3'-0"	286°-30'	728'-6"	1	SPARE	60
36			3"	3'-0"	285°0'	729'-0"	1	SPARE	80
37	X44	4"	12"	3'-0"	281°30'	727'-6"	1	SEAL WATER OUT	55
38	X107	SEE 47W331-1			300°-30'	717'-11½"		RHR SUPPLY	X
39	X42	3"	6"	3'-0"	301°-0'	723'-6"	1	PRIM WTR TO PRT	70
40	X52	6"	10"	3'-0"	299°30'	722'-6"	1	COMPONENT COOLING SYS	60
41	X53	6"	10"	3'-0"	301°0'	721'-6"	1	COMPONENT COOLING SYS	60
42	X35	6"	10"	3'-0"	301°-30'	720'-6"	1	COMPONENT COOLING SYS	60
43	X34	2"	8"	3'-0"	299°30'	720'-6"	1	CONTROL AIR	65
44	X25	**	8"	3'-0"	296°-0'	723'-6"	1	I & C	65
45			8"	3'-0"	293°-0'	723'-6"	1	SPARE	65
46	X27	**	8"	3'-0"	292°-0'	723'-6"	1	I & C	65
47	X41	2"	6"	3'-0"	294°-0'	719'-6"	1	FLOOR SUMP PMP DISCH	70
48			3"	3'-0"	255° 21'	733'-3"	1	SPARE	80
49	X31	4"	10"	3'-0"	289°30'	722'-0"	1	FIRE PROTECTION	60
50	X29	6"	10"	3'-0"	288°30'	720'-0"	1	COMPONENT COOLING SYS	60
51			3"	3'-0"	287°30'	721'-0"	1	SPARE DET L3	80
52	NONE		8"	3'-0"	0°-0'	729'-6"	1	SPARE	65
53	NONE		8"	3'-0"	180°-0'	729'-6"	1	SPARE	65
54	X28		4"	3'-0"	259°-0'	730'-3"	1	SPARE	75
55	X39	**	10"	3'-0"	280°0'	720'-6"	1	I & C	60
56	X91	2"	6"	3'-0"	291°-0'	723'-6"	1	CONTROL AIR	70
57			10"	3'-0"	290°-0'	723'-6"	1	SPARE	60
58	X93	3/8"	3"	3'-0"	289°-0'	723'-6"	1	I & C	80
59	X94	**	6"	3'-0"	294°0'	741'-0"	1	RADIATION MONITORING	70
60	X95	**	6"	3'-0"	295°0'	741'-0"	1	RADIATION MONITORING	70

SLEEVE DATA

SLEEVE MK. NO.	PENETRATION NO.	PIPE SIZE	SLEEVE SIZE	LENGTH	AZIMUTH	& ELEVATION	QTY. UNIT 2	SYSTEM	B.M. MK. NO.
61			3"	3'-0"	253° 4'	734'3"	1	SPARE	80
62	X3	20"	24"	3'0"	262°'0'	711'-2¾"	1	FUEL TRANSFER-DET A3	85
63	X13D	32"	66" BY OTHERS (SEE 48N946)			737'-0"	1	MAIN STEAM - DETAIL C3	
64	X13C	32"	66" BY OTHERS (SEE 48N946)			737'-0"	1	MAIN STEAM - DETAIL C3	
65	X13B	32"	66" BY OTHERS (SEE 48N946)			737'-0"	1	MAIN STEAM - DETAIL C3	
66	X13A	32"	66" BY OTHERS (SEE 48N946)			737'-0"	1	MAIN STEAM - DETAIL C3	
67	X12D	16"	40" BY OTHERS (SEE 48N946)			731'-0"	1	FEEDWATER - DETAIL C3	
68	X12C	16"	40" BY OTHERS (SEE 48N946)			731'-0"	1	FEEDWATER - DETAIL C3	
69	X12B	16"	40" BY OTHERS (SEE 48N946)			731'-0"	1	FEEDWATER - DETAIL C3	
70	X12A	16"	40" BY OTHERS (SEE 48N946)			731'-0"	1	FEEDWATER - DETAIL C3	
71			8"	3'-0"	286°30'	720'-0"	1	SPARE	65
72	X54		8"	3'-6"	90°-0'	740'-0"	1	THIMBLE RENEWAL SEE DETAIL B3	88
73	NONE		4"	3'-0"					75
74	X62A	8"	10"	3'-0"	7°-0'	706'-6"	1	LOWER CONT ERCW SUPPLY	60
75	X63A	8"	10"	3'-0"	5°-0'	706'-6"	1	LOWER CONT ERCW RETURN	60
76	X61A	8"	10"	3'-0"	9°-0'	706'-6"	1	LOWER CONT ERCW RETURN	60
77	X60A	8"	10"	3'-0"	11°-0'	706'-6"	1	LOWER CONT ERCW SUPPLY	60
78	X59A	6"	10"	3'-0"	351°-0'	708'-0"	1	LOWER CONT ERCW RETURN	60
79	X58A	6"	10"	3'-0"	349°30'	706'-6"	1	LOWER CONT ERCW SUPPLY	60
80	X56A	6	10"	3'-0"	348°-0'	708'-0"	1	LOWER CONT ERCW SUPPLY	60
81	X57A	6"	10"	3'-0"	339°-0'	706'-6"	1	LOWER CONT ERCW RETURN	60
82	X71	2"	4"	3'-0"	325°-0'	730'-0"	1	UPPER CONT ERCW RETURN	75
83	X75	2"	4"	3'-0"	324°-0'	731'-0"	1	UPPER CONT ERCW SUPPLY	75
84	X70	2"	4"	3'-0"	323°-0'	730'-0"	1	UPPER CONT ERCW RETURN	75
85	X74	2"	4"	3'-0"	322°-0'	731'-0"	1	UPPER CONT ERCW SUPPLY	75
86	X69	2"	4"	3'-0"	321°-0'	730'-0"	1	UPPER CONT ERCW SUPPLY	75
87	X73	2"	4"	3'-0"	320°-0'	731'-0"	1	UPPER CONT ERCW RETURN	75
88	X68	2"	4"	3'-0"	319°-0'	730'-0"	1	UPPER CONT ERCW SUPPLY	75
89	X72	2"	4"	3'-0"	318°-0'	731'-0"	1	UPPER CONT ERCW RETURN	75
90	X65	2"	6"	3'-0"	352°30'	706'-6"	1	INST RM CH WATER RETURN	70

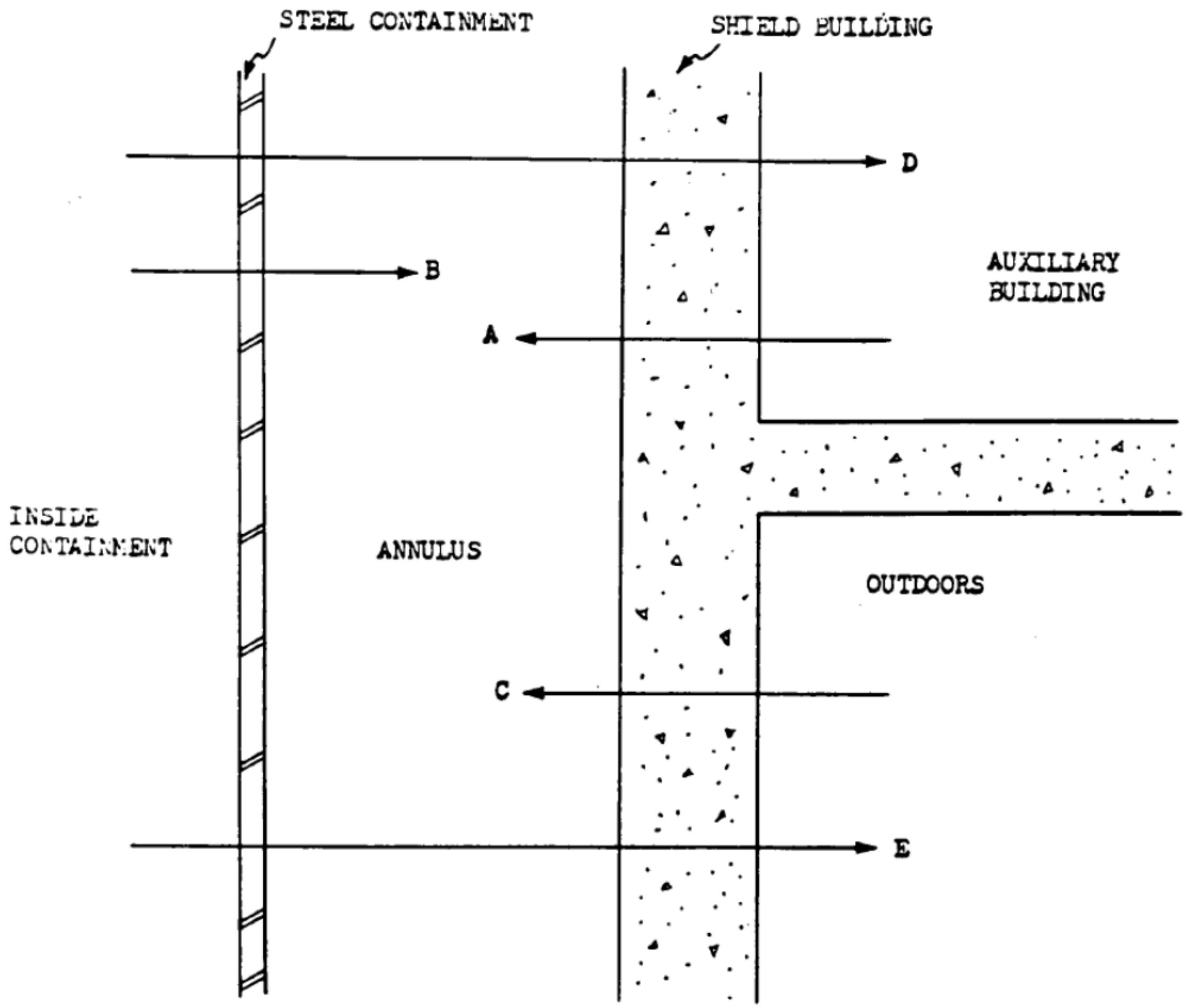
SLEEVE DATA

SLEEVE MK. NO.	PENETRATION NO.	PIPE SIZE	SLEEVE SIZE	LENGTH	AZIMUTH	& ELEVATION	QTY. UNIT 2	SYSTEM	B.M. MK. NO.
91	X67	2"	6"	3'-0"	354°-0'	708'-0"	1	INST RM CH WATER SUPPLY	70
92	X64		6"	3'-0"	357°-0'	708'-0"	1		70
93	X66	2"	6"	3'-0"	355°30'	706'-6"	1	INST RM CH WATER SUPPLY	70
94	X82	6"	10"	3'-0"	282°30'	718'-0"	1	REFUELING CAVITY CLEANUP PUMP SUCTION	60
95	X83	4"	8"	3'-0"	294°-0'	733'-0"	1	REFUELING CAVITY CLEANUP PUMP DISCHARGE	65
96	X77	2"	6"	3'-0"	299°0'	708'-9"	1	DEMIN WATER	70
97	X76	2"	6"	3'-0"	300°-0'	708'-9"	1	SERVICE AIR	70
98	X78	4"	6"	3'-0"	301°-0'	708'-9"	1	FIRE PROTECTION	70
99	X84	3/8"	6"	3'-0"	307°30'	723'-0"	1	GAS ANALYZER	70
100	X85	3/8"	6"	3'-0"	306°0'	723'-0"	1	I & C SAMPLE	70
101	X84	MULTI	6"	3'-0"	307°30'	721'-6"	1	RVLIS	70
102	X87	MULTI	6"	3'-0"	306°-0'	721'-6"	1	RVLIS	70
103		4"	8"	3'-0"	277°-30'	733'-0"	1	FIRE PROTECTION	65
104		4"	8"	3'-0"	278°30'	732'-0"	1	FIRE PROTECTION	65
105	X90	2"	4"	3'-0"	280°-30'	729'-6"	1	CONTROL AIR	75
106			6"	3'-0"	292°-0'	741'-0"	1	SPARE	70
107			8"	3'-0"	291°0'	741'-0"	1	SPARE	80
108			3"	3'-0"	290°-0'	741'-0"	1	SPARE	80
109			3"	3'-0"	236°-0'	740'-0"	1	SPARE	80
110	X114	2"	6"	3'-0"	300°-0'	775'-0"	1	GLYCOL FLOOR COOLING	70
111	X115	2"	6"	3'-0"	300°-0'	771'-6"	1	GLYCOL FLOOR COOLING	70
112	X8A	6"	30"	3'-0"	—	737'-0"	1	FEEDWATER-DETAIL C3	38
113	X8B	6"	30"	3'-0"	—	737'-0"	1	FEEDWATER - DETAIL C3	38
114	X8C	6"	30"	3'-0"	—	737'-0"	1	FEEDWATER - DETAIL C3	38
115	X8D	6"	30"	3'-0"	—	737'-0"	1	FEEDWATER - DETAIL C3	38
116	NONE		4"	3'-0"					75
				(CONT ON SHEET 7)					

UFSAR AMENDMENT 1

WATTS BAR
FINAL SAFETY
ANALYSIS REPORT

POWERHOUSE
UNIT 2
MECHANICAL
SLEEVES - SHIELD
BUILDING
TVA DWG NO. 2-47W470-2 R2
FIGURE 6.2.4-19(U2)

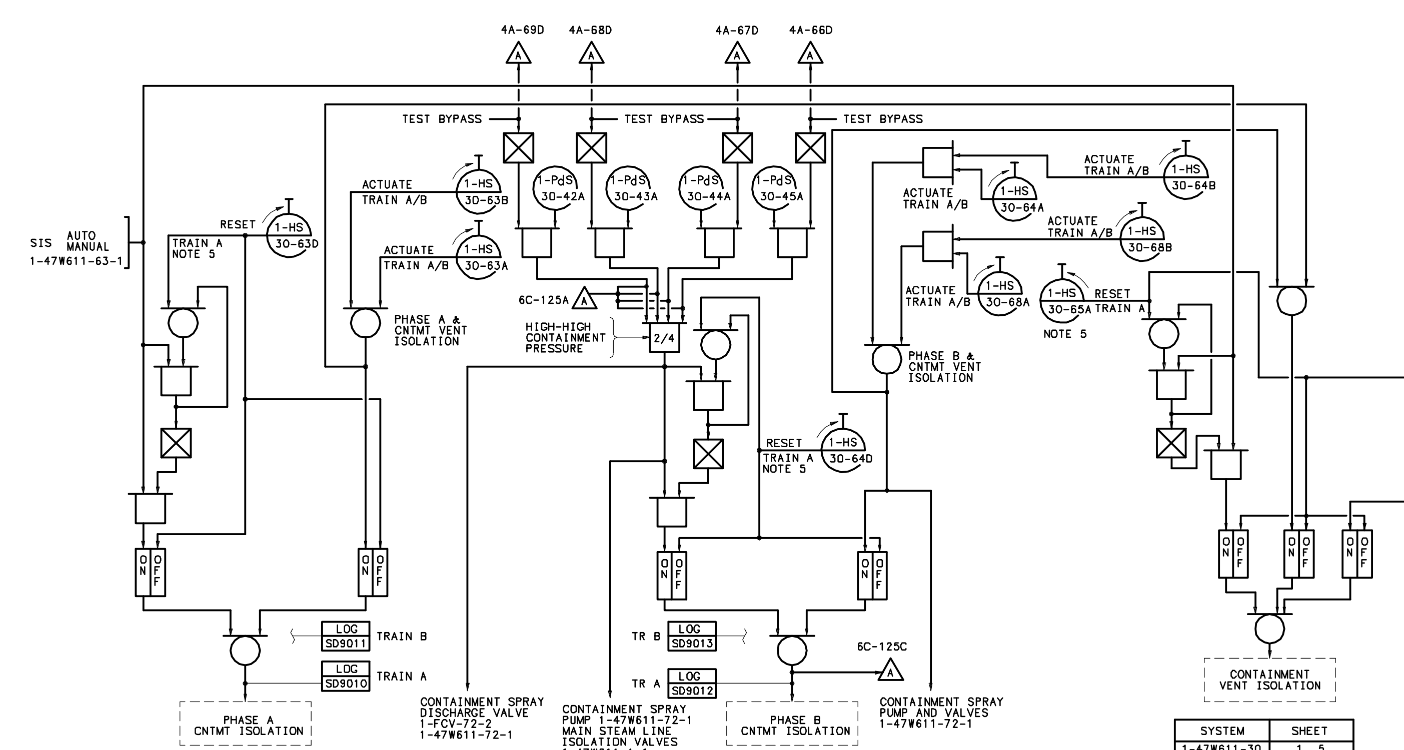


WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

Schematic Diagram
of Leakage Paths

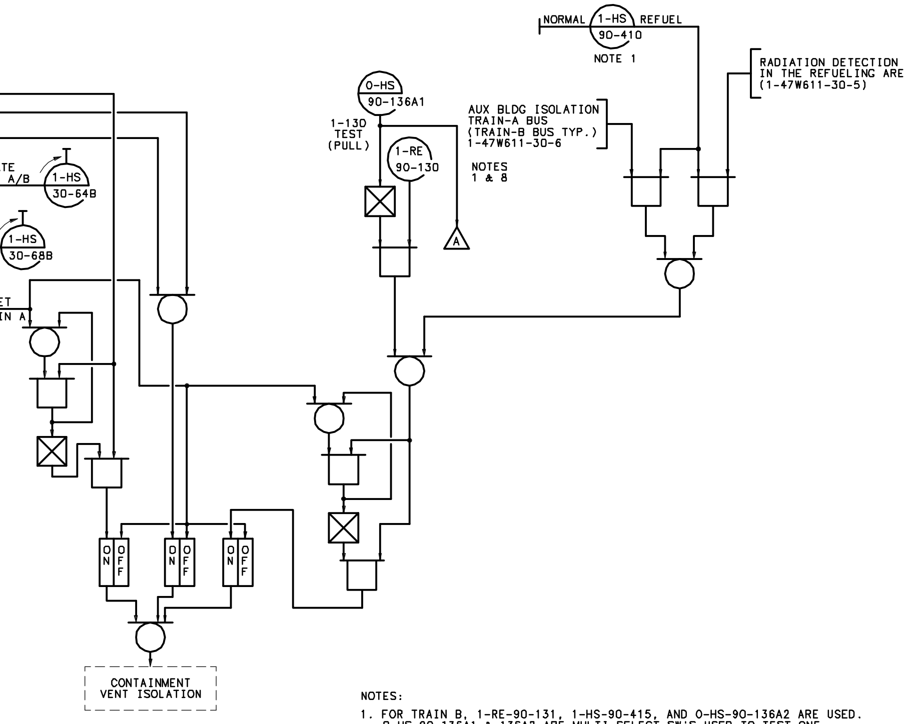
Figure 6.2.4-20

CAD MAINTAINED DRAWING

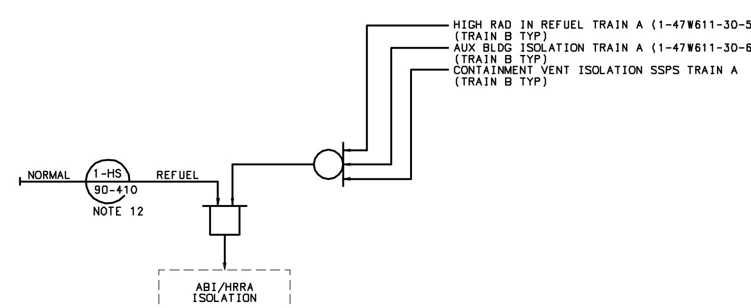


SYSTEM	SHEET
1-47W611-30	1, 6
1-47W611-61	2
1-47W611-62	1
1-47W611-63	3, 5, 6, 8
1-47W611-65	1, 2, 3
1-47W611-68	1
1-47W611-70	3
1-47W611-77	1, 5
1-47W611-81	1
1-47W611-1	3
1-47W611-31	7
1-47W611-88	1
1-47W611-26	3

SYSTEM	SHEET
1-47W611-30	3, 4
1-47W611-70	3
1-47W611-32	2
1-47W611-67	3



SYSTEM	SHEET
1-47W611-30	1, 5
1-47W611-88	1

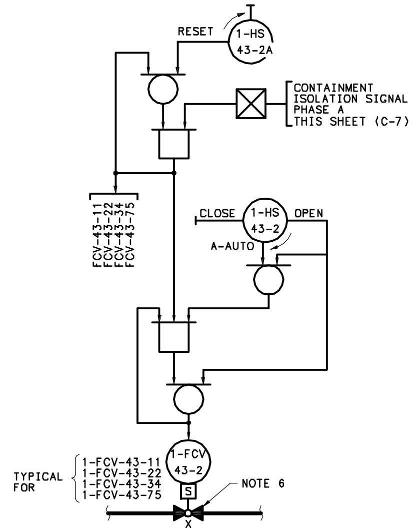


SYSTEM	SHEET
1-47W611-30	1

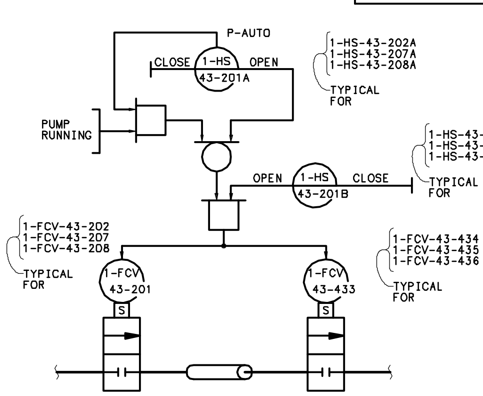
NOTES:

1. FOR TRAIN B, 1-RE-90-131, 1-HS-90-415, AND 0-HS-90-136A2 ARE USED. 0-HS-90-136A1 & 136A2 ARE MULTI SELECT SW'S USED TO TEST ONE CHANNEL AT A TIME.
3. ALL ISOLATION SIGNALS AND CIRCUITRY THAT ARE PART OF ENGINEERED SAFEGUARDS ARE REDUNDANT.
4. NOTE DELETED.
5. EACH TRAIN A RESET HAS AN ASSOCIATED TRAIN B RESET. SEE TVA DRAWING 1-47W610-30-1 FOR TRAIN B "RESET" HANDSWITCH NUMBERS.
6. VALVES 1-FCV-43-22 AND 1-FCV-43-23 MAY BE NORMALLY OPEN WITH 1-FSV-43-23 ENERGIZED TO ALLOW CONTINUOUS RCS FLOW DUE TO THERMAL CYCLING CONCERNS (REF. EDC-51134).
8. 1-RE-90-130 & 131, WHICH HOUSE GAS RADIATION DETECTORS, MONITOR THE CONTAINMENT PURGE EXHAUST.
9. DIGITAL AND ANALOG LOGIC SYMBOLS ARE USED ON LOGIC DIAGRAMS TO FUNCTIONALLY DESCRIBE THE PROCESS CONTROL. REFER TO THE ASSOCIATED WIRING SCHEMATIC FOR THE ELECTRICAL COMPONENTS USED TO IMPLEMENT THE CONTROL SCHEME.
11. FOR SYMBOLS SEE INSTRUMENTION &IDENTIFICATION STANDARDS, LATEST ISSUE.
12. FOR TRAIN B, 1-HS-90-415 IS USED.

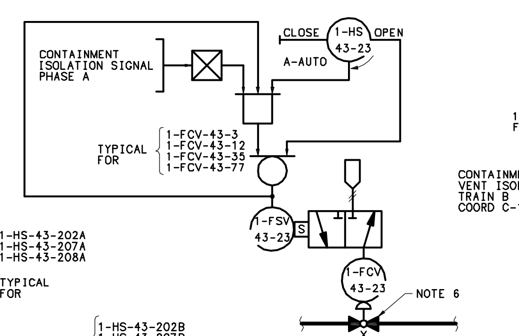
REFERENCE DRAWINGS:
1-47W611-0-1 ---------- LOGIC INDEX & SYMBOLS
47B601-SERIES ------- INSTRUMENT TABULATION
1-45W600-43-2 --------- SAMPLING AND WATER QUALITY SYSTEM SCHEMATIC DIAGRAM
1-45W600-90-1 --------- RADIATION MONITORING SYSTEM SCHEMATIC DIAGRAM
1-45W600-57-8 --------- SEPARATION & MISC AUX RELAYS SCHEMATIC DIAGRAMS
1-47W610-30-1 THRU 6 -- CONTAINMENT VENTILATION SYSTEM
1-47W610-90-1 THRU 3 -- RADIATION MONITORING SYSTEM CONTROL DIAGRAM
1-47W610-43-3, -5, -6 - SAMPLING & WATER QUALITY SYSTEM CONTROL DIAGRAM
1-47W611-1-1 ---------- MAIN STEAM SYSTEM LOGIC DIAGRAM
1-47W611-63-1 --------- SAFETY INJECTION SYSTEM LOGIC DIAGRAM
1-47W611-72-1 --------- CONTAINMENT SPRAY SYSTEM LOGIC DIAGRAM



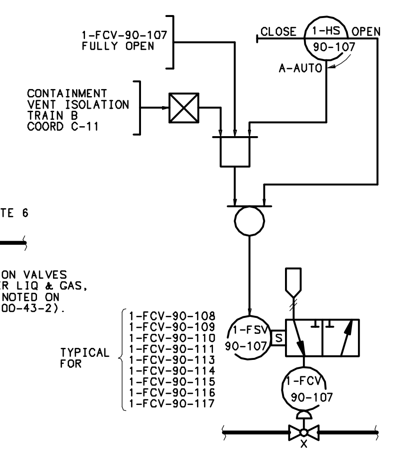
TYPICAL SAMPLING SYSTEM ISOLATION VALVES FOR REACTOR COOLANT, PRESSURIZER LIQ & GAS, AND ACCUM TANKS. (PWR TRAIN AS NOTED ON 1-47W610-43-1, -3 & -5 & 1-45W600-43-2).



CONTAINMENT H_2 SAMPLE SYSTEM (PWR TRAIN AS NOTED ON 1-47W610-43-6 & 1-45W600-43-2)



TYPICAL SAMPLING SYSTEM ISOLATION VALVES FOR REACTOR COOLANT, PRESSURIZER LIQ & GAS, AND ACCUM TANKS. (PWR TRAIN AS NOTED ON 1-47W610-43-1, -3 & -5 & 1-45W600-43-2).



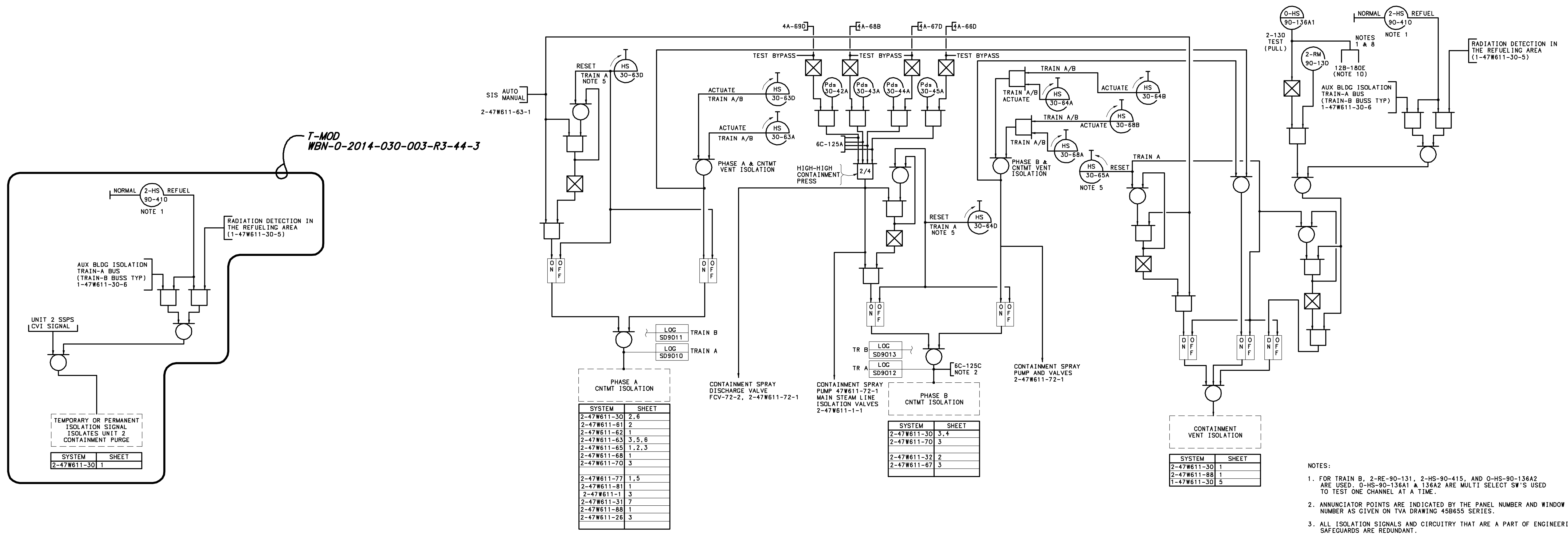
TYPICAL CONTAINMENT BLDG UPPER & LWR COMPARTMENT AIR MONITOR ISOL VLV (PWR TRAIN AS NOTED ON 1-47W610-90-3 & 1-45W600-90-1)

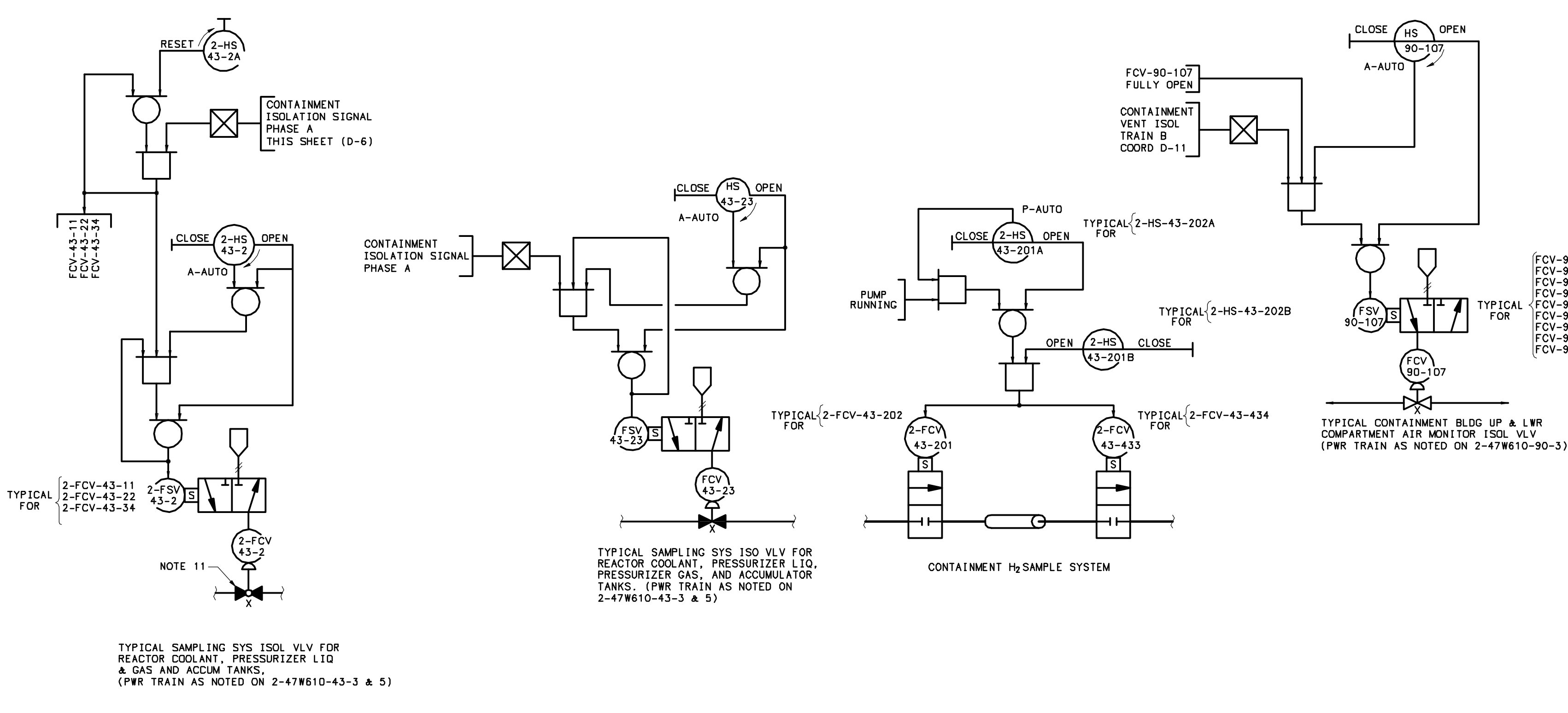
UFSAR AMENDMENT 1

WATTS BAR
FINAL SAFETY
ANALYSIS REPORT

POWERHOUSE
UNIT 1
ELECTRICAL
LOGIC DIAGRAM
CONTAINMENT ISOLATION
TVA DWG NO. 1-47W611-88-1 R28
FIGURE 6.2.4-21

CAD MAINTAINED DRAWING





NOTES:

1. FOR TRAIN B, 2-RE-90-131, 2-HS-90-415, AND O-HS-90-136A2 ARE USED. O-HS-90-136A1 & 136A2 ARE MULTI SELECT SW'S USED TO TEST ONE CHANNEL AT A TIME.
2. ANNUNCIATOR POINTS ARE INDICATED BY THE PANEL NUMBER AND WINDOW NUMBER AS GIVEN ON TVA DRAWING 45B655 SERIES.
3. ALL ISOLATION SIGNALS AND CIRCUITRY THAT ARE A PART OF ENGINEERING SAFEGUARDS ARE REDUNDANT.
4. UNIT 2 CONTROL AIR ISOLATION VALVES ARE TAGGED FCV-32-81,103,111.
5. ASSOCIATED WITH EACH TRAIN A RESET EXISTS A TRAIN B RESET. SEE TVA DRAWING 47W610-30-1 FOR TRAIN B "RESET" HANDSWITCH NUMBERS.
8. RE-90-130 & 131, WHICH HOUSE GAS RADIATION DETECTORS, MONITOR THE CONTAINMENT PURGE EXHAUST.
9. "DIGITAL AND ANALOG LOGIC SYMBOLS ARE USED ON LOGIC DIAGRAMS TO FUNCTIONALLY DESCRIBE THE PROCESS CONTROL. REFER TO THE ASSOCIATED WIRING SCHEMATIC FOR THE ELECTRICAL COMPONENTS USED TO IMPLEMENT THE CONTROL SCHEME."
10. ANNUNCIATOR WINDOW FOR HS-90-136A2 IS 12B-180E.
11. VALVES 2-FCV-43-22 AND 2-FCV-43-23 MAY BE NORMALLY OPEN WITH 2-FSV-43-22 ENERGIZED TO ALLOW CONTINUOUS RCS DUE TO THERMAL CYCLING CONCERNS.

REFERENCES:
2-47W610-90-1 THRU 3 RADIATION MONITORING SYSTEM FSAR 11.4.
2-47W610-30-1 THRU 6 CONTAINMENT VENTILATION SYSTEM.

UFSAR AMENDMENT 1

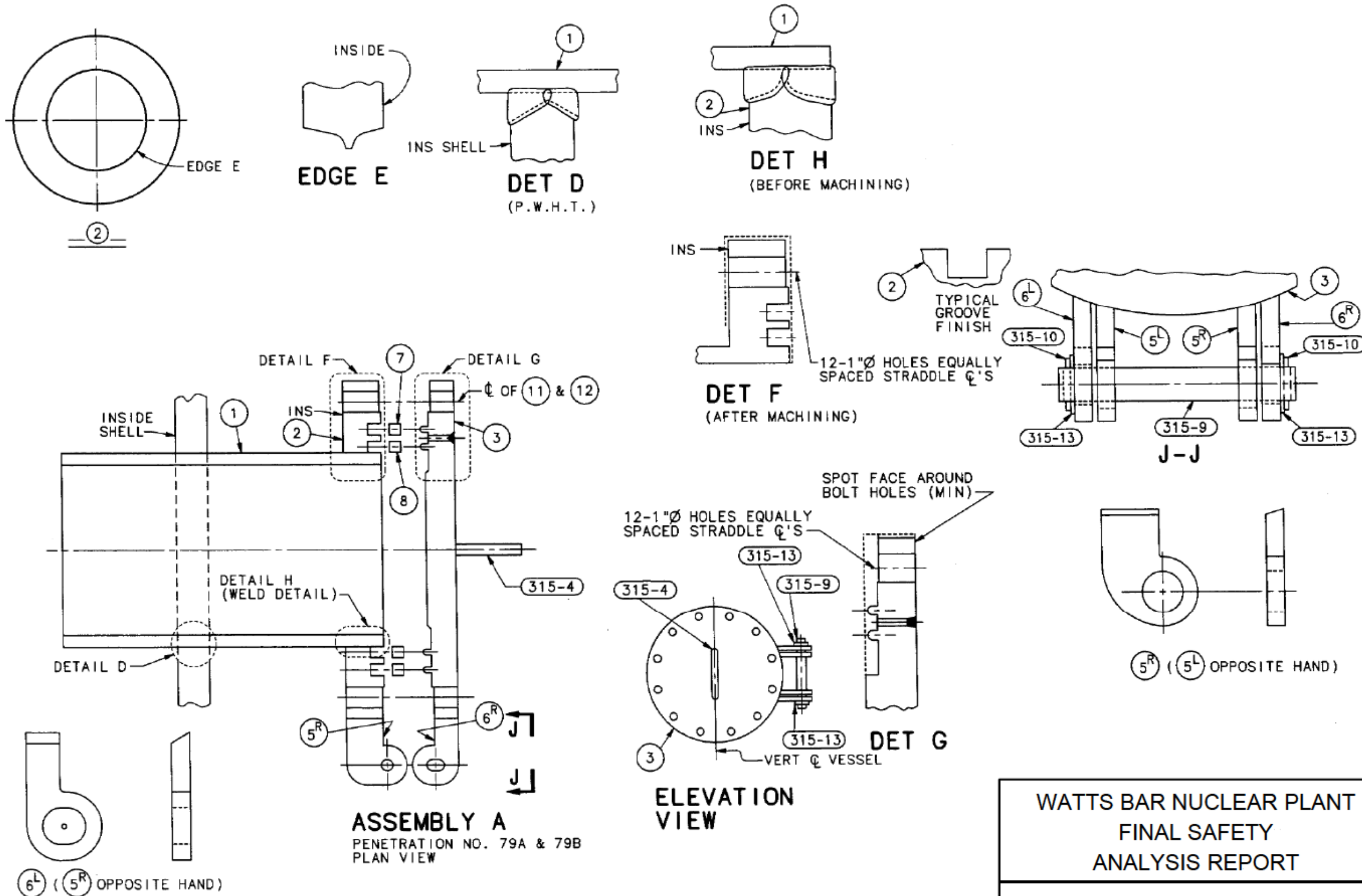
WATTS BAR
FINAL SAFETY
ANALYSIS REPORT

POWERHOUSE
UNIT 2
ELECTRICAL
LOGIC DIAGRAM
CONTAINMENT ISOLATION
TVA DWG NO. 2-47W611-88-1 R10
FIGURE 6.2.4-21(U2)

FIGURE 6.2.4-22A THRU

6.2.4-22-II

DELETED



WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

Ice Blowing and Negative
Return Lines-Blind Flange
Details
FIGURE 6.2.4-23

6.2.5 COMBUSTIBLE GAS CONTROL IN CONTAINMENT

The containment combustible gas control system is designed to control the concentration of hydrogen that may be released into the containment following a beyond-design-basis accident to ensure that containment structural integrity is maintained. The combustible gas control system consists of the containment air return system, the hydrogen analyzer system (HAS) and the hydrogen mitigation system (HMS) which conform to 10 CFR 50.44 requirements.

6.2.5.1 Design Bases

In an accident more severe than the design-basis loss-of-coolant accident, combustible gas is predominantly generated within containment as a result of the following:

1. Fuel clad-coolant reaction between the fuel cladding and the reactor coolant.

2. Molten core-concrete interaction in a severe core melt sequence with a failed reactor vessel.

If a sufficient amount of combustible gas is generated, it may react with the oxygen present in the containment at a rate rapid enough to lead to a containment breach or a leakage rate in excess of Technical Specification limits. Additionally, damage to systems and components essential to continued control of the post-accident conditions could occur.

The systems provided for combustible gas control have the following functional and mechanical requirements:

1. The air return fans enhance the ice condenser and containment spray heat removal operation by circulating air from the upper compartment to the lower compartment through the ice condenser, and then back to the upper compartment. Hydrogen concentration is limited in potentially stagnant regions by providing air flow in these regions.

2. The HAS provides the capability for extracting a sample and obtaining the measurement necessary to determine the volume percent concentration for hydrogen present in the sample. The system provides indication and alarms of volume percent concentrations in the main control room. Indication is also provided at the remote control center.

3. The HMS is designed to increase the containment capability to accommodate hydrogen that could be released during a degraged core accident. The system is based on the concept of controlled ignition using thermal igniters.

4. The air return fans and HAS are designed to operate continuously during accident conditions. The HMS igniter assemblies are qualified for a 30 year life of operational cycles.

5. The combustible gas control system is designed for periodic testing and inspection.

6.2.5.2 System Design

Containment Air Return System

Mixing of the containment atmosphere is accomplished by the containment air return system described in Section 6.8. The air return fans start automatically 9 + 1 minutes after receipt of a containment Phase B isolation signal (see 7.3.1.1.1). In addition, the fans may be started manually.

The associated ductwork, which must remain intact following a beyond-design-basis accident to assure that no localized hydrogen concentration exceeds 4%, consists of (1) two 12-inch ducts (one associated with each air return fan intake) which draw air from the containment dome region, (2) one 8-inch duct which circles the containment removing air from accumulator rooms and other dead-ended spaces and terminates at each air return fan housing, (3) two 12-inch ducts which circle the crane wall, removing air from the steam generator and pressurizer compartments and terminate through two 8-inch ducts at each air return fan housing, (4) two 8-inch pipes (one connected to each air return fan housing) which remove air from the refueling canal, and (5) the main duct between upper and lower compartment through the divider deck, including the non-return dampers.

The ductwork described above is embedded in concrete, where possible, to prevent damage from buildup of pressure during a beyond-design-basis accident. Ductwork not protected by embedment is also designed to withstand the beyond-design-basis accident environment. The air return system also includes heavy-duty backdraft dampers to prevent back flow from the lower compartment to the upper compartment under a differential pressure of 15 psig. These dampers prevent steam from bypassing the ice condenser during the initial blowdown. Figures 6.2.5-3, 6.2.5-4 and 6.2.5-5 are provided to show the routing of the recirculation ducts.

During the post-blowdown period, the pressure gradient between containment compartments is almost nonexistent and hydrogen can accumulate in potentially stagnant regions. The regions of concern are the ten dead-ended compartments: the four steam generator enclosures; the pressurizer enclosure; the four accumulator spaces; and the instrument room. The air return fans provide recirculation flow through the dead-ended compartments to prevent excessive hydrogen buildup. Each fan will mix 1,960 cfm form the enclosed areas in the lower compartment to the general lower compartment atmosphere.

Hydrogen Analyzer

The HAS provides the capability to extract air samples from containment and to determine volume percent concentration of hydrogen. The primary functions of the system include continuous sampling from a remote location, measurement of hydrogen partial pressure, visual indication of volume percent concentrations and hydrogen concentration alarms in the main control room.

The sampling system consists of a single, non-trained detection loop. The analyzer is fed by one process sample line and returns to containment on one process effluent line. This line is equipped with two manually controlled isolation valves on both the sample and return lines. The system is installed at two locations. The sample loop station, mounted inside the Annulus, consists of the detection chamber, pressure transducer, solenoid valves, condensate management components and check valves for directing sample through the loop.

The HAS remote control center is mounted is a mild environment in the Auxiliary Building where access is always permitted. The major components of the center include a PLC based signal conditioner/system controller, a remote touch screen display and the system switch panel.

Containment isolation valve hand switches, a fan control hand switch, a hydrogen concentration indicator and alarms are provided in the main control room.

Upon actuation of the system, sample enters the loop assembly and passes through a “T” which separates the liquid and collects it in a condensate trap. The sample continues through a flow switch into the detection chamber. Inside the detection chamber, partial and total pressure measurements are obtained. The sample exits the detection chamber, passing through the sample pump which then drives the sample out of the loop assembly.

The analyzer is designed to continually measure hydrogen concentration following a beyond-design-basis accident. The analyzer is calibrated to measure hydrogen concentrations between zero and ten percent with an accuracy of +0.2% hydrogen. A sample loop flow diagram is shown in Figure 6.2.5-6.

The hydrogen analyzer components are seismically supported.

Hydrogen Mitigation System

To assure that any hydrogen release would be ignited at containment locations as soon as the concentration exceeded the lower flammability limit, durable thermal igniters, capable of maintaining an adequate surface temperature, are used. An igniter developed by Tayco Engineering, operating at a nominal plant voltage of 120V AC, is used. The igniter has been shown by experiment to be capable of maintaining surface temperatures in excess of the required minimum for extended periods, initiating combustion and continuing to operate in various combustion environments.

The igniters in the HMS are equally divided into two redundant groups, each with independent and separate controls, power supplies and locations, to ensure adequate coverage even in the event of a single failure. Manual control of each group of igniters is provided in the main control room and the status (on-off) of each group is indicated there. A separate train of Class IE 480V AC auxiliary power is provided for each group of igniters and is backed by automatic loading onto the diesel generators upon loss of offsite power. Each individual circuit powers two igniters. See Figures 6.2.5-8 through 6.2.5-12 for igniter locations.

To assure adequate spatial coverage, 68 igniters are distributed throughout the various regions of the containment in which hydrogen could be released or to which it could flow in significant quantities (see Figures 6.2.5-8 through 6.2.5-12). There are at least two igniters, controlled and powered redundantly located in each of these regions. Following a degraded core accident, any hydrogen which is produced is released into the lower compartment inside the crane wall. To cover this region, 22 igniters (equally divided between trains) are provided. Eight of these are distributed on the reactor cavity wall exterior and crane wall interior at an intermediate elevation to ensure the partial burning that accompanies upward flame propagation.

Two igniters are located at the lower edge of each of the five enclosures for the four steam generators and the pressurizer, two in the top of the pressurizer enclosure and another pair above the reactor vessel in the cavity. These 22 lower compartment igniters help prevent flammable mixtures from entering the ice condenser. Any hydrogen not burned in the lower compartment is carried up through the ice condenser and into its upper plenum. Since steam is removed from the mixture as it is passed through the ice bed, mixtures that were nonflammable in the lower compartment tend to be flammable in the ice condenser upper plenum. This phenomenon is supported by the CLASIX containment analysis code which predicts more sequential burns to occur in the upper plenum than in any other region. Four igniters are

located around the upper compartment dome, four at intermediate elevations on the outside of the steam generator enclosures, four more around the top inside of the crane wall and on above each of the two air return fans. The air return fans provide recirculation flow from the upper compartment through several dead-ended compartments (see Section 6.2.1.3.3) back into the main part of the lower compartment. To cover this region, there are pairs of igniters in each of the eight rooms (a total of 16 igniters) through which the recirculation flow passes. The location of the HMS igniters is shown in Figures 6.2.5-8 through 6.2.5-12.

The components of the HMS inside containment are seismically supported.

6.2.5.3 Design Evaluation

Containment pressure during the post blowdown phase of a loss-of-coolant accident is calculated with the LOTIC code which models the containment structural heat sinks and containment safeguards system. The long-term containment pressure analysis accounts for hydrogen partial pressure. This transient is discussed in Section 6.2.1 and 15.4.1.2.

The containment air return system is designed to reduce containment pressure after blowdown to prevent excessive hydrogen concentrations in pocketed areas, and circulate air through the ice condenser. The air return fans automatically start on a Phase B containment isolation signal and can be started manually. The fans provide a continuous mixing of the containment compartment atmosphere for the long-term post–blowdown environment. The system has redundancy, is single-failure-proof and will remain operable with a loss of onsite or offsite power.

The hydrogen analyzer is a highly reliable commercial grade, Category 3 instrument as defined in Regulatory Guide 1.97 and permitted by Regulatory Guide 1.7. The system is capable of being energized and fully operational within 90 minutes. It is required to operate correctly and continuously following a beyond-design-basis accident.

The HMS, due to its igniter type and locations, redundancy, capability of functioning in a post-accident environment, seismic support, main control room actuation, and remote surveillance, performs its intended function in a manner that provides adequate safety margins. The Unit 2 containment structures can survive the effects of credible degraded core accidents when hydrogen hazards are mitigated by HMS.

6.2.5.4 Testing and Inspections

The combustible gas control system is subjected to periodic testing and inspection to demonstrate its availability.

The periodic test program for the containment air return system and the HMS are described in the Technical Specifications.

The periodic test program for the HAS is described in the Technical Requirements Manual.

Preoperational tests are described in Chapter 14.

6.2.5.5 Instrumentation Application

The instrumentation design details of the air return fans are shown on Figures 9.4-30 and 9.4-33. The logic, controls and instrumentation of this engineered safety feature system are such

that a single failure of any component does not result in the loss of functional capability for the system.

The HAS instrument range is configured to measure from 0-10% hydrogen by volume. There are two configurable hydrogen concentration alarms, one on the remote control center and one in the main control room. The HAS also provides a trouble alarm to indicate failures of the sensor and pump. An equipment status display monitors pump pressure and flow.

Annunciation is provided in the main control room upon loss of power or undervoltage to the igniters. Each of the two HMS groups is placed in service from the main control room by a handswitch. The igniters are manually energized following any accident which indicates inadequate core cooling. This is done without waiting for a potential hydrogen buildup.

6.2.5A Hydrogen Mitigation System

6.2.5A.1 Design Basis

The hydrogen mitigation system (HMS) is designed to increase the containment capability to accommodate hydrogen that could be released during a degraded core accident. This system, which is based on the concept of controlled ignition using thermal igniters, has been designed to be redundant, capable of functioning in a post accident environment, seismically supported, and capable of actuation from the main control room. In addition, the system is designed to have an ample number of igniters distributed throughout the containment to mitigate the effects of hydrogen releases in containment.

6.2.5A.2 System Description

To assure that any hydrogen released would be ignited at different containment locations as soon as the concentration exceeded the lower flammability limit, durable thermal igniters capable of maintaining an adequate surface temperature are used. An igniter developed by Tayco Engineering, operating at a nominal plant voltage of 120V ac, is used. The igniter has been shown by experiment to be capable of maintaining surface temperatures in excess of the required minimum for extended periods, initiating combustion, and continuing to operate in various combustion environments.

The igniters in the HMS are equally divided into two redundant groups, each with independent and separate controls, power supplies, and locations, to ensure adequate coverage even in the event of a single failure. Manual control of each group of igniters is provided in the main control room and the status (on-off) of each group is indicated there. A separate train of Class 1E 480V ac auxiliary power is provided for each group of igniters and is backed by automatic loading onto the diesel generators upon loss of offsite power. Each individual circuit powers two igniters.

To assure adequate spatial coverage, 68 igniters are distributed throughout the various regions of the containment in which hydrogen could be released or to which it could flow in significant quantities (see Figure 6.2.5A-1 through 6.2.5A-5). There are at least two igniters, controlled and powered redundantly located in each of these regions. Following a degraded core accident, any hydrogen which is produced is released into the lower compartment inside the crane wall. To cover this region, 22 igniters (equally divided between trains) are provided. Eight of these are distributed on the reactor cavity wall exterior and crane wall interior at an intermediate elevation to ensure the partial burning that accompanies upward flame propagation.

Two igniters are located at the lower edge of each of the five enclosures for the four steam generators and the pressurizer, two in the top of the pressurizer enclosure, and another pair above the reactor vessel in the cavity. These 22 lower compartment igniters help prevent flammable mixtures from entering the ice condenser. Any hydrogen not burned in the lower compartment is carried up through the ice condenser and into its upper plenum. Since steam is removed from the mixture as it is passed through the ice bed, mixtures that were nonflammable in the lower compartment tend to be flammable in the ice condenser upper plenum. This phenomenon is supported by the CLASIX containment analysis code which predicts more sequential burns to occur in the upper plenum than in any other region. Therefore, the system is designed to take advantage of the favorable combustion characteristics of the upper plenum by the provision of 16 igniters equally spaced around it. Four igniters are located around the upper compartment dome, four at intermediate elevations on the outside of the steam generator enclosures, four more around the top inside of the crane wall, and one above each of the two air return fans. The air return fans provide recirculation flow from the upper compartment through several dead-ended compartments (see Section 6.2.1.3.3) back into the main part of the lower compartment. To cover this region, there are pairs of igniters in each of the eight rooms (a total of 16 igniters) through which the recirculation flow passes. The location of the HMS igniters is shown in Figures 6.2.5A-1 through 6.2.5A-5.

The components of the HMS inside containment are seismically supported.

6.2.5A.3 Operation

The HMS is energized manually from the main control room following any accident upon the occurrence of any condition which indicates inadequate core cooling. This is done without waiting for a potential hydrogen buildup.

6.2.5A.4 Safety Evaluation

The HMS due to its igniter type and locations, redundancy, capability of functioning in a post accident environment, seismic support, main control room actuation, and remote surveillance, performs its intended function in a manner that provides adequate safety margins. The containment structures can survive the effects of credible degraded core accidents when hydrogen hazards are mitigated by HMS.

6.2.5A.5 Testing

Surveillance testing for the HMS consists of energizing the system from the main control room and taking voltage and current readings from each circuit at the distribution panels located in the Auxiliary Building. These readings are then compared to acceptance criteria developed from testing at Watts Bar Nuclear Plant and TVA's central laboratory facility to determine whether or not both igniters on each circuit are operational. This form of testing does not require containment entry. The operability of at least 33 of the 34 igniters per train would conservatively guarantee an effective coverage throughout the containment.

TABLE 6.2.5-1

UNIT 1

DELETED

Table 6.2.5-2 (Sheet 1 of 9)
UNIT 1 ONLY

COMBUSTIBLE GAS CONTROL SYSTEM

FAILURE MODE AND EFFECTS ANALYSIS

No.	Component Identification	Function	Failure Mode	Potential Cause	Method of Failure Detection	Effect on System	Effect on Plant	Remarks
1.	Emergency power to Train A	Provide power to hydrogen analyzer	Power Train A fails	Loss of power at 480V Reactor MOV Board 1A2-A	Power indicating light locally or annunciation	Loss of monitoring capability	None-redundant subsystem	Only one train is required to monitor hydrogen concentration
2.	Hydrogen analyzer system 1-H_2AN-43-200 (Train A)	To detect and provide remote continuous indication of presence and concentration of H_2 in the primary containment atmosphere within 30 minutes after a DBA	Fails	See analyzer internal component analyses (Item Nos. 6-14 in this table)				
3.	Emergency power to Train B	Provide power to hydrogen analyzer	Power Train B fails	Loss of power at 480V Reactor MOV Board 1B2-B	Power indicating light locally or annunciation	Loss of monitoring capability	None-redundant subsystem	Only one train is required to monitor hydrogen concentration
4.	Hydrogen analyzer system 1-H_2AN-43-210 (Train B)	To detect and provide remote continuous indication of	Fails	See analyzer internal component analyses				

Table 6.2.5-2 (Sheet 2 of 9)
UNIT 1 ONLY

COMBUSTIBLE GAS CONTROL SYSTEM

FAILURE MODE AND EFFECTS ANALYSIS

No.	Component Identification	Function	Failure Mode	Potential Cause	Method of Failure Detection	Effect on System	Effect on Plant	Remarks
		presence and concentration of H_2 in the primary containment atmosphere within 30 minutes after a DBA		(Item Nos. 6-14 in this table)				
5.	Shut-off valves (inlet and outlet)	Manually isolates hydrogen analyzer subsystem from containment	Closed	Left closed during maintenance	Low cell flow indication in MCR	Loss of monitoring capability	None-redundant system	Manual valve used during maintenance
6.	Sample cooler	Cools sample gas to approximately 150°F	Plugged		Low cell flow indication in MCR	Loss of monitoring capability	None-redundant subsystem	
7.	Pressure regulator (R_1)	Maintains downstream vacuum to provide a constant sample gas pressure for the hydrogen analyzer	Closed	Plugged	Low cell flow indication in MCR	Loss of monitoring capability	None-redundant subsystem	
			Open	Mechanical failure	Conflicting hydrogen concentration indications in	Incorrect hydrogen concentration indications	None-redundant subsystem	Operator action would be required to determine

Table 6.2.5-2 (Sheet 3 of 9)
UNIT 1 ONLY

COMBUSTIBLE GAS CONTROL SYSTEM

FAILURE MODE AND EFFECTS ANALYSIS

No.	Component Identification	Function	Failure Mode	Potential Cause	Method of Failure Detection	Effect on System	Effect on Plant	Remarks
					MCR			which train was indicating correctly
8.	Sample pump	Draws a gas sample from containment and returns sample gas to containment	Broken diaphragm	Mechanical failure or sample cooler failure	Low cell flow indication in MCR	Loss of monitoring capability	None-redundant subsystem	
9.	Pump motor	Drives sample pump	Shorted stator	Fuse blow	Low cell flow indication in MCR	Loss of monitoring capability	None-redundant subsystem	
			Broken shaft	Mechanical failure	Low cell flow indication in MCR	Loss of monitoring capability	None-redundant subsystem	
10.	Moisture separator	Provide a dry gas sample to the hydrogen analyzer	Plugged inlet	Mechanical failure	Low cell flow indication in MCR	Loss of monitoring capability	None-redundant subsystem	
			Plugged gas outlet	Mechanical failure	Low cell flow indication in MCR	Loss of monitoring capability	None-redundant subsystem	
			Plugged moisture outlet	Mechanical failure	Low cell flow indication in MCR	Loss of monitoring capability	None-redundant subsystem	Analyzer would be flooded

Table 6.2.5-2 (Sheet 4 of 9)
UNIT 1 ONLY

COMBUSTIBLE GAS CONTROL SYSTEM

FAILURE MODE AND EFFECTS ANALYSIS

No.	Component Identification	Function	Failure Mode	Potential Cause	Method of Failure Detection	Effect on System	Effect on Plant	Remarks
11.	Pressure regulator (R_3)	Regulates cell flow in conjunction with fixed orifice	Closed	Plugged	Low cell flow indication in MCR	Loss of monitoring capability	None-redundant subsystem	
			Open	Mechanical failure	Conflicting hydrogen concentration indications in MCR	Incorrect hydrogen concentration indications	None-redundant subsystem	Operator action would be required to determine which train was indicating correctly
12.	Fixed orifice	Regulates cell in conjunction with fixed orifice	Closed	Plugged	Low cell flow indication in MCR	Loss of monitoring capability	None-redundant subsystem	
13.	Hydrogen analyzer	Determines percent of hydrogen in the sample gas	Closed	Plugged	Low cell flow indication in MCR	Loss of monitoring capability	None-redundant subsystem	
14.	Pressure regulator (R_2)	Maintains moisture separator condensate downstream vacuum	Closed	Plugged	Low cell flow indication in MCR	Loss of monitoring capability	None-redundant subsystem	Analyzer would be flooded
			Open	Mechanical failure	Low cell flow indication in	Loss of monitoring	None-redundant	

Table 6.2.5-2 (Sheet 5 of 9)
UNIT 1 ONLY

COMBUSTIBLE GAS CONTROL SYSTEM

FAILURE MODE AND EFFECTS ANALYSIS

No.	Component Identification	Function	Failure Mode	Potential Cause	Method of Failure Detection	Effect on System	Effect on Plant	Remarks
					MCR	capability	subsystem	
15.	FCV-43-201-A	Hydrogen sample line inboard containment isolation valve	Closed	Mechanical failure or loss of DC power or loss of hydrogen analyzer pump	Valve position indication via switch HS-43-201A HS-43-201B	Loss of monitoring capability	None-redundant subsystem	
16.	FCV-43-433-A	Hydrogen sample line outboard containment isolation valve	Closed	Mechanical failure or loss of DC power or loss of hydrogen analyzer pump	Valve position indication via switch HS-43-201A HS-43-201B	Loss of monitoring capability	None-redundant subsystem	
17.	FCV-43-202-A	Hydrogen sample line inboard containment isolation valve	Closed	Mechanical failure or loss of DC power or loss of hydrogen analyzer pump	Valve position indication via switch HS-43-202A HS-43-202B	Loss of monitoring capability	None-redundant subsystem	
18.	FCV-43-434-A	Hydrogen sample line outboard containment isolation valve	Closed	Mechanical failure or loss of DC power or loss of hydrogen	Valve position indication via switch HS-43-202A	Loss of monitoring capability	None-redundant subsystem	

Table 6.2.5-2 (Sheet 6 of 9)
UNIT 1 ONLY

COMBUSTIBLE GAS CONTROL SYSTEM

FAILURE MODE AND EFFECTS ANALYSIS

No.	Component Identification	Function	Failure Mode	Potential Cause	Method of Failure Detection	Effect on System	Effect on Plant	Remarks
				analyzer pump	HS-43-202B			
19.	Sample line Train A heat tracing	To maintain sample line temperature outside containment	Fails	Loss of power	Alarm “SI heat trace PNL A1/A2 abnormal” or indicating light "Hydrogen analyzer Heat Trace System A abnormal temperature”	Loss of monitoring capability	None-redundant subsystem	
20.	FCV-43-207-B	Hydrogen sample line inboard containment isolation valve	Closed	Mechanical failure or loss of DC power or loss of hydrogen analyzer pump	Valve position indication via switch HS-43-207A HS-43-207B	Loss of monitoring capability	None-redundant subsystem	
21.	FCV-43-435-B	Hydrogen sample line outboard containment isolation valve	Closed	Mechanical failure or loss of DC power or loss of hydrogen analyzer pump	Valve position indication via switch HS-43-207A HS-43-207B	Loss of monitoring capability	None-redundant subsystem	

Table 6.2.5-2 (Sheet 7 of 9)
UNIT 1 ONLY

COMBUSTIBLE GAS CONTROL SYSTEM

FAILURE MODE AND EFFECTS ANALYSIS

No.	Component Identification	Function	Failure Mode	Potential Cause	Method of Failure Detection	Effect on System	Effect on Plant	Remarks
22.	FCV-43-436-B	Hydrogen sample line outboard containment isolation valve	Closed	Mechanical failure or loss of DC power or loss of hydrogen analyzer pump	Valve position indication via switch HS-43-208A HS-43-208B	Loss of monitoring capability	None-redundant subsystem	
23.	FCV-43-208-B	Hydrogen sample line inboard containment isolation valve	Closed	Mechanical failure or loss of DC power or loss of hydrogen analyzer pump	Valve position indication via switch HS-43-208A HS-43-208B	Loss of monitoring capability	None-redundant subsystem	
24.	Sample line Train B heat tracing	To maintain sample line temperature outside containment	Fails	Loss of power	Alarm “SI heat trace PNL B1/B2 abnormal” or indicating light "Hydrogen analyzer Heat Trace System B abnormal temperature”	Loss of monitoring capability	None-redundant subsystem	
25.	Emergency power to	Provide power to	Fails	Loss of	Indicating	HMS function	None-	Only one

Table 6.2.5-2 (Sheet 8 of 9)
UNIT 1 ONLY

COMBUSTIBLE GAS CONTROL SYSTEM

FAILURE MODE AND EFFECTS ANALYSIS

No.	Component Identification	Function	Failure Mode	Potential Cause	Method of Failure Detection	Effect on System	Effect on Plant	Remarks
	Train A igniters	HMS Train A		power at 480V Cont and Aux Bldg Vent Board 1A-A	light in control room	to mitigate hydrogen lost	redundant subsystem	train is required to mitigate hydrogen if degraded core condition occurs. System is not required immediately and activated manually by the operator
26.	Igniters Train A (each train is made up of 34 igniters)	To ignite hydrogen concentration level of 5% to 8% by volume	Fails	Fuse blow	Indicating light	HMS function to mitigate hydrogen lost	None-redundant subsystem	
				Distribution panel breaker trips	Annunciator	HMS function to mitigate hydrogen lost	None-redundant subsystem	
27.	Emergency power to Train B igniters	Provide power to HMS Train B	Fails	Loss of power at 480V Cont and Aux Bldg Vent Board	Indicating light in control room	HMS function to mitigate hydrogen lost	None-redundant subsystem	Only one train is required to mitigate hydrogen if

Table 6.2.5-2 (Sheet 9 of 9)
UNIT 1 ONLY

COMBUSTIBLE GAS CONTROL SYSTEM

FAILURE MODE AND EFFECTS ANALYSIS

No.	Component Identification	Function	Failure Mode	Potential Cause	Method of Failure Detection	Effect on System	Effect on Plant	Remarks
				1B-B				degraded core condition occurs. System is not required immediately and activated manually by the operator
28.	Igniters Train B (each train is made up of 34 igniters)	To ignite hydrogen concentration level of 5% to 8% by volume	Fails	Fuse blow	Indicating light	HMS function to mitigate hydrogen lost	None-redundant subsystem	
				Distribution panel breaker trips	Annunciator	HMS function to mitigate hydrogen lost	None-redundant subsystem	

FIGURE 6.2.5-1

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FIGURE 6.2.5-2

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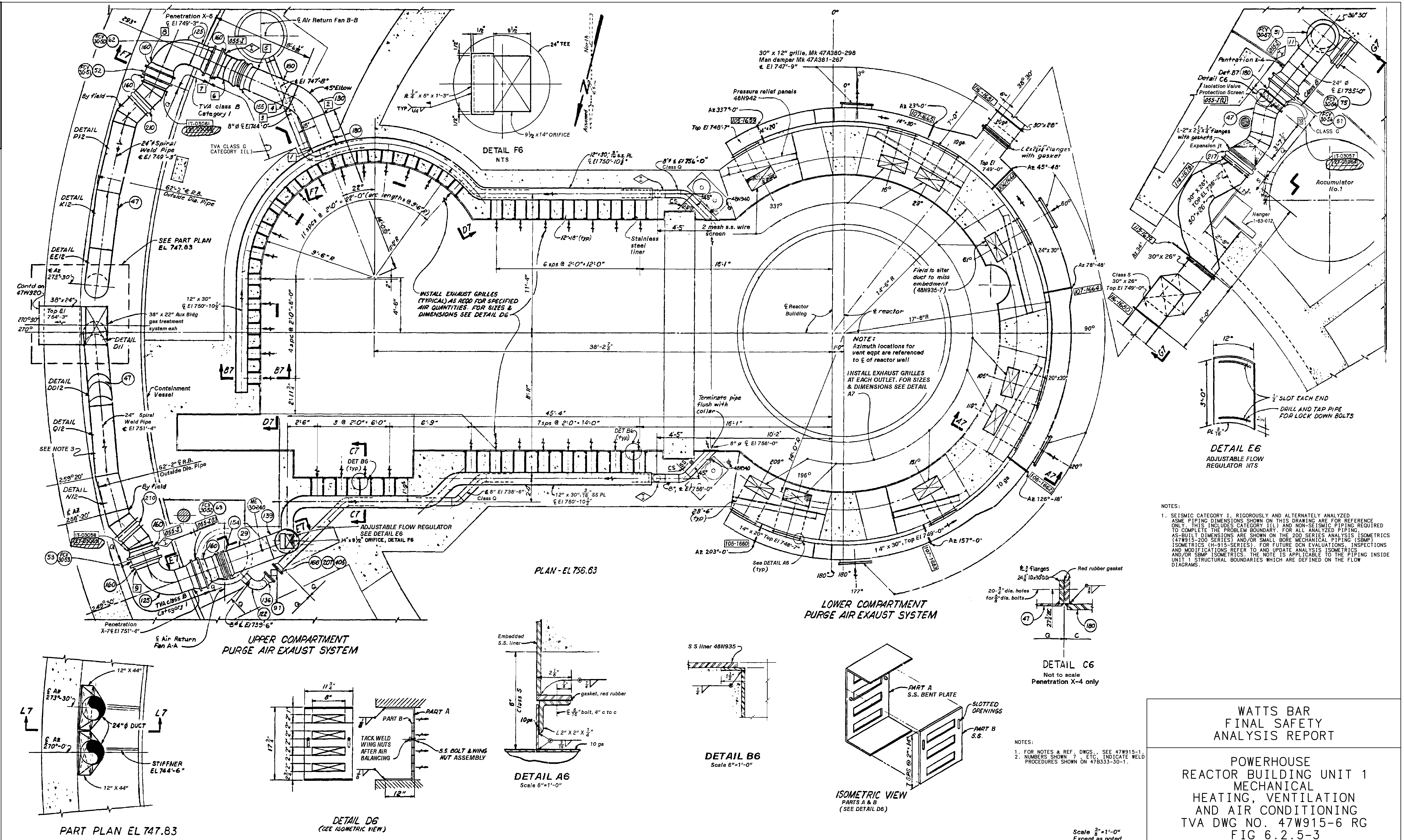
CAD MAINTAINED DRAWING



NOTES:

1. SEISMIC CATEGORY I, RIGOROUSLY AND ALTERNATELY ANALYZED ASME PIPING DIMENSIONS SHOWN ON THIS DRAWING ARE FOR REFERENCE ONLY. THIS INCLUDES CATEGORY I(L) AND NON-SEISMIC PIPING REQUIRED TO COMPLETE THE PROBLEM BOUNDARY. FOR ALL ANALYZED PIPING, AS-BUILT DIMENSIONS ARE SHOWN ON THE 200 SERIES ANALYSIS ISOMETRICS (47W915-200 SERIES) AND/OR SMALL BORE MECHANICAL PIPING (SBMP) ISOMETRICS (H-915-SERIES). FOR FUTURE DCN EVALUATIONS, INSPECTIONS AND MODIFICATIONS REFER TO AND UPDATE ANALYSIS ISOMETRICS AND/OR SBMP ISOMETRICS. THE NOTE IS APPLICABLE TO THE PIPING INSIDE UNIT 1 STRUCTURAL BOUNDARIES WHICH ARE DEFINED ON THE FLOW DIAGRAMS.

NOTES:

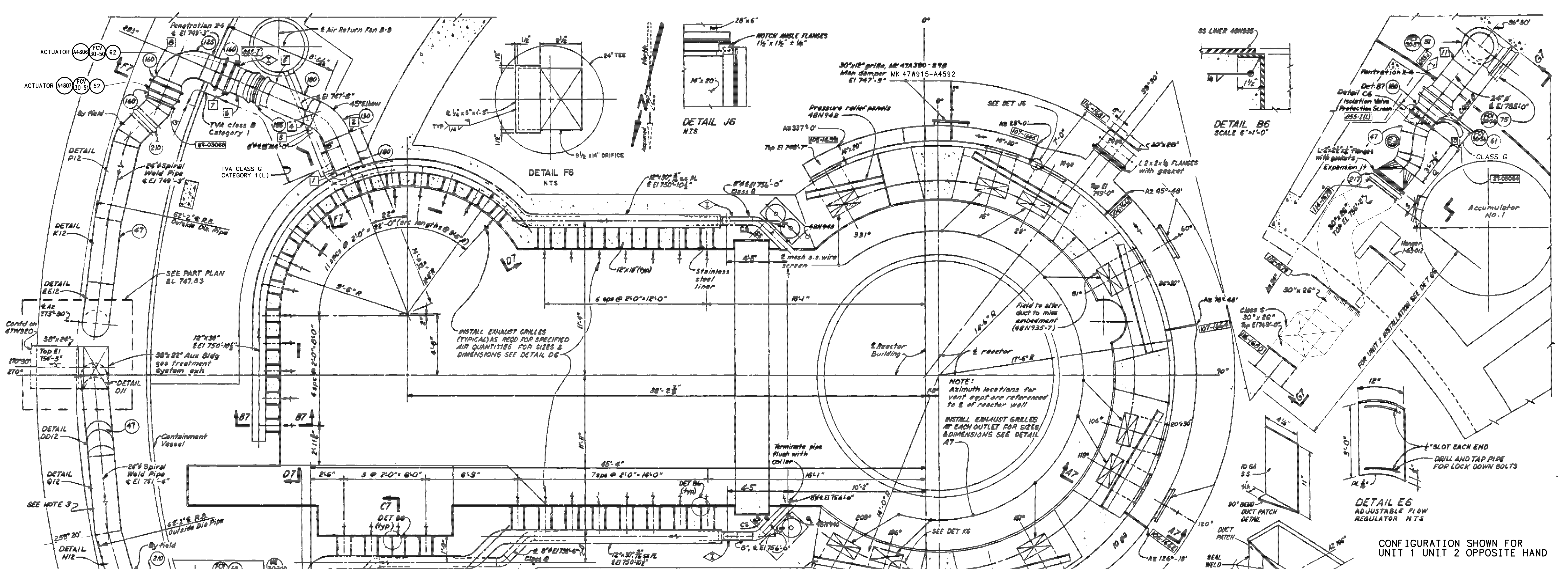
1. FOR NOTES & REF. DWGS., SEE 47W915-1.
2. NUMBERS SHOWN 7, ETC. INDICATE WELD PROCEDURES SHOWN ON 47B333-30-1.

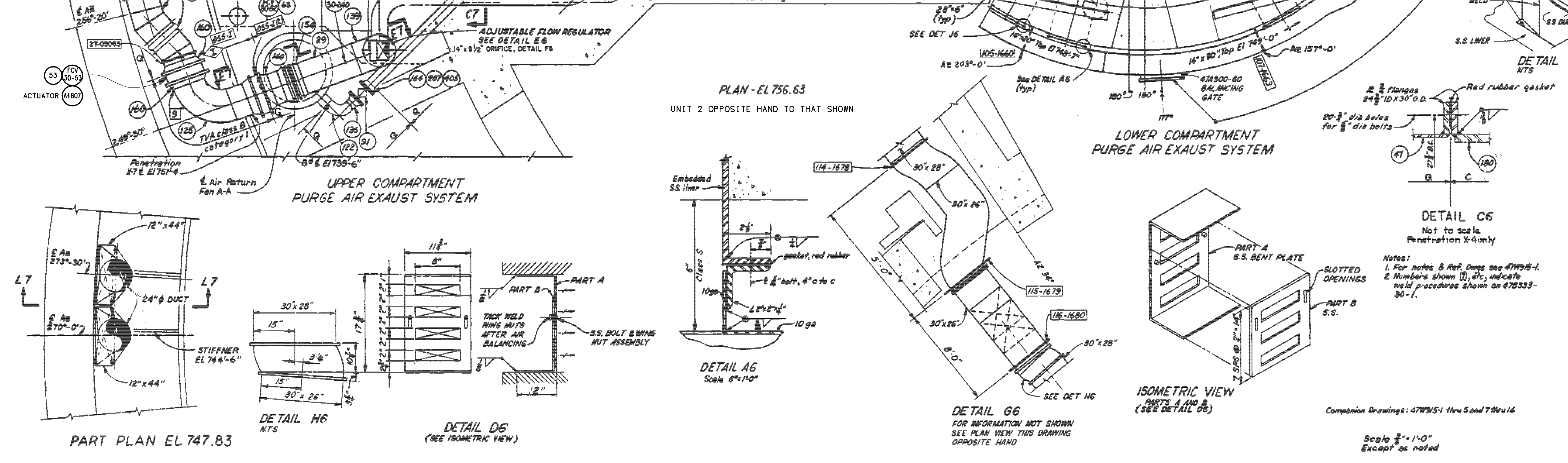
Scale 3/8"=1'-0"
Except as noted

WATTS BAR
FINAL SAFETY
ANALYSIS REPORT

POWERHOUSE
REACTOR BUILDING UNIT 1
MECHANICAL
HEATING, VENTILATION
AND AIR CONDITIONING
TVA DWG NO. 47W915-6 RG
FIG 6.2.5-3

CAD MAINTAINED DRAWING



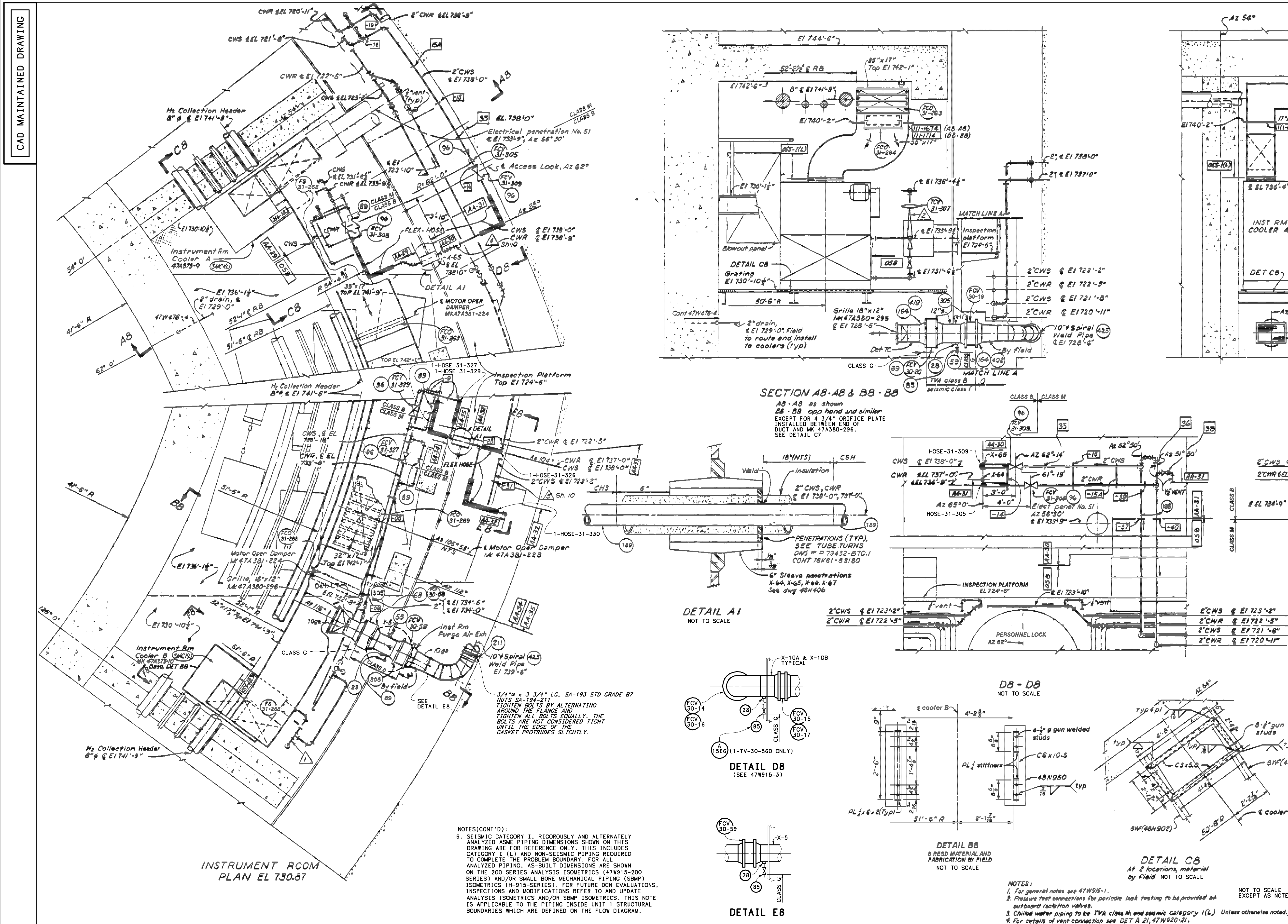


NOTES:
1. SEISMIC CATEGORY I, RIGOROUSLY AND ALTERNATELY ANALYZED ASME PIPING DIMENSIONS SHOWN ON THIS DRAWING ARE FOR REFERENCE ONLY. THIS INCLUDES CATEGORY I(L) AND NON-SEISMIC PIPING REQUIRED TO COMPLETE THE PROBLEM BOUNDARY. FOR ALL ANALYZED PIPING, AS-BUILT DIMENSIONS ARE SHOWN ON THE 100 SERIES ANALYSIS ISOMETRICS (47W915-100 SERIES) AND/OR SMALL BORE MECHANICAL PIPING (SBMP) ISOMETRICS (H-915-SERIES). FOR FUTURE DCN EVALUATIONS, INSPECTIONS AND MODIFICATIONS REFER TO AND UPDATE ANALYSIS ISOMETRICS AND/OR SBMP ISOMETRICS. THIS NOTE IS APPLICABLE TO THE PIPING INSIDE UNIT 2 STRUCTURAL BOUNDARIES WHICH ARE DEFINED ON THE FLOW DIAGRAMS.
2. REFER TO DRAWING 2-47W915-24 FOR HARDWARE MATERIAL REPLACED ON ASME VALVES.

USFAR AMENDMENT 1

WATTS BAR
FINAL SAFETY
ANALYSIS REPORT

POWERHOUSE
UNIT 2
MECHANICAL
HEATING, VENTILATION
AND AIR CONDITIONING
TVA DWG NO. 2-47W915-6 R2
FIG 6.2.5-3(U2)



NOTES(CONT'D):

6. SEISMIC CATEGORY I, RIGOROUSLY AND ALTERNATELY ANALYZED ASME PIPING DIMENSIONS SHOWN ON THIS DRAWING ARE FOR REFERENCE ONLY. THIS INCLUDES CATEGORY I (L) AND NON-SEISMIC PIPING REQUIRED TO COMPLETE THE PROBLEM BOUNDARY. FOR ALL ANALYZED PIPING, AS-BUILT DIMENSIONS ARE SHOWN ON THE 200 SERIES ANALYSIS ISOMETRICS (47W915-200 SERIES) AND/OR SMALL BORE MECHANICAL PIPING (SBMP) ISOMETRICS (H-915-SERIES). FOR FUTURE DCN EVALUATIONS, INSPECTIONS AND MODIFICATIONS REFER TO AND UPDATE ANALYSIS ISOMETRICS AND/OR SBMP ISOMETRICS. THIS NOTE IS APPLICABLE TO THE PIPING INSIDE UNIT 1 STRUCTURAL BOUNDARIES WHICH ARE DEFINED ON THE FLOW DIAGRAM.

NOTES:

1. For general notes see 47W915-1.
2. Pressure test connections for pericdic leak testing to be provided at outboard isolation valves.
3. Chilled water piping to be TVA class M and seismic category I(L) Unless otherwise noted.
4. For details of vent connection see DET A 21, 47W920-21.
5. Hanger numbers shown thus -8 are detailed on 47A915-8 series

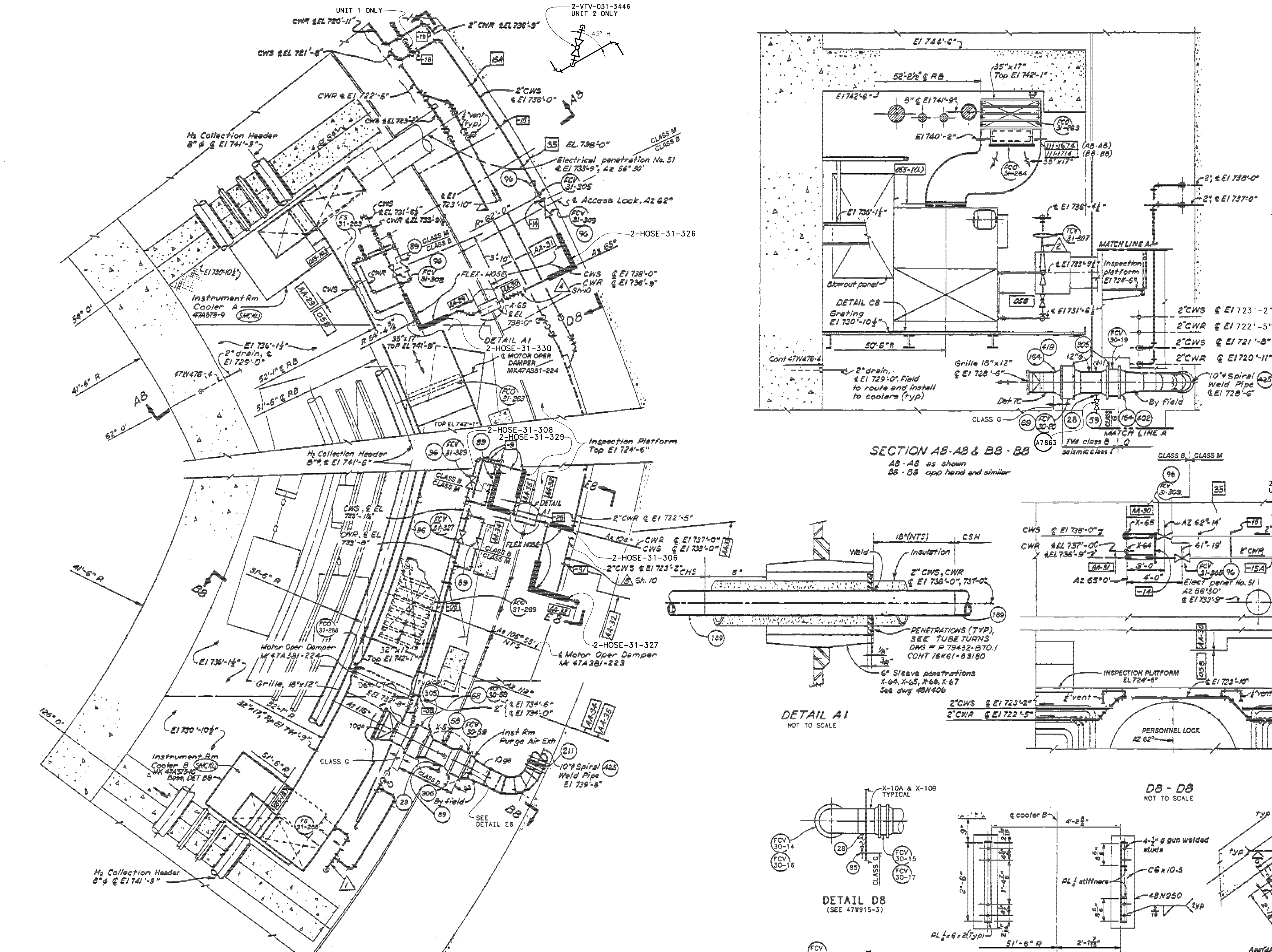
NOT TO SCALE
EXCEPT AS NOTED

USFAR AMENDMENT 1

WATTS BAR
FINAL SAFETY
ANALYSIS REPORT

POWERHOUSE
REACTOR BUILDING UNIT 1
MECHANICAL
HEATING, VENTILATING
AND AIR CONDITIONING
TVA DWG NO. 47W915-8 RN
FIGURE 6.2.5-4

CAD MAINTAINED DRAWING

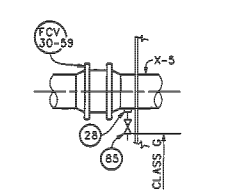


INSTRUMENT ROOM
PLAN EL 730.87
UNIT 2 OPPOSITE HAND
TO THAT SHOWN

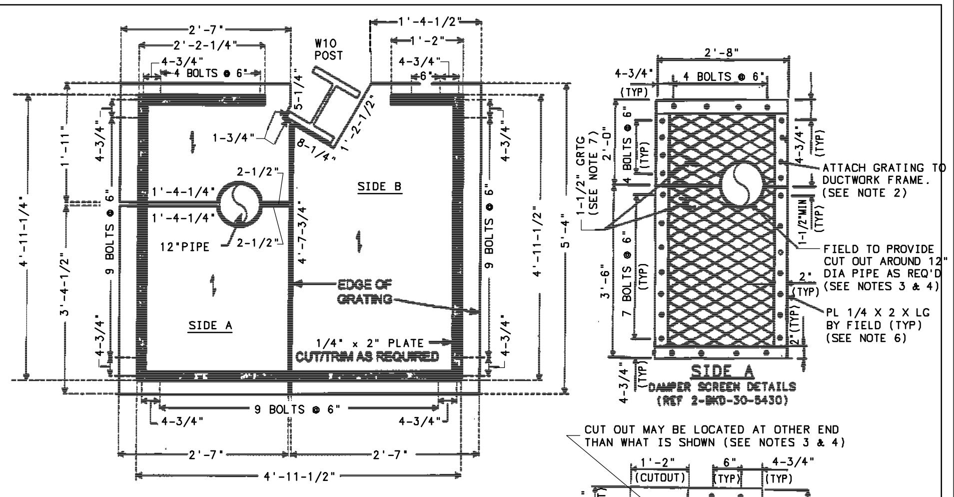
SECTION A8-A8 & B8-B8
A8-A8 as shown
B8-B8 opp hand and similar

DETAIL A1
NOT TO SCALE

DETAIL D8
(SEE 47W915-3)



DETAIL E8

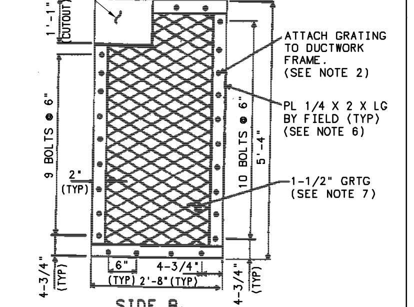


SIDE A
DAMPER SCREEN DETAILS
(REF 2-BKD-30-5430)

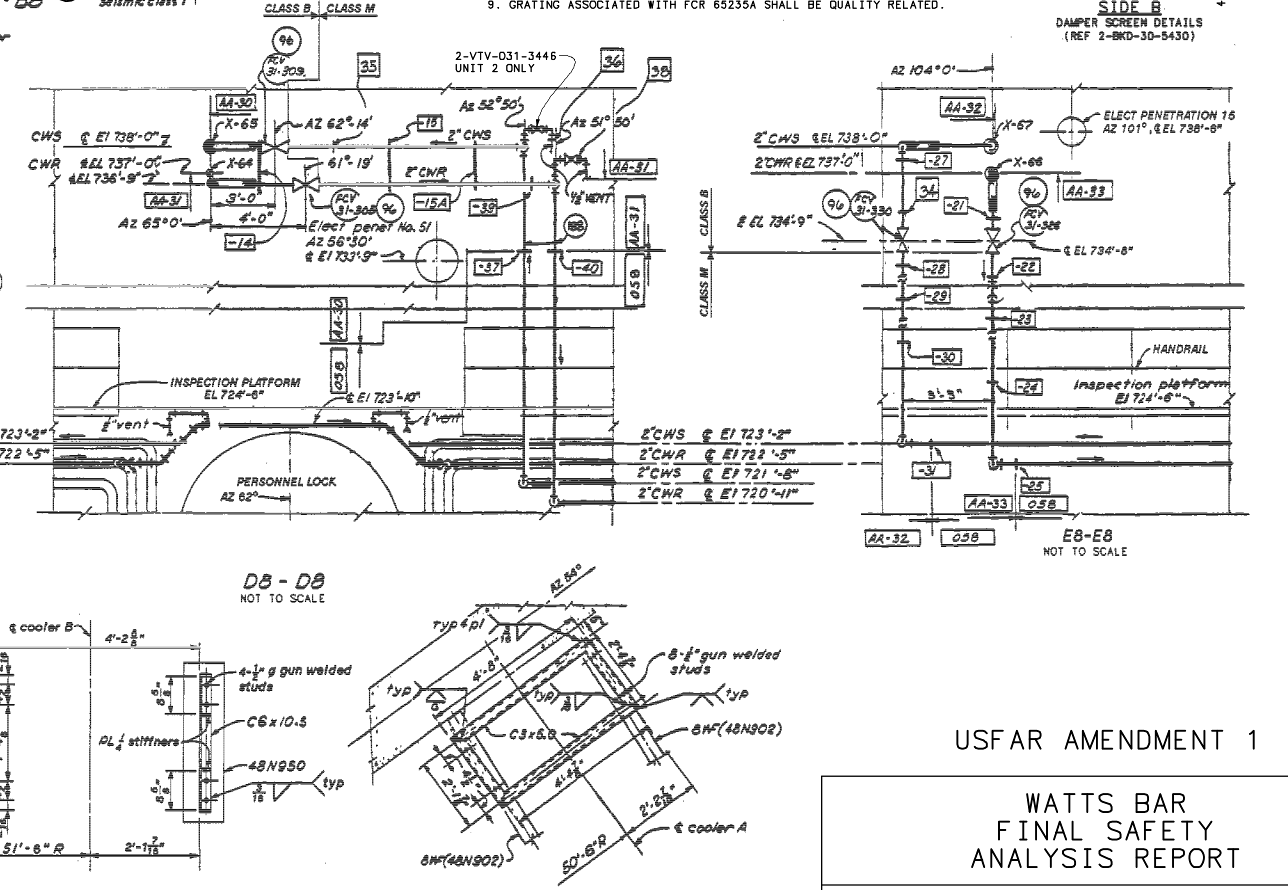
DETAIL F8

DETAIL F8 NOTES:

1. DAMPER SCREEN CONSIST OF TWO SECTIONS (SIDE A) AND ONE SECTION (SIDE B) AND SHALL BE INSTALLED TO CONFORM TO THE DIMENSIONS OF THE DUCTWORK FRAME.
2. FIELD TO USE 1/2"Ø BOLTS (ASTM 307, GR A) AND CORRESPONDING NUTS/WASHERS FOR GRATING HOLD DOWN (SEE DETAILS).
3. FIELD TO PROVIDE CUT-OUT TO CLEAR EXISTING COMMODITIES AS SHOWN. "SIDE A" CUT-OUT REQ'D DUE TO EXISTING 12" PIPE. "SIDE B" CUT-OUT REQ'D DUE TO EXISTING W10 POST. FOR CUT-OUTS MADE BY FIELD, RAW BAR ENDS OF GRATING ARE TO BE ZINC SPRAYED.
4. PROVIDE 0" TO 2" CLEARANCE BETWEEN GRATING FRAMEWORK AND EXISTING 12" PIPE (SIDE A) & W10 POST (SIDE B).
5. DIMENSIONAL TOLERANCE SHALL BE ±1"(UNO).
6. BACKING/COVER PLATE MATERIAL SHALL BE ASTM A36, CUT TO SUIT (QUANTITIES BY FIELD).
7. GRATING CALL OUT PICTORIALLY SHOWN AS GRILLE. FIELD TO INSTALL 1-1/2" GRATING.
8. GRATING SHALL BE CARBON STEEL (BY FIELD).
9. GRATING ASSOCIATED WITH FCR 65235A SHALL BE QUALITY RELATED.



SIDE B
DAMPER SCREEN DETAILS
(REF 2-BKD-30-5430)



D8-D8
NOT TO SCALE

E8-E8
NOT TO SCALE

DETAIL B8
8 REQD MATERIAL AND FABRICATION BY FIELD
NOT TO SCALE

DETAIL C8
At 2 locations, material by field NOT TO SCALE

NOTES(CONT'D):

6. SEISMIC CATEGORY I, RIGOROUSLY AND ALTERNATELY ANALYZED ASME PIPING DIMENSIONS SHOWN ON THIS DRAWING ARE FOR REFERENCE ONLY. THIS INCLUDES CATEGORY I(L) AND NON-SEISMIC PIPING REQUIRED TO COMPLETE THE PROBLEM BOUNDARY, FOR ALL ANALYZED PIPING, AS-BUILT DIMENSIONS ARE SHOWN ON THE 100 SERIES ANALYSIS ISOMETRICS (47W915-100 SERIES) AND/OR SMALL BORE MECHANICAL PIPING (SBMP) ISOMETRICS (H-915-SERIES). FOR FUTURE DCN EVALUATIONS, INSPECTIONS AND MODIFICATIONS REFER TO AND UPDATE ANALYSIS ISOMETRICS AND/OR SBMP ISOMETRICS. THIS NOTE IS APPLICABLE TO THE PIPING INSIDE UNIT 2 STRUCTURAL BOUNDARIES WHICH ARE DEFINED ON THE FLOW DIAGRAMS.

7. FOR UNIT 2 PIPING DETAILS SEE 2-47W915-17

NOTES:

1. For general notes see 47W915-1.
2. Pressure test connections for periodic leak testing to be provided at outboard isolation valves.
3. Chilled water piping to be TVA class M and seismic category I(L) Unless otherwise noted.
4. For details of vent connection see DET A 21, 47W920-21.
5. Hanger numbers shown thus -18 are detailed on 47A915-8 series

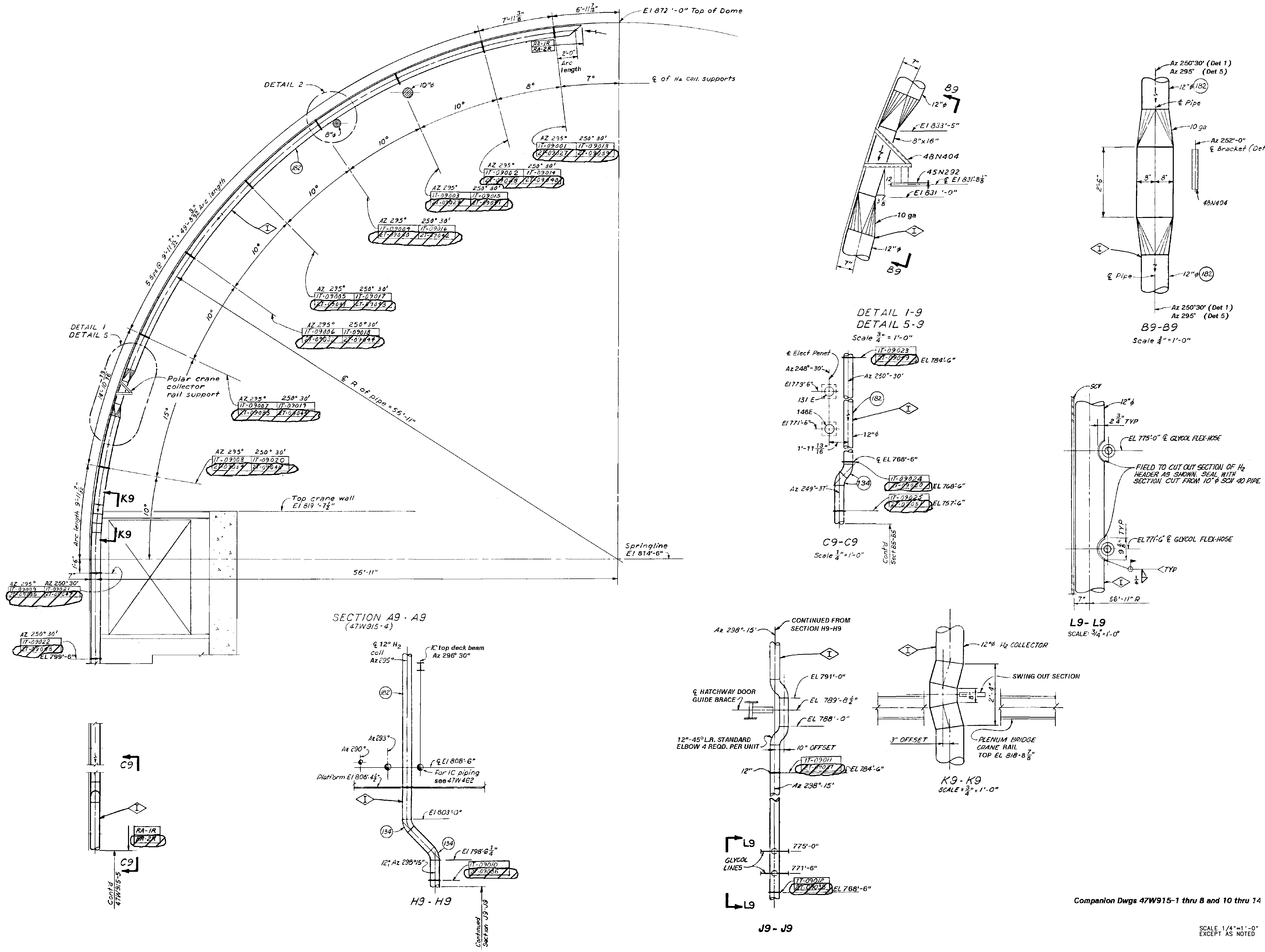
NOT TO SCALE
EXCEPT AS NOTED

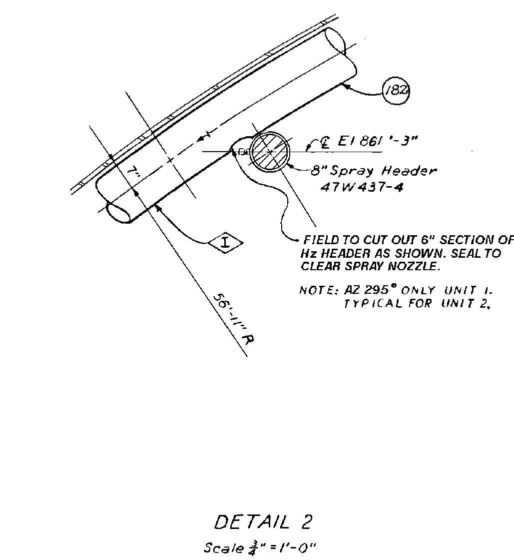
USFAR AMENDMENT 1

WATTS BAR
FINAL SAFETY
ANALYSIS REPORT

POWERHOUSE
REACTOR BUILDING UNIT 2
MECHANICAL
HEATING, VENT & AIR COND
TVA DWG NO. 2-47W915-8 R2
FIGURE 6.2.5-4(U2)

CAD MAINTAINED DRAWING





NOTES:

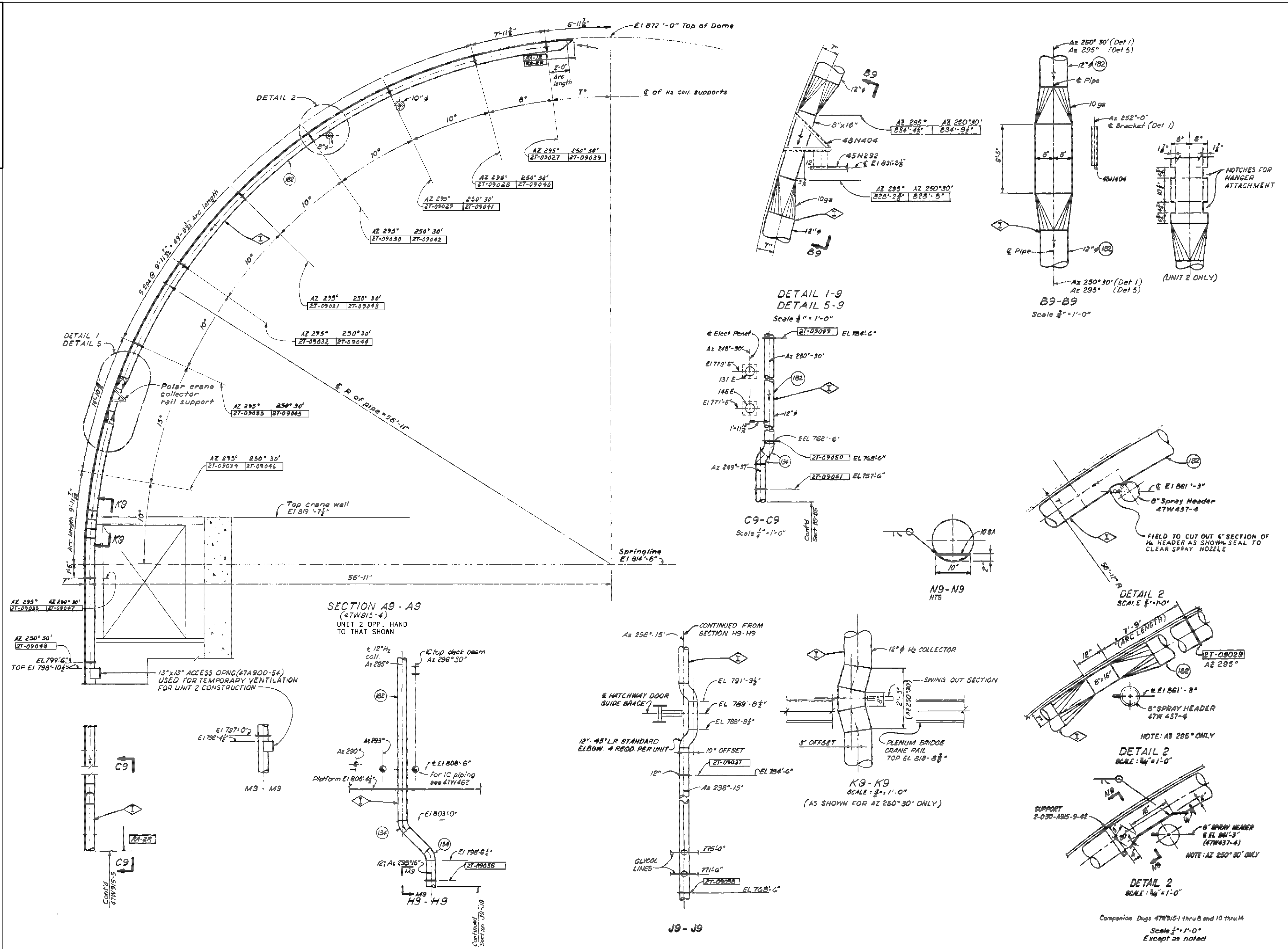
1. FOR GENERAL NOTES SEE 47W915-1.
2. COLLECTOR PIPES ARE TVA CLASS Q & SEISMIC CATEGORY 1.
3. ALL PLATES TO BE ASTM A36.
4. REMOVE ALL EXISTING LUGS AND REPLACE WITH SUPPORTS 47A915-9-1 THRU 52.
5. IT-09001 INDICATES SUPPORT LOCATIONS.
   - HANGER DWG. NO.
   - PIPING SHEET NO.
   - DESIGN ORGANIZATION
   - UNIT NO.
6. WELDING AND WELD INSPECTION PERFORMED ON DUCT AFTER JANUARY 12, 1987 SHALL BE IN ACCORDANCE WITH REQUIREMENTS OF CONSTRUCTION SPECIFICATION N3M-914, BEFORE JANUARY 12, 1987 CLASSES Q & S CATEGORY I DUCT WELDS HAVE BEEN QUALIFIED BY CORRECTIVE ACTIONS OF SCRS WBN 7707-S WBN MEB 8714, 8721, AND 8722.
7. SEISMIC CATEGORY I, RIGOROUSLY AND ALTERNATELY ANALYZED ASME PIPING DIMENSIONS SHOWN ON THIS DRAWING ARE FOR REFERENCE ONLY. THIS INCLUDES CATEGORY I(L) AND NON-SEISMIC PIPING REQUIRED TO COMPLETE THE PROBLEM BOUNDARY. FOR ALL ANALYZED PIPING, AS-BUILT DIMENSIONS ARE SHOWN ON THE 200 SERIES ANALYSIS ISOMETRICS (47W915-200 SERIES) AND/OR SMALL BORE MECHANICAL PIPING (SBMP) ISOMETRICS (H-915-SERIES). FOR FUTURE DCN EVALUATIONS, INSPECTIONS AND MODIFICATIONS REFER TO AND UPDATE ANALYSIS ISOMETRICS AND/OR SBMP ISOMETRICS. THIS NOTE IS APPLICABLE TO THE PIPING INSIDE UNIT 1 STRUCTURAL BOUNDARIES WHICH ARE DEFINED ON THE FLOW DIAGRAMS.

SCALE 1/4"=1'-0"
EXCEPT AS NOTED

WATTS BAR
FINAL SAFETY
ANALYSIS REPORT

POWERHOUSE
REACTOR BUILDING UNIT 1
MECHANICAL
HEATING, VENTILATION
& AIR CONDITIONING
TVA DWG NO. 47W915-9 RC
FIGURE 6.2.5-5

CAD MAINTAINED DRAWING



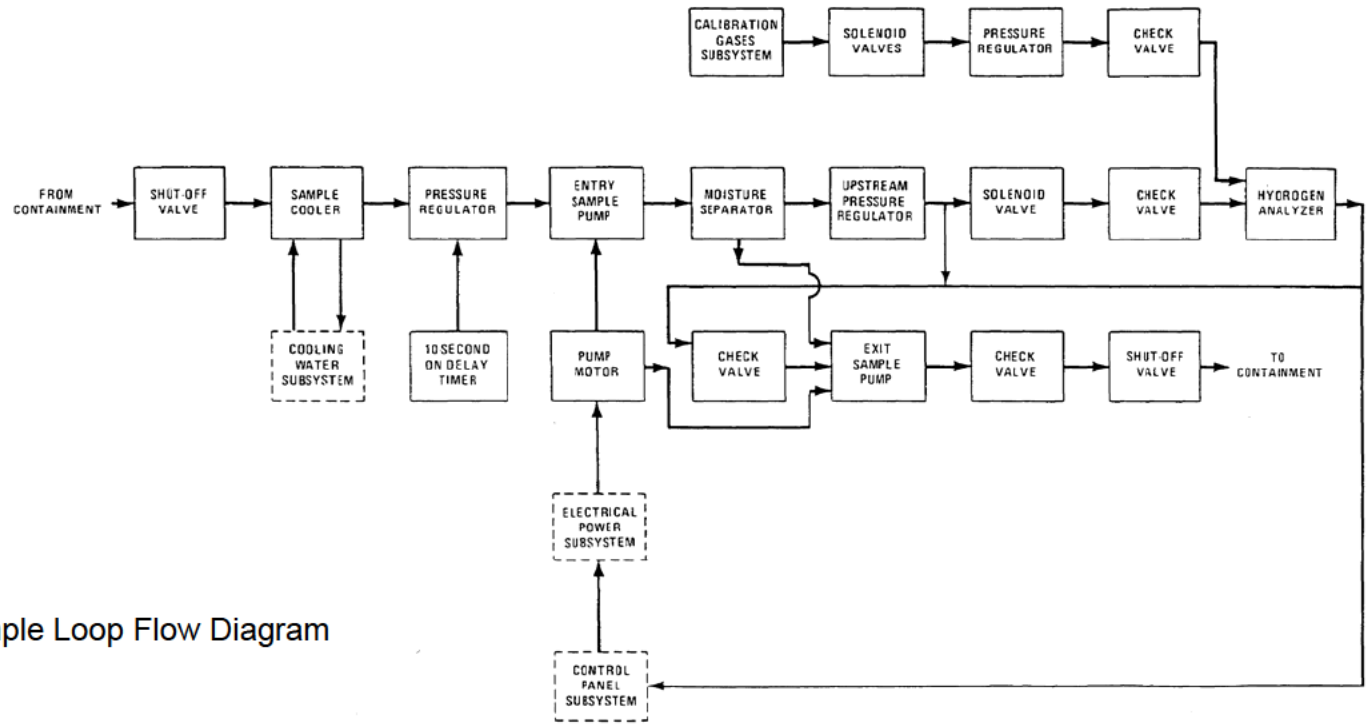
NOTES:

1. FOR GENERAL NOTES SEE 47W915-1.
2. COLLECTOR PIPES ARE TVA CLASS Q & SEISMIC CATEGORY I.
3. ALL PLATES TO BE ASTM A36
4. REMOVE ALL EXISTING LUGS AND REPLACE WITH SUPPORTS 47A915-9-1 THRU 52
5. 17-03001 INDICATES SUPPORT LOCATIONS.
   - HANGER DWG. NO.
   - PIPING SHEET NO.
   - DESIGN ORGANIZATION
   - UNIT NO.
6. WELDING AND WELD INSPECTION PERFORMED ON DUCT AFTER JANUARY 12,1987 SHALL BE IN ACCORDANCE WITH REQUIREMENTS OF CONSTRUCTION SPECIFICATION N3M-914. BEFORE JANUARY 12, 1987 CLASSES Q&S CATEGORY I DUCT WELDS HAVE BEEN QUALIFIED BY CORRECTIVE ACTIONS OF SCRS WBN 7707-3 WBN MEB 8714, 8721, AND 8722.
7. SEISMIC CATEGORY I, RIGOROUSLY AND ALTERNATELY ANALYZED ASME PIPING DIMENSIONS SHOWN ON THIS DRAWING ARE FOR REFERENCE ONLY. THIS INCLUDES CATEGORY I(L) AND NON-SEISMIC PIPING REQUIRED TO COMPLETE THE PROBLEM BOUNDARY. FOR ALL ANALYZED PIPING, AS-BUILT DIMENSIONS ARE SHOWN ON THE 100 SERIES ANALYSIS ISOMETRICS (47W915-100 SERIES) AND/OR SMALL BORE MECHANICAL PIPING (SBMP) ISOMETRICS (H-915-SERIES). FOR FUTURE DCN EVALUATIONS, INSPECTIONS AND MODIFICATIONS REFER TO AND UPDATE ANALYSIS ISOMETRICS AND/OR SBMP ISOMETRICS. THIS NOTE IS APPLICABLE TO THE PIPING INSIDE UNIT 2 STRUCTURAL BOUNDARIES WHICH ARE DEFINED ON THE FLOW DIAGRAMS.

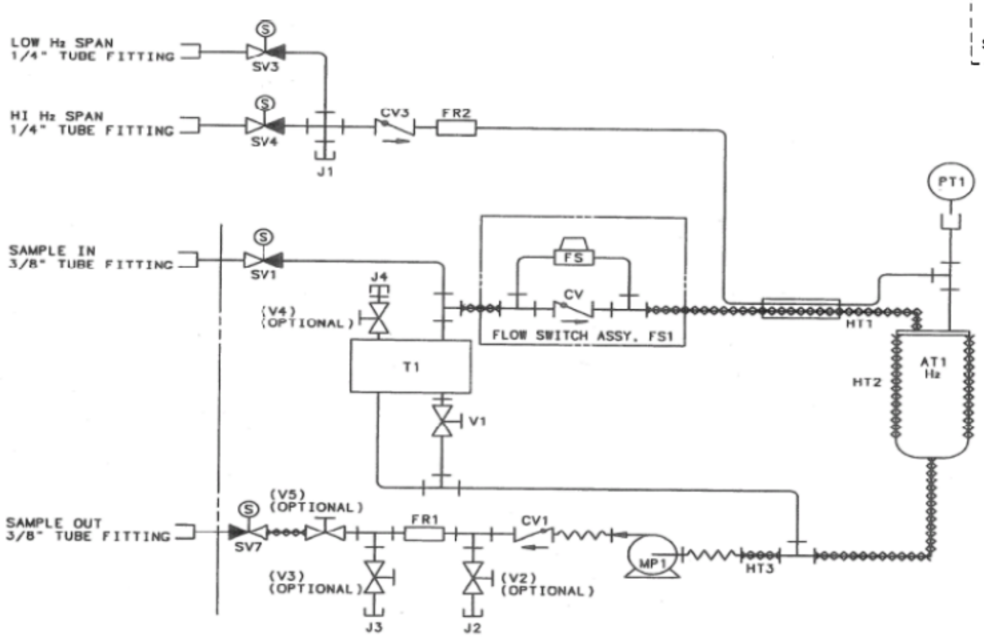
WATTS BAR
FINAL SAFETY
ANALYSIS REPORT

POWERHOUSE
REACTOR BUILDING UNIT 2
MECHANICAL
HEATING, VENTILATION
& AIR CONDITIONING
TVA DWG NO. 2-47W915-9 RO
FIGURE 6.2.5-5(U2)

Unit 1 – Function Flow Block Diagram – Containment Gas Monitor Subsystem



Unit 2 – Sample Loop Flow Diagram



WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

Monitoring System

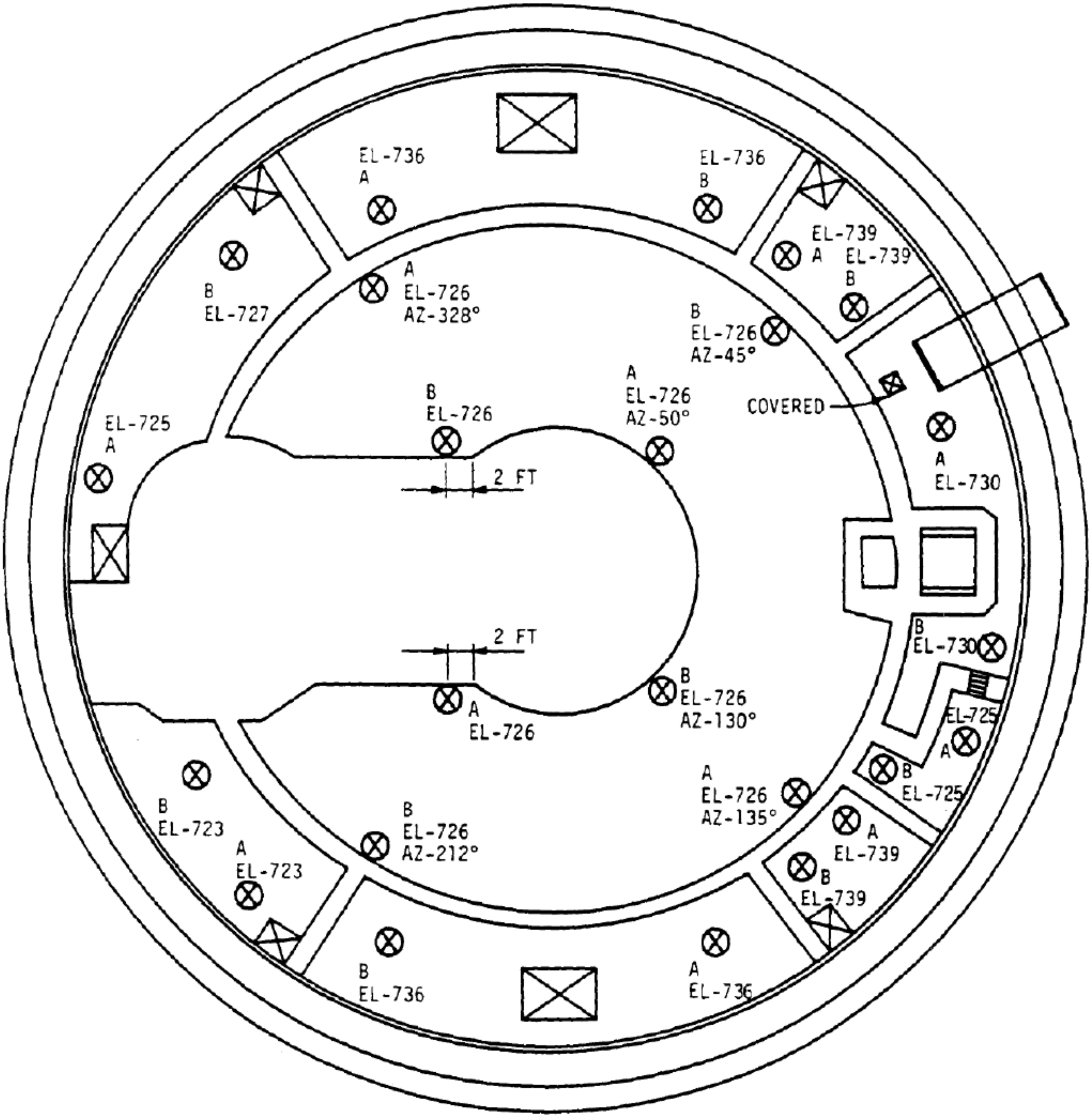
FIGURE 6.2.5-6

FIGURE 6.2.5-7

DELETED

FIGURE 6.2.5-7a

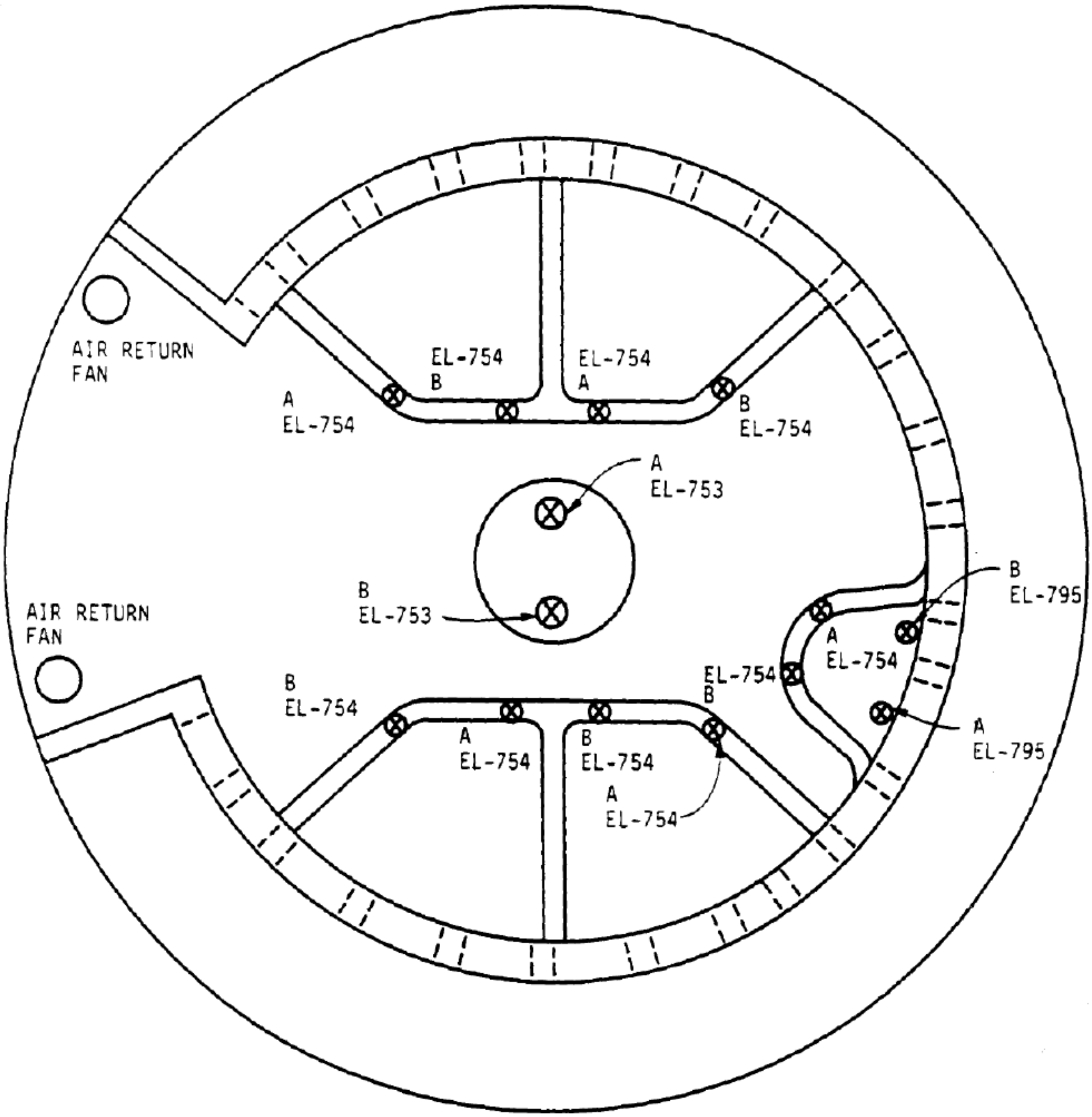
DELETED



WATTS BAR NUCLEAR PLANT FINAL SAFETY ANALYSIS REPORT

Igniter Locations Lower Compartment and Dead Ended Compartments

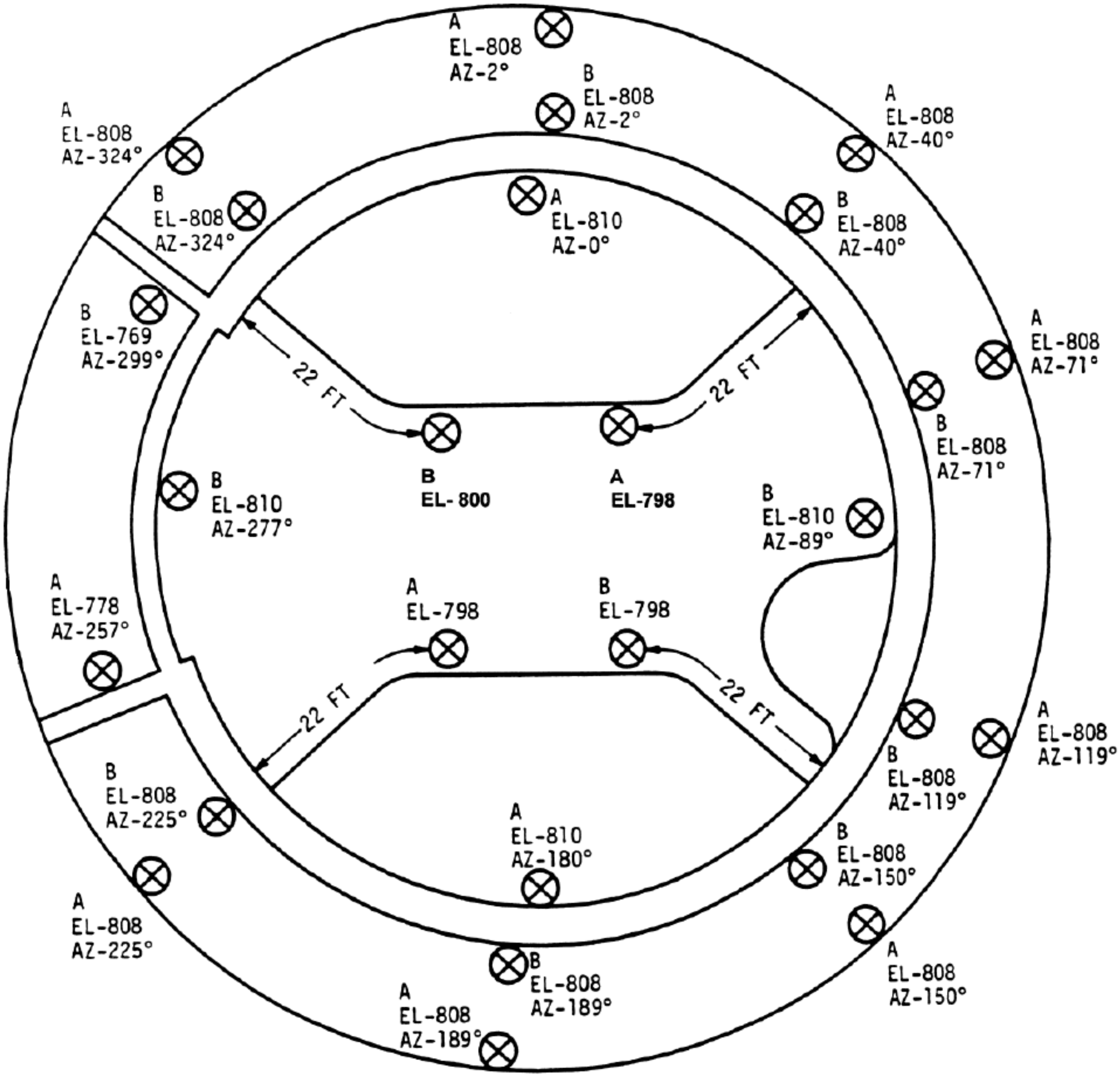
Figure 6.2.5-8



WATTS BAR NUCLEAR PLANT FINAL SAFETY ANALYSIS REPORT

Igniter Locations Lower Compartments

Figure 6.2.5-9



WATTS BAR NUCLEAR PLANT
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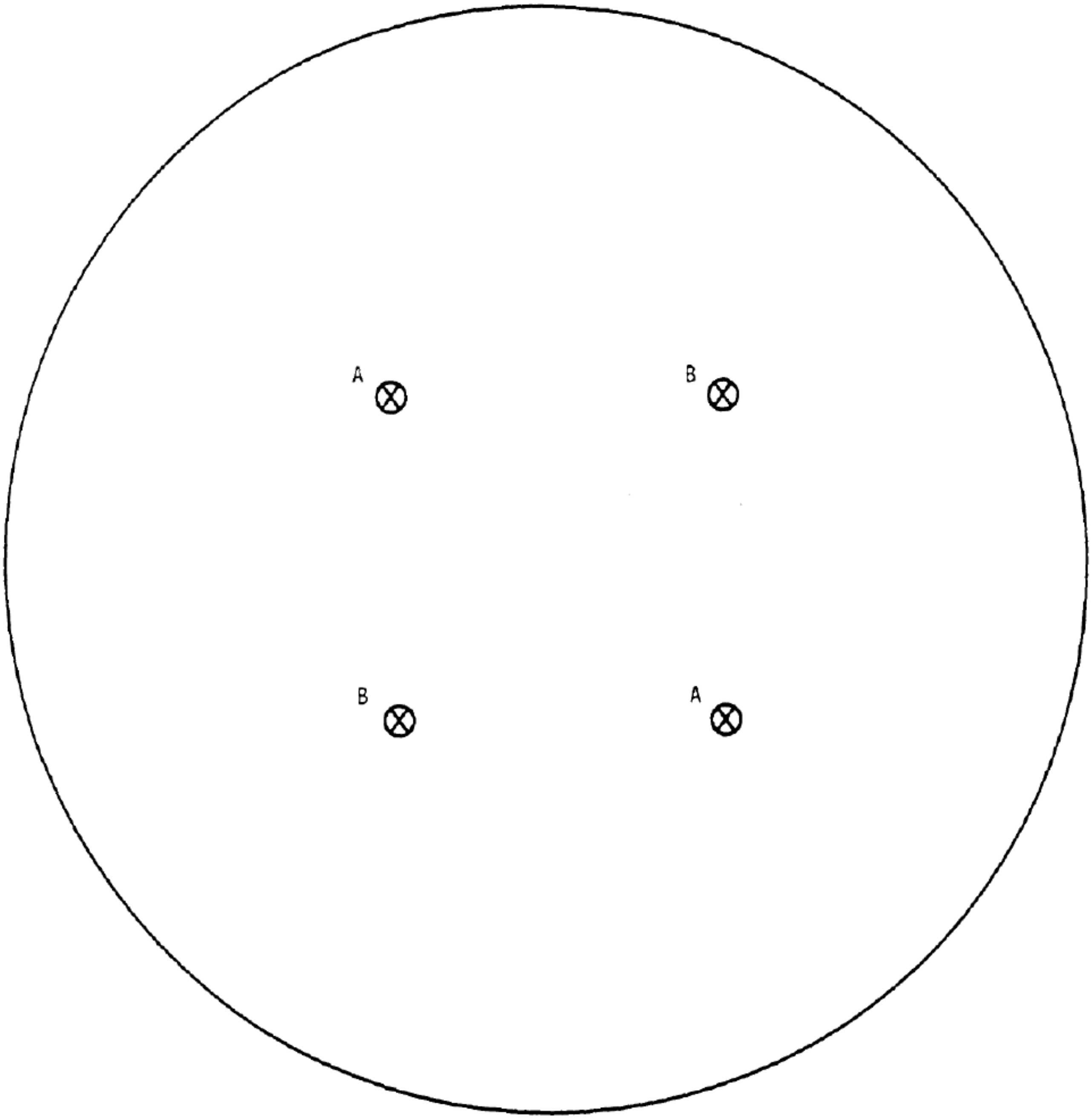
ANALYSIS REPORT

Igniter Locations
Upper Plenum and
Upper Containment

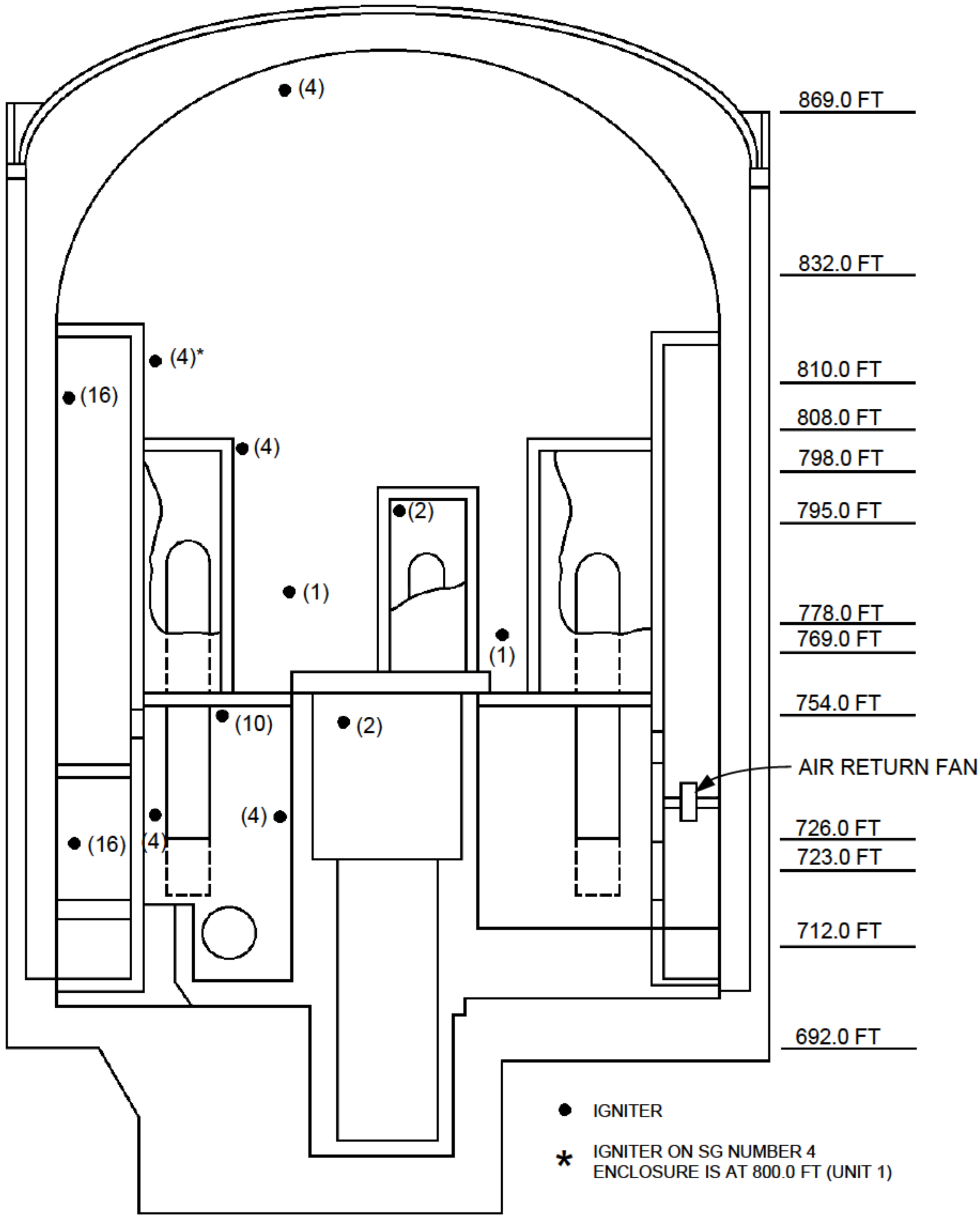
Figure 6.2.5-10



WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

Igniter Locations
Dome

Figure 6.2.5-11



WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

Igniter Locations
Elevation

Figure 6.2.5-12

6.2.6 CONTAINMENT LEAKAGE TESTING

Primary containment leakage tests and containment isolation system valve operability tests will be performed periodically to verify that leakage from the containment is maintained within acceptable limits set forth in the Technical Specifications. The types of leakage tests are as follows:

1. Test Type A

   Tests to measure the reactor primary containment overall integrated leakage rate. The containment leak rate test will be conducted in accordance with 10 CFR 50, Appendix J, Option B.

2. Test Type B

   Tests to detect and measure local leaks of containment penetrations, hatches, and personnel locks as required by 10 CFR 50, Appendix J, Option B.

3. Test Type C

   Test to detect and measure containment isolation valve leakage as described by 10 CFR 50, Appendix J, Option B.

The leakage rate testing pressure for the above tests, P_a (as defined in 10CFR50 Appendix J), has a nominal value of 15.0 psig with allowance for instrument error. Exceptions to this test pressure are noted elsewhere in this section.

6.2.6.1 Containment Integrated Leak Rate Test

The maximum allowable containment leakage rate for the Watts Bar Nuclear Plant is 0.25 weight percent per day as specified in the Technical Specifications. The preoperational testing will be conducted in full compliance with 10CFR50, Appendix J as shown in Table 14.2-1 (historical information). Subsequent periodic testing will also be performed in accordance with Appendix J, Option B with approved exemptions.

Prior to conducting the integrated leak rate test, those lines which penetrate primary containment are aligned as shown in Table 6.2.6-3.

The containment is then pressurized in accordance with 10 CFR 50, Appendix J, and the Technical Specification requirements. When test pressure is reached, the containment is isolated from its pressure source and the following parameters are recorded at periodic intervals:

1. Containment absolute pressure

2. Dry bulb temperatures

3. Water vapor pressures

4. Outside containment pressure and temperature conditions

During the test, ventilation inside the containment is operated as necessary to enhance an even air temperature distribution. The test data are processed at periodic intervals during the test to determine test status and leakage conditions. If it appears that the leakage is excessive, the pressure plateau is either maintained on the test or aborted to perform repairs. The test is run for a prescribed time period to obtain assurance of the test leak rate.

Following the leak rate test, a second leak rate is performed to verify the information obtained in the first test. This verification test consists of slowly bleeding off pressure from containment at a known rate and measuring the total containment leak rate. The superimposed, measured flow is adjusted to a value which causes a change in the weight of air in the containment that is in the same order of magnitude as the allowable leakage rate.

The mass point equations are used to determine the integrated leak rate. The mass point equations were for preoperational testing as discussed in Table 14.2-1 (historical information).

6.2.6.2 Containment Penetration Leakage Rate Test

Table 6.2.4-1 lists penetrations in the primary containment. The Type B test is performed on all operational electrical equipment and personnel hatch, fuel transfer tube, thimble renewal (Unit 1) or Incore Instrument Thimble Assembly (Unit 2), and ice blowing penetrations, and penetration bellows in accordance with 10 CFR 50, Appendix J, Option B. The dual-ply bellows on containment penetration will be tested at P_a by applying the pressure between the plies. Airlock door seals are tested at 6.0 psig per Technical Specification requirements. Experience has shown that pressurizing the space between the seals to greater than 6.5 psig on personnel airlock doors of the design used at Watts Bar will lift the door and induce gross leakage unless strong-backs are used. Since the door seal test is intended to prove integrity of the seals, it is our position that a test conducted at 6.0 psig will conservatively demonstrate that seal integrity is maintained.

Table 6.2.6-1 lists all penetrations subjected to type B testing. Spare electrical penetrations will be subjected to Type B testing as they become operational. Tables 6.2.4-1 through 6.2.4-4 and Figures 8.3-44 and 8.3-45 give details on these penetrations. The test is performed in full compliance with 10 CFR 50, Appendix J, Option B. The acceptance criteria as required by Appendix J, Option B, are specified in the Technical Specifications.

Table 6.2.4-1 lists containment isolation valves. Table 6.2.6-2a identifies those valves that are tested during a Type C test.

Isolation valves that are part of closed systems that are in use after a design basis event and valves that are water sealed for at least 30 days after a design basis event are not tested in the Type C test program. Table 6.2.6-2b lists the valves exempted from type C leak testing. Bases for exemptions and exceptions from type C leakage rate testing on a penetration by penetration basis are as follows:

I. Exemptions

1. Feedwater Bypass - X-8A, X-8B, X-8C, X-8D
Feedwater - X-12A, X-12B, X-12C, X-12D
Main Steam - X-13A, X-13B, X-13C, X-13D
Steam Generator Blowdown - X-14A, X-14B, X-14C, X-14D
Steam Generator Blowdown Sample Line - X-27A, X-27B, X-27C, X-27D
Auxiliary Feedwater - X-40A, X-40B

These penetrations are directly connected to the secondary side of the steam generator. The main steam and feedwater lines of PWR containments are not required to be Type C tested (see definition of Type C test in 10 CFR 50, Appendix J). These lines are assumed not to rupture as a result of an accident (missile protected). Any leakage through these lines would be identified during operation by the leakage detection program. In addition, during a design basis accident, the secondary side of the steam generator is filled with water and would be at a higher pressure than the containment atmosphere thus preventing outleakage from containment. The integrity of the inside piping is also verified during the Type A test.

2. CVCS Normal Charging Line, X-16

This penetration uses an inboard check valve and a closed loop outside containment (CLOC) as the means of containment isolation. Type C testing for this path is not required due to the seal pressure greater than 1.1 P_a and the 30-day water seal inventory as specified in 10 CFR 50, Appendix J. A positive pressure preventing air outleakage is assured by the pressure applied against FCV-62-90 and FCV-62-91 (both of which receive a Safety Injection signal) by the high head SI pumps. Water testing for piping integrity is performed in accordance with ASME XI.

3. RHR Hot Leg Injection, X-17

This penetration uses inboard containment isolation valves and a CLOC for containment boundaries. Type C testing for this penetration is not required since a continuous water seal will be provided at a pressure greater than 1.1 P_a and a 30-day water seal is provided, as specified in 10 CFR 50, Appendix J. Water testing for piping integrity is performed in accordance with ASME XI.

4. RHR Cold Leg Injection, X-20A, X-20B

Same as for X-17.

5. SIS Hot Leg Injection, X-21, X-32

These lines make use of inboard containment isolation valves and a CLOC for containment boundaries. These lines are postulated to be in-service post accident and the high head pumps will maintain a pressure seal greater than 1.1 P_a for greater than 30 days, as specified in 10 CFR 50, Appendix J.

6. Charging Pump Discharge, X-22

This line makes use of inboard containment isolation valves and a CLOC for containment boundaries. Type C testing is not required for the same reasons as X-21 and X-32. Water seal is provided by the high head pumps.

7. SIS Cold Leg Injection, X-33

Same as X-21 and X-32

8. RCP Seal Injection, X-43A, X-43B, X-43C, X-43D

These lines make use of an inboard containment isolation valve and a CLOC for containment boundaries. Type C test is not required for the same reason as given for X-16.

II. Exceptions

1. Sump Suction to RHR, X-19A, X-19B

These lines make use of a containment isolation valve located outside of containment and a CLOC for containment isolation boundaries. During a design basis accident, these lines would be submerged under water which would preclude air outleakage. In addition, these valves are exposed to P_a during each Type A test.

2. SI Relief Valve Discharge to PRT, X-24

The line uses an inboard containment isolation valve and a CLOC for containment boundaries. The maximum calculated containment accident pressure per 10 CFR 50 Appendix J (P_a) during the design basis accident would not create a substantial outleakage driving force and, in any case, tends to cause the relief valves to seat rather than lift. The systems feeding this line are ECCS and, due either to operating pressure or static head, outleakage would be prevented. Therefore, no Type C testing will be performed for this line. However, this line is exposed to the P_a test pressure during the Type A test.

3. Containment Spray, X-48A, X-48B and RHR Spray, X-49A, X-49B

These lines make use of an inboard containment isolation check valve and a CLOC for containment boundaries of each line. Additionally, a water leg seal exists against a system valve outside containment in each line. This seal prevents the outleakage of containment atmosphere. An inventory test is performed to ensure a 30-day seal water inventory at a pressure greater than 1.1 P_a, should one spray system shut down. The seal water inventory leakage rate test is performed in lieu of a Type C air leakage rate test of the containment isolation check valves.

4. RHR Pump Supply, X-107

This line makes use of inboard containment isolation valves and a CLOC for containment boundaries. This penetration satisfies ANSI-N271-1976. In addition, an ASME Section XI water leakage test is performed to verify system integrity. Thus, no Type C test is required for this line.

Test connections and pressurizing means are provided to test isolation valves or barriers for leak tightness. Either air or nitrogen is used as the pressurizing medium, depending on the physical location and service of each line. Leak testing of individual valves and penetrations is accomplished by one of the following methods:

1. Method 1, Pressure Decay

The test volume initially shall be pressurized with air or nitrogen to a test pressure greater than the pressure under accident conditions, P_a. The final test pressure shall be greater than P_a. The pressure and temperature shall be recorded at the start and end of the test. The leakage rate shall be calculated from the following formula:

$$L_i = \frac{TV}{t}\left[\frac{P_1}{T_1} - \frac{P_2}{T_2}\right]\frac{T_{stp}}{P_{stp}}$$

Where:

L_i = Local leak rate, cfm

TV = Test volume, ft^3

P_1 = Initial pressure, psia

P_2 = Final pressure, psia

T_1 = Initial temperature, °R

T_2 = Final temperature, °R

T_{stp} = Standard atmospheric temperature, °R

P_{stp} = Standard atmospheric pressure, psia

t = Test duration, min.

2. Method 2, Makeup Flow Rate

   The test volume shall be pressurized and maintained to at least 15 psig using a pressure regulator to maintain test pressure. Makeup fluid flow to the test volume required to maintain test pressure shall be used as the leakage rate of the barrier under test.

The acceptance criteria for the Type C test are given in the Technical Specification which complies with 10 CFR 50 Appendix J, Option B.

6.2.6.3 Scheduling and Reporting of Periodic Tests

Type A integrated containment leakage tests are performed prior to operation of the plant and subsequently in accordance with the Containment Leakage Rate Testing Program.

Type B and Type C leakage rate tests are performed in accordance with the Containment Leakage Rate Testing Program.

Test reports are made in accordance with 10 CFR 50 Appendix J, Option B.

REFERENCES

1. Title 10, Code of Federal Regulations, Part 50, Appendix J, Option B, “Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors - Performance-Based Requirements.”

2. Regulatory Guide 1.163, “Performance-Based Containment Leak-Test Program,” September 1995.

3. NEI 94-01, Revision 0, “Industry Guideline for Implementing Performance-Based Option of 10CFR Part 50, Appendix J.”

6.2.6 CONTAINMENT LEAKAGE TESTING

Primary containment leakage tests and containment isolation system valve operability tests will be performed periodically to verify that leakage from the containment is maintained within acceptable limits set forth in the Technical Specifications. The types of leakage tests are as follows:

1. Test Type A

   Tests to measure the reactor primary containment overall integrated leakage rate. The containment leak rate test will be conducted in accordance with 10 CFR 50, Appendix J, Option B.

2. Test Type B

   Tests to detect and measure local leaks of containment penetrations, hatches, and personnel locks as required by 10 CFR 50, Appendix J, Option B.

3. Test Type C

   Test to detect and measure containment isolation valve leakage as described by 10 CFR 50, Appendix J, Option B.

The leakage rate testing pressure for the above tests, P_a (as defined in 10CFR50 Appendix J), has a nominal value of 15.0 psig with allowance for instrument error. Exceptions to this test pressure are noted elsewhere in this section.

6.2.6.1 Containment Integrated Leak Rate Test

The maximum allowable containment leakage rate for the Watts Bar Nuclear Plant is 0.25 weight percent per day as specified in the Technical Specifications. The preoperational testing will be conducted in full compliance with 10 CFR 50, Appendix J as shown in Table 14.2-1. Subsequent periodic testing will also be performed in accordance with Appendix J, Option B with approved exemptions.

Prior to conducting the integrated leak rate test, those lines which penetrate primary containment are aligned as shown in Table 6.2.6-4.

The containment is then pressurized in accordance with 10 CFR 50, Appendix J, and the Technical Specification requirements. When test pressure is reached, the containment is isolated from its pressure source and the following parameters are recorded at periodic intervals:

1. Containment absolute pressure

2. Dry bulb temperatures

3. Water vapor pressures

4. Outside containment pressure and temperature conditions

During the test, ventilation inside the containment is operated as necessary to enhance an even air temperature distribution. The test data are processed at periodic intervals during the test to determine test status and leakage conditions. If it appears that the leakage is excessive, the pressure plateau is either maintained on the test or aborted to perform repairs. The test is run for a prescribed time period to obtain assurance of the test leak rate.

Following the leak rate test, a second leak rate is performed to verify the information obtained in the first test. This verification test consists of slowly bleeding off pressure from containment at a known rate and measuring the total containment leak rate. The superimposed, measured flow is adjusted to a value which causes a change in the weight of air in the containment that is in the same order of magnitude as the allowable leakage rate.

The mass point equations are used to determine the integrated leak rate. The mass point equations will be used for preoperational testing as discussed in Table 14.2-1.

6.2.6.2 Containment Penetration Leakage Rate Test

Table 6.2.4-1 lists penetrations in the primary containment. The Type B test is performed on all operational electrical equipment and personnel hatch, fuel transfer tube, Incore Instrumentation Thimble Assembly renewal and ice blowing penetrations, and penetration bellows in accordance with 10 CFR 50, Appendix J, Option B. The dual-ply bellows on containment penetration will be tested at P_a by applying the pressure between the plies. Airlock door seals are tested at 6.0 psig per Technical Specification requirements. Experience has shown that pressurizing the space between the seals to greater than 6.5 psig on personnel airlock doors of the design used at Watts Bar will lift the door and induce gross leakage unless strongbacks are used. Since the door seal test is intended to prove integrity of the seals, it is our position that a test conducted at 6.0 psig will conservatively demonstrate that seal integrity is maintained.

Table 6.2.6-1 lists all penetrations subjected to type B testing. Spare electrical penetrations will be subjected to Type B testing as they become operational. Tables 6.2.4-1 through 6.2.4-4 and Figures 8.3-44 and 8.3-45 give details on these penetrations. The test is performed in full compliance with 10 CFR 50, Appendix J, Option B. The acceptance criteria as required by Appendix J, Option B are specified in the Technical Specifications.

Table 6.2.4-1 lists containment isolation valves. Table 6.2.6-2 identifies those valves that are tested during a Type C test.

Isolation valves that are part of closed systems that are in use after a design basis event and valves that are water sealed for at least 30 days after a design basis event are not tested in the Type C test program (References 1, 2, and 3). Table 6.2.6-3 lists the valves exempted from type C leak testing. Bases for exemptions and exceptions from type C leakage rate testing on a penetration by penetration basis are as follows:

I. <u>Exemptions</u>

1. Feedwater Bypass - X-8A, X-8B, X-8C, X-8D
Feedwater - X-12A, X-12B, X-12C, X-12D
Main Steam - X-13A, X-13B, X-13C, X-13D
Steam Generator Blowdown - X-14A, X-14B, X-14C, X-14D
Steam Generator Blowdown Sample Line - X-27A, X-27B, X-27C, X-27D
Auxiliary Feedwater - X-40A, X-40B

These penetrations are directly connected to the secondary side of the steam generator. The main steam and feedwater lines of PWR containments are not required to be Type C tested (see definition of Type C test in 10 CFR 50, Appendix J). These lines are assumed not to rupture as a result of an accident (missile protected). Any leakage through these lines would be identified during operation by the leakage detection program. In addition, during a design basis accident, the secondary side of the steam generator is filled with water and would be at a higher pressure than the containment atmosphere thus preventing outleakage from containment. The integrity of the inside piping is also verified during the Type A test.

2. <u>CVCS Normal Charging Line, X-16</u>

   This penetration uses an inboard check valve and a closed loop outside containment (CLOC) as the means of containment isolation. Type C testing for this path is not required due to the seal pressure greater than 1.1 P_a and the 30-day water seal inventory. A positive pressure preventing air outleakage is assured by the pressure applied against FCV-62-90 and FCV-62-91 (both of which receive a safety injection signal to close) by the high head SI pumps. Water testing for piping integrity is performed in accordance with ASME XI.

3. <u>RHR Hot Leg Injection, X-17</u>

   This penetration uses inboard containment isolation valves and a CLOC for containment boundaries. Type C testing for this penetration is not required since a continuous water seal will be provided at a pressure greater than 1.1 P_a and a 30-day water seal is provided. Water testing for piping integrity is performed in accordance with ASME XI.

4. <u>RHR Cold Leg Injection, X-20A, X-20B</u>

   Same as for X-17.

5. <u>SIS Hot Leg Injection, X-21, X-32</u>

   These lines make use of inboard containment isolation valves and a CLOC for containment boundaries. These lines are postulated to be in-service post accident and the high head pumps will maintain a pressure seal greater than 1.1 P_a for greater than 30 days.

6. <u>Charging Pump Discharge, X-22</u>

   This line makes use of inboard containment isolation valves and a CLOC for containment boundaries. Type C testing is not required for the same reasons as X-21 and X-32. Water seal is provided by the high head pumps.

7. <u>SIS Cold Leg Injection, X-33</u>

   Same as X-21 and X-32

8. <u>RCP Seal Injection, X-43A, X-43B, X-43C, X-43D</u>

These lines make use of an inboard containment isolation valve and a CLOC for containment boundaries. Type C test is not required for the same reason as given for X-16.

II. Exceptions

1. Sump Suction to RHR, X-19A, X-19B

These lines make use of a containment isolation valve located outside of containment and a CLOC for containment isolation boundaries. During a design basis accident, these lines would be submerged under water which would preclude air outleakage. In addition, these valves are exposed to P_a during each Type A test.

2. Relief Valve Discharge, X-24

The line uses an inboard containment isolation valve and a CLOC for containment boundaries. The maximum calculated containment accident pressure per 10 CFR 50 Appendix J (P_a) during the design basis accident would not create a substantial outleakage driving force and, in any case, tends to cause the relief valves to seat rather than lift. The systems feeding this line are ECCS and, due either to operating pressure or static head, outleakage would be prevented. Therefore, no Type C testing will be performed for this line. However, this line is exposed to the P_a test pressure during the Type A test.

3. Containment Spray, X-48A, X-48B and RHR Spray, X-49A, X-49B

These lines make use of an inboard containment isolation check valve and a CLOC for containment boundaries of each line. Additionally, a water leg seal exists against a system valve outside containment in each line. This seal prevents the outleakage of containment atmosphere. An inventory test is performed to ensure a 30-day seal water inventory at a pressure greater than 1.1 P_a, should one spray system shut down. The seal water inventory leakage rate test is performed in lieu of a Type C air leakage rate test of the containment isolation check valves.

4. RHR Pump Supply, X-107

This line makes use of inboard containment isolation valves and a CLOC for containment boundaries. This penetration satisfies ANSI-N271-1976. In addition, an ASME Section XI water leakage test is performed to verify system integrity. Thus, no Type C test is required for this line.

Test connections and pressurizing means are provided to test isolation valves or barriers for leaktightness. Either air or nitrogen is used as the pressurizing medium, depending on the physical location and service of each line. Leak testing of individual valves and penetrations is accomplished by one of the following methods:

1. Method 1, Pressure Decay

The test volume initially shall be pressurized with air or nitrogen to a test pressure greater than the pressure under accident conditions, Pa. The final test pressure shall be

greater than Pa. The pressure and temperature shall be recorded at the start and end of the test. The leakage rate shall be calculated from the following formula:

$$L_i = \frac{TV}{t}\left[\frac{P_1}{T_1} - \frac{P_2}{T_2}\right]\frac{T_{stp}}{P_{stp}}$$

Where:

L_i	=	Local leak rate, cfm
TV	=	Test volume, ft^3
P_1	=	Initial pressure, psia
P_2	=	Final pressure, psia
T_1	=	Initial temperature, °R
T_2	=	Final temperature, °R
T_{stp}	=	Standard atmospheric temperature, °R
P_{stp}	=	Standard atmospheric pressure, psia
t	=	Test duration, min.

2. <u>Method 2, Makeup Flow Rate</u>

The test volume shall be pressurized and maintained to at least 15 psig using a pressure regulator to maintain test pressure. Makeup fluid flow to the test volume required to maintain test pressure shall be used as the leakage rate of the barrier under test.

The acceptance criteria for the Type C test are given in the Technical Specification which complies with 10 CFR 50 Appendix J, Option B.

6.2.6.3 Scheduling and Reporting of Periodic Tests

Type A integrated containment leakage tests are performed prior to operation of the plant and subsequently in accordance with the Containment Leakage Rate Testing Program.

Type B and Type C leakage rate tests are performed in accordance with the Containment Leakage Rate Testing Program.

Test reports are made in accordance with 10 CFR 50 Appendix J, Option B.

REFERENCES

1. Title 10, Code of Federal Regulations, Part 50, Appendix J, Option B, “Primary Reactor

Containment Leakage Testing for Water-Cooled Power Reactors - Performance-Based Requirements."

2. Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," September 1995.

3. NEI 94-01, Revision 0, "Industry Guideline for Implementing Performance-Based Option of 10CFR Part 50, Appendix J."

WBN

TABLE 6.2.6-1 (Sheet 1 of 3)

PENETRATIONS SUBJECTED TO TYPE B TESTING

Penetration	Description
X-121E	RCP No. 1 - Non-Div.
X-122E	RCP No. 2 - Non-Div.
X-123E	RCP No. 3 - Non-Div.
X-124E	RCP No. 4 - Non-Div.
X-125E	480V Power - Non-Div.
X-126E	480V Power A
X-127E	480V Power B
X-128E	480V Power A
X-129E	480V Power B
X-130E	Control - Non-Div.
X-131E	480V Power - Non-Div.
X-132E	Control Rod Drive Power
X-133E	Control Rod Drive Power
X-134E	480V Power A
X-135E	480V Power A
X-136E	480V Power B
X-137E	480V Power B
X-138E	Low Level - Non-Div.
X-139E	Process Instr. Protection
X-140E	Incore Instrumentation
X-141E	480V Power A
X-142E	Incore Instrumentation
X-143E	NIS Channel III
X-144E	480V Power - Non-Div.
X-145E	Control Rod Pos. Detection
X-146E	Control Rod Drive Power
X-147E	Control A
X-148E	Process Inst. Control
X-149E	Low Level - Non-Div.
X-150E	Annunciation and Communication
X-151E	NIS Channel IV
X-152E	480V Power - Non-Div.
X-153E	Low Level - Non-Div.
X-154E	Process Inst. Control
X-155E	480V Power - Non-Div.
X-156E	Control B
X-157E	Annunciation
X-158E	Process Inst. Protection
X-159E	Process Inst. Control
X-160E	Annunciation and Communication
X-161E	480V Power - Non-Div.
X-163E	NIS Channel I
X-164E	Control A

TABLE 6.2.6-1 (Sheet 2 of 3)

PENETRATIONS SUBJECTED TO TYPE B TESTING

Penetration	Description
X-165E	Process Inst. Protection
X-166E	Control - Non-Div.
X-167E	480V Power - Non-Div.
X-168E	Control - Non-Div.
X-169E	Process Inst. Protection
X-170E	Process Inst. Control
X-171E	Control - Non-Div.
X-172E	Control B
X-173E	Control - Non-Div.
X-174E	NIS Channel II
X-1	Equipment Hatch (Resilient Seal)
X-2A	Personnel Hatch (Resilient Seal)
X-2B	Personnel Hatch (Resilient Seal)
X-3	Fuel Transfer Tube (Resilient Seal)
X-54	Thimble Renewal (Resilient Seal)
X-79A	Ice Blowing (Resilient Seal)
X-13A	Main Steam Line (Bellows)
X-13B	Main Steam Line (Bellows)
X-13C	Main Steam Line (Bellows)
X-13D	Main Steam Line (Bellows)
X-12A	Main Feedwater Line (Bellows)
X-12B	Main Feedwater Line (Bellows)
X-12C	Main Feedwater Line (Bellows)
X-12D	Main Feedwater Line (Bellows)
X-17	RHR Pump Return Line (Bellows)
X-107	RHR Pump Supply Line (Bellows)
X-14A	Steam Generator Blowdown (Bellows)
X-14B	Steam Generator Blowdown (Bellows)
X-14C	Steam Generator Blowdown (Bellows)
X-14D	Steam Generator Blowdown (Bellows)
X-15	Chemical and Volume Control System (Bellows)
X-20A	Low Head Safety Injection System (Bellows)
X-20B	Low Head Safety Injection System (Bellows)
X-21	Safety Injection Hot Legs (Bellows)
X-22	BIT Charging Pump Discharge (Bellows)
X-24	SIS Relief Valve Discharge (Bellows)
X-40D	Hydrogen Purge (Resilient Seal)
X-79B	Ice Blowing (Resilient Seal)
X-8A	Feedwater Bypass (Bellows)
X-8B	Feedwater Bypass (Bellows)
X-8C	Feedwater Bypass (Bellows)
X-8D	Feedwater Bypass (Bellows)
X-30	Accum. To Holdup Tank (Bellows)
X-32	High Head Safety Injection System (Bellows)
X-33	High Head Safety Injection System (Bellows)

TABLE 6.2.6-1 (Sheet 3 of 3)

PENETRATIONS SUBJECTED TO TYPE B TESTING

Penetration	Description
X-45	RC Drain Tank (Bellows)
X-46	RC Drain Tank (Bellows)
X-47A	Glycol (Bellows)
X-47B	Glycol (Bellows)
X-81	RC Drain Tank to Anal. (Bellows)
X-108	Testable Spare (Resilient Seal)
X-108	Maintenance Port (Bellows)
X-109	Testable Spare (Resilient Seal)
X-109	Maintenance Port (Bellows)
X-36	Steam Generator Cleanup (Resilient Seal)
X-37	Maintenance Port (Resilient Seal)
X-117	Maintenance Port (Resilient Seal)
X-118	Layup Water (Resilient Seal)

TABLE 6.2.6-2a (Sheet 1 of 7)

CONTAINMENT ISOLATION VALVES SUBJECTED TO TYPE C TESTING

Penetration No.	Isolation Valve No.	Description
X-4	30-56(1) 30-57 30-555(1)	Lower Compt Purge Air Exhaust
X-5	30-58(1) 30-59 30-554(1)	Inst Room Purge Air Exhaust
X-6	30-50(1) 30-51 30-558(1)	Upper Compt Purge Air Exhaust
X-7	30-52(1) 30-53 30-557(1)	Upper Compt Purge Air Exhaust
X-9A	30-7 30-8(1) 30-563(1)	Upper Compt Purge Air Supply
X-9B	30-9 30-10(1) 30-562(1)	Upper Compt Purge Air Supply
X-10A	30-14 30-15(1) 30-561(1)	Lower Compt Purge Air Supply
X-10B	30-16 30-17(1) 30-560(1)	Lower Compt Purge Air Supply
X-11	30-19 30-20(1) 30-559(1)	Inst Room Purge Air Supply
X-15	62-72 62-73 62-74 62-76 62-77 62-662(1)	Chem and Vol System Letdown
X-23	43-318 (Unit 1) 43-319 (Unit 1)	PAS Cont Air Intk TR-B
X-25A	43-11 43-12	Pressurizer Liquid Sample

TABLE 6.2.6-2a (Sheet 2 of 7)

CONTAINMENT ISOLATION VALVES SUBJECTED TO TYPE C TESTING

Penetration No.	Isolation Valve No.	Description
X-25D	43-2 43-3	Pressurizer Steam Sample
X-26B	52-500 52-504	ILRT Sensor Line
X-26A	52-501 52-505	ILRT Sensor Line
X-28	43-341 (Unit 1) 43-834 (Unit 1)	PAS Cont Sump Rtrn TR-B
X-29	70-89 70-92 70-698	CCS from RC Pump Coolers
X-30	63-71 63-84 63-23 63-28	Accum to Holdup Tank
X-31	26-243 26-1296	Fire Protection
X-34	32-110 (Unit 1) 32-111(Unit 2) 32-288 32-293	Control Air
X-35	70-85 70-703(1)	CCS from Excess Letdn HX CCS to Excess Letdn HX
X-39A	63-64 63-868	N_2 to Accumulators

TABLE 6.2.6-2a (Sheet 3 of 7)

CONTAINMENT ISOLATION VALVES SUBJECTED TO TYPE C TESTING

Penetration No.	Isolation Valve No.	Description
X-39B	68-305 68-849	N_2 to Press Relief Tank
X-41	77-127 77-128 77-2875	Floor Sump Pump Disch
X-42	81-12 81-502	Press Rel Tank Makeup
X-44	62-61 62-63 62-639	From RC Pump Seals
X-45	77-18 77-19 77-20	RC Drain Tank and Prt to VH
X-46	77-9 77-10 84-530	RC Drain Tank Pump Disch
X-47A	61-191 61-192 61-533	Glycol Supply
X-47B	61-193 61-194 61-680	Glycol Return
X-50A	70-87 70-90 70-687	RCP Therm Barrier Return
X-50B	70-679 70-134	RCP Therm Barrier Supply
X-52	70-140 1-70-100 1-70-790	CCS to RC Pump Coolers
X-53	70-143	CCS to Excess Letdown HX
X-56A	67-107 1-67-113 1-67-1054D	Lower Cont ERCW Supply

TABLE 6.2.6-2a (Sheet 4 of 7)

CONTAINMENT ISOLATION VALVES SUBJECTED TO TYPE C TESTING

Penetration No.	Isolation Valve No.	Description
X-57A	67-111 67-112 67-575D	Lower Cont ERCW Return
X-58A	67-83 67-89 67-1054A	Lower Cont ERCW Supply
X-59A	67-87 67-88 67-575A	Lower Cont ERCW Return
X-60A	67-99 67-105 67-1054B	Lower Cont ERCW Supply
X-61A	67-103 67-104 67-575B	Lower Cont ERCW Return
X-62A	67-91 67-97 67-1054C	Lower Cont ERCW Supply
X-63A	67-95 67-96 67-575C	Lower Cont ERCW Return
X-68	67-141 67-580D	Upper Cont ERCW Supply
X-69	67-130 67-580A	Upper Cont ERCW Supply

TABLE 6.2.6-2a (Sheet 5 of 7)

CONTAINMENT ISOLATION VALVES SUBJECTED TO TYPE C TESTING

Penetration No.	Isolation Valve No.	Description
X-70	67-139 67-297 67-585B	Upper Cont ERCW Return
X-71	67-134 67-296 67-585C	Upper Cont ERCW Return
X-72	67-142 67-298 67-585D	Upper Cont ERCW Return
X-73	67-131 67-295 67-585A	Upper Cont ERCW Return
X-74	67-138 67-580B	Upper Cont ERCW Supply
X-75	67-133 67-580C	Upper Cont ERCW Supply
X-76	33-713 (Unit 1) 33-714 (Unit 1) 33-732 (Unit 2) 33-733 (Unit 2)	Service Air
X-77	59-522 59-698	Demineralized Water
X-78	26-240 26-1260	Fire Protection
X-80	30-37 30-40 30-556[(1)]	Lower Comp Press Relief
X-81	77-16 77-17	RC Drain Tk to Gas Analyzer
X-82	78-560 78-561	Refuel Cav Purification Pump Suction
X-83	78-557 78-558	Refuel Cav Purification Pump Discharge
X-84A	68-307 68-308	Prt to Gas Analyzer
X-85A	43-75 43-77	Ex Lt Dn Hx to Boron Anal

TABLE 6.2.6-2a (Sheet 6 of 7)

CONTAINMENT ISOLATION VALVES SUBJECTED TO TYPE C TESTING

Penetration No.	Isolation Valve No.	Description
X-85B	43-22 43-23	Hot Leg Sample - Loops 1 & 3
X-86A	43-288 (Unit 1) 43-287 (Unit 1)	PAS Cont Air Intk TR-A
X-86B	43-883 (Unit 1) 43-307 (Unit 1)	PAS Cont Air Rtrn TR-A
X-86C	43-342 43-841	PAS Cont Sump Rtrn TR-A
X-90	32-102 32-308 32-313	Control Air TR-B
X-91	32-80 (Unit 1) 32-81 (Unit 2) 32-298 (Unit 1) 32-328 (Unit 2) 32-303 (Unit 1) 32-333 (Unit 2)	Control Air TR-A
X-92A	43-207 (Unit 1) 43-435 (unit 1)	Hydrogen Analyzer TR-B
X-92B	43-208 (Unit 1) 43-436 (Unit 1)	Hydrogen Analyzer TR-B
X-92C	43-250 (Unit 1) 43-251 (Unit 1)	PAS Hot Leg 1 TR-A
X-93	43-34 43-35	Accum Sample
X-94B	90-110 90-111	Upper Comp Air Mon Intake
X-94C	90-107 90-108 90-109	Upper Comp Air Mon Return
X-95B	90-116 90-117	Lower Comp Air Mon Intake
X-95C	90-113 90-114 90-115	Lower Comp Air Mon Return

TABLE 6.2.6-2a (Sheet 7 of 7)

CONTAINMENT ISOLATION VALVES SUBJECTED TO TYPE C TESTING

Penetration No.	Isolation Valve No.	Description
X-96A	52-506 52-502	ILRT Sensor Line
X-96B	52-507 52-503	ILRT Sensor Line
X-97	30-134 (Unit 1) 30-135 (Unit 1)	Containment ΔP Sensor
X-99	43-202 43-434	Hydrogen Analyzer TR-A
X-100	43-201 43-433	Hydrogen Analyzer TR-A
X-105	43-325 (Unit 1) 43-884 (Unit 1)	PAS Cont Air Rtrn TR-B
X-106	43-310 (Unit 1) 43-309 (Unit 1)	PAS Hot Leg 3 TR-B
X-114	61-110 61-122 61-745	Glycol Floor Cooling (From)
X-115	61-96 61-97 61-692	Glycol Floor Cooling (To)

Notes:

(1) These isolation valves are leakage rate tested in the reverse direction. This is acceptable since the results are equivalent or more conservative.

TABLE 6.2.6-2b (Sheet 1 of 5)

VALVES EXEMPTED FROM TYPE C LEAK TESTING

Penetration No.	Valve No.	System	Justification
X-8A	FCV 3-236	Feedwater Bypass	Note 1
	LCV 3-164	Auxiliary Feedwater	Note 1
	LCV 3-174	Auxiliary Feedwater	Note 1
	LCV 3-164A	Auxiliary Feedwater	Note 1
X-8B	FCV 3-239	Feedwater Bypass	Note 1
X-8C	FCV 3-242	Feedwater Bypass	Note 1
X-8D	FCV 3-245	Feedwater Bypass	Note 1
	LCV 3-171	Auxiliary Feedwater	Note 1
	LCV 3-175	Auxiliary Feedwater	Note 1
	LCV 3-171A	Auxiliary Feedwater	Note 1
X-12A	FCV 3-33	Main Feedwater	Note 1
X-12B	FCV 3-47	Main Feedwater	Note 1
X-12C	FCV 3-87	Main Feedwater	Note 1
X-12D	FCV 3-100	Main Feedwater	Note 1
X-13A	FCV 1-4	Main Steam	Note 1
	FCV 1-147	Main Steam	Note 1
	FCV 1-15	Main Steam	Note 1
	DRV 1-536	Main Steam	Note 1
	PCV 1-5	Main Steam	Note 1
	SFV 1-522	Main Steam	Note 1
	SFV 1-523	Main Steam	Note 1
	SFV 1-524	Main Steam	Note 1
	SFV 1-525	Main Steam	Note 1
	SFV 1-526	Main Steam	Note 1
X-13B	FCV 1-11	Main Steam	Note 1
	FCV 1-148	Main Steam	Note 1
	PCV 1-12	Main Steam	Note 1
	DRV 1-534	Main Steam	Note 1
	SFV 1-517	Main Steam	Note 1
	SFV 1-518	Main Steam	Note 1
	SFV 1-519	Main Steam	Note 1
	SFV 1-520	Main Steam	Note 1
	SFV 1-521	Main Steam	Note 1
X-13C	FCV 1-22	Main Steam	Note 1
	PCV 1-23	Main Steam	Note 1
	FCV 1-149	Main Steam	Note 1
	DRV 1-532	Main Steam	Note 1
	SFV 1-512	Main Steam	Note 1
	SFV 1-513	Main Steam	Note 1
	SFV 1-514	Main Steam	Note 1
	SFV 1-515	Main Steam	Note 1
	SFV 1-516	Main Steam	Note 1
X-13D	FCV 1-29	Main Steam	Note 1
	FCV 1-150	Main Steam	Note 1
	PCV 1-30	Main Steam	Note 1
	DRV 1-538	Main Steam	Note 1
	FCV 1-16	Main Steam	Note 1

TABLE 6.2.6-2b (Sheet 2 of 5)

VALVES EXEMPTED FROM TYPE C LEAK TESTING

Penetration No.	Valve No.	System	Justification
	SFV 1-527	Main Steam	Note 1
	SFV 1-528	Main Steam	Note 1
	SFV 1-529	Main Steam	Note 1
	FCV 1-530	Main Steam	Note 1
	SFV 1-531	Main Steam	Note 1
X-14A	FCV 1-14	Steam Generator Blowdown	Note 1
X-14B	FCV 1-32	Steam Generator Blowdown	Note 1
X-14C	FCV 1-25	Steam Generator Blowdown	Note 1
X-14D	FCV 1-7	Steam Generator Blowdown	Note 1
X-16	62-543*	Normal Charging	Note 3
X-17	63-640*	RHR (Low HD SIS)	Note 2
	63-643*	RHR (Low HD SIS)	Note 2
	FCV 63-158	RHR (Low HD SIS)	Note 2
X-19A	FCV 63-72	SIS (Sump Suction)	Note 5
	FCV 72-44	CS (Sump Suction)	Note 5
X-19B	FCV 63-73	SIS (Sump Suction)	Note 5
	FCV 72-45	CS (Sump Suction)	Note 5
X-20A	FCV 63-112	RHR (Low HD SIS)	Note 2
	63-633*	RHR (Low HD SIS)	Note 2
	63-635*	RHR (Low HD SIS)	Note 2
X-20B	FCV 63-111	RHR (Low HD SIS)	Note 2
	63-632*	RHR (Low HD SIS)	Note 2
	63-634*	RHR (Low HD SIS)	Note 2
X-21	FCV 63-167	SIS (Safety Injection)	Note 6
	63-547*		Note 6
	63-549*		Note 6
X-22	FCV 63-174	SIS (Charging)	Note 7
	63-581*	SIS (Charging)	Note 7
X-24	68-559*	Reactor Coolant	Note 8
X-25B	30-42C1 30-42C2	Containment ΔP Sensor Unit 1 Only	Note 11
X-25C	30-30CC1 30-30CC2	Containment ΔP Sensor Unit 1 Only	Note 11
X-26C	30-43C1 30-43C2 30-310C1 30-310C2	Containment ΔP Sensor Unit 1 Only	Note 11
X-27A	FCV 43-55	SG Blowdown Sample	Note 1

TABLE 6.2.6-2b (Sheet 3 of 5)

VALVES EXEMPTED FROM TYPE C LEAK TESTING

Penetration No.	Valve No.	System	Justification
		Line	
X-27B	FCV 43-58	SG Blowdown Sample Line	Note 1
X-27C	FCV 43-61	SG Blowdown Sample Line	Note 1
X-27D	FCV 43-64	SG Blowdown Sample Line	Note 1
X-32	FCV 63-21 63-543* 63-545*	SIS (Safety Injection) SIS (Safety Injection) SIS (Safety Injection)	Note 6 Note 6 Note 6
X-33	FCV 63-121 63-551 63-553 63-555 63-557	SIS (Safety Injection)	Note 6 Note 6 Note 6 Note 6 Note 6
X-40A	LCV 3-156 LCV 3-173 LCV 3-156A	Auxiliary Feedwater Auxiliary Feedwater Auxiliary Feedwater	Note 1 Note 1 Note 1
X-40B	LCV 3-148 LCV 3-172 LCV 3-148A	Auxiliary Feedwater Auxiliary Feedwater Auxiliary Feedwater	Note 1 Note 1 Note 1
X-43A	62-562*	CVCS (Pump Seal Injection)	Note 9
X-43B	62-561*	CVCS (Pump Seal Injection)	Note 9
X-43C	62-563*	CVCS (Pump Seal Injection)	Note 9
X-43D	62-560*	CVCS (Pump Seal Injection)	Note 9
X-48A	72-548*	Containment Spray	Note 10
X-48B	72-549*	Containment Spray	Note 10
X-49A	72-562*	RHR (RHR Spray)	Note 10
X-49B	72-563*	RHR (RHR Spray)	Note 10
X-85C	30-45C1 30-45C2	Containment ΔP Sensor Unit 1 Only	Note 11
X-96C	30-44C1 30-44C2 30-311C1 30-311C2	Containment ΔP Sensor Unit 1 Only	Note 11
X-107	FCV 74-2 FCV 74-8 RFV 74-505 FCV 63-185	RHR RHR RHR SIS	Note 4 Note 4 Note 4 Note 4

TABLE 6.2.6-2b (Sheet 4 of 5)

VALVES EXEMPTED FROM TYPE C LEAK TESTING

*Check Valve

Note 1. This penetration is directly connected to the secondary side of the steam generator. The main steam, feedwater, and steam generator blowdown lines of PWR containments are not required to be tested (see definition of Type C test in 10 CFR 50, Appendix J). These lines are assumed not to rupture as a result of an accident (missile Protected). Any leakage through these lines would be identified during operation by the leakage detection program. In addition, during a design basis accident, the secondary side would be at a higher pressure than the containment atmosphere, thus preventing outleakage from containment. The integrity of the inside piping is also verified during the Type A test.

Note 2. This penetration uses inboard containment isolation valves and a CLOC for containment boundaries. Type C testing for this penetration is not required since a continuous water seal will be provided at a pressure greater than 1.1 P_a and a guaranteed 30-day water inventory. Water testing for piping integrity will be performed in accordance with ASME XI.

Note 3. This penetration uses an inboard check valve and a closed loop outside containment (CLOC) as the means of containment isolation. Type C testing for this path is not required due to the presence of a 1.1 P_a pressure and a 30-day inventory criteria as specified in 10 CFR 50, Appendix J. A positive pressure preventing air outleakage is assured by the pressure applied against FCV-62-90 and FCV-62-91 (both of which receive a phase A signal) by the high-head SI pumps. Water testing for piping integrity is performed in accordance with ASME XI.

Note 4. This line makes use of inboard containment isolation valves an a CLOC for containment boundaries. This penetration satisfies ANSI-N271-1976. In addition, an ASME Section XI water leakage test will be performed to verify system integrity.

Note 5. This line makes use of a containment isolation valve located outside of containment and a CLOC for containment isolation boundaries. During a design basis accident, this line would be submerged under water which would preclude air outleakage. In addition, these valves are exposed to P_a during each Type A test.

Note 6. This line makes use of inboard containment isolation valves and a CLOC for containment boundaries. These lines are postulated to be inservice post-accident and when not in use, the pumps maintain a pressure seal greater than 1.1 P_a.

Note 7. This line makes use of inboard containment isolation valves and a CLOC for containment boundaries. Type C testing is not required for the same reasons as X-21 and X-32 as stated in Note 6. Water seal is provided by the high-head pumps.

Note 8. This line uses an inboard containment isolation valve and a CLOC for containment boundaries. P_a during the design basis accident would not create a substantial outleakage driving force and, in any case, tend to cause the relief valves in the CLOC to seat rather than lift. The systems feeding this line are ECCS and, due either to operating pressure or static head outleakage would be prevented. This line is exposed to the P_a test pressure during the Type A test.

Note 9. This line makes use of an inboard containment isolation valve and a CLOC for containment boundaries. Type C testing is not required for the same reason as given for X-16 in Note 3.

TABLE 6.2.6-2b (Sheet 5 of 5)

VALVES EXEMPTED FROM TYPE C LEAK TESTING

Note 10. This line makes use of an inboard containment isolation valve and a CLOC for containment boundaries. An inventory test is performed to ensure a 30-day inventory at a pressure greater than 1.1 P_a exists, should one spray system shut down. This will prevent outleakage of containment atmosphere.

Note 11. This instrument line has a CLOC for containment boundary. This design is required as discussed in Section 6.2.4. This instrument is tested at P_a during the Type A test. (Unit 1 Only)

TABLE 6.2.6-3 (Sheet 1 of 7)

CONTAINMENT VESSEL PRESSURE AND LEAK TEST
REACTOR BUILDING CONTAINMENT PENETRATION STATUS

A. PENETRATION STATUS DURING TEST PERFORMANCE

Penetration	Description	Status
X-1	Equipment Hatch	Closed
X-2A	Elevation 719'- 4" Air Lock	Closed
X-2B	Elevation 760'- 4" Air Lock	Closed
X-3	Fuel Transfer Tube	Closed
X-4	Heating and Ventilating Air Flow	Vented
X-5	Heating and Ventilating Air Flow	Vented
X-6	Heating and Ventilating Air Flow	Vented
X-7	Heating and Ventilating Air Flow	Vented
X-8	Seal Welded Spare	Vented (see Note 1)
X-8A	Feedwater Bypass	Normal Lineup
X-8B	Feedwater Bypass	Normal Lineup
X-8C	Feedwater Bypass	Normal Lineup
X-8D	Feedwater Bypass	Normal Lineup
X-9A	Heating and Ventilating Air Flow	Vented
X-9B	Heating and Ventilating Air Flow	Vented
X-10A	Heating and Ventilating Air Flow	Vented
X-10B	Heating and Ventilating Air Flow	Vented
X-11	Heating and Ventilating Air Flow	Vented
X-12A	Feedwater System	Normal Lineup
X-12B	Feedwater System	Normal Lineup
X-12C	Feedwater System	Normal Lineup
X-12D	Feedwater System	Normal Lineup
X-13A	Main and Reheat Steam System	Normal Lineup
X-13B	Main and Reheat Steam System	Normal Lineup
X-13C	Main and Reheat Steam System	Normal Lineup
X-13D	Main and Reheat Steam System	Normal Lineup
X-14A	Steam Generator Blowdown System	Normal Lineup
X-14B	Steam Generator Blowdown System	Normal Lineup
X-14C	Steam Generator Blowdown System	Normal Lineup
X-14D	Steam Generator Blowdown System	Normal Lineup
X-15	Chemical and Volume Control System	See Note 3
X-16	Chemical and Volume Control System	Normal Lineup
X-17	Residual Heat Removal System	Normal Lineup
X-18	Seal Welded Spare	Vented (see Note 1)
X-19A	Safety Injection System	Normal Lineup
X-19B	Safety Injection System	Normal Lineup
X-20A	Safety Injection System	Normal Lineup
X-20B	Safety Injection System	Normal Lineup
X-21	Safety Injection System	Normal Lineup
X-22	Safety Injection System	Normal Lineup
X-23	PAS Cont. Air Intk LC Tr-B	Vented
X-24	Reactor Coolant System	Vented
X-25A	Radiation System	See Note 3
X-25B	Containment Delta P Sensor (PdT30-42)	Vented
X-25C	Containment Delta P Sensor (PdT30-30c)	Vented

TABLE 6.2.6-3 (Sheet 2 of 7)

CONTAINMENT VESSEL PRESSURE AND LEAK TEST
REACTOR BUILDING CONTAINMENT PENETRATION STATUS

A. PENETRATION STATUS DURING TEST PERFORMANCE

Penetration	Description	Status
X-25D	Radiation Sampling System	See Note 3
X-26A	ILRT Sensor Line	In Use (see Note 2)
X-26B	ILRT Sensor Line	In Use (see Note 2)
X-26C	Containment Delta P Sensor (PdT30-43)	Vented
X-27A	Radiation Sampling System	Normal Lineup
X-27B	Radiation Sampling System	Normal Lineup
X-27C	Radiation Sampling System	Normal Lineup
X-27D	Radiation Sampling System	Normal Lineup
X-28	PAS Cont. Sump Return Tr-B	See Note 3
X-29	Component Cooling System	See Note 3
X-30	Safety Injection System	See Note 3
X-31	Fire Protection	See Note 3
X-32	Safety Injection System	Normal Lineup
X-33	Safety Injection System	Normal Lineup
X-34	Control Air System	Vented
X-35	Component Cooling Water System	See Note 3
X-36	SG Chem. Cleaning	Vented (see Note 1)
X-37	Maintenance Port	Vented (see Note 1)
X-38	Seal Welded Spare	Vented (see Note 1)
X-39A	Waste Disposal System	Vented
X-39B	Waste Disposal System	Vented
X-39C	Seal Welded Spare	Vented (see Note 1)
X-39D	Seal Welded Spare	Vented (see Note 1)
X-40A	Auxiliary Feedwater System	Normal Lineup
X-40B	Auxiliary Feedwater System	Normal Lineup
X-40C	Seal Welded Spare	Vented (see Note 1)
X-40D	Hydrogen Purge	Vented
X-41	Waste Disposal System	See Note 3
X-42	Primary Water System	See Note 3
X-43A	Chemical and Volume Control System	Normal Lineup
X-43B	Chemical and Volume Control System	Normal Lineup
X-43C	Chemical and Volume Control System	Normal Lineup
X-43D	Chemical and Volume Control System	Normal Lineup
X-44	Chemical and Volume Control System	See Note 3
X-45	Waste Disposal System	Vented
X-46	Waste Disposal System	See Note 3
X-47A	Ice Condenser System	In Use
X-47B	Ice Condenser System	In Use
X-48A	Containment Spray System	Normal Lineup
X-48B	Containment Spray System	Normal Lineup
X-49A	Containment Spray System	Normal Lineup
X-49B	Containment Spray System	Normal Lineup
X-50A	Component Cooling System	See Note 3
X-50B	Component Cooling System	See Note 3
X-51	Seal Welded Spare	Vented (see Note 1)

TABLE 6.2.6-3 (Sheet 3 of 7)

CONTAINMENT VESSEL PRESSURE AND LEAK TEST
REACTOR BUILDING CONTAINMENT PENETRATION STATUS

A. PENETRATION STATUS DURING TEST PERFORMANCE

Penetration	Description	Status
X-52	Component Cooling System	See Note 3
X-53	Component Cooling System	See Note 3
X-54	Thimble Renewal	In Use
X-55	Seal Welded Spare	Vented (see Note 1)
X-56A	Essential Raw Cooling Water	See Note 3
X-56B	Seal Welded Spare	Vented (see Note 1)
X-57A	Essential Raw Cooling Water	See Note 3
X-57B	Seal Welded Spare	Vented (see Note 1)
X-58A	Essential Raw Cooling Water	See Note 3
X-58B	RCS Pressure Sensor	Normal Lineup
X-59A	Essential Raw Cooling Water	See Note 3
X-59B	Seal Welded Spare	Vented (see Note 1)
X-60A	Essential Raw Cooling Water	See Note 3
X-60B	Seal Welded Spare	Vented (see Note 1)
X-61A	Essential Raw Cooling Water	See Note 3
X-61B	Seal Welded Spare	Vented (see Note 1)
X-62A	Essential Raw Cooling Water	See Note 3
X-62B	Seal Welded Spare	Vented (see Note 1)
X-63A	Essential Raw Cooling Water	See Note 3
X-63B	Seal Welded Spare	Vented (see Note 1)
X-64	Air Conditioning System	See Note 4
X-65	Air Conditioning System	See Note 4
X-66	Air Conditioning System	See Note 4
X-67	Air Conditioning System	See Note 4
X-68	Essential Raw Cooling Water	See Note 3
X-69	Essential Raw Cooling Water	See Note 3
X-70	Essential Raw Cooling Water	See Note 3
X-71	Essential Raw Cooling Water	See Note 3
X-72	Essential Raw Cooling Water	See Note 3
X-73	Essential Raw Cooling Water	See Note 3
X-74	Essential Raw Cooling Water	See Note 3
X-75	Essential Raw Cooling Water	See Note 3
X-76	Control and Service Air System	Vented
X-77	Demineralized Water and Cask Decon	See Note 3
X-78	Fire Protection	See Note 3
X-79A	Ice Blowing	Vented
X-79B	Negative Return	Vented
X-80	Heating and Ventilating Air Flow	Vented
X-81	Waste Disposal System	See Note 3
X-82	Fuel Pool Cooling and Cleaning System	See Note 3
X-83	Fuel Pool Cooling and Cleaning System	See Note 3
X-84A	Radiation Sampling System	See Note 3
X-84B	RVLIS	Normal Lineup
X-84C	RVLIS	Normal Lineup
X-84D	RVLIS	Normal Lineup

TABLE 6.2.6-3 (Sheet 4 of 7)

CONTAINMENT VESSEL PRESSURE AND LEAK TEST
REACTOR BUILDING CONTAINMENT PENETRATION STATUS

A. PENETRATION STATUS DURING TEST PERFORMANCE

Penetration	Description	Status
X-85A	Radiation Sampling System	See Note 3
X-85B	Radiation Sampling System	See Note 3
X-85C	Containment Delta P Sensor (PdT30-45)	Vented
X-85D	Seal Welded Spare	Vented (see Note 1)
X-86A	PAS Cont. Air Intk UC Tr-A	Vented
X-86B	PAS Cont. Air Rtrn Tr-A	Vented
X-86C	PAS Cont. Sump Rtrn Tr-A	See Note 3
X-86D	Seal Welded Spare	Vented (see Note 1)
X-87A	Seal Welded Spare	Vented (see Note 1)
X-87B	RVLIS	Normal Lineup
X-87C	RVLIS	Normal Lineup
X-87D	RVLIS	Normal Lineup
X-88	Seal Welded Spare	Vented (see Note 1)
X-89	Seal Welded Spare	Vented (see Note 1)
X-90	Control Air System Tr-B	Vented
X-91	Control Air System Tr-A	Vented
X-92A	Hydrogen Analyzer Tr-B	Vented
X-92B	Hydrogen Analyzer Tr-B	Vented
X-92C	PAS Hot Leg 1 Tr-A	See Note 3
X-92D	Seal Welded Spare	Vented (see Note 1)
X-93	Radiation Sampling System	See Note 3
X-94A	Seal Welded Spare	Vented (see Note 1)
X-94B	Radiation Monitoring System	Vented
X-94C	Radiation Monitoring System	Vented
X-95A	Seal Welded Spare	Vented (see Note 1)
X-95B	Radiation Monitoring System	Vented
X-95C	Radiation Monitoring System	Vented
X-96A	ILRT Sensor Line	In Use (see Note 2)
X-96B	ILRT Sensor Line	In Use (see Note 2)
X-96C	Containment Delta P Sensor (PDT 30-44)	Vented
X-97	Containment Delta P Sensor (PDT 30-133)	Vented
X-98	Seal Welded Spare	Vented (see Note 1)
X-99	Hydrogen Analyzer Tr-A	Vented
X-100	Hydrogen Analyzer Tr-A	Vented
X-101	Seal Welded Spare	Vented (see Note 1)
X-102	Seal Welded Spare	Vented (see Note 1)
X-103	Seal Welded Spare	Vented (see Note 1)
X-104	Seal Welded Spare	Vented (see Note 1)
X-105	PAS Cont. Air Rtrn Tr-B	Vented
X-106	PAS Hot Leg 3 Tr-B	See Note 3
X-107	Residual Heat Removal System	Normal Lineup
X-108	Maintenance Port	Vented (see Note 1)
X-109	Maintenance Port	Vented (see Note 1)
X-110	Seal Welded Spare	Vented (see Note 1)
X-111	Seal Welded Spare	Vented (see Note 1)

TABLE 6.2.6-3 (Sheet 5 of 7)

CONTAINMENT VESSEL PRESSURE AND LEAK TEST
REACTOR BUILDING CONTAINMENT PENETRATION STATUS

A. PENETRATION STATUS DURING TEST PERFORMANCE

Penetration	Description	Status
X-112	Seal Welded Spare	Vented (see Note 1)
X-113	Seal Welded Spare	Vented (see Note 1)
X-114	Ice Condenser System	In Use
X-115	Ice Condenser System	In Use
X-116	Seal Welded Spare	Vented (see Note 1)
X-117	Maintenance Port	Vented (see Note 1)
X-118	Layup Water Treatment	In Use
X-119	Seal Welded Spare	Vented (see Note 1)
X-120	Seal Welded Spare	Vented (see Note 1)
X-121E	Electrical Penetration	Vented (see Note 1)
X-122E	Electrical Penetration	Vented (see Note 1)
X-123E	Electrical Penetration	Vented (see Note 1)
X-124E	Electrical Penetration	Vented (see Note 1)
X-125E	Electrical Penetration	Vented (see Note 1)
X-126E	Electrical Penetration	Vented (see Note 1)
X-127E	Electrical Penetration	Vented (see Note 1)
X-128E	Electrical Penetration	Vented (see Note 1)
X-129E	Electrical Penetration	Vented (see Note 1)
X-130E	Electrical Penetration	Vented (see Note 1)
X-131E	Electrical Penetration	Vented (see Note 1)
X-132E	Electrical Penetration	Vented (see Note 1)
X-133E	Electrical Penetration	Vented (see Note 1)
X-134E	Electrical Penetration	Vented (see Note 1)
X-135E	Electrical Penetration	Vented (see Note 1)
X-136E	Electrical Penetration	Vented (see Note 1)
X-137E	Electrical Penetration	Vented (see Note 1)
X-138E	Electrical Penetration	Vented (see Note 1)
X-139E	Electrical Penetration	Vented (see Note 1)
X-140E	Electrical Penetration	Vented (see Note 1)
X-141E	Electrical Penetration	Vented (see Note 1)
X-142E	Electrical Penetration	Vented (see Note 1)
X-143E	Electrical Penetration	Vented (see Note 1)
X-144E	Electrical Penetration	Vented (see Note 1)
X-145E	Electrical Penetration	Vented (see Note 1)
X-146E	Electrical Penetration	Vented (see Note 1)
X-147E	Electrical Penetration	Vented (see Note 1)
X-148E	Electrical Penetration	Vented (see Note 1)
X-149E	Electrical Penetration	Vented (see Note 1)
X-150E	Electrical Penetration	Vented (see Note 1)
X-151E	Electrical Penetration	Vented (see Note 1)
X-152E	Electrical Penetration	Vented (see Note 1)
X-153E	Electrical Penetration	Vented (see Note 1)
X-154E	Electrical Penetration	Vented (see Note 1)
X-155E	Electrical Penetration	Vented (see Note 1)
X-156E	Electrical Penetration	Vented (see Note 1)

TABLE 6.2.6-3 (Sheet 6 of 7)

CONTAINMENT VESSEL PRESSURE AND LEAK TEST
REACTOR BUILDING CONTAINMENT PENETRATION STATUS

A. PENETRATION STATUS DURING TEST PERFORMANCE

Penetration	Description	Status
X-157E	Electrical Penetration	Vented (see Note 1)
X-158E	Electrical Penetration	Vented (see Note 1)
X-159E	Electrical Penetration	Vented (see Note 1)
X-160E	Electrical Penetration	Vented (see Note 1)
X-161E	Electrical Penetration	Vented (see Note 1)
X-162E	Seal Welded Spare	Vented (see Note 1)
X-163E	Electrical Penetration	Vented (see Note 1)
X-164E	Electrical Penetration	Vented (see Note 1)
X-165E	Electrical Penetration	Vented (see Note 1)
X-166E	Electrical Penetration	Vented (see Note 1)
X-167E	Electrical Penetration	Vented (see Note 1)
X-168E	Electrical Penetration	Vented (see Note 1)
X-169E	Electrical Penetration	Vented (see Note 1)
X-170E	Electrical Penetration	Vented (see Note 1)
X-171F	Electrical Penetration	Vented (see Note 1)
X-172E	Electrical Penetration	Vented (see Note 1)
X-173E	Electrical Penetration	Vented (see Note 1)
X-174E	Electrical Penetration	Vented (see Note 1)

TABLE 6.2.6-3 (Sheet 7 of 7)

CONTAINMENT VESSEL PRESSURE AND LEAK TEST
REACTOR BUILDING CONTAINMENT PENETRATION STATUS

B. TESTABLE PENETRATIONS REQUIRED TO BE INSERVICE DURING TEST PERFORMANCE

Penetration	Description	Status
X-26A X-26B	Integrated Leak Rate Test	Isolation valves required to be open to monitor containment pressure (see Note 2)
X-47A	Ice Condenser System	Glycol cooling supply to air handling units in ice condenser required to ensure ice condition is maintained
X-47B	Ice Condenser System	Glycol cooling return from air handling units in ice condenser required to ensure ice condition is maintained.
X-54	Thimble Renewal	Used as pressurization point for air compressors
X-96A X-96B	Integrated Leak Rate Test	Isolation valves required to be open to monitor containment pressure (see Note 2)
X-107	Residual Heat Removal System	Residual heat removal system required inservice to remove decay heat from fuel
X-114	Ice Condenser System	Glycol return from floor cooling coils required to ensure ice condition is maintained
X-115	Ice Condenser System	Glycol supply line to floor cooling coils required to ensure ice condition is maintained.
X-118	Hatch	Used as source for verification flow and post-test depressurization

Notes:

1. These penetrations are closed. Venting is provided by the design of the penetration such that any leakage is detectable by the integrated leak rate test.

2. These penetrations are designed to facilitate ILRT performance. It may not be necessary to utilize all of the penetrations. If not in use, the penetration is vented.

3. These penetrations may remain water filled and/or unvented to facilitate planning and scheduling or for ALARA considerations, provided they have been Type B or C tested within the previous 24 months.

4. The process piping for these penetrations have been cut and capped.

6.3 EMERGENCY CORE COOLING SYSTEM

The emergency core cooling system (ECCS) is discussed in detail in this section. For additional information on the ECCS see the following sections:

1. Compliance with the 10 CFR 50.46 acceptance criteria is discussed in Section 15.4.1.

2. Components which are necessary following a postulated loss-of-coolant accident (LOCA) over the entire range of break sizes are discussed in Sections 15.3 and 15.4.

3. External forces and their effect on the operation of the ECCS are treated in Sections 3.7 and 3.9.

4. Preoperational system testing is discussed in Chapter 14 (historical information).

5. The actuation of the ECCS following a LOCA is discussed in detail in Section 7.3.

6. Instrumentation available to the operator to monitor conditions after a LOCA is found in Section 7.5.

7. Testing intervals are discussed in the Technical Specifications.

6.3.1 Design Bases

6.3.1.1 Range of Coolant Ruptures and Leaks

The ECCS is designed to cool the reactor core as well as to provide additional shutdown capability following initiation of the following accident conditions:

1. A pipe break or spurious valve lifting in the reactor coolant system (RCS) which causes a discharge larger than that which can be made up by the normal makeup system, up to and including the instantaneous circumferential rupture of the largest pipe in the reactor coolant system.

2. Rupture of a control rod drive mechanism causing a rod cluster control assembly ejection accident.

3. A pipe break or spurious valve lifting 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 acceptance criteria for the consequences of each of these accidents is described in Chapter 15 in the respective accident analyses sections.

6.3.1.2 Fission Product Decay Heat

The primary function of the ECCS following a LOCA is to remove 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 the accidents, as well as analyses of the accidents are provided in Chapter 15.

6.3.1.3 Reactivity Required for Cold Shutdown

The ECCS provides shutdown capability for the accidents listed above by means of chemical poison (boron) injection. The most critical accident for shutdown capability is the steam line break and for this accident the ECCS meets the criteria defined in Chapter 15. During a steam line break outside containment, the refueling water storage tank (RWST) is assumed to rupture. This could be due to a tornado induced steam line break.

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. This subject is detailed in Section 6.3.2.11.

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

The ECCS is designed to perform its function of ensuring core cooling shutdown capability following an accident under simultaneous safe shutdown earthquake loading. The seismic requirements are defined in Section 3.7.

6.3.2 System Design

6.3.2.1 Schematic Piping and Instrumentation Diagrams

Flow diagrams of the ECCS are shown in Figure 6.3-1 Sheet 1.

6.3.2.2 Equipment and Component Design

Pertinent design and operating parameters for the components of the ECCS are given in Table 6.3-1. The codes and standards to which the individual components of the ECCS are designed are listed in Table 3.2-2a.

The component design and operating conditions are specified as the most severe conditions to which each respective 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 structural integrity of the ECCS components is maintained. These components are designed to withstand the appropriate seismic loadings in accordance with their safety class as given in Table 3.2-2a.

Cold Leg Injection Accumulators

These accumulators are pressure vessels filled with borated water and pressurized with nitrogen gas. One accumulator is attached to each of the cold legs of the RCS. 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. 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 contents of only three tanks need to be injected in order to meet initial core cooling requirements. The contents of the fourth accumulator is assumed to spill through the break.

Connections are provided for remotely adjusting the level and boron concentration of the borated water in each accumulator during normal plant operation as required. Accumulator water level may be adjusted either by draining to the holdup tank or the reactor coolant drain tank or by pumping borated water from the RWST to the accumulator using a safety injection pump.

Accumulator pressure is provided by a supply of nitrogen gas within its own volume, and can be adjusted as required during normal plant operation; however, the accumulators are normally isolated from the source of this nitrogen supply. Gas relief valves on the accumulators protect them from pressures in excess of design pressure.

The accumulators are located within the containment but outside of the secondary shield wall which protects them from missiles. Since the accumulators are located within the containment, a release of the nitrogen gas in the accumulators would cause an increase in normal containment pressure. Containment pressure following release of the gas from all accumulators when evaluated in accordance with the ideal gas law, is well below the containment pressure setpoint for ECCS actuation.

Release of accumulator gas would be detected by the accumulator pressure indicators and alarms. Thus the operator could take action promptly as required to maintain plant operation within the requirements of the Technical Specification covering accumulator operability and containment pressure.

The complete listing of the design parameters for the cold leg injection accumulator is presented in Table 6.3-1.

Pumps

Residual Heat Removal (RHR) Pumps

RHR pumps are provided to deliver water from the RWST or the containment sump to the RCS should the RCS pressure fall below their shutoff head.

Each RHR pump is a single stage, vertical position, centrifugal pump. It has an integral motor-pump shaft, driven by an induction motor. The unit has a self contained mechanical seal cooling system. Component cooling water system (CCS) is the heat exchange medium. The pumps start on receipt of a safety injection signal.

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.

The RHR pumps are also discussed in Section 5.5.7.

Centrifugal Charging Pumps

These pumps deliver water from the RWST through the boron injection tank to the RCS at the prevailing RCS pressure. Each centrifugal charging pump is a multistage, diffuser design, barrel type casing with vertical suction and discharge nozzles. The pump is driven through a speed increaser connected to an induction motor. The pump and speed increaser have self-contained lubrication systems with CCS as the heat exchanger medium. The pump has a mechanical seal cooling system. System process water is the normal heat exchange medium for the mechanical seal. The pumps start on receipt of a safety injection signal.

A minimum flow bypass line is provided on each pump discharge to recirculate flow back to the pump suction after cooling in the seal water heat exchanger. This is required to protect the pumps at the shutoff head. The minimum flow bypass line contains two valves in series which are provided for isolation of the mini-flow line. These valves are normally open with power to the valve operators locked out at each valve breaker to prevent inadvertent isolation of the mini-flow line. The charging pumps may be tested during normal operation through the use of the minimum flow bypass line. The centrifugal charging pumps are also discussed in Section 9.3.4.

Safety Injection Pumps

The safety injection pumps deliver water from the RWST to the RCS after the reactor coolant pressure is reduced below their shutoff head. Each high head safety injection pump is a multistage, centrifugal pump. The pump is driven directly by an induction motor. The unit has a self-contained lubrication system with CCS as the heat exchanger medium. The pump also has a mechanical seal cooling system. System process water is the heat exchange medium for the mechanical seals. The pumps start on receipt of a safety injection signal.

A minimum flow bypass line is provided on each pump discharge to recirculate flow to the RWST in the event the pumps are started with the normal flow paths blocked. This line also permits pump testing during normal operation. Redundant motor operated valves (MOVs) in series are provided for isolation of this line. These valves are closed by operator action during the recirculation mode.

Residual Heat Exchangers

The residual heat exchangers are conventional shell and U-tube type units. During normal operation of the RHR system, reactor coolant flows through the tube side while component cooling water flows through the shell side. During emergency core cooling recirculation operation, water from the containment sump flows through the tube side. The tubes are seal welded to the tube sheet.

A further discussion of the residual heat exchangers is found in Section 5.5.7.

Valves

Design parameters for all types of valves used in the ECCS are given in Table 6.3-1.

Design features employed to minimize valve leakage include:

1. Where possible, packless valves are used.

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

3. Relief valves are enclosed, i.e., they are provided with a closed bonnet.

4. Some control valves and MOVs (2 inches and above) exposed to recirculation flow have double packed stuffing boxes and stem leakoff connections to the waste processing system. Other valves may have their leakoff line connections plugged after the packing has been upgraded with graphite packing rings. This packing configuration will reduce stem leakage to essentially zero.

Table 6.3-10 provides a list of the principal ECCS valves and their respective positions during normal and all ECCS modes of operation.

Motor-Operated Valves

The seating design of all motor operated gate valves is of the parallel disc design or the flexible wedge design. These designs release the mechanical holding force during the first increment of travel so that the motor operator works 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 to provide ease of gate movement. The seating surfaces are hard faced to prevent galling and to reduce wear.

Where a gasket is employed for the body to bonnet joint, it is either a fully trapped, controlled compression, spiral wound gasket with provisions for seal welding, or it is of the pressure seal design with provisions for seal welding. The valve stuffing boxes are designed with a lantern ring leakoff connection with a minimum of a full set of packing below the lantern ring and a minimum of one-half of a set of packing above the lantern ring. A full set of packing is defined as a depth equal to 1-1/2 times the stem diameter.

The motor operator incorporates a "hammer blow" feature that allows the motor to impact the discs away from the backseat upon opening or closing. This "hammer blow" feature not only impacts the disc but allows the motor to attain its operational speed prior to impact. Valves which must function against system pressure are designed such that they function with a pressure drop equal to full system pressure across the valve disc.

Manual Globe, Gate, and Check Valves

Gate valves are either wedge design or parallel disc and are straight through. The wedge is either split or solid. All gate valves have backseat and outside screw and yoke.

Globe valves of the "T" and "Y" styles are full-ported.

Check valves are spring-loaded lift piston types for sizes 2 inches and smaller and swing type or tilting disc type for size 3 inches and larger. Stainless steel check valves have no body penetrations other than the inlet, outlet and bonnet. The check hinge is serviced through the bonnet.

The stem packing and gasket of the stainless steel manual globe and gate valves are similar to those described above for MOVs.

Diaphragm Valves

The diaphragm valves are of the Saunders-patent type which uses the diaphragm member for shut off with even weir bodies. These valves are used in systems not exceeding 200°F and 220 psig design temperature and pressure, respectively.

Accumulator Check Valves

The cold leg accumulator check valve is designed with a low pressure drop configuration with all operating parts contained within the body. The disc is permitted to rotate, providing a new seating surface after each valve opening.

Design considerations and analyses which assure that leakage across all the check valves located in each accumulator injection line will not impair accumulator availability are as follows:

1. During normal operation the check valves are in the closed position with a nominal differential pressure across the disc. The differential pressure is approximately 1650 psi for the check valves in the cold leg lines. Since the valves remain in this position except for testing or when called upon to function, and are not, therefore, subject to the abuses of flow operation or impact loads caused by sudden flow reversal and seating, they do not experience significant wear of the moving parts. Hence, they are expected to function with minimal leakage.

2. The check valves are tested for leakage when the RCS is being pressurized during the normal plant heat-up operation. This test confirms the seating of the disc and whether or not there has been an increase in the leakage since the last test. After this test is completed and prior to 1000 psig, the discharge line motor operated isolation valves are opened and the RCS pressure increase is continued. There should be no increase in leakage from this point on since increasing reactor coolant pressure increases the seating force and decreases the probability of leakage.

3. Experience derived from check valves employed in emergency injection systems indicates that they are reliable and workable. This is substantiated by the satisfactory experience obtained from their operation at other plants where the usage of check valves is identical to their application at Watts Bar.

4. The accumulators can accept some in-leakage from the RCS without affecting availability.

   In-leakage requires, however, that the accumulator water volume and boron concentration be adjusted accordingly to remain within Technical Specification requirements. An accumulator high water level alarm is provided as an added safeguard to warn the operator that excessive accumulator in-leakage is occurring.

Relief Valves

The accumulator relief valves are sized to pass nitrogen gas at a rate in excess of the accumulator gas fill line delivery rate. The relief valves will also pass water in excess of the expected accumulator inleakage rate. This is not considered to be necessary because the time required to fill the gas space gives the operator ample opportunity to correct the situation.

Other relief valves are installed in various sections of the ECCS to protect lines which have a lower design pressure than the RCS. Relief valves normally discharge to the pressurizer relief tank. The boron injection tank relief discharges to the CVCS holdup tank (HUT). The seal water heat exchanger (centrifugal charging pump (CCP) miniflow path) discharges to the volume control tank which in turn discharges to the CVCS HUT. Table 3.9-20 lists the system's relief valves with their capacities and setpoints.

Butterfly Valves

Each main RHR line has an air-operated butterfly valve which is normally open and is designed to fail in the open position. These valves are in the full-open position during normal operation to maximize flow from this system to the RCS during the injection mode of the ECCS operation.

<u>Piping</u>

Piping joints are welded except where disassembly of the joint may be required. In order to assure structural integrity, pipe weld connections are fabricated to satisfy ASME Code requirements.

Minimum piping and fitting wall thicknesses are increased to account for the manufacturer's permissible tolerance on the nominal wall and an appropriate allowance for wall thinning on the external radius during any pipe bending operations in the shop fabrication of the subassemblies. The wall thicknesses are determined by formula from the 1971 ASME Code, Section III, Summer 1973 Addenda.

System Operation

The operation of the ECCS, following a loss-of-coolant accident, can be divided into two distinct modes:

1. The injection mode in which any reactivity increase following the postulated accidents is terminated, initial cooling of the core is accomplished, and coolant lost from the primary system in the case of a LOCA is replenished, and

2. The recirculation mode in which long-term core cooling is provided during the accident recovery period.

A discussion of these modes follows.

Break Spectrum Coverage

The principal mechanical components of the ECCS which provide core cooling immediately following a LOCA are the accumulators, the safety injection pumps, the centrifugal charging pumps, the RHR pumps, RWST, and the associated valves and piping.

For large pipe ruptures, the RCS would be depressurized and voided of coolant rapidly, and a high flow rate of emergency coolant is required to quickly cover the exposed fuel rods and limit possible core damage. This high flow is provided by the passive cold leg accumulators, the charging pumps, safety injection pumps, and the RHR pumps discharging into the cold legs of the RCS. The RHR and safety injection pumps deliver into the accumulator injection lines, between the two check valves, during the injection mode. The charging pumps deliver coolant to the cold legs during the injection mode.

Emergency cooling is provided for small pipe ruptures primarily by the high-head injection pumps. The charging pumps and safety injection pumps are commonly referred to as "high-head pumps" and the RHR pumps as "low-head pumps." Likewise, the term "high-head injection" is used to denote charging pump and safety injection pump injection and "low-head injection" refers to RHR pump injection. Small pipe ruptures are those, with an equivalent diameter of 6 inches or less, which do not immediately depressurize the RCS below the accumulator discharge pressure. The centrifugal charging pumps are designed to deliver borated water at the prevailing RCS pressure. During the injection mode, the charging pumps take suction from the RWST.

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The safety injection pumps also take suction from the RWST and deliver borated water to the cold legs of the RCS. The safety injection pumps begin to deliver water to the RCS after the pressure has fallen below the pump shutoff head.

Core protection is afforded with the minimum engineered safety feature equipment. The minimum engineered safety feature equipment is defined by consideration of the single failure criteria as discussed in Sections 6.3.1.4 and 3.1. The minimum design case ensures the entire break spectrum is accounted for and core cooling design bases of Section 6.3.1 are met. The analyses for this case are presented in Sections 15.3 and 15.4.

In the minimum design case for large RCS ruptures, the cold leg accumulators and one train of active high-head and low-head pumping components serve to complete the core refill. One RHR loop is required for long-term recirculation along with components of the auxiliary heat removal system, which are required to transfer heat from the ECCS to the component cooling system and essential raw cooling water system.

If the break is small (6-inch equivalent diameter or less) the accumulators with one charging pump and one safety injection pump ensure adequate cooling during the injection mode. Long-term recirculation requires one RHR loop and components of the auxiliary heat removal systems. The loss-of-coolant analyses are presented in Section 15.3 and 15.4.

Certain deviations (i.e., reduced component availability) from the normal operating status as given in Table 6.3-4 of the ECCS are permissible without appreciably impairing the reliability of the ECCS to provide adequate core cooling capability. Accordingly, Technical Specifications have been established to identify these types of deviations and restrict the time period that a given deviation may exist.

The Technical Specifications permit one cold leg accumulator and various pumps of the ECCS to be inoperable during power operation for a period of time. The permissible time periods for which accumulators, ECCS pumps, and associated equipment may be inoperable are listed and their bases described in the Technical Specifications.

The minimum active components will be capable of delivering full rated flow within a specified time interval after process parameters reach the setpoints for the safety injection signal. Response of the system is automatic, with appropriate allowances for delays in actuation of circuitry and active components. The active portions of the system are actuated by the safety injection signal directly, with the exception of the isolation valves for the hydrogen vent lines on the charging pump suction piping. These valves are electrically interlocked to the volume control tank outlet valves. In analyses of system performance, delays in reaching the programmed trip points and in actuation of components are established on the basis that only emergency onsite power is available. The starting sequence is detailed in Table 8.3-3.

In the loss-of-coolant accident analyses presented in Sections 15.3 and 15.4 no credit is assumed for partial flow prior to the establishment of full flow and no credit is assumed for the availability of normal offsite power sources.

For smaller loss-of-coolant accidents, there is some additional delay before the process variables reach their respective programmed trip setpoints since this is a function of the severity of the transient imposed by the accident. This is allowed for in the analyses of the range of loss-of-coolant accidents.

Accumulator injection occurs immediately when the RCS is depressurized below accumulator operating pressure. For the cold leg injection accumulator this setpoint will be reached only in event of a large rupture.

The cold leg injection accumulators can be isolated from the RCS by closure of their motor-operated isolation valves. Since these accumulators operate only after considerable RCS pressure loss, the injection of pressurized nitrogen via the cold legs is not considered a problem.

Injection Mode After Loss of Primary Coolant

The injection mode of emergency core cooling is initiated by the safety injection signal ("SI" signal). This signal is initiated by any of the following:

1. Low pressurizer pressure

2. High containment pressure

3. Low steamline pressure in any steamline.

4. Manual actuation

Operation of the ECCS during the injection mode is completely automatic. Refer to Figure 7.3-3 (Sheet 3) for safety injection signal logic. The safety injection signal in addition to activating the ESF equipment automatically initiates the following actions:

1. Starts the diesel generators and trips the diesel generator feeder breaker if the diesel generator is in test with the offsite power source. They will be aligned to the 6.9 kV shutdown boards if power is lost to the respective board.

2. Starts the charging pumps, the safety injection pumps, and the RHR pumps.

3. Aligns the charging pumps for injection by:

   a. Closing the valves in the charging pump discharge line to the normal charging line.

   b. Opening the valves in the charging pumps suction line from the RWST.

   c. Closing the valves in the charging pump normal suction line from the volume control tank when either of the two RWST suction valves are fully open. Closing these valves initiates the closing of the isolation valves at the hydrogen vent line for the charging pumps suction side piping.

   d. Opening the isolation valves located in the discharge line from the boron injection tank.

The switchover to recirculation mode is initiated when the low level is reached in the RWST coincident with a high level in the containment sump.

Recirculation Mode

The injection mode continues until the RHR pumps have been realigned to the recirculation mode. During the injection mode, all pumps take suction from the RWST until a low level signal from the RWST in conjunction with the "SI" signal and a high sump level signal aligns the RHR pumps to take suction from the containment sump. The RHR RWST isolation valves (FCV-74-3 and FCV-74-21) are automatically closed coincident with the opening of the sump isolation valves (FCV-63-72 and FCV-63-73). The automatic positioning of these valves is initiated only in the event that actuation signals are generated by the safeguards protection logic ("SI" signal), two of four RWST low level protection logic signals, and two out of four sump high level signals. It has been determined that the RHR pumps continue to receive adequate suction flow during this automatic changeover, thus there is no possibility of pump damage due to loss of suction. Alarms on RWST low level and level indications from both the sump and RWST are used by the operator to appraise the accident situation and complete the remainder of switchover sequence.

Table 6.3-3 describes the sequence of changeover operation from injection to recirculation.

The switchover initiation point and minimum assured final volume in the RWST before completion of switchover are selected on the basis of maximizing the allowable operator action time for accompanying manual operations and total water injected to the RCS while avoiding the potential problems due to low levels in either the active sump inside containment or in the RWST. Crane wall penetrations inside containment are sealed as necessary between Elevations 702.78 and 716 to initially retain more water in the active sump, thereby maximizing the active sump water level at the onset of the recirculation switchover.

The sequence (as delineated in Table 6.3-3) is followed regardless of which power supply is available (offsite or emergency onsite).

The time required to complete the sequence is essentially the time required for the operator to perform the accompanying manual operations. Controls for ECCS components are grouped together on the main control board. The component position lights indicate equipment position/status.

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 low head recirculation flow path consists of the RHR pumps taking suction from the containment sump and discharging the flow directly to the RCS through the residual heat exchangers and cold leg injection lines. The high head recirculation flow path consists of the RHR pumps taking suction from the containment sump and discharging the flow through the residual heat exchangers to the suction of the centrifugal charging pumps and safety injection pumps. The flow from the centrifugal charging pumps and safety injection pumps is returned to the RCS through the cold leg injection lines. 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.

Approximately 3 hours after event initiation, hot leg recirculation will be initiated to assure against an excessive buildup of boric acid in the core.

The containment sump isolation valve is interlocked with its respective pump suction/RWST isolation valve to the RHR system. The interlock is provided with redundant signals from each isolation valve. This interlock prevents remote manual opening the sump isolation valve when the RWST isolation valves are open and thus prevents dumping the RWST contents into the containment sump. However, when an accident signal is present, this interlock is bypassed to allow initiation of the switchover sequence.

The RWST is protected from back flow of reactor coolant from the RCS. All connections to the RWST are provided with check valves to prevent back flow. When the RCS is hot and pressurized there is no direct connection between the RWST and the RCS. When the RCS is being cooled and the RHR system is placed in service, the RHR system is isolated from the RWST by a motor-operated valve in addition to a check valve.

Redundancy in the external recirculation loop is provided by the inclusion of duplicate charging, safety injection, and RHR pumps and residual heat exchangers. Inside the containment, the charging pump and safety injection pump discharge is piped separately into all four cold legs for the charging pumps and into all four cold legs and all four hot legs for the safety injection pumps. The low head pumps take suction through redundant lines from the containment sump and discharge through separate paths to the RCS. The containment sump design is shown in Figures 6.3-6 and 6.3-6A.

The containment sump is located in the containment floor (Elevation 702.78 ft, 270°) below the refueling canal to provide protection from high energy pipe failures. A strainer assembly is installed on the top of the sump suction pit to prevent debris that may be present after a design basis accident from degrading the performance of the ECCS and containment spray system. A horizontal grating is located one foot below the ceiling to eliminate vortexing. This pit is surrounded by a six-inch high curb which is used to prevent sediment from entering the pit. A fine mesh (¼-inch) screen located in the containment sump suction pit is used to divide the sump into two suction volumes.

The sump design does not comply fully with Regulatory Guide 1.82 Revision 0 because the plant design was well advanced when the Regulatory Guide was issued; the plant construction permit preceded the Regulatory Guide by 17 months. The design does comply fully, however, except as itemized below:

Position C.1: Dual sumps are not provided. The single sump, however, has adequate capacity and inlet area.

Position C.3: The containment sump suction pit has a six-inch high curb with a metal strainer assembly atop it. This arrangement provides vortex suppression. The sump intake is not protected by two screens. It is protected by an advanced design strainer assembly that can withstand the expected debris loading and differential pressure. The perforation size is a 0.085 inch diameter hole which prevents debris from entering the inner sump.

Position C.4: The floor is level, but a six-inch high curb is provided around the sump suction pit inlet, thus providing an effect comparable to a sloped floor.

Position C.6: The trash rack and inner screen have been replaced with an advanced design strainer which has been designed to be strong enough to withstand the expected debris loading and differential pressure.

Position C.7: The strainer assembly has mainly horizontal straining surfaces, but has been designed to obtain low approach velocity due to significant surface area. The design configuration of the entire assembly impedes the deposition or settling of debris on the strainer surfaces.

The limiting size particle which may be circulated by the ECCS and containment spray systems without causing system damage is a function of several physical parameters including:

A. limiting system clearance,
B. particle concentration,
C. particle abrasive properties, and
D. particle hardness.

The smallest component which must be prevented from being plugged has been determined to be the loop injection throttle valves which have a minimum clearance of 0.0925 inches. Soft material of larger dimensions could easily be passed by the pumps, but would cause blockage of these components. Damage to the pumps could occur if significant concentrations of hard, dense, smaller debris were allowed to cause surface abrasion or binding between moving pump parts. To eliminate all such particles by smaller screen sizing would increase the threat of screen blockage. Hence, the design philosophy for the sump is to size the sump screens according to the limiting clearance and to otherwise take advantage of settling properties to eliminate the threat of damage from the smaller, more dense particles.

No debris is expected to reach the sump and cause blockage of the sump screens during or after a LOCA. However, the effective screen opening areas are many times larger than the combined flow area of both sump suction pipes to allow appreciable blockage from unspecified debris before any significant pressure drop is developed across the screens.

The lower containment is an open, one level area and no drains are used to route water to the sump (except the two large refueling cavity drains and two small accumulator room drains that route water away from the strainer assembly). The water simply fills the floor area and covers the sump entrance. Debris with a specific gravity greater than one will largely settle before reaching the sump inlet.

The containment sump suction pit (located in the inner sump area) has a six-inch high curb with the strainer assembly atop it. The strainer perforations are 0.085 inch diameter holes which prevent debris from entering the sump. It also serves to maintain a low inlet velocity. Debris generated during the initial blowdown of a LOCA will have an approximate minimum time of ten minutes to settle before any suction is taken from the sump.

Peeling paint has been identified as a possible hazard to the containment sump. To prevent this hazard, surface coatings will meet the requirements as described in Section 6.1.2. Further, paint used inside containment produces a hardened film having a specific gravity appreciably greater than one. Hence, gravitational settling helps assure protection against plugging of the sump with paint particles.

Equipment insulation has been designed to assure against it becoming a source of sump blockage. This has been done by providing metallic insulation on reactor coolant pressure boundary vessels and piping where required. The systems and components utilizing metallic insulation include the reactor vessel, steam generators, pressurizer, reactor coolant pumps and piping, RHR piping, safety injection piping, chemical and volume control piping, and main steam and feedwater piping. The insulation is not designed to withstand the blowdown and jet impingement forces associated with pipe breaks and sections of insulation may be stripped away if a break occurs. However, the strainer assembly will prevent the metallic insulation from entering the containment sump.

Small sections of pipe inside containment are insulated with stainless-steel-jacketed mass-type insulation, as needed, to avoid interferences with and potential overheating of electrical cable and components. As is the case with the metallic insulation, this insulation is not designed to withstand the blowdown and jet impingement forces associated with pipe breaks. However, it also poses no significant hazard to the containment sump if a pipe break were to occur. The metal jacketing may sink if located in the vicinity of the sump, but it is prevented from entering the sump by the strainer assembly. The mass-type insulation floats since its specific gravity is appreciably less than one, and it is hydrophobic grade to prevent absorption of water.

Small amounts of rockwool insulation are located inside the guard pipes of containment penetrations with flued heads as shown in Figures 6.2.4-1 through 6.2.4-4 and 6.2.4-10. This insulation is enclosed and protected by the guard pipes.

Ice condenser insulation is specifically designed not to create debris during a LOCA. Materials that could become segmented or decompose are enclosed. Solid insulation such as foam concrete is used where practical and protection is provided to prevent insulation damage from the blowdown effects of a LOCA. All equipment inside containment is designed to prevent its becoming a source of blockage to the sump.

Each recirculation line from the sump is run outside the containment to a sump isolation valve. This valve is surrounded with a steel enclosure and the section of piping joining it to the sump is run within a guard pipe welded to the enclosure. Any excessive leakage or passive failure downstream of the sump valves can be controlled and isolated by closure of the sump valve in the affected train.

NRC Generic Letter 2004-02

The ECCS containment sump design addresses the potential post-LOCA ECCS performance issues provided in NRC Generic Letter 2004-02, “Potential Impact of Debris Blockage on Emergency Recirculation during Design Basis Accidents at Pressurized-Water Reactors.” The containment sump design has been tested and analyzed based on guidance provided in NEI-04-07, “Pressurized Water Reactor Sump Performance Evaluation Methodology”, as supplemented by the NRC in the “Safety Evaluation by the Office of Nuclear Reactor Regulation Related to NRC GL 2004-07.” Downstream effects were evaluated in accordance with Topical Report WCAP-16406-P, “Evaluation of Downstream Sump Debris Effects in

Support of GSI-191." Chemical effects were evaluated based on testing, WCAP-16530-NP, "Evaluation of Post-Accident Chemical Effects in Containment Sump Fluids to Support GSI-191," and WCAP-19793-NP, "Evaluation of Long-Term Cooling Considering Particulate, Fibrous and Chemical Debris in the Recirculating Fluid." The tests and evaluations confirm that the required safety functions of the containment sump and ECCS system will be met during design basis accidents considering the effects of debris.

External Recirculation Loop

The ECCS recirculation loop piping and components external to containment are surrounded by shielding. This shielding is designed to permit access for maintenance to a component such as a pump while the redundant pump is recirculating sump fluid.

Pressure relieving devices from portions of the ECCS located outside containment which might contain radioactivity discharge to the pressurizer relief tank, except for the BIT and seal water heat exchanger relief valves, which discharge to the CVCS holdup tank and VCT, respectively.

An analysis has been performed to evaluate the radiological effects of recirculation loop leakage as discussed in Section 15.5.3.

During recirculation, significant margin exists between the design and operating conditions (in terms of pressure and temperature) of the ECCS components.

Since redundant flow paths are provided during recirculation, a leaking component in one of the flow paths may be isolated. This action curtails any further leakage and renders the component available for corrective maintenance. Maximum potential leakage from components during recirculation mode operation is given in Table 6.3-6.

6.3.2.3 Applicable Codes and Classifications

The codes and standards to which the individual components of the ECCS are designed are listed in Table 3.2-2a.

6.3.2.4 Materials Specifications and Compatibility

Materials employed for components of the ECCS are given in Table 6.3-2. Materials are selected to meet the applicable material requirements of the codes in Table 3.2-2a and the following additional requirements:

1. The parts of components in contact with sump solution during recirculation are fabricated of austenitic stainless steel or equivalent corrosion resistant material.

2. The parts of components in contact with borated water are fabricated of, or clad with, austenitic stainless steel or equivalent corrosion resistant material, with the exception of pump seals and valve packing.

3. Valve seating surfaces are hard-faced with Stellite No. 6 or equivalent to prevent galling and 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 is well within the design temperature of all the 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 the long term recirculation operations.

Environmental qualification of the ECCS equipment inside the containment, which is required to operate following a LOCA is discussed in Section 3.11. The results of the program indicate that the safety features will operate satisfactorily during and following exposure to the combined containment post-accident environments of temperature, pressure, chemistry, and radiation.

6.3.2.5 Design Pressures and Temperatures

The component design pressures and temperatures are given in Table 6.3-1. These pressure and temperature conditions are specified as the most severe conditions to which each respective 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 (see Section 3.2) and with due consideration for the design and operating conditions, the fundamental assurance of structural integrity of the ECCS components is maintained.

6.3.2.6 Coolant Quantity

The minimum storage volume for the accumulators and the RWST is given in Table 6.3-4. The minimum storage volume in the RWST and the accumulators is sufficient to ensure that, after a RCS break, sufficient water is injected and is available within the containment to permit recirculation cooling flow to the core, and to meet the net positive suction head requirements of the RHR pumps. A further discussion of coolant requirements is contained in Sections 15.3 and 15.4.

6.3.2.7 Pump Characteristics

Design parameters for the ECCS pumps are given in Table 6.3-1.

6.3.2.8 Heat Exchanger Characteristics

Residual heat exchanger characteristics are found in Section 5.5.7.

6.3.2.9 ECCS Flow Diagrams

The SIS flow diagram is given as Figure 6.3-1 Sheet 1.

6.3.2.10 Relief Valves

The ECCS relief valves, their capacities and settings are given in Table 3.9-20.

6.3.2.11 System Reliability

6.3.2.11.1 Definitions

Period of Recovery

The period of recovery is the time necessary to bring the plant to a cold shutdown and regain access to faulted equipment. The recovery period is the sum of the short and long term periods defined below.

Incident

An incident is any natural or accidental event of infrequent occurrence and its related consequences which affect plant operation and require the use of engineered safeguards systems. Such events, which are analyzed independently and are not assumed to occur simultaneously, include the loss-of-coolant accident, steam line ruptures, steam generator tube ruptures, etc. A loss of offsite power event may be an isolated occurrence or may be concurrent with any event requiring engineered safeguards systems use.

Short Term

The short-term time period is the time immediately following the incident during which automatic actions are performed, system responses are checked, type of incident is identified and preparations for long-term recovery operation are made. The short term is the injection phase for LOCA and is the first 24 hours following initiation of the event for all others.

Long Term

The long term time period is the remainder of the recovery period following the short term. In comparison with the short term, where the main concern is to prevent or limit site release, the long-term period of operation involves bringing the plant to cold shutdown conditions where access to the containment can be gained and repairs effected.

Active Failure

An active failure is the failure of a powered component, such as a piece of mechanical equipment or a component of the electrical supply system or instrumentation and control equipment, to act on command to perform its design function. Examples include the failure of a motor-operated valve to move to its correct position, the failure of an electrical breaker or relay to respond, the failure of a pump, fan or diesel generator to start, etc.

Passive Failure

A passive failure is the structural failure of a static component which limits the component's effectiveness in carrying out its design function. When applied to a fluid system, this means a break in the pressure boundary resulting in abnormal leakage not exceeding 50 gpm for 30 minutes. Such leak rates are consistent with limited cracks in pipes, sprung flanges, valve packing leaks or pump seal failures.

6.3.2.11.2 Active and Passive Failure Criteria

Active Failure Criteria

The ECCS is designed to accept any single failure at any time following the incident without loss of its protective function. The system design will tolerate the failure of any single active component in the ECCS itself or in the necessary associated service systems at any time during the period of required system operations following the incident.

A single active failure analysis is presented in Table 6.3-8, and demonstrates that the ECCS can sustain the failure of any single active component in either the short or long term and still meet the level of performance for core cooling.

Since the initial operation of the active components of the ECCS following a steam line rupture is identical to that following a LOCA, the same analysis is applicable and the ECCS can sustain the failure of any single active component and still meet the level of performance for the addition of shutdown reactivity. Passive failure is not considered for the short term.

Passive Failure Criteria

The following philosophy provides for necessary redundancy in component and system arrangement to meet the intent of the General Design Criteria on single failure as it specifically applies to failure of passive components in the ECCS. Thus, for the long term, the system is based on accepting either a passive or an active failure.

Redundancy of Flow Paths and Components for Long-Term Emergency Core Cooling

In the design of the ECCS, Westinghouse utilized the following criteria:

1. During the long-term cooling period following a loss of coolant, the emergency core cooling flow paths are separable into two subsystems, either of which can provide minimum core cooling functions and return spilled water from the floor of the containment back to the RCS.

2. Either of the two subsystems can be isolated and removed from service in the event of a leak outside the containment.

3. Adequate redundancy of check valves is provided to tolerate failure of a check valve during the long term as a passive component.

4. Should one of these subsystems be isolated in this long-term period, the other subsystem remains operable.

5. Provisions are also made in the design to detect leakage from components outside the containment, collect this leakage and to provide for limited maintenance of the affected equipment.

Thus, for the long-term emergency core cooling function, adequate core cooling capacity exists with one flow path removed from service whether isolated due to a leak, because of blocking of one flow path, or because failure in the containment results in a spill of the delivery of one injection flow path.

The design of the ECCS includes the provision for diversion of a portion of the RHR pump flow from the low head injection path to auxiliary spray headers in the upper containment volume. For this mode the RHR pumps continue to supply recirculation flow from the containment sump to the core via the safety injection and centrifugal charging pumps.

The diversion of the RHR flow from the low head injection path to the auxiliary spray headers occurs only after the switchover to the recirculation mode and no earlier than 1 hour after initiation of the LOCA. When RHR spray is required, the operator is provided with a detailed procedure (Table 6.3-3) to follow in aligning the system for RHR spray operation. This procedure requires that low head safety injection flow to the core be terminated under single train operating condition prior to initiating RHR spray flow.

6.3.2.11.3 Subsequent Leakage from Components in Safeguards Systems

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 leak rate assuming only the presence of a seal retention ring around the pump shaft showed 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 significantly less severe than the pump seal failure.

Larger leaks in the ECCS are prevented by the following:

1. The piping is classified in accordance with ANS Safety Class 2 and receives the ASME Class 2 quality assurance program associated with this safety class.

2. The piping, equipment and supports are designed to ensure no loss of function for the safe shutdown earthquake.

3. The system piping is located within a controlled area on the plant site.

4. The piping system receives periodic pressure tests and is accessible for periodic visual inspection.

5. The piping is austenitic stainless steel which, due to its ductility, can withstand severe distortion without failure.

Based on this review, the design of the Auxiliary Building and related equipment is based upon handling of ECCS leaks up to a maximum of 50 gpm. To assure adequate core cooling, design features are provided to prevent this limiting passive failure from causing any loss of function in the other train of the ECCS equipment due to flooding of redundant components or loss of NPSH to the ECCS pumps. Three independent means are available to provide information to the operator for use in identifying ECCS leakage into certain locations in the Auxiliary Building. These means include the Auxiliary Building flood detection system, the instrumentation and alarms associated with the drainage and waste processing systems which normally handle drainage into these areas, and redundant level indicators in the Auxiliary Building passive sump.

A flood detection system, utilizing water level detector devices, is used to monitor and actuate alarms for ECCS and other leakage at locations throughout the Auxiliary Building. Individual detectors are located in each ECCS pump compartment, in the ECCS heat exchanger rooms, and in the pipe gallery (Elevation 676). A common alarm in the main control room will alert the operator when any of these flood detectors are tripped. A flood detector indicator panel, located immediately outside the control room, then identifies the exact location of the tripped detector. The detectors were preoperationally tested to verify initial operability and will be periodically tested as a part of the plant maintenance program.

Since each ECCS pump compartment is monitored by a level detection device, the operator may immediately identify which subsystem must be shut down and secured to terminate the leak.

The operator can readily accomplish this action from the main control room by stopping the appropriate subsystem pump, and by closing the corresponding sump isolation valves and individual pump discharge valves. The time necessary for the operator to detect leakage in a pump compartment is dependent on the leakage rate. A limiting 50 gpm leak in the largest ECCS pump compartment can be detected within 30 minutes. Slower leaks may require proportionally longer detection times.

Leakage into these ECCS pump compartments is piped to the floor drain collector tank for the safety injection and centrifugal charging pumps and to the Auxiliary Building floor and equipment drain sump for the RHR and containment spray pumps. The drain in each of these rooms is provided with a standpipe which assures that the setpoint for the level detector is reached prior to draining the leakage from the room. However, the standpipes each have two ½-inch drilled holes to allow minor normal leakage to drain from the room.

ECCS leakage into the Auxiliary Building locations other than the ECCS pump compartments is piped to the Auxiliary Building floor and equipment drain sump. This sump is provided with redundant 50 gpm pumps which are indicated in the main control room. The floor drain collector tank is provided with overflow piping which discharges to the Auxiliary Building floor and equipment drain sump. Leakage into these areas can be detected by the flood detection system described above, by indication of sump pump operation, or by high level alarm from the sump or the floor drain collector tank. However, the exact location of the leak, if other than the ECCS pump compartment, or the subsystem from which leakage occurs, may not be immediately identified. Since ECCS leaks other than a pump seal failure are of a nature to develop very slowly and are less severe than a seal failure, the operator has an extended time period to detect and isolate the leak. Isolation of these minor leaks can be accomplished by arbitrarily selecting and isolating an ECCS subsystem and evaluating the response of the flood detector system.

The flood detection system described above is not designed to meet the requirements of IEEE-279. The detectors, indicator panel, and control room alarm are powered from nondivisional boards and do not meet the single failure criteria. However, the system is designed such that a loss of power to any individual detector will actuate the main control room common alarm. Additionally, the nondivisional boards which supply the flood detector system are powered from a Class 1E power board which is automatically loaded on the diesel generators. This ensures continued power availability to the flood detection system after an accident.

In addition to the flood detection and normal drainage processing systems described above, redundant level sensors which do meet the requirements of IEEE-279 are provided in the Auxiliary Building passive sump. These sensors, which are a part of the post accident monitoring (PAM) system as described in Section 7.5, are designed to continuously indicate and able to trend the passive sump level. Also, the Auxiliary Building is provided with redundant ESF grade air cleanup and filtration systems as described in Section 6.5.1.

With these design ground rules, continued function of the ECCS meets minimum core cooling requirements. A single passive failure analysis is presented in Table 6.3-9. It demonstrates that the ECCS can sustain a single passive failure during the long term phase and still retain an intact flow path to the core to supply sufficient flow to maintain the core covered and affect the removal of decay heat. An event resulting in maximum leakage would have an insignificant impact on ECCS capability since a 100% capacity redundant train is available to assure the ECCS capability and since the maximum leakage represents less than 0.5% of system flow capacity.

6.3.2.12 Protection Provisions

The provisions taken to protect the system from damage that might result from dynamic effects on piping systems are discussed in Section 3.6. The provisions taken to protect the system from missiles are discussed in Section 3.5. The provisions to protect the system from seismic damage are discussed in Sections 3.7, 3.9, and 3.10. Thermal stresses on the RCS are discussed in Section 5.2.

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 Net Positive Suction Head

The ECCS is designed so that adequate net positive suction head is provided to system pumps. See Table 6.3-12, Available NPSH during ECCS Operation, for a complete listing of available NPSH for all pumps in ECCS while operating. Adequate net positive suction head is shown to be available for all pumps as follows:

1. Residual Heat Removal Pumps

The net positive suction head of the RHR pumps is evaluated for normal plant shutdown operation, and for both the injection and recirculation modes of operation for the design basis accident. Recirculation operation gives the limiting net positive suction head requirement, and the net positive suction head available is determined from the containment pressure, vapor pressure of liquid in the sump, containment sump level relative to the pump elevation and the pressure drop in the suction piping from the sump to the pumps. No credit is taken for water level above the RHR sump strainer assembly, and no credit is taken for containment over pressure. The net positive suction head evaluation is based on all pumps operating at the maximum design basis accident flow rates. The RHR pump head-capacity curves are given in Figure 6.3-2.

2. Safety Injection and Centrifugal Charging Pumps

The net positive suction head for the safety injection pumps and the centrifugal charging pumps is evaluated for both the injection and recirculation modes of operation for the design basis accident. The end of the injection mode of operation gives the limiting net positive suction head available. The net positive suction head available is determined from the elevation head and vapor pressure of the water in the RWST, which is at atmospheric pressure, and the pressure drop in the suction piping from the tank to the pumps. No credit is taken for RWST water level above the floor plate. At the end of the injection mode when suction from the RWST is terminated, adequate net positive suction head is supplied from the containment sump by the booster action of the low head pumps. The net positive suction head evaluation is based on all pumps operating at the maximum design flow rates. The head-capacity curve for the safety injection pumps is given in Figure 6.3-3. The head-capacity curve for the charging pumps is given in Figure 6.3-4. Available NPSH parameters are given in Table 6.3-12.

6.3.2.15 Control of Motor-Operated Isolation Valves

The cold leg accumulator (CLA) valves are opened and their power removed prior to RCS pressure exceeding 1000 psig. This action assures that the CLAs are available for all plant operating conditions in which passive CLA discharge is required for accident mitigation. Power is removed by opening a shunt trip breaker, allowing the control circuit and indication to remain functional. The interlock for the CLA accumulator discharge valves to open upon receipt of the safety injection or P-11 signal remains from the original design, but this control function is obviated by removal of power and is no longer required for the accumulators to perform their safety function. A main control room alarm is actuated if any of the CLA valves are not fully open and the RCS pressure is above the P-11 permissive setpoint. A further discussion of these valves is given in Section 6.3.5.5.

6.3.2.16 Motor-Operated Valves and Controls

Certain remotely operated valves for the injection mode which are under manual control (i.e., critical valves normally in the ready position not requiring an SIS signal) have an audible alarm which is sounded in the main control room if a valve is not in the ready position for injection.

6.3.2.17 Manual Actions

No manual actions are required during the injection phase. The only actions required by the operator for proper ECCS operation following injection are those required to realign the system for cold leg recirculation and, approximately 3 hours after event initiation, its hot leg recirculation mode of operation.

6.3.2.18 Process Instrumentation

Process instrumentation is available to the operator in the control room to assist in assessing post LOCA conditions are tabulated in Section 7.5.

6.3.2.19 Materials

Materials employed for components of the ECCS are given in Table 6.3-2. These materials are chosen based upon their ability to resist radiolytic and pyrolytic decomposition (see Section 6.3.2.4). Coatings specified for use on the ECCS components (mainly, the cold leg

accumulators) are listed in Section 6.1.2.

6.3.3 Performance Evaluation

6.3.3.1 Evaluation Model

The analyses reflected in Section 15.4 were performed to ensure that the limits on core behavior following various pipe ruptures, etc., are met by the ECCS operating with minimum design equipment. The flow delivered to the RCS by the ECCS as a function of reactor coolant pressure with the operation of minimum design equipment is analyzed in Section 15.4.

The design basis performance characteristic is derived from the specified performance characteristic for each pump with a conservative estimate of system piping resistance, based upon piping layout.

The performance characteristic utilized in the accident analyses includes a 3-5% decrease in the design head for margin. When the initiating incident is assumed to be the severance of an injection line the injection curve utilized in the analysis accounts for the loss of injection water through the broken line.

6.3.3.2 ECCS Performance

The large pipe break analysis is used to evaluate the initial core thermal transient for a spectrum of pipe ruptures up to the double-ended rupture of the largest pipe in the RCS. (See Section 15.4.1 for size).

The injection flow from active components is required to control the cladding temperature subsequent to accumulator injection, complete reactor vessel refill, and will eventually return the core to a subcooled state. The results of the large break analysis indicate that the maximum cladding temperature attained at any point in the core is such that the limits on core behavior as specified in Section 15.4 are met.

6.3.3.3 Alternate Analysis Methods

Small Pipe Break

The small pipe break analysis is used to evaluate the initial core thermal transient for a spectrum of pipe rupture from 3/8-inch up to and including the ruptures defined in Section 15.3. For breaks 3/8-inch or smaller, the charging system can maintain the pressurizer level at the RCS operating pressure and the ECCS would not be actuated.

The results of the small pipe break analysis indicate that the limits on core behavior are adequately met, as shown in Section 15.3.

Main Steam System Single Active Failure

Analyses of reactor behavior following any single active failure in the main steam system which results in an uncontrolled release of steam are included in Section 15.2. The analyses assume that a single valve (largest of the safety, relief, or bypass valves) opens and fails to close, which results in an uncontrolled cooldown of the RCS. The ECCS provides adequate protection for this incident.

Steam Line Rupture

Following a steamline rupture, the ECCS is automatically actuated to deliver borated water from the RWST to the RCS. The response of the ECCS following a steam line break is identical to its response during the injection mode of operation following a LOCA.

This accident is discussed in detail in Section 15.4. The limiting steam line rupture is a complete line severance.

In the case of a steam line rupture when offsite power is not assumed lost, credit is taken for the uninterrupted availability of power for the ECCS components.

The results of the analysis in Section 15.4 indicate that the design basis criteria are met. Thus, the ECCS adequately fulfills its shutdown reactivity addition function.

The safety injection actuation signal initiates identical actions as described for the injection mode of the loss-of-coolant accident, even though not all of these actions are required following a steam line rupture; e.g., the RHR pumps are not required since the RCS pressure will remain above their shutoff head.

The delivery of the borated water from the charging pump results in a negative reactivity change to counteract the increase in reactivity caused by the system cooldown. The charging pumps continue to deliver borated water from the RWST, until enough water has been added to the RCS to make up for the shrinkage due to cooldown. The safety injection pumps also deliver borated water from the RWST for the interval when the RCS pressure is less than the shutoff head of the safety injection pumps. After pressurizer water level has been restored, the operator will verify that the criteria for "Safety Injection Termination" as defined in the Emergency Instructions are satisfied before manually terminating injection flow.

The sequence of events following a postulated steam line break is described in Section 15.4.

6.3.3.4 Fuel Rod Perforations

Discussions of peak clad temperature and metal-water reactions appear in Sections 15.3.1 and 15.4.1. Analyses of the radiological consequences of RCS pipe ruptures also are presented in Section 15.5.3.

6.3.3.5 Effects of ECCS Operation on the Core

The effects of the ECCS on the reactor core are discussed in Sections 15.3 and 15.4.

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 in each category are as follows:

1. Components of the ECCS which perform no other functions are:

    a. One accumulator for each loop which discharges borated water into its respective cold leg of the reactor coolant loop piping.

    b. Two safety injection pumps which supply borated water for core cooling to the RCS and makeup to the accumulators.

    c. Associated piping, valves, and instrumentation.

2. Components which also have a normal operating function are as follows:

    a. The RHR pumps and the residual heat exchangers: These components are normally used during reactor cooldown and heatup and when the reactor is at cold shutdown or refueling for core decay heat removal. However, during all other plant operating periods, they are aligned to perform the low head injection function.

    b. The centrifugal charging pumps: These pumps are normally aligned for charging service. As a part of the chemical and volume control system, the normal operation of these pumps is discussed in Section 9.3.4.

    c. The RWST: This tank is used to fill the refueling canal for refueling operations. However, during all other plant operating periods it is aligned to the suction of the safety injection pumps and the RHR pumps. The charging pumps are aligned to the suction of the RWST upon receipt of a safety injection signal.

An evaluation of all components required for operation of the ECCS demonstrated 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; if not aligned to its accident function, two valves in parallel are provided to align the system for injection, and two valves in series are provided to isolate portions of the system not utilized for injection. These valves are automatically actuated by a safety injection signal, except in the case of the two isolation valves in series on the hydrogen vent line for the charging pumps suction-side piping. These vent valves are actuated by closing the valves in the charging pump normal suction line from the volume control tank, which is initiated by a safety injection signal.

Table 6.3-5 indicates the alignment of components during normal operation, and the realignment required to perform the accident function.

<u>Dependence on Other Systems</u>

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 charging pumps, the safety injection pumps, and the RHR pumps during the injection and recirculation

modes of operation.

2. The essential raw water system provides cooling water to the component cooling heat exchangers and the ESF equipment room coolers.

3. The electrical systems provide 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 Operations

See the Technical Specification 3.0 for the details concerning the limiting conditions for maintenance during operations.

6.3.3.7 Lag Times

The minimum active components will be capable of delivering full rated flow within a specified time interval after process parameters reach the setpoints for the safety injection signal. Response of the system is automatic, with appropriate allowances for delays in actuation of circuitry and active components. The active portions of the system are actuated by the safety injection signal directly, with the exception of the isolation valves for the hydrogen vent lines on the charging pump suction piping. These valves are electrically interlocked to the volume control tank outlet valves. In analyses of system performance, delays in reaching the programmed trip points and in actuation of components are established on the basis that only emergency onsite power is available. A further discussion of the starting sequence is given in Section 8.3.1.

In the LOCA analysis presented in Sections 15.3 and 15.4 no credit is assumed for partial flow prior to the establishment of full flow and no credit is assumed for the availability of offsite power sources.

For smaller LOCAs, there are some additional delays before the process variables reach their respective programmed trip setpoints since this is a function of the severity imposed by the accident. Allowances are made for this in the analyses of the spectrum of reactor coolant pipe breaks.

6.3.3.8 Thermal Shock Considerations

Thermal shock considerations are discussed in Section 5.2.

6.3.3.9 Limits on System Parameters

A comprehensive qualification program has been undertaken to demonstrate that the ECCS components and associated instrumentation and electrical equipment applicable to ECCS will operate for the time period required in the combined post-loss-of-coolant accident conditions of temperature, pressure, humidity, radiation, and chemistry (See Section 3.11).

The specification of individual parameters as given in Table 6.3-1 includes due consideration of

allowances for margins over and above the required performance value (e.g., pump flow and net positive suction head), and the most severe conditions to which the component could be subjected (e.g., pressure, temperature, and flow).

6.3.3.10 Use of RHR Spray

No earlier than one hour after initiation of the LOCA, the low head RHR flow may be diverted from the core low head injection path to the RHR spray headers. For minimum safeguards, one high head safety injection pump and one centrifugal charging pump would supply the coolant to the core after realignment of a portion of the RHR pump discharge to the RHR spray headers. The amount of water which would be supplied to the core at a RCS pressure of 15 psig (which is the peak containment design pressure) is approximately 93 lbm/sec (Unit 1) or 105 lbm/sec (Unit 2). At one hour after a hypothetical LOCA the core has been quenched so that effluent carryover has been terminated. The time that effluent carryover or entrainment from the core ends is conservatively assumed to occur when the core mixture height reaches the 10 foot elevation (at approximately 150 seconds). At one hour, the thin and thick metal sensible heat has been removed and temperatures reduced to the saturation temperature for the containment pressure. The only heat generation at this time is decay heat.

The decay heat mass boil off at one hour, which is the minimum time that the RHR low head flow can be diverted to the RHR spray, is 61.5 lbm/sec based on the following assumptions:

1. 100.6% of engineered safeguards design power rating of 3579 MWt (Unit 1). 102% of engineered safeguards design power rateing of 3579 Mwt. (Unit 2).

2. ANS infinite decay heat with 20% margin (10 CFR 50.46 and 10 CFR Appendix K). Refer to Table 6.3-11.

3. Coolant entering the core is subcooled by 60 Btu/lbm.

Therefore, the coolant entering the RCS piping is roughly 1.5 to 2 times that required by conservative calculation of the decay heat mass boiloff.

It should be noted that the minimum time given above for diversion of RHR low head flow to the containment spray system is consistent with the containment pressure analysis presented in Section 6.2.1.

6.3.4 Tests and Inspections

In order to demonstrate the readiness and operability of the ECCS, the components are subjected to periodic tests and inspections. Performance tests of the components were performed in the manufacturer's shop prior to delivery. A comprehensive preoperational test program on the ECCS and its components was performed prior to initial fuel loading to provide assurance that the ECCS will accomplish its intended function when required.

6.3.4.1 Preoperational Tests - *Historical Information*

Preoperational testing of each system and component of the ECCS is to be performed in compliance with the requirements of Regulatory Guide 1.79, "Preoperational Testing of Emergency Core Cooling Systems for Pressurized Water Reactors," with the exception of the following nonconformance items.

Section	Description per Reg. Guide 1.79	Comments
C-1-b-(2)	"The testing should include taking suction from the sump to verify vortex control and acceptable pressure drops across screening and suction lines and valves."	Scale model testing has been performed to verify vortex control and demonstrate insignificant reduction of pump NPSH due to screen and trashrack pressure drops. Calculations have been performed to verify acceptable pressure drops across suction lines and valves. A flowpath from the sump will be verified by water blasting during flushing operations.

The preoperational test of the ECCS and components is discussed in more detail in Chapter 14.

6.3.4.2 Component Testing

Routine periodic testing of the ECCS components and necessary support systems is detailed in Technical Specifications as clarified below. Valves, which operate after a loss-of-coolant accident, are operated through a complete cycle where practical, and pumps are operated individually in this test on their mini-flow lines. If such testing indicates a need for corrective maintenance, the redundancy of equipment in these systems permits such maintenance to be performed without shutting down or reducing load under certain conditions. These conditions include considerations such as the period within which the component should be restored to service and the capability of the remaining equipment to provide the minimum required level of performance during such a period. The inservice component tests of ECCS pumps and valves conform, to the extent practicable allowed by plant design, to the guidelines of the 2001 Edition of the ASME OM Code with Addenda through 2003 for inservice testing of pumps and valves. Performance testing of the Auxiliary Building ECCS pump room coolers is conducted in accordance with TVA's Generic Letter 89-13 commitments.

6.3.4.3 Periodic System Testing

System testing can be conducted during plant shutdown to demonstrate proper automatic operation of the ECCS. The test program demonstrates the operation of the diesel generator loading sequence, the valves, pumps, and automatic circuitry. The accumulator isolation valve motors are deenergized and the centrifugal charging and safety injection pumps are maintained in recirculation flow for this test so that flow is not introduced into the RCS. The breakers supplying power to the safeguards busses are then tripped manually to simulate a loss of offsite power which initiates a blackout signal from the board’s undervoltage relays. Load shedding and diesel generator start signals are verified. A test emergency core-cooling signal is then applied to initiate diesel start and diesel loading. The pump and valve responses are then verified. The ability of the diesel generators to reject 600 kW without tripping is then tested. After the blackout signals (from the simulated loss of offsite power) are reset, plant configuration is restored in accordance with established procedures.

The test is considered satisfactory if control board indication and visual observation indicate all components have operated and sequenced properly. Periodic ECCS testing is detailed in Technical Specifications. The inservice inspection program described in Sections 5.2.8 and 6.6 provides further confirmation that no significant deterioration is occurring in the emergency core

cooling system fluid boundary.

6.3.4.3.1 ECCS Gas

With the exception of the operating centrifugal charging pump, the ECCS pumps are normally in a standby, nonoperating mode. As such, flow path piping has the potential to develop voids and pockets of entrained gases. Maintaining the piping from the ECCS pumps to the RCS full of water by venting the ECCS pump casings and accessible suction and discharge piping high points ensures the system will perform properly, injecting its full capacity into the RCS upon demand. This will also prevent water hammer, pump cavitation, and pumping of noncondensible gas (e.g., air, nitrogen, or hydrogen) into the reactor vessel following an SI signal or during shutdown cooling.

While lower levels of gas are ideal, due to the robust design of the ECCS system, it can still perform its design functions despite gas volumes below a specifically defined value being present in the system. An evaluation was performed on the effects of the gas on system performance which included transient effects on piping, components, and supports. The gas was also evaluated for the potential to delay the ECCS flow delivery to ensure the analyzed delivery times remained valid. An allowable gas volume for ECCS piping was calculated based on this analysis. (Reference 5)

6.3.5 Instrumentation Application

Instrumentation and associated process protection and logic channels employed for initiation of ECCS operation is discussed in Section 7.3. This section describes the instrumentation employed for monitoring ECCS components during normal plant operation and post-accident operation.

6.3.5.1 Temperature Indication

Residual Heat Exchanger Temperature

The fluid temperature at the inlet and outlet of each residual heat exchanger is recorded in the main control room.

Refueling Water Storage Tank (RWST) Temperature

Two temperature channels are provided to monitor the RWST temperature. Both are indicated in the main control room.

6.3.5.2 Pressure Indication

Safety Injection Header Pressure

Safety injection pump discharge header pressure is indicated in the control room.

Cold Leg Accumulator Pressure

Duplicate pressure channels are installed on each cold leg accumulator. Pressure indication in the control room and a common high and low pressure alarms are provided by each channel. An additional channel for each accumulator provides pressure indication and high pressure

alarm in the auxiliary control room.

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 main control room. A common high pressure main control room alarm is actuated by each channel.

6.3.5.3 Flow Indication

Charging Pump Injection Flow

Injection header flow to the reactor cold legs is indicated in the main control room.

Residual Heat Removal Pump Flow

Flow through the RHR injection header and recirculation header is indicated in the main 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 Minimum Flow

A local flowmeter installed in each RHR pump discharge header provides control for the valve located in the pump minimum flow line.

Loss of RHR Flow

An alarm is provided in the main control room to detect low RHR flow. The alarm will detect a miniflow condition coincident with the RHR pump running.

Safety Injection Pump Flow

Injection header flow to the reactor hot and cold legs is indicated in the main control room.

6.3.5.4 Level Indication

Refueling Water Storage Tank Level

Four water level channels which indicate and alarm RWST level in the main control room are provided. The low level setpoint is used in the automatic switchover (sequence described in Table 6.3-3) in a 2/4 logic. Each channel inputs to a common alarm on low and low-low water levels and is indicated on the main control board. Two additional water level channels monitor the upper tank level and provide indication and alarms in the main control room. These high

and low alarms are used to ensure adequate RWST inventory and preclude overfilling.

Cold Leg Accumulator Level

Two water level channels are provided for each tank which indicate and alarm the water level in the main control room. The common low and high level alarms ensure adequate accumulator water level.

Containment Sump Water Level

Four containment sump water level indicator channels provide the control room with water level indication and also provide a permissive signal (2 out of 4 logic) to initiate the auto-switchover from the injection to recirculation mode. A common main control room alarm is used to identify a high containment water level condition.

6.3.5.5 Valve Position Indication

The majority of the engineered safety features remote-operated valves have red and green lights on the control board to indicate valve position. The exceptions to this are discussed in Section 7.3.

Accumulator Isolation Valve Position Indication

The accumulator isolation valves are provided with red (open) and green (closed) position indication lights located on the main control room hand switch for each valve. These lights are powered by valve control power and actuated by valve motor operator limit switches.

Refueling Water Storage Tank Isolation Valve

The RWST isolation valve is provided with red (open) and green (closed) position indication lights located on the main control room hand switch. These lights are powered by valve control power and actuated by valve motor operator limit switch.

REFERENCES

1. NEI-04-07, “Pressurized Water Reactor Sump Performance Evaluation Methodology”.

2. Andreychek, Timothy S., et al., "Evaluation of Downstream Sump Debris Effects in Support of GSI 191," WCAP-16406-P, R1 (Proprietary), August 2007.

3. Ann E. Lane. et al., "Evaluation of Post-Accident Chemical Effects in Containment Sump Fluids to Support GSI-191," WCAP-16530-NP, R0 (Non-Proprietary) February 2006.

4. Andreychek, Timothy S., et al., "Evaluation of Long-Term Cooling Considering Particulate, Fibrous and Chemical Debris in the Recirculating Fluid," WCAP-16793-NP, R2-A (Non-Proprietary) September 2011.

5. NEI 09-10, “ Guidelines for Effective Prevention and Management of System Gas Accumulation.”

6.3 EMERGENCY CORE COOLING SYSTEM

The emergency core cooling system (ECCS) is discussed in detail in this section. For additional information on the ECCS see the following sections:

1. Compliance with the 10 CFR 50.46 acceptance criteria is discussed in Section 15.4.1.

2. Components which are necessary following a postulated loss-of-coolant accident (LOCA) over the entire range of break sizes are discussed in Sections 15.3 and 15.4.

3. External forces and their effect on the operation of the ECCS are treated in Sections 3.7 and 3.9.

4. Preoperational system testing is discussed in Chapter 14.

5. The actuation of the ECCS following a LOCA is discussed in detail in Section 7.3.

6. Instrumentation available to the operator to monitor conditions after a LOCA is found in Section 7.5.

7. Testing intervals are discussed in the Technical Specifications.

6.3.1 Design Bases

6.3.1.1 Range of Coolant Ruptures and Leaks

The ECCS is designed to cool the reactor core as well as to provide additional shutdown capability following initiation of the following accident conditions:

1. A pipe break or spurious valve lifting in the reactor coolant system (RCS) which causes a discharge larger than that which can be made up by the normal makeup system, up to and including the instantaneous circumferential rupture of the largest pipe in the reactor coolant system.

2. Rupture of a control rod drive mechanism causing a rod cluster control assembly ejection accident.

3. A pipe break or spurious valve lifting 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 acceptance criteria for the consequences of each of these accidents is described in Chapter 15 in the respective accident analyses sections.

6.3.1.2 Fission Product Decay Heat

The primary function of the ECCS following a LOCA is to remove 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 the accidents, as well as analyses of the accidents are provided in Chapter 15.

6.3.1.3 Reactivity Required for Cold Shutdown

The ECCS provides shutdown capability for the accidents listed above by means of chemical poison (boron) injection. The most critical accident for shutdown capability is the steam line break and for this accident the emergency core cooling system meets the criteria defined in Chapter 15. During a steam line break outside containment, the refueling water storage tank (RWST) is assumed to rupture. This could be due to a tornado induced steamline break.

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. This subject is detailed in Section 6.3.2.11.

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

The ECCS is designed to perform its function of ensuring core cooling shutdown capability following an accident under simultaneous safe shutdown earthquake loading. The seismic requirements are defined in Section 3.7.

6.3.2 System Design

6.3.2.1 Schematic Piping and Instrumentation Diagrams

Flow diagrams of the ECCS are shown in Figure 6.3-1 Sheet 1.

6.3.2.2 Equipment and Component Design

Pertinent design and operating parameters for the components of the ECCS are given in Table 6.3-1. The codes and standards to which the individual components of the ECCS are designed are listed in Table 3.2-2a.

The component design and operating conditions are specified as the most severe conditions to which each respective 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 structural integrity of the ECS components is maintained. These components are designed to withstand the appropriate seismic loadings in accordance with their safety class as given in Table 3.2-2a.

Cold Leg Injection Accumulators

These accumulators are pressure vessels filled with borated water and pressurized with nitrogen gas. One accumulator is attached to each of the cold legs of the RCS. During normal operation each accumulator is isolated from the reactor coolant system by two check valves in

series. Should the RCS pressure fall below the accumulator pressure, the check valves open and borated water is forced into the RCS. Mechanical operation of the swing-disc check valves is the only action required to open the injection path from the accumulators to the core via the cold leg. The contents of only three tanks need to be injected in order to meet initial core cooling requirements. The contents of the fourth accumulator is assumed to spill through the break.

Connections are provided for remotely adjusting the level and boron concentration of the borated water in each accumulator during normal plant operation as required. Accumulator water level may be adjusted either by draining to the holdup tank or the reactor coolant drain tank or by pumping borated water from the RWST to the accumulator using a safety injection pump.

Accumulator pressure is provided by a supply of nitrogen gas within its own volume, and can be adjusted as required during normal plant operation; however, the accumulators are normally isolated from the source of this nitrogen supply. Gas relief valves on the accumulators protect them from pressures in excess of design pressure.

The accumulators are located within the containment but outside of the secondary shield wall which protects them from missiles. Since the accumulators are located within the containment, a release of the nitrogen gas in the accumulators would cause an increase in normal containment pressure. Containment pressure following release of the gas from all accumulators when evaluated in accordance with the ideal gas law, is well below the containment pressure setpoint for ECCS actuation.

Release of accumulator gas would be detected by the accumulator pressure indicators and alarms. Thus the operator could take action promptly as required to maintain plant operation within the requirements of the Technical Specification covering accumulator operability and containment pressure.

The complete listing of the design parameters for the cold leg injection accumulator is presented in Table 6.3-1.

Pumps

Residual Heat Removal (RHR) Pumps

RHR pumps are provided to deliver water from the RWST or the containment sump to the RCS should the RCS pressure fall below their shutoff head.

Each RHR pump is a single stage, vertical position, centrifugal pump. It has an integral motor-pump shaft, driven by an induction motor. The unit has a self contained mechanical seal cooling system. Component cooling water system (CCS) is the heat exchange medium. The pumps start on receipt of a safety injection signal.

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.

The RHR pumps are also discussed in Section 5.5.7.

Centrifugal Charging Pumps

These pumps deliver water from the RWST through the boron injection tank to the RCS at the prevailing RCS pressure. Each centrifugal charging pump is a multistage, diffuser design, barrel type casing with vertical suction and discharge nozzles. The pump is driven through a speed increaser connected to an induction motor. The pump and speed increaser have self-contained lubrication systems with CCS as the heat exchanger medium. The pump has a mechanical seal cooling system. System process water is the normal heat exchange medium for the mechanical seal. The pumps start on receipt of a safety injection signal.

A minimum flow bypass line is provided on each pump discharge to recirculate flow back to the pump suction after cooling in the seal water heat exchanger. This is required to protect the pumps at the shutoff head. The minimum flow bypass line contains two valves in series which are provided for isolation of the mini-flow line. These valves are normally open with power to the valve operators locked out at each valve breaker to prevent inadvertent isolation of the mini-flow line. The charging pumps may be tested during normal operation through the use of the minimum flow bypass line. The centrifugal charging pumps are also discussed in Section 9.3.4.

Safety Injection Pumps

The safety injection pumps deliver water from the RWST to the RCS after the reactor coolant pressure is reduced below their shutoff head. Each high head safety injection pump is a multistage, centrifugal pump. The pump is driven directly by an induction motor. The unit has a self-contained lubrication system with CCS as the heat exchanger medium. The pump also has a mechanical seal cooling system. System process water is the heat exchange medium for the mechanical seals. The pumps start on receipt of a safety injection signal.

A minimum flow bypass line is provided on each pump discharge to recirculate flow to the RWST in the event the pumps are started with the normal flow paths blocked. This line also permits pump testing during normal operation. Redundant motor operated valves (MOVs) in series are provided for isolation of this line. These valves are closed by operator action during the recirculation mode.

Residual Heat Exchangers

The residual heat exchangers are conventional shell and U-tube type units. During normal operation of the RHR system, reactor coolant flows through the tube side while component cooling water flows through the shell side. During emergency core cooling recirculation operation, water from the containment sump flows through the tube side. The tubes are seal welded to the tube sheet.

A further discussion of the residual heat exchangers is found in Section 5.5.7.

<u>Valves</u>

Design parameters for all types of valves used in the ECCS are given in Table 6.3-1.

Design features employed to minimize valve leakage include:

1. Where possible, packless valves are used.

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

3. Relief valves are enclosed, i.e., they are provided with a closed bonnet.

4. Some control valves and MOVs (2 inches and above) exposed to recirculation flow have double packed stuffing boxes and stem leakoff connections to the waste processing system. Other valves may have their leakoff line connections plugged after the packing has been upgraded with graphite packing rings. This packing configuration will reduce stem leakage to essentially zero.

Table 6.3-10 provides a list of the principal ECCS valves and their respective positions during normal and all ECCS modes of operation.

Motor-Operated Valves

The seating design of all motor operated gate valves is of the parallel disc design or the flexible wedge design. These designs release the mechanical holding force during the first increment of travel so that the motor operator works 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 to provide ease of gate movement. The seating surfaces are hard faced to prevent galling and to reduce wear.

Where a gasket is employed for the body to bonnet joint, it is either a fully trapped, controlled compression, spiral wound gasket with provisions for seal welding, or it is of the pressure seal design with provisions for seal welding. The valve stuffing boxes are designed with a lantern ring leakoff connection with a minimum of a full set of packing below the lantern ring and a minimum of one-half of a set of packing above the lantern ring. A full set of packing is defined as a depth equal to 1-1/2 times the stem diameter.

The motor operator incorporates a "hammer blow" feature that allows the motor to impact the discs away from the backseat upon opening or closing. This "hammer blow" feature not only impacts the disc but allows the motor to attain its operational speed prior to impact. Valves which must function against system pressure are designed such that they function with a pressure drop equal to full system pressure across the valve disc.

Manual Globe, Gate, and Check Valves

Gate valves are either wedge design or parallel disc and are straight through. The wedge is either split or solid. All gate valves have backseat and outside screw and yoke.

Globe valves of the "T" and "Y" styles are full-ported with outside screw and yoke construction.

Check valves are spring-loaded lift piston types for sizes 2 inches and smaller and swing type or tilting disc type for size 3 inches and larger. Stainless steel check valves have no body penetrations other than the inlet, outlet and bonnet. The check hinge is serviced through the bonnet.

The stem packing and gasket of the stainless steel manual globe and gate valves are similar to those described above for MOVs.

Diaphragm Valves

The diaphragm valves are of the Saunders-patent type which uses the diaphragm member for shut off with even weir bodies. These valves are used in systems not exceeding 200°F and 220 psig design temperature and pressure, respectively.

Accumulator Check Valves

The cold leg accumulator check valve is designed with a low pressure drop configuration with all operating parts contained within the body. The disc is permitted to rotate, providing a new seating surface after each valve opening. Design considerations and analyses which assure that leakage across all the check valves located in each accumulator injection line will not impair accumulator availability are as follows:

1. During normal operation the check valves are in the closed position with a nominal differential pressure across the disc. The differential pressure is approximately 1650 psi for the check valves in the cold leg lines. Since the valves remain in this position except for testing or when called upon to function, and are not, therefore, subject to the abuses of flow operation or impact loads caused by sudden flow reversal and seating, they do not experience significant wear of the moving parts. Hence, they are expected to function with minimal leakage.

2. The check valves are tested for leakage when the RCS is being pressurized during the normal plant heat-up operation. This test confirms the seating of the disc and whether or not there has been an increase in the leakage since the last test. After this test is completed and prior to 1000 psig, the discharge line motor operated isolation valves are opened and the RCS pressure increase is continued. There should be no increase in leakage from this point on since increasing reactor coolant pressure increases the seating force and decreases the probability of leakage.

3. Experience derived from check valves employed in emergency injection systems indicates that they are reliable and workable. This is substantiated by the satisfactory experience obtained from their operation at other plants where the usage of check valves is identical to their application at Watts Bar.

4. The accumulators can accept some in-leakage from the RCS without affecting availability.

In-leakage requires, however, that the accumulator water volume and boron concentration be adjusted accordingly to remain within Technical Specification requirements. An accumulator high water level alarm is provided as an added safeguard to warn the operator that excessive accumulator in-leakage is occurring.

Relief Valves

The accumulator relief valves are sized to pass nitrogen gas at a rate in excess of the accumulator gas fill line delivery rate. The relief valves will also pass water in excess of the

expected accumulator inleakage rate. This is not considered to be necessary because the time required to fill the gas space gives the operator ample opportunity to correct the situation.

Other relief valves are installed in various sections of the ECCS to protect lines which have a lower design pressure than the RCS. Relief valves normally discharge to the pressurizer relief tank. The boron injection tank relief discharges to the CVCS holdup tank (HUT). The seal water heat exchanger (centrifugal charging pump (CCP) miniflow path) discharges to the volume control tank which in turn discharges to the CVCS HUT.

Table 3.9-20 lists the system's relief valves with their capacities and setpoints.

Butterfly Valves

Each main RHR line has an air-operated butterfly valve which is normally open and is designed to fail in the open position. These valves are in the full-open position during normal operation to maximize flow from this system to the RCS during the injection mode of the ECCS operation.

<u>Piping</u>

Piping joints are welded except where disassembly of the joint may be required. In order to assure structural integrity, pipe weld connections are fabricated to satisfy ASME Code requirements.

Minimum piping and fitting wall thicknesses are increased to account for the manufacturer's permissible tolerance on the nominal wall and an appropriate allowance for wall thinning on the external radius during any pipe bending operations in the shop fabrication of the subassemblies. The wall thicknesses are determine by formula from the 1971 ASME Code, Section III, Summer 1973 Addenda.

System Operation

The operation of the ECCS, following a loss-of-coolant accident, can be divided into two distinct modes:

1. The injection mode in which any reactivity increase following the postulated accidents is terminated, initial cooling of the core is accomplished, and coolant lost from the primary system in the case of a LOCA is replenished, and

2. The recirculation mode in which long-term core cooling is provided during the accident recovery period.

A discussion of these modes follows.

Break Spectrum Coverage

The principal mechanical components of the ECCS which provide core cooling immediately following a LOCA are the accumulators, the safety injection pumps, the centrifugal charging pumps, the RHR pumps, RWST, and the associated valves and piping.

For large pipe ruptures, the RCS would be depressurized and voided of coolant rapidly, and a high flow rate of emergency coolant is required to quickly cover the exposed fuel rods and limit

possible core damage. This high flow is provided by the passive cold leg accumulators, the charging pumps, safety injection pumps, and the RHR pumps discharging into the cold legs of the RCS. The RHR and safety injection pumps deliver into the accumulator injection lines, between the two check valves, during the injection mode. The charging pumps deliver coolant to the cold legs during the injection mode. Emergency cooling is provided for small pipe ruptures primarily by the high-head injection pumps. The charging pumps and safety injection pumps are commonly referred to as "high-head pumps" and the RHR pumps as "low-head pumps." Likewise, the term "high-head injection" is used to denote charging pump and safety injection pump injection and "low-head injection" refers to RHR pump injection. Small pipe ruptures are those, with an equivalent diameter of 6 inches or less, which do not immediately depressurize the RCS below the accumulator discharge pressure. The centrifugal charging pumps are designed to deliver borated water at the prevailing RCS pressure. During the injection mode, the charging pumps take suction from the RWST. The safety injection pumps also take suction from the RWST and deliver borated water to the cold legs of the RCS. The safety injection pumps begin to deliver water to the RCS after the pressure has fallen below the pump shutoff head.

Core protection is afforded with the minimum engineered safety feature equipment. The minimum engineered safety feature equipment is defined by consideration of the single failure criteria as discussed in Sections 6.3.1.4 and 3.1. The minimum design case ensures the entire break spectrum is accounted for and core cooling design bases of Section 6.3.1 are met. The analyses for this case are presented in Sections 15.3 and 15.4.

In the minimum design case for large RCS ruptures, the cold leg accumulators and one train of active high-head and low-head pumping components serve to complete the core refill. One RHR loop is required for long-term recirculation along with components of the auxiliary heat removal system, which are required to transfer heat from the ECCS to the component cooling system and essential raw cooling water system.

If the break is small (6-inch equivalent diameter or less) the accumulators with one charging pump and one safety injection pump ensure adequate cooling during the injection mode. Long-term recirculation requires one RHR loop and components of the auxiliary heat removal systems. The loss-of-coolant analyses are presented in Section 15.3 and 15.4.

Certain deviations (i.e., reduced component availability) from the normal operating status as given in Table 6.3-4 of the ECCS are permissible without appreciably impairing the reliability of the ECCS to provide adequate core cooling capability. Accordingly, Technical Specifications have been established to identify these types of deviations and restrict the time period that a given deviation may exist.

The Technical Specifications permit one cold leg accumulator and various pumps of the ECCS to be inoperable during power operation for a period of time. The permissible time periods for which accumulators, ECCS pumps, and associated equipment may be inoperable are listed and their bases described in the Technical Specifications.

The minimum active components will be capable of delivering full rated flow within a specified time interval after process parameters reach the setpoints for the safety injection signal. Response of the system is automatic, with appropriate allowances for delays in actuation of circuitry and active components. The active portions of the system are actuated by the safety injection signal directly, with the exception of the isolation valves for the hydrogen vent lines on the charging pump suction piping.

These valves are electrically interlocked to the volume control tank outlet valves. In analyses of system performance, delays in reaching the programmed trip points and in actuation of components are established on the basis that only emergency onsite power is available. The starting sequence is detailed in Table 8.3-3.

In the loss-of-coolant accident analyses presented in Sections 15.3 and 15.4 no credit is assumed for partial flow prior to the establishment of full flow and no credit is assumed for the availability of normal offsite power sources. For smaller loss-of-coolant accidents, there is some additional delay before the process variables reach their respective programmed trip setpoints since this is a function of the severity of the transient imposed by the accident. This is allowed for in the analyses of the range of loss-of-coolant accidents.

Accumulator injection occurs immediately when the RCS is depressurized below accumulator operating pressure. For the cold leg injection accumulator this setpoint will be reached only in event of a large rupture.

The cold leg injection accumulators can be isolated from the RCS by closure of their motor-operated isolation valves. Since these accumulators operate only after considerable RCS pressure loss, the injection of pressurized nitrogen via the cold legs is not considered a problem.

Injection Mode After Loss of Primary Coolant

The injection mode of emergency core cooling is initiated by the safety injection signal ("SI" signal). This signal is initiated by any of the following:

1. Low pressurizer pressure

2. High containment pressure

3. Low steamline pressure in any steamline.

4. Manual actuation

Operation of the ECCS during the injection mode is completely automatic. Refer to Figure 7.3-3 (Sheet 3) for safety injection signal logic. The safety injection signal in addition to activating the ESF equipment automatically initiates the following actions:

1. Starts the diesel generators and trips the diesel generator feeder breaker if the diesel generator is in test with the offsite power source. They will be aligned to the 6.9 kV shutdown boards if power is lost to the respective board.

2. Starts the charging pumps, the safety injection pumps, and the RHR pumps. (3) Aligns the charging pumps for injection by:

    a. Closing the valves in the charging pump discharge line to the normal charging line.

    b. Opening the valves in the charging pumps suction line from the RWST.

    c. Closing the valves in the charging pump normal suction line from the volume control tank when either of the two RWST suction valves are fully open. Closing these valves

initiates the closing of the isolation valves at the hydrogen vent line for the charging pumps suction side piping.

d. Opening the isolation valves located in the discharge line from the boron injection tank.

The injection mode continues until the low level is reached in the RWST coincident with a high level in the containment sump. Then the recirculation mode is initiated.

Recirculation Mode

The injection mode continues until the RHR pumps have been realigned to the recirculation mode. During the injection mode, all pumps take suction from the RWST until a low level signal from the RWST in conjunction with the "SI" signal and a high sump level signal aligns the RHR pumps to take suction from the containment sump. The RHR RWST isolation valves (FCV-74-3 and -21) are automatically closed coincident with the opening of the sump isolation valves (FCV-63-72 and -73). The automatic positioning of these valves is initiated only in the event that actuation signals are generated by the safeguards protection logic ("SI" signal), two of four RWST low level protection logic signals, and two out of four sump high level signals. It has been determined that the RHR pumps continue to receive adequate suction flow during this automatic changeover, thus there is no possibility of pump damage due to loss of suction. Alarms on RWST low level and level indications from both the sump and RWST are used by the operator to appraise the accident situation and complete the remainder of switchover sequence.

Table 6.3-3 describes the sequence of changeover operation from injection to recirculation.

The switchover initiation point and minimum assured final volume in the RWST before completion of switchover are selected on the basis of maximizing the allowable operator action time for accompanying manual operations and total water injected to the RCS while avoiding the potential problems due to low levels in either the active sump inside containment or in the RWST. Crane wall penetrations inside containment are sealed as necessary between elevations 702.78 and 716 to initially retain more water in the active sump, thereby maximizing the active sump water level at the onset of the recirculation switchover.

The sequence (as delineated in Table 6.3-3) is followed regardless of which power supply is available (offsite or emergency onsite).

The time required to complete the sequence is essentially the time required for the operator to perform the accompanying manual operations. Controls for ECCS components are grouped together on the main control board. The component position lights indicate equipment position / status.

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 low head recirculation flow path consists of the RHR pumps taking suction from the containment sump and discharging the flow directly to the RCS through the residual heat exchangers and cold leg injection lines. The high head recirculation flow path consists of the residual heat removal pumps taking suction from the containment sump and discharging the flow through the residual heat exchangers to the suction of the centrifugal charging pumps and safety injection pumps. The flow from the centrifugal charging pumps and safety injection pumps is returned to the RCS through the cold leg

injection lines. The latter mode of operation assures flow in the event of a small rupture where the depressurization proceeds more slowly such that the reactor coolant system pressure is still in excess of the shutoff head of the residual heat removal pumps at the onset of recirculation.

Approximately 3 hours after the event initiation, hot leg recirculation will be initiated to assure against an excessive buildup of boric acid in the core.

The containment sump isolation valve is interlocked with its respective pump suction/RWST isolation valve to the RHR system. The interlock is provided with redundant signals from each isolation valve. This interlock prevents remote manual opening the sump isolation valve when the RWST isolation valves are open and thus prevents dumping the RWST contents into the containment sump. However, when an accident signal is present, this interlock is bypassed to allow initiation of the switchover sequence.

The RWST is protected from back flow of reactor coolant from the RCS. All connections to the RWST are provided with check valves to prevent back flow. When the RCS is hot and pressurized there is no direct connection between the RWST and the RCS. When the RCS is being cooled and the RHR system is placed in service, the RHR system is isolated from the RWST by a motor-operated valve in addition to a check valve.

Redundancy in the external recirculation loop is provided by the inclusion of duplicate charging, safety injection, and RHR pumps and residual heat exchangers. Inside the containment, the charging pump and safety injection pump discharge is piped separately into all four cold legs for the charging pumps and into all four cold legs and all four hot legs for the safety injection pumps. The low head pumps take suction through redundant lines from the containment sump and discharge through separate paths to the RCS. The containment sump design is shown in Figure 6.3-6A.

The containment sump is located in the containment floor (el 702.78 ft, 270°) below the refueling canal to provide protection from high energy pipe failures. A strainer assembly is installed on top of the sump suction pit to prevent debris that may be present after a design basis accident from degrading performance of the ECCS and containment spray system. A horizontal grating is located one foot below the ceiling to eliminate vortexing. This pit is surrounded by a six-inch high curb which is used to prevent sediment from entering the pit. A fine mesh (1/4-inch) screen located in the containment sump suction pit is used to divide the sump into two suction volumes. The sump design does not comply fully with Regulatory Guide 1.82, Revision 0 because the plant design was well advanced when the Regulatory Guide was issued; the plant construction permit preceded the Regulatory Guide by 17 months. The design does comply fully, however, except as itemized below:

Position C.1: Dual sumps are not provided. The single sump, however, has adequate capacity and inlet area.

Position C.3: The containment sump pit has a six-inch high curb with a metal strainer assembly on top of it. This arrangement provides vortex suppression. The sump intake is not protected by two screens. It is protected by an advanced design strainer assembly that can withstand the expected debris loading and differential pressure. The perforation size is a 0.085 inch diameter hole which prevents debris from entering the inner sump.

Position C.4: The floor is level, but a six-inch high curb is provided around the sump suction pit inlet, thus providing an effect comparable to a sloped floor.

Position C.6: The trash rack and inner screen have been replaced with an advanced design strainer which has been designed to be strong enough to withstand the expected debris loading and differential pressure.

Position C.7: The strainer assembly has mainly horizontal straining surfaces, but has been designed to obtain low approach velocity due to significant surface area. The design configuration of the entire assembly impedes the deposition or settling of debris on the strainer surfaces.

The limiting size particle which may be circulated by the ECCS and containment spray systems without causing system damage is a function of several physical parameters including:

A. limiting system clearance,

B. particle concentration,

C. particle abrasive properties, and

D. particle hardness.

The sump screen openings are sized sufficiently small to protect other components from debris plugging that would challenge the operability of mitigative systems. Soft material of larger dimensions could easily be passed by the pumps, but would cause blockage of these components. Damage to the pumps could occur if significant concentrations of hard, dense, smaller debris were allowed to cause surface abrasion or binding between moving pump parts. To eliminate all such particles by smaller screen sizing would increase the threat of screen blockage. Hence, the design philosophy for the sump is to size the sump screens according to the limiting clearance and to otherwise take advantage of settling properties to eliminate the threat of damage from the smaller, more dense particles.

No debris is expected to reach the sump and cause blockage of the sump screens during or after a LOCA. However, the effective screen opening areas are many times larger than the combined flow area of both sump suction pipes to allow appreciable blockage from unspecified debris before any significant pressure drop is developed across the screens.

The lower containment is an open, one level area and no drains are used to route water to the sump (except the two large refueling cavity drains and two small accumulator room drains that route water away from the strainer assembly). The water simply fills the floor area and covers the sump entrance. Debris with a specific gravity greater than one will largely settle before reaching the sump inlet.

The containment sump suction pit (located in the inner sump area) has a six-inch high curb with the strainer assembly atop it. The strainer perforations are 0.085 inch diameter holes which prevent debris from entering the sump. It also serves to maintain a low inlet velocity. Debris generated during the initial blowdown of a LOCA will have an approximate minimum time of ten minutes to settle before any suction is taken from the sump.

Peeling paint has been identified as a possible hazard to the containment sump. To prevent this hazard, surface coatings will meet the requirements as described in Section 6.1.2. Further, paint used inside containment produces a hardened film having a specific gravity appreciably greater than one. Hence, gravitational settling helps assure protection against plugging of the sump with paint particles.

Equipment insulation has been designed to assure against it becoming a source of sump blockage. This has been done by providing metallic insulation on reactor coolant pressure boundary vessels and piping where required. The systems and components utilizing metallic insulation include the reactor vessel, steam generators, pressurizer, reactor coolant pumps and piping, RHR piping, safety injection piping, chemical and volume control piping, and main steam and feedwater piping. The insulation is not designed to withstand the blowdown and jet impingement forces associated with pipe breaks and sections of insulation may be stripped away if a break occurs. However, the strainer assembly will prevent the metallic insulation from entering the containment sump.

Small sections of pipe inside containment are insulated with stainless-steel-jacketed mass-type insulation, as needed, to avoid interferences with and potential overheating of electrical cable and components. As is the case with metallic insulation, this insulation is not designed to withstand the blowdown and jet impingement forces associated with pipe breaks. However, it also poses no significant hazard to the containment sump if a pipe break were to occur. The metal jacketing may sink if located in the vicinity of the sump, but it is prevented from entering the sump by the strainer assembly. The mass-type insulation floats since its specific gravity is appreciably less than one, and it is hydrophobic grade to prevent absorption of water.

Small amounts of rockwool insulation are located inside the guard pipes of containment penetrations with flued heads as shown in Figures 6.2.4-1 through 6.2.4-4 and 6.2.4-10. This insulation is enclosed and protected by the guard pipes.

Ice condenser insulation is specifically designed not to create debris during a LOCA. Materials that could become segmented or decompose are enclosed. Solid insulation such as foam concrete is used where practical and protection is provided to prevent insulation damage from the blowdown effects of a LOCA. All equipment inside containment is designed to prevent its becoming a source of blockage to the sump. Each recirculation line from the sump is run outside the containment to a sump isolation valve. This valve is surrounded with a steel enclosure and the section of piping joining it to the sump is run within a guard pipe welded to the enclosure. Any excessive leakage or passive failure downstream of the sump valves can be controlled and isolated by closure of the sump valve in the affected train.

NRC Generic Letter 2004-02

The ECCS containment sump design addresses the potential post-LOCA ECCS performance issues provided in NRC Generic Letter 2004-02, “Potential Impact of Debris Blockage on Emergency Recirculation during Design Basis Accidents at Pressurized-Water Reactors.” The containment sump design has been tested and analyzed based on guidance provided in NEI-04-07, “Pressurized Water Reactor Sump Performance Evaluation Methodology”, as supplemented by the NRC in the “Safety Evaluation by the Office of Nuclear Reactor Regulation Related to NRC GL 2004-07.” Downstream effects were evaluated in accordance with Topical Report WCAP-16406-P, “Evaluation of Downstream Sump Debris Effects in Support of GSI-191.” Chemical effects were evaluated based on testing, WCAP-16530-NP, “Evaluation of Post-Accident Chemical Effects in Containment Sump Fluids to Support GSI-

191," and WCAP-19793-NP, "Evaluation of Long-Term Cooling Considering Particulate, Fibrous and Chemical Debris in the Recirculating Fluid." The tests and evaluations confirm that the required safety functions of the containment sump and ECCS system will be met during design basis accidents considering the effects of debris.

External Recirculation Loop

The ECCS recirculation loop piping and components external to containment are surrounded by shielding. This shielding is designed to permit access for maintenance to a component such as a pump while the redundant pump is recirculating sump fluid. Pressure relieving devices from portions of the ECCS located outside containment which might contain radioactivity discharge to the pressurizer relief tank, except for the BIT and seal water heat exchanger relief valves, which discharge to the CVCS holdup tank and VCT, respectively.

An analysis has been performed to evaluate the radiological effects of recirculation loop leakage as discussed in Section 15.5.3.

During recirculation, significant margin exists between the design and operating conditions (in terms of pressure and temperature) of the ECCS components.

Since redundant flow paths are provided during recirculation, a leaking component in one of the flow paths may be isolated. This action curtails any further leakage and renders the component available for corrective maintenance. Maximum potential leakage from components during recirculation mode operation is given in Table 6.3-6.

6.3.2.3 Applicable Codes and Classifications

The codes and standards to which the individual components of the ECCS are designed are listed in Table 3.2-2a.

6.3.2.4 Materials Specifications and Compatibility

Materials employed for components of the ECCS are given in Table 6.3-2. Materials are selected to meet the applicable material requirements of the codes in Table 3.2-2a and the following additional requirements:

1. The parts of components in contact with sump solution during recirculation are fabricated of austenitic stainless steel or equivalent corrosion resistant material.

2. The parts of components in contact with borated water are fabricated of, or clad with, austenitic stainless steel or equivalent corrosion resistant material, with the exception of pump seals and valve packing.

3. Valve seating surfaces are hard-faced with Stellite No. 6 or equivalent to prevent galling and 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 is well within the design temperature of all the 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 the long term recirculation operations.

Environmental qualification of the ECCS equipment inside the containment, which is required to operate following a LOCA is discussed in Section 3.11. The results of the program indicate that the safety features will operate satisfactorily during and following exposure to the combined containment post-accident environments of temperature, pressure, chemistry, and radiation.

6.3.2.5 Design Pressures and Temperatures

The component design pressures and temperatures are given in Table 6.3-1. These pressure and temperature conditions are specified as the most severe conditions to which each respective 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 (see Section 3.2) and with due consideration for the design and operating conditions, the fundamental assurance of structural integrity of the ECCS components is maintained.

6.3.2.6 Coolant Quantity

The minimum storage volume for the accumulators and the RWST is given in Table 6.3-4. The minimum storage volume in the RWST and the accumulators is sufficient to ensure that, after a RCS break, sufficient water is injected and is available within the containment to permit recirculation cooling flow to the core, and to meet the net positive suction head requirements of the RHR pumps. A further discussion of coolant requirements is contained in Sections 15.3 and 15.4.

6.3.2.7 Pump Characteristics

Design parameters for the ECCS pumps are given in Table 6.3-1.

6.3.2.8 Heat Exchanger Characteristics

Residual heat exchanger characteristics are found in Section 5.5.7.

6.3.2.9 ECCS Flow Diagrams

The SIS flow diagram is given as Figure 6.3-1 Sheet 1.

6.3.2.10 Relief Valves

The ECCS relief valves, their capacities and settings are given in Table 3.9-20.

6.3.2.11 System Reliability

6.3.2.11.1 Definitions

Period of Recovery

The period of recovery is the time necessary to bring the plant to a cold shutdown and regain access to faulted equipment. The recovery period is the sum of the short and long term periods defined below.

Incident

An incident is any natural or accidental event of infrequent occurrence and its related consequences which affect plant operation and require the use of engineered safeguards systems. Such events, which are analyzed independently and are not assumed to occur simultaneously, include the loss-of-coolant accident, steam line ruptures, steam generator tube ruptures, etc. A loss of off site power event may be an isolated occurrence or may be concurrent with any event requiring engineered safeguards systems use.

Short Term

The short-term time period is the time immediately following the incident during which automatic actions are performed, system responses are checked, type of incident is identified and preparations for long-term recovery operation are made. The short term is the injection phase for LOCA and is the first 24 hours following initiation of the event for all others.

Long Term

The long term time period is the remainder of the recovery period following the short term. In comparison with the short term, where the main concern is to prevent or limit site release, the long-term period of operation involves bringing the plant to cold shutdown conditions where access to the containment can be gained and repairs effected.

Active Failure

An active failure is the failure of a powered component, such as a piece of mechanical equipment or a component of the electrical supply system or instrumentation and control equipment, to act on command to perform its design function. Examples include the failure of a motor-operated valve to move to its correct position, the failure of an electrical breaker or relay to respond, the failure of a pump, fan or diesel generator to start, etc.

Passive Failure

A passive failure is the structural failure of a static component which limits the component's effectiveness in carrying out its design function. When applied to a fluid system, this means a break in the pressure boundary resulting in abnormal leakage not exceeding 50 gpm for 30 minutes. Such leak rates are consistent with limited cracks in pipes, sprung flanges, valve packing leaks or pump seal failures.

6.3.2.11.2 Active and Passive Failure Criteria

Active Failure Criteria

The ECCS is designed to accept any single failure at any time following the incident without loss of its protective function. The system design will tolerate the failure of any single active component in the ECCS itself or in the necessary associated service systems at any time during the period of required system operations following the incident.

A single active failure analysis is presented in Table 6.3-8, and demonstrates that the ECCS can sustain the failure of any single active component in either the short or long term and still meet the level of performance for core cooling.

Since the initial operation of the active components of the ECCS following a steam line rupture is identical to that following a LOCA, the same analysis is applicable and the ECCS can sustain the failure of any single active component and still meet the level of performance for the addition of shutdown reactivity. Passive failure is not considered for the short term.

Passive Failure Criteria

The following philosophy provides for necessary redundancy in component and system arrangement to meet the intent of the General Design Criteria on single failure as it specifically applies to failure of passive components in the ECCS. Thus, for the long term, the system is based on accepting either a passive or an active failure.

Redundancy of Flow Paths and Components for Long-Term Emergency Core Cooling

In the design of the ECCS, Westinghouse utilized the following criteria:

1. During the long-term cooling period following a loss of coolant, the emergency core cooling flow paths are separable into two subsystems, either of which can provide minimum core cooling functions and return spilled water from the floor of the containment back to the RCS.

2. Either of the two subsystems can be isolated and removed from service in the event of a leak outside the containment.

3. Adequate redundancy of check valves is provided to tolerate failure of a check valve during the long term as a passive component.

4. Should one of these subsystems be isolated in this long-term period, the other subsystem remains operable.

5. Provisions are also made in the design to detect leakage from components outside the containment, collect this leakage and to provide for limited maintenance of the affected equipment.

Thus, for the long-term emergency core cooling function, adequate core cooling capacity exists with one flow path removed from service whether isolated due to a leak, because of blocking of one flow path, or because failure in the containment results in a spill of the delivery of one injection flow path.

The design of the ECCS includes the provision for diversion of a portion of the RHR pump flow from the low head injection path to auxiliary spray headers in the upper containment volume. For this mode the RHR pumps continue to supply recirculation flow from the containment sump to the core via the safety injection and centrifugal charging pumps.

The diversion of the RHR flow from the low head injection path to the auxiliary spray headers occurs only after the switchover to the recirculation mode and no earlier than 1 hour after initiation of the LOCA. When RHR spray is required, the operator is provided with a detailed

procedure (Table 6.3-3) to follow in aligning the system for RHR spray operation. This procedure requires that low head safety injection flow to the core be terminated under single train operating condition prior to initiating RHR spray flow.

6.3.2.11.3 Subsequent Leakage from Components in Safeguards Systems

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 leak rate assuming only the presence of a seal retention ring around the pump shaft showed 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 significantly less severe than the pump seal failure.

Larger leaks in the ECCS are prevented by the following:

1. The piping is classified in accordance with ANS Safety Class 2 and receives the ASME Class 2 quality assurance program associated with this safety class.

2. The piping, equipment and supports are designed to ensure no loss of function for the safe shutdown earthquake.

3. The system piping is located within a controlled area on the plant site.

4. The piping system receives periodic pressure tests and is accessible for periodic visual inspection.

5. The piping is austenitic stainless steel which, due to its ductility, can withstand severe distortion without failure.

Based on this review, the design of the Auxiliary Building and related equipment is based upon handling of ECCS leaks up to a maximum of 50 gpm. To assure adequate core cooling, design features are provided to prevent this limiting passive failure from causing any loss of function in the other train of the ECCS equipment due to flooding of redundant components or loss of NPSH to the ECCS pumps. Three independent means are available to provide information to the operator for use in identifying ECCS leakage into certain locations in the Auxiliary Building. These means include the Auxiliary Building flood detection system, the instrumentation and alarms associated with the drainage and waste processing systems which normally handle drainage into these areas, and redundant level indicators in the Auxiliary Building passive sump.

A flood detection system, utilizing water level detector devices, is used to monitor and actuate alarms for ECCS and other leakage at locations throughout the Auxiliary Building. Individual detectors are located in each ECCS pump compartment, in the ECCS heat exchanger rooms, and in the pipe gallery (elevation 676). A common alarm in the main control room will alert the operator when any of these flood detectors are tripped. A flood detector indicator panel, located immediately outside the control room, then identifies the exact location of the tripped detector. The detectors were preoperationally tested to verify initial operability and will be periodically tested as a part of the plant maintenance program.

Since each ECCS pump compartment is monitored by a level detection device, the operator may immediately identify which subsystem must be shut down and secured to terminate the leak.

The operator can readily accomplish this action from the main control room by stopping the appropriate subsystem pump, and by closing the corresponding sump isolation valves and individual pump discharge valves. The time necessary for the operator to detect leakage in a pump compartment is dependent on the leakage rate. A limiting 50 gpm leak in the largest ECCS pump compartment can be detected within 30 minutes. Slower leaks may require proportionally longer detection times.

Leakage into these ECCS pump compartments is piped to the floor drain collector tank for the safety injection and centrifugal charging pumps and to the Auxiliary Building floor and equipment drain sump for the RHR and containment spray pumps. The drain in each of these rooms is provided with a standpipe which assures that the setpoint for the level detector is reached prior to draining the leakage from the room. However, the standpipes each have two 1/2-inch drilled holes to allow minor normal leakage to drain from the room.

ECCS leakage into the Auxiliary Building locations other than the ECCS pump compartments is piped to the Auxiliary Building floor and equipment drain sump. This sump is provided with redundant 50 gpm pumps which are indicated in the main control room. The floor drain collector tank is provided with overflow piping which discharges to the Auxiliary Building floor and equipment drain sump. Leakage into these areas can be detected by the flood detection system described above, by indication of sump pump operation, or by high level alarm from the sump or the floor drain collector tank. However, the exact location of the leak, if other than the ECCS pump compartment, or the subsystem from which leakage occurs, may not be immediately identified. Since ECCS leaks other than a pump seal failure are of a nature to develop very slowly and are less severe than a seal failure, the operator has an extended time period to detect and isolate the leak. Isolation of these minor leaks can be accomplished by arbitrarily selecting and isolating an ECCS subsystem and evaluating the response of the flood detector system.

The flood detection system described above is not designed to meet the requirements of IEEE-279. The detectors, indicator panel, and control room alarm are powered from nondivisional boards and do not meet the single failure criteria. However, the system is designed such that a loss of power to any individual detector will actuate the main control room common alarm. Additionally, the nondivisional boards which supply the flood detector system are powered from a Class 1E power board which is automatically loaded on the diesel generators. This ensures continued power availability to the flood detection system after an accident.

In addition to the flood detection and normal drainage processing systems described above, redundant level sensors which do meet the requirements of IEEE-279 are provided in the Auxiliary Building passive sump. These sensors, which are a part of the post accident monitoring (PAM) system as described in Section 7.5, are designed to continuously indicate and record the passive sump level. Also, the Auxiliary Building is provided with redundant ESF grade air cleanup and filtration systems as described in Section 6.5.1.

With these design ground rules, continued function of the ECCS meets minimum core cooling requirements. A single passive failure analysis is presented in Table 6.3-9. It demonstrates that the ECCS can sustain a single passive failure during the long term phase and still retain an intact flow path to the core to supply sufficient flow to maintain the core covered and affect the

removal of decay heat. An event resulting in maximum leakage would have an insignificant impact on ECCS capability since a 100% capacity redundant train is available to assure the ECCS capability and since the maximum leakage represents less than 0.5% of system flow capacity.

6.3.2.12 Protection Provisions

The provisions taken to protect the system from damage that might result from dynamic effects on piping systems are discussed in Section 3.6. The provisions taken to protect the system from missiles are discussed in Section 3.5. The provisions to protect the system from seismic damage are discussed in Sections 3.7, 3.9, and 3.10. Thermal stresses on the reactor coolant system are discussed in Section 5.2.

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 Net Positive Suction Head

The ECCS is designed so that adequate net positive suction head is provided to system pumps. See Table 6.3-12, Available NPSH During ECCS Operation, for a complete listing of available NPSH for all pumps in ECCS while operating. Adequate net positive suction head is shown to be available for all pumps as follows:

1. Residual Heat Removal Pumps

The net positive suction head of the RHR pumps is evaluated for normal plant shutdown operation, and for both the injection and recirculation modes of operation for the design basis accident. Recirculation operation gives the limiting net positive suction head requirement, and the net positive suction head available is determined from the containment pressure, vapor pressure of liquid in the sump, containment sump level relative to the pump elevation and the pressure drop in the suction piping from the sump to the pumps. A containment pressure of zero psig (building pressure of 14.3 psia) is used in calculating the most limiting (minimum) NPSHA. No credit is taken for water level above the RHR sump strainer assembly, and no credit is taken for containment accident pressure. The net positive suction head evaluation is based on all pumps operating at the maximum design basis accident flow rates. The RHR pump head-capacity curves are given in Figure 6.3-2.

2. Safety Injection and Centrifugal Charging Pumps

The net positive suction head for the safety injection pumps and the centrifugal charging pumps is evaluated for both the injection and recirculation modes of operation for the design basis accident. The end of the injection mode of operation gives the limiting net positive suction head available. The net positive suction head available is determined from the elevation head and vapor pressure of the water in the RWST, which is at atmospheric pressure, and the pressure drop in the suction piping from the tank to the pumps. At the end of the injection mode when suction from the RWST is terminated, adequate net positive suction head is supplied from the containment sump by the booster action of the low head pumps. The net positive suction head evaluation is based on all pumps operating at the maximum design flow rates. The head-capacity curve for the safety injection pumps is given in Figure 6.3-3. The head-capacity curve

for the charging pumps is given in Figure 6.3-4. Available NPSH parameters are given in Table 6.3-12.

6.3.2.15 Control of Motor-Operated Isolation Valves

The cold leg accumulator (CLA) valves are opened and their power removed prior to RCS pressure exceeding 1000 psig. This action assures that the CLAs are available for all plant operating conditions in which passive CLA discharge is required for accident mitigation. Power is removed by opening a shunt trip breaker, allowing the control circuit and indication to remain functional. The interlock for the CLA accumulator discharge valves to open upon receipt of the safety injection or P-11 signal remains from the original design, but this control function is obviated by removal of power and is no longer required for the accumulators to perform their safety function. A main control room alarm is actuated if any of the CLA valves are not fully open and the RCS pressure is above the P-11 permissive setpoint. A further discussion of these valves is given in Section 6.3.5.5.

6.3.2.16 Motor-Operated Valves and Controls

Certain remotely operated valves for the injection mode which are under manual control (i.e., critical valves normally in the ready position not requiring an SIS signal) have an audible alarm which is sounded in the main control room if a valve is not in the ready position for injection.

6.3.2.17 Manual Actions

No manual actions are required to support the injection phase. The only actions required by the operator for proper ECCS operation during injection are those required to realign the system for cold leg recirculation and, approximately 3 hours after event initiation, its hot leg recirculation mode of operation.

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 7.5.

6.3.2.19 Materials

Materials employed for components of the ECCS are given in Table 6.3-2. These materials are chosen based upon their ability to resist radiolytic and pyrolytic decomposition (see Section 6.3.2.4). Coatings specified for use on the ECCS components (mainly, the cold leg accumulators) are listed in Section 6.1.2.

6.3.3 Performance Evaluation

6.3.3.1 Evaluation Model

The analyses reflected in Section 15.4 were performed to ensure that the limits on core behavior following various pipe ruptures, etc., are met by the ECCS operating with minimum design equipment. The flow delivered to the RCS by the ECCS as a function of reactor coolant pressure with the operation of minimum design equipment is analyzed in Section 15.4.

The design basis performance characteristic is derived from the specified performance characteristic for each pump with a conservative estimate of system piping resistance, based upon piping layout.

The performance characteristic utilized in the accident analyses includes a 5 percent decrease in the design head for margin. When the initiating incident is assumed to be the severance of an injection line the injection curve utilized in the analysis accounts for the loss of injection water through the broken line.

6.3.3.2 ECCS Performance

The large pipe break analysis is used to evaluate the initial core thermal transient for a spectrum of pipe ruptures up to the double-ended rupture of the largest pipe in the RCS. (See Section 15.4.1 for size).

The injection flow from active components is required to control the cladding temperature subsequent to accumulator injection, complete reactor vessel refill, and will eventually return the core to a subcooled state. The results of the large break analysis indicate that the maximum cladding temperature attained at any point in the core is such that the limits on core behavior as specified in Section 15.4 are met.

6.3.3.3 Alternate Analysis Methods

Small Pipe Break

The small pipe break analysis is used to evaluate the initial core thermal transient for a spectrum of pipe rupture from 3/8-inch up to and including the ruptures defined in Section 15.3. For breaks 3/8-inch or smaller, the charging system can maintain the pressurizer level at the RCS operating pressure and the ECCS would not be actuated.

The results of the small pipe break analysis indicate that the limits on core behavior are adequately met, as shown in Section 15.3.

Main Steam System Single Active Failure

Analyses of reactor behavior following any single active failure in the main steam system which results in an uncontrolled release of steam are included in Section 15.2. The analyses assume that a single valve (largest of the safety, relief, or bypass valves) opens and fails to close, which results in an uncontrolled cooldown of the RCS. The ECCS provides adequate protection for this incident.

Steam Line Rupture

Following a steamline rupture, the ECCS is automatically actuated to deliver borated water from the RWST to the RCS. The response of the ECCS following a steam line break is identical to its response during the injection mode of operation following a LOCA.

This accident is discussed in detail in Section 15.4. The limiting steam line rupture is a complete line severance.

In the case of a steam line rupture when offsite power is not assumed lost, credit is taken for the uninterrupted availability of power for the ECCS components.

The results of the analysis in Section 15.4 indicate that the design basis criteria are met. Thus, the ECCS adequately fulfills its shutdown reactivity addition function.

The safety injection actuation signal initiates identical actions as described for the injection mode of the loss-of-coolant accident, even though not all of these actions are required following a steam line rupture; e.g., the RHR pumps are not required since the reactor coolant system pressure will remain above their shutoff head.

The delivery of the borated water from the charging pump results in a negative reactivity change to counteract the increase in reactivity caused by the system cooldown. The charging pumps continue to deliver borated water from the RWST, until enough water has been added to the RCS to make up for the shrinkage due to cooldown. The safety injection pumps also deliver borated water from the RWST for the interval when the RCS pressure is less than the shutoff head of the safety injection pumps. After pressurizer water level has been restored, the operator will verify that the criteria for "Safety Injection Termination" as defined in the Emergency Instructions are satisfied before manually terminating injection flow.

The sequence of events following a postulated steam line break is described in Section 15.4.

6.3.3.4 Fuel Rod Perforations

Discussions of peak clad temperature and metal-water reactions appear in Sections 15.3.1 and 15.4.1. Analyses of the radiological consequences of RCS pipe ruptures also are presented in Section 15.5.3.

6.3.3.5 Effects of ECCS Operation on the Core

The effects of the ECCS on the reactor core are discussed in Sections 15.3 and 15.4.

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 in each category are as follows:

1. Components of the ECCS which perform no other functions are:

   a. One accumulator for each loop which discharges borated water into its respective cold leg of the reactor coolant loop piping.

   b. Two safety injection pumps which supply borated water for core cooling to the reactor coolant system and makeup to the accumulators.

   c. Associated piping, valves, and instrumentation.

2. Components which also have a normal operating function are as follows:

a. The RHR pumps and the residual heat exchangers: These components are normally used during reactor cooldown and heatup and when the reactor is at cold shutdown or refueling for core decay heat removal. However, during all other plant operating periods, they are aligned to perform the low head injection function.

b. The centrifugal charging pumps: These pumps are normally aligned for charging service. As a part of the chemical and volume control system, the normal operation of these pumps is discussed in Section 9.3.4.

c. The RWST: This tank is used to fill the refueling canal for refueling operations. However, during all other plant operating periods it is aligned to the suction of the safety injection pumps and the RHR pumps. The charging pumps are aligned to the suction of the RWST upon receipt of a safety injection signal.

An evaluation of all components required for operation of the ECCS demonstrated 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; if not aligned to its accident function, two valves in parallel are provided to align the system for injection, and two valves in series are provided to isolate portions of the system not utilized for injection. These valves are automatically actuated by a safety injection signal, except in the case of the two isolation valves in series on the hydrogen vent line for the charging pumps suction-side piping. These vent valves are actuated by closing the valves in the charging pump normal suction line from the volume control tank, which is initiated by a safety injection signal.

Table 6.3-5 indicates the alignment of components during normal operation, and the realignment required to perform the accident function.

<u>Dependence on Other Systems</u>

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 charging pumps, the safety injection pumps, and the RHR pumps during the injection and recirculation modes of operation.

2. The essential raw water system provides cooling water to the component cooling heat exchangers and the ESF equipment room coolers.

3. The electrical systems provide 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.

<u>Limiting Conditions for Maintenance During Operations</u>

See the Technical Specification 3.0 for the details concerning the limiting conditions for maintenance during operations.

6.3.3.7 Lag Times

The minimum active components will be capable of delivering full rated flow within a specified time interval after process parameters reach the setpoints for the safety injection signal. Response of the system is automatic, with appropriate allowances for delays in actuation of circuitry and active components. The active portions of the system are actuated by the safety injection signal directly, with the exception of the isolation valves for the hydrogen vent lines on the charging pump suction piping. These valves are electrically interlocked to the volume control tank outlet valves. In analyses of system performance, delays in reaching the programmed trip points and in actuation of components are established on the basis that only emergency onsite power is available. A further discussion of the starting sequence is given in Section 8.3.1.

In the LOCA analysis presented in Sections 15.3 and 15.4 no credit is assumed for partial flow prior to the establishment of full flow and no credit is assumed for the availability of offsite power sources.

For smaller LOCA, there are some additional delays before the process variables reach their respective programmed trip setpoints since this is a function of the severity imposed by the accident. Allowances are made for this in the analyses of the spectrum of reactor coolant pipe breaks.

6.3.3.8 Thermal Shock Considerations

Thermal shock considerations are discussed in Section 5.2.

6.3.3.9 Limits on System Parameters

A comprehensive qualification program has been undertaken to demonstrate that the ECCS components and associated instrumentation and electrical equipment applicable to ECCS will operate for the time period required in the combined post-loss-of-coolant accident conditions of temperature, pressure, humidity, radiation, and chemistry (See Section 3.11).

The specification of individual parameters as given in Table 6.3-1 includes due consideration of allowances for margins over and above the required performance value (e.g., pump flow and net positive suction head), and the most severe conditions to which the component could be subjected (e.g., pressure, temperature, and flow).

6.3.3.10 Use of RHR Spray

No earlier than one hour after initiation of the LOCA, the low head RHR flow may be diverted from the core low head injection path to the RHR spray headers. For minimum safeguards, one high head safety injection pump and one centrifugal charging pump would supply the coolant to the core after realignment of a portion of the RHR pump discharge to the RHR spray headers. The amount of water which would be supplied to the core at a RCS pressure of 15 psig (which is the peak containment design pressure) is approximately 105 lbm/sec. At one hour after a hypothetical LOCA the core has been quenched so that effluent carryover has been terminated.

The time that effluent carryover or entrainment from the core ends is conservatively assumed to occur when the core mixture height reaches the 10 foot elevation (at approximately 150 seconds). At one hour, the thin and thick metal sensible heat has been removed and temperatures reduced to the saturation temperature for the containment pressure. The only heat generation at this time is decay heat.

The decay heat mass boiloff at one hour, which is the minimum time that the RHR low head flow can be diverted to the RHR spray, is 61.5 lbm/sec based on the following assumptions:

1. 102% of engineered safeguards design power rating of 3579 Mwt.

2. ANS infinite decay heat with 20% margin (10 CFR 50.46 Appendix K). Refer to Table 6.3-11.

3. Coolant entering the core is subcooled by 60 Btu/lbm.

Therefore, the coolant entering the RCS piping is roughly twice that required by conservative calculation of the decay heat mass boiloff.

It should be noted that the minimum time given above for diversion of RHR low head flow to the containment spray system is consistent with the containment pressure analysis presented in Section 6.2.1.

6.3.4 Tests and Inspections

In order to demonstrate the readiness and operability of the ECCS, the components are subjected to periodic tests and inspections. Performance tests of the components were performed in the manufacturer's shop prior to delivery. A comprehensive preoperational test program on the ECCS and its components were performed prior to initial fuel loading to provide assurance that the ECCS will accomplish its intended function when required.

6.3.4.1 Preoperational Tests

Preoperational testing of each system and component of the ECCS is to be performed in compliance with the requirements of Regulatory Guide 1.79, "Preoperational Testing of Emergency Core Cooling Systems for Pressurized Water Reactors," with the exception of the following nonconformance items.

Section	Description per Reg. Guide 1.79	Comments
C-1-b-(2)	"The testing should include taking suction from the sump to verify vortex control and acceptable pressure drops across screening and suction lines and valves."	Scale model testing has been performed to verify vortex control and demonstrate insignificant reduction of pump NPSH due to screen and trashrack pressure drops. Calculations have been performed to verify acceptable pressure drops across suction lines and valves. A flowpath from the sump will be verified by water blasting during flushing operations.

The preoperational test of the ECCS and components is discussed in more detail in Chapter 14.

6.3.4.2 Component Testing

Routine periodic testing of the ECCS components and necessary support systems is detailed in Technical Specifications as clarified below. Valves, which operate after a loss-of-coolant accident, are operated through a complete cycle where practical, and pumps are operated individually in this test on their mini-flow lines. If such testing indicates a need for corrective maintenance, the redundancy of equipment in these systems permits such maintenance to be performed without shutting down or reducing load under certain conditions. These conditions include considerations such as the period within which the component should be restored to service and the capability of the remaining equipment to provide the minimum required level of performance during such a period. The inservice component tests of ECCS pumps and valves conform, to the extent practicable allowed by plant design, to the guidelines of the latest edition and addenda of the ASME OM Code incorporated by reference in 10 CFR 50.55a(b) on the date 12 months before the date of issuance of the operating license for Unit 2 for inservice testing of pumps and valves. Performance testing of the Auxiliary Building ECCS pump room coolers is conducted in accordance with TVA's Generic Letter 89- 13 commitments.

6.3.4.3 Periodic System Testing

System testing can be conducted during plant shutdown to demonstrate proper automatic operation of the ECCS. The test program demonstrates the operation of the diesel generator loading sequence, the valves, pumps, and automatic circuitry. The accumulator isolation valve motors are deenergized and the centrifugal charging and safety injection pumps are maintained in recirculation flow for this test so that flow is not introduced into the RCS. The breakers supplying power to the safeguards busses are then tripped manually to simulate a loss of offsite power. Load shedding and diesel generator start signals are verified. A test emergency core-cooling signal is then applied to initiate diesel start and diesel loading. The pump and valve responses are then verified. The ability of the diesel generators to reject 600 kW without tripping is then tested. After the blackout signals are reset, plant configuration is restored in accordance with established procedures.

The test is considered satisfactory if control board indication and visual observation indicate all components have operated and sequenced properly. Periodic ECCS testing is detailed in Technical Specifications. The inservice inspection program described in Sections 5.2.8 and 6.6 provides further confirmation that no significant deterioration is occurring in the emergency core cooling system fluid boundary.

6.3.5 Instrumentation Application

Instrumentation and associated process protection and logic channels employed for initiation of ECCS operation is discussed in Section 7.3. This section describes the instrumentation employed for monitoring emergency core cooling system components during normal plant operation and post-accident operation.

6.3.5.1 Temperature Indication

Residual Heat Exchanger Temperature

The fluid temperature at the inlet and outlet of each residual heat exchanger is recorded in the main control room.

Refueling Water Storage Tank (RWST) Temperature

Two temperature channels are provided to monitor the RWST temperature. Both are indicated in the main control room.

6.3.5.2 Pressure Indication

Safety Injection Header Pressure

Safety injection pump discharge header pressure is indicated in the control room.

Cold Leg Accumulator Pressure

Duplicate pressure channels are installed on each cold leg accumulator. Pressure indication in the control room and a common high and low pressure alarms are provided by each channel. An additional channel for each accumulator provides pressure indication and high pressure alarm in the auxiliary control room.

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 main control room. A common high pressure main control room alarm is actuated by each channel.

6.3.5.3 Flow Indication

Charging Pump Injection Flow

Injection header flow to the reactor cold legs is indicated in the main control room.

Residual Heat Removal Pump Flow

Flow through the RHR injection header and recirculation header is indicated in the main 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 Minimum Flow

A local flowmeter installed in each RHR pump discharge header provides control for the valve located in the pump minimum flow line.

Loss of RHR Flow

An alarm is provided in the main control room to detect low RHR flow. The alarm will detect a miniflow condition coincident with the RHR pump running.

Safety Injection Pump Flow

Injection header flow to the reactor hot and cold legs is indicated in the main control room.

6.3.5.4 Level Indication

Refueling Water Storage Tank Level

Four water level channels which indicate and alarm RWST level in the main control room are provided. The low level setpoint is used in the automatic switchover (sequence described in Table 6.3-3) in a 2/4 logic. Each channel inputs to a common alarm on low and low-low water levels and is indicated on the main control board. Two additional water level channels monitor the upper tank level and provide indication and alarms in the main control room. These high and low alarms are used to ensure adequate RWST inventory and preclude overfilling.

Cold Leg Accumulator Level

Two water level channels are provided for each tank which indicate and alarm the water level in the main control room. The common low and high level alarms ensure adequate accumulator water level.

Containment Sump Water Level

Four containment sump water level indicator channels provide the control room with water level indication and also provide a permissive signal (2 out of 4 logic) to initiate the auto-switchover from the injection to recirculation mode. A common main control room alarm is used to identify a high containment water level condition.

6.3.5.5 Valve Position Indication

The majority of the engineered safety features remote-operated valves have red and green lights on the control board to indicate valve position. The exceptions to this are discussed in Section 7.3.

Accumulator Isolation Valve Position Indication

The accumulator isolation valves are provided with red (open) and green (closed) position indication lights located on the main control room hand switch for each valve. These lights are powered by valve control power and actuated by valve motor operator limit switches.

Refueling Water Storage Tank Isolation Valve

The RWST isolation valve is provided with red (open) and green (closed) position indication lights located on the main control room hand switch. These lights are powered by valve control power and actuated by valve motor operator limit switch.

REFERENCES

1. NEI-04-07, “Pressurized Water Reactor Sump Performance Evaluation Methodology”.

2. Andreychek, Timothy S., et al., "Evaluation of Downstream Sump Debris Effects in Support of GSI 191," WCAP-16406-P, R1 (Proprietary), August 2007.

3. Ann E. Lane. et al., "Evaluation of Post-Accident Chemical Effects in Containment Sump Fluids to Support GSI-191," WCAP-16530-NP, R0 (Non-Proprietary) February 2006.

4. Andreychek, Timothy S., et al.,"Evaluation of Long-Term Cooling Considering Particulate, Fibrous and Chemical Debris in the Recirculating Fluid," WCAP-16793-NP, R2-A (Non-Proprietary) September 2011.

TABLE 6.3-1 (Sheet 1 of 4)

EMERGENCY CORE COOLING SYSTEM
COMPONENT PARAMETERS

Component	Parameters	
Cold Leg Injection Accumulators	Number	4
	Design Pressure, psig	700
	Design Temperature, °F	300
	Operating Temperature, °F	60-150
	Minimum Safety Analysis Limit Pressure, psig	585
	Nominal Total Volume, ft^3	1356 each
	Nominal Water Volume, ft^3	1050 each
	Nominal Volume N_2 Gas, ft^3	306
	Boric Acid Concentration	
	nominal, ppm	3150
	minimum, ppm	3000
	maximum, ppm	3300
	Hi level Alarm, ft^3	1165
	Relief Valve Setpoint, psig	700
Centrifugal Charging Pumps	Number	2
	Design Pressure, psig	2800
	Design Temperature, °F	300
	**Design Flow Rate, gpm (original design point)	150
	Design Head, ft (original design point)	5800
	Max. Flow Rate, gpm (inj. mode/recirc. mode)	550/560
	Head Required at Max. Flow Rate, ft (injection mode)	1342
	NPSH Required at Max. Flow Rate, ft (injection mode)	25
	Motor Rating, hp (original/upgraded)	600/720****
	Maximum Starting Time, sec	5

TABLE 6.3-1 (Sheet 2 of 4)

EMERGENCY CORE COOLING SYSTEM
COMPONENT PARAMETERS

Component	Parameters	
Safety Injection Pumps	Number	2
	Design Pressure, psig	1750
	Design Temperature, °F	300
	Design Flow Rate, gpm (original design point)	400
	Design Head, ft (original design point)	2700
	Max. Flow Rate, gpm (inj. mode/recirc. mode)	650/675
	Head Required at Max. Flow Rate, ft (injection mode)	1808
	NPSH Required at Max. Flow Rate, ft (injection mode)	30 (Train A) 28 (Train B)
	***Motor Rating, hp	400
	Maximum Starting Time, sec	5
Residual Heat Removal Pumps	Refer to Section 5.5.7 for parameter information	
Residual Heat Exchangers	Refer to Section 5.5.7 for parameter information	
Boron Injection Tank	Number	1
	Total Volume, gal	900
	Useable Volume at Operating Conditions, gal	900
	Design Pressure, psig	2735
	Design Temperature, °F	300
Refueling Water Storage Tanks	Number	1
	Total Volume, gal	400,000
	Volume at Overflow, gal	380,000
	Minimum Volume, gal	370,000
	Normal Pressure, psig	Atmospheric
	Design Pressure, psig	Atmospheric
	Design Temperature, °F	200
	Boron Concentrations (as boric acid), ppm	3200 nominal 3100 minimum 3300 maximum

TABLE 6.3-1 (Sheet 3 of 4)

EMERGENCY CORE COOLING SYSTEM
COMPONENT PARAMETERS

Component	Parameters	
Valves*		
Valve Number	Valve Description	Maximum Stroke Time
FCV-63-25, -26	BIT Outlet	20 sec
FCV-63-152, -153	SIPs CL Injection Crosstie	15 sec
LCV-62-132, -133	VCT Outlet Isolation to CCPs	10 sec
LCV-62-135, -136	RWST to CCPs Suction	15 sec
FCV-62-90, -91	Charging Line Isolation	10 sec
FCV-63-5	RWST to SIPs Suction	14 sec
FCV-63-47, -48	SIP Suction	15 sec
FCV-63-3, -4, -175	SIP Miniflow Isol	10 sec
FCV-63-156, -157	SIP HL Injection Isol	17 sec
FCV-63-1	RWST to RHRPs Suction	60 sec
FCV-74-3	RWST to RHRPs Suction	60 sec
FCV-74-21	RWST to RHRPs Suction	60 sec
FCV-74-33, -35	RHR Discharge Header Crosstie	15 sec
FCV-63-8, -11	RHR HXs to SIPs and CCPs Suctions	28 sec
FCV-63-93, -94	RHRP CL Injection	40 sec
FCV-63-172	RHRP HL Injection	120 sec
FCV-63-72, -73	Cntmt Sump to RHRPs	60 sec
FCV-63-23, -71, -84, -64	SIS Test Valves and N_2 Supply Valves	10 sec
FCV-63-6, -7	SIPs to CCPs Suctions	10 sec
FCV-63-185	Leak Test Line Isolation	10 sec

TABLE 6.3-1 (Sheet 4 of 4)

EMERGENCY CORE COOLING SYSTEM
COMPONENT PARAMETERS

Valve Design-Basis Leakage

Leakage by Valve Type	Leakage Allowed Unit 1	Leakage Allowed Unit 2
a. Conventional Globe Valves		
disc leakage per inch of nominal valve size	3 cc/hr	0-10 cc/hr
back seat leakage per inch of stem diameter	1 cc/hr	1 cc/hr
b. Gate Valves		
disc leakage per inch of nominal valve size	3 cc/hr	0-10 cc/hr
back seat leakage per inch of stem diameter	1 cc/hr	1 cc/hr
c. Check Valves		
disc leakage per inch of nominal valve size	3 cc/hr	0-10 cc/hr
d. Diaphragm Valves		
disk leakage	0 cc/hr	0 cc/hr
e. Pressure Relief Valves		
disc leakage, maximum	10 cc/hr	10 cc/hr
f. Accumulator Check Valves		
disc leakage per inch of nominal valve size	3 cc/hr	3 cc/hr

* FCV-63-22, FCV-62-98, and FCV-62-99 are not listed since they are considered passive valves.

** Includes miniflow.

*** Service factor of 1.15 not included.

**** Actual bhp requirements are based on installed pump rotating element consistent with analysis.

***** Specific allowable leak rates are defined on valve data sheets. (Unit 2 Only)

TABLE 6.3-2 (Sheet 1 of 2)

EMERGENCY CORE COOLING SYSTEM COMPONENTS

Component	Material
Cold Leg Accumulators	Carbon Steel, Clad with Austenitic Stainless Steel
Boron Injection Tank	Austenitic Stainless Steel
Pumps	
Safety Injection	Austenitic Stainless Steel
Centrifugal Charging	Austenitic Stainless Steel
Residual Heat Removal	Austenitic Stainless Steel
Residual Heat Exchangers	
Shell	Carbon Steel
Shell End Cap	Carbon Steel
Tubes	Austenitic Stainless Steel
Channel	Austenitic Stainless Steel
Channel Cover	Austenitic Stainless Steel
Tube Sheet	Austenitic Stainless Steel
Valves	
Motor-Operated Valves Containing Radioactive Fluids:	
Pressure Containing Parts	Austenitic Stainless Steel or Equivalent
Body-to-Bonnet Bolting and Nuts	High Alloy Steel
Seating Surfaces	Stellite No. 6 or Equivalent
Stems	Austenitic Stainless Steel or 17-4 PH Stainless

TABLE 6.3-2 (Sheet 2 of 2)

MATERIALS EMPLOYED FOR
EMERGENCY CORE COOLING SYSTEM COMPONENTS

Component	Material
Diaphragm Valves	Austenitic Stainless Steel
Accumulator Check Valves	
Parts Contacting Borated Water	Austenitic Stainless Steel
Clapper Arm Shaft Pin	Nickel Alloy
Relief Valves	
Bodies	Stainless Steel
All Nozzles, Discs, and Guides Spindles	Austenitic Stainless Steel, Nickel Alloy, and stellite high alloy steel
Bonnets for Stainless Steel Valves without a Balancing Bellows	Stainless Steel
All Other Bonnets	Carbon Steel
Piping	
All Piping in Contact with Borated Water	Austenitic Stainless Steel
Refueling Water Storage Tank	Austenitic Stainless Steel

TABLE 6.3-3 (Sheet 1 of 3)
UNIT 1

SEQUENCE OF CHANGE-OVER OPERATION, INJECTION TO RECIRCULATION

The following automatic phase of switchover from the injection to the recirculation mode is initiated when the RWST is at low level and the containment sump level has risen to its level switch actuation point. (Westinghouse flow diagram valve numbers are shown in brackets.) The component cooling water isolation valve to each RHR heat exchanger (FCV 70-153, -156) is opened manually prior to or during this switchover.

1. The valves that admit suction from the containment sump to the RHR pumps (FCV 63-72 & 73) open while the RHR pumps continue to run. [8811A and B]

2. The valves that were open and permitting suction for the RHR pumps to be taken from the RWST (FCV 74-3 & -21) start to close when the valves in Step 1 start to open. [8700A and B]

The manual operations below are accomplished following the automatic switchover phase.

1. Verify completion of the automatic valve realignments above. If an RHR pump has failed to switchover to the sump, stop that pump.

2. Verify SIP flow to the RCS (e.g., large break case) and close the three safety injection pump miniflow valves (FCV 63-3, -4, -175). [8811, 8920, 8813]

3. Close the two valves in the crossover line downstream of the RHR heat exchangers (FCV 74-33, -35). [8716A and B]

4. Open the two parallel valves in the common suction line between the charging pump suction and the safety injection pump suction (FCV 63-6, -7). [8807A and B]. Ensure FCV-63-177 is open [8924].

5. Open the valve in the line from the train A RHR pump discharge to the charging pump suction (FCV 63-8) and the valve in the line from the train B RHR pump discharge to the safety injection pump suction (FCV 63-11). [8804A and B]

6. Reset the SIS actuation signal and close the two parallel valves in the line from the RWST to the charging pump suction (FCV 62-135, -136). [LCV-112D and E] Place corresponding valve handswitches in A-Auto.

7. Close the valve in the line from the RWST to the safety injection pump suction (FCV 63-5). [8806]

TABLE 6.3-3 (Sheet 2 of 3)
UNIT 1

SEQUENCE OF CHANGE-OVER OPERATION, INJECTION TO RECIRCULATION

8. Restore power and close the valve in the common line from the RWST to both RHR pumps (FCV 63-1). [8812]

Upon reaching the RWST low-low level setpoint, as indicated on the qualified PAM indicator channels, the operator shall realign the containment spray system. The following steps are required for the realignment of the containment spray system from the injection to the recirculation mode. First, reset containment spray actuation signal.

1. Stop both containment spray pumps ("pull to lock in stop" to preclude the possibility of pump restart while realigning suction valves).

2. Close the spray pump/RWST isolation valve at the suction of each containment spray pump (FCV 72-22 and -21). [9017A and B]

3. Open the essential raw cooling water isolation valves to each containment spray heat exchanger (FCV 67-125, -126, -123, -124).

4. Open the sump isolation valve at the suction of each containment spray pump (FCV 72-44 and -45) after the valves in Step 2 have completed their travel. [9020A and B]

5. Verify that the valve realignments in Steps 2 through 4 have been completed.

6. Restart both containment spray pumps, if containment pressure is greater than or equal to 2.0 psig.

If the ECCS and the containment spray system are both operating in the recirculation mode and at least one hour has elapsed since the initiation of the LOCA, a portion of the RHR flow may be diverted to the RHR spray headers for additional containment cooling. (Only one RHR train is used for RHR spray.)

To align No. 1 RHR pump to spray (complete Steps 1 and 2).

1. Close the valve between the No. 1 RHR pump discharge and two RCS cold legs (FCV 63-93).

2. Open the valve between the discharge of the No. 1 RHR pump and the No. 1 RHR spray header (FCV-72-40).

   Note: The valve between the discharge of the No. 1 RHR pump and the No. 1 RHR spray header is interlocked such that it cannot be opened unless the containment sump valve to the No. 1 RHR pump is open.

To align No. 2 RHR pump to spray (complete Steps 3 and 4).

3. Close the valve between the No. 2 RHR pump discharge and two RCS cold legs (FCV-63-94).

TABLE 6.3-3 (Sheet 3 of 3)
UNIT 1

SEQUENCE OF CHANGE-OVER OPERATION, INJECTION TO RECIRCULATION

4. Open the valve between the discharge of the No. 2 RHR pump and the No. 2 RHR spray header (FCV-72-41).

   Note: The valve between the discharge of the No. 2 RHR pump and the No. 2 RHR spray header is interlocked such that it cannot be opened unless the containment sump valve to the No. 2 RHR pump is open.

The following switching operation should be used approximately 3 hours after event initiation when realigning the ECCS from the cold leg recirculation mode to the hot leg recirculation mode. In the event a train cannot be aligned in the hot leg recirculation mode, it will be realigned to the cold leg recirculation.

If RHR hot leg injection is desired:

1. Close train A or B cold leg injection valve (FCV 63-93 or -94). [8809A, 8809B]

2. Open the RHR crossover valve (FCV 74-33 or -35). [8716B, 8716A]

3. Open the RHR hot leg injection valve (FCV 63-172). [8840]

4. Close the other train RHR pump cold leg injection valve (FCV-63-94 or -93). [8809B, 8809A]

5. Stop the safety injection pumps.

6. Close the safety injection pumps discharge crossover valves (FCV 63-152 and -153). [8821A and B]

7. Open the safety injection pump hot leg injection valves (FCV 63-156 and -157). [8802A and B]

8. Start the safety injection pumps.

9. If available, close the safety injection pump cold leg injection valve (FCV 63-22). [8835]

TABLE 6.3-3 (Sheet 1 of 4)
UNIT 2

SEQUENCE OF CHANGE-OVER OPERATION, INJECTION TO RECIRCULATION

The following steps are performed while the ECCS is in the injection mode in preparation for the recirculation mode:

1. Prepare ERCW for recirculation mode by:
   a. Position the “CCS HX A OUTLET ERCW FLOW CNTL” valve (1-FCV-67-146) to Position A or B (As appropriate)

   b. Close the ‘CCS HX A OUTLET ERCW FLOW CNTRL BYP” valve (1-FCV-67-143)

   c. Position the “CCS HX B OUTLET ERCW FLOW CNTL” valve (2-FCV-67-146) to Position A or B (as appropriate)

   d. Close the “CCS HX B OUTLET ERCW FLOW CNTL BYP” valve (2*FCV-67-143)

   e. Position the ‘CCS HX C OUTLET ERCW FLOW CNT” valve (0-FCV-67-153) to Position A or B (as appropriate)

   f. Close the “CCS HX C OUTLET ERCW FLOW CNTL BYP” valve (0-FCV-67-144)

   g. Open the “CNTMT SPRAY NX 1(2)A INLET” (1(2)-FCV-64-125) and “CNTMT SPRAY HX 1(2)A RETURN: (1(2)-FCV-67-126) valves

   h. Open the “CNTMT SPRAY HX 1(2)B INLET” (1(2)-FCV-67-123) and “CNTMT SPRAY HX 1(2)B RETURN” (1(2)-FCV-67-124) valves

   i. Bypass ERCW Pump Lockout for ERCW pumps powered by diesel generators associated with the non-accident unit.

   j. Ensure two ERCW pumps are running on each train.

   k. If available, start a third ERCW pump on each train.

2. Prepare CCS for recirculation mode by:

   a. Start all available CCS pumps.

   b. Ensure RHR HX 1(2)A OUTLET 1(2)-FCV-70-156 valve(s) are open.

   c. Ensure RHR HX 1(2)B OUTLET 1(2)-FCV-70-153 valve is throttled to 5,000 gpm on Accident Unit.

   d1. If the non-accident is using RHR and with only one (1) CCS pump supplying CCS HX C, while maintaining at least 5,000 gpm CCS flow to the RHR HX on the Accident Unit, ensure that Non-Accident Unit 2 RHR HX 1(2)B OUTLET 1(2)-FCV-70-153 valve is throttled to approximately 2,725 gpm (or whatever is achievable).

   d2. If the non-accident is using RHR and with two (2) CCP pumps sullying CCS HX C, while

TABLE 6.3-3 (Sheet 2 of 4)
UNIT 2

SEQUENCE OF CHANGE-OVER OPERATION, INJECTION TO RECIRCULATION

maintaining at least 5,000 Gpm CCS Flow to the RHR HX on eh Accident Unit, ensure the NOn-Accident Unit RHR HX 1(2)B OUTLET 1(2)-FCV-70-153 valve is throttled to approximately 5,000 gpm.

e. For a LOCA on Unit 1 Only, if both CCS pumps 1A-A and 1B-B (aligned to CCS Header 1A) are not running, close SFP HEAT EXHANGER A CCP SUPPLY(0-FCV-70-197).

The following automatic phase of switchover from the injection to the recirculation mode is initiated when the RWST is at low level and the containment sump level has risen to its level switch actuation point. (Westinghouse flow diagram valve numbers are shown in brackets.) The component cooling water isolation valve to each RHR heat exchanger (FCV 70-153, -156) is opened during this switchover or immediately thereafter.

1. The valves that admit suction from the containment sump to the RHR pumps (FCV 63-72 & 73) open while the RHR pumps continue to run. [8811A and B]

2. The valves that were open and permitting suction for the RHR pumps to be taken from the RWST (FCV 74-3 & -21) start to close when the valves in Step 1 start to open. [8700A and B]

The manual operations below are accomplished following the automatic switchover phase.

1. Verify completion of the automatic valve realignments above. If an RHR pump has failed to switchover to the sump, stop that pump.

2. Verify SIP flow to the RCS (e.g., large break case) and close the three safety injection pump miniflow valves (FCV 63-3, -4, -175). [8811, 8920, 8813]

3. Close the two valves in the crossover line downstream of the RHR heat exchangers (FCV 74-33,-35). [8716A and B]

4. Open the two parallel valves in the common suction line between the charging pump suction and the safety injection pump suction (FCV 63-6, -7). [8807A and B] Ensure FCV-63-177 is open [8924].

5. Open the valve in the line from the train A RHR pump discharge to the charging pump suction (FCV-63-8) and the valve in the line from the train B RHR pump discharge to the safety injection pump suction (FCV 63-11). [8804A and B]

6. Reset the SIS actuation signal and close the two parallel valves in the line from the RWST to the charging pump suction (FCV 62-135, -136). [LCV-112D and E] Place corresponding valve handswitches in A-Auto.

7. Close the valve in the line from the RWST to the safety injection pump suction (FCV 63-5). [8806]

8. Restore power and close the valve in the common line from the RWST to both RHR pumps (FCV-63-1). [8812]

TABLE 6.3-3 (Sheet 3 of 4)
UNIT 2

SEQUENCE OF CHANGE-OVER OPERATION, INJECTION TO RECIRCULATION

Upon reaching the RWST low-low level setpoint, as indicated on the qualified PAM indicator channels, the operator shall realign the containment spray system. The following steps are required for the realignment of the containment spray system from the injection to the recirculation mode.

First, reset containment spray actuation signal.

1. Stop both containment spray pumps ("pull to lock in stop" to preclude the possibility of pump restart while realigning suction valves).

2. Close the spray pump/RWST isolation valve at the suction of each containment spray pump (FCV-72-22 and -21). [9017A and B]

3. Open the sump isolation valve at the suction of each containment spray pump (FCV-72-44 and -45) after the valves in Step 2 have completed their travel. [9020A and B]

4. Verify that the valve realignments in Steps 2 through 4 have been completed.

5. Restart both containment spray pumps, if containment pressure is greater than or equal to 2.0psig.

If the ECCS and the containment spray system are both operating in the recirculation mode and at least one hour has elapsed since the initiation of the LOCA, a portion of the RHR flow may be diverted to the RHR spray headers for additional containment cooling. (Only one RHR train is used for RHR spray.)

To align No. 1 RHR pump to spray (complete Steps 1 and 2).

1. Close the valve between the No. 1 RHR pump discharge and two RCS cold legs (FCV-63-93).

2. Open the valve between the discharge of the No. 1 RHR pump and the No. 1 RHR spray header
(FCV-72-40).

Note: The valve between the discharge of the No. 1 RHR pump and the No. 1 RHR spray header is
interlocked such that it cannot be opened unless the containment sump valve to the No. 1 RHR pump is open.

To align No. 2 RHR pump to spray (complete Steps 3 and 4).

3. Close the valve between the No. 2 RHR pump discharge and two RCS cold legs (FCV-63-94).

4. Open the valve between the discharge of the No. 2 RHR pump and the No. 2 RHR spray header
(FCV-72-41).

TABLE 6.3-3 (Sheet 4 of 4)
UNIT 2

SEQUENCE OF CHANGE-OVER OPERATION, INJECTION TO RECIRCULATION

Note: The valve between the discharge of the No. 2 RHR pump and the No. 2 RHR spray header is interlocked such that it cannot be opened unless the containment sump valve to the No. 2 RHR pump is open.

The following switching operation should be used approximately 3 hours after event initiation when realigning the ECCS from the cold leg recirculation mode to the hot leg recirculation mode. In the event a train cannot be aligned in the hot leg recirculation mode, it will be realigned to the cold leg recirculation.

If RHR hot leg injection is desired:

1. Close train A or B cold leg injection valve (FCV-63-93 or -94). [8809A, 8809B]

2. Open the RHR crossover valve (FCV-74-33 or -35). [8716B, 8716A]

3. Open the RHR hot leg injection valve (FCV-63-172). [8840]

4. Close the other train RHR pump cold leg injection valve (FCV-63-94 or -93). [8809B, 8809A]

5. Stop the safety injection pumps.

6. Close the safety injection pumps discharge crossover valves (FCV-63-152 and -153). [8821A and B]

7. Open the safety injection pump hot leg injection valves (FCV-63-156 and -157). [8802A and B]

8. Start the safety injection pumps.

9. If available, close the safety injection pump cold leg injection valve (FCV-63-22). [8835]

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TABLE 6.3-3a

EVALUATION OF TIME SEQUENCE ASSOCIATED WITH CHANGEOVER
OPERATION FROM INJECTION TO RECIRCULATION

1) Minimum time to switchover initiation	10 minutes (measured from time of ESFAS actuation due to LOCA)
2) Automatic switchover completed	1 minute*
3) Stop RHR pump if it fails to switch over	1.5 minutes*
4) Complete manual switchover to the point where FCV-63-8/-11 are open	5.5 minutes*
5) Stop CS pump after receipt of low-low level in RWST	10 seconds

* Time measured from initiation of auto switchover.

TABLE 6.3-4

NORMAL OPERATING STATUS OF EMERGENCY CORE COOLING

SYSTEM COMPONENTS FOR CORE COOLING

Number of Safety Injection Pumps Operable	2
Number of Charging Pumps Operable	2
Number of Residual Heat Removal Pumps Operable	2
Number of Residual Heat Exchangers Operable	2
Refueling Water Storage Tank Volume, gal (minimum)	370,000
Boron Concentration in Refueling Water Storage Tanks, minimum ppm	3,100
Boron Concentration in Cold Leg Accumulator, minimum ppm	3,000
Number of Accumulators	4
Minimum Cold Leg Accumulator Pressure, psig (Safety Analysis)	585
Maximum Cold Leg Accumulator Pressure, psig (Safety Analysis)	690
Nominal Cold Leg Accumulator Water Volume, ft^3	1,050
System Valves, Interlocks, and Piping Required for the Above Components which are Operable	All

WBN

TABLE 6.3-5

EMERGENCY CORE COOLING SYSTEM SHARED FUNCTIONS EVALUATION

Component	Normal Operating Arrangement	Accident
Refueling Water Storage Tank	Lined up to suction of safety injection, containment spray, and residual heat removal pumps.	Lined up to suction of centrifugal charging, safety injection, residual heat removal pumps, and containment spray pumps. Valves for realignment meet single failure criteria.
Centrifugal Charging Pumps	Lined up for charging service Suction from volume control tanks	Lined up to inlet of boron injection tank and outlet of RWST. Valves for realignment meet single failure criteria.
Residual Heat Removal Pumps	Lined up to cold legs of reactor coolant piping.	Lined up to cold legs of reactor coolant piping.
Residual Heat Exchangers	Lined up for residual heat removal pump operation.	Lined up for residual heat removal pump operations.
Safety Injection Pumps	Lined up to cold legs of reactor coolant piping.	Lined up to cold legs of reactor coolant piping.
Accumulators	Lined up to cold legs of reactor coolant piping	Lined up to cold legs of reactor coolant piping.

TABLE 6.3-6

MAXIMUM RECIRCULATION LOOP LEAKAGE EXTERNAL TO CONTAINMENT

Item	Type of Leakage Control and Unit Leakage Rate Used in the Analysis	Leakage to Atmosphere cc/hr	Leakage to Drain Tank cc/hr
1. Residual Heat Removal Pumps (Low Head Safety Injection)	Mechanical seal	*	0
2. Safety Injection Pumps	Same as residual heat removal pump	*	0
3. Charging Pumps	Same as residual heat removal pump	*	0
4. Flanges:			
a. Pumps	Gasket - adjusted to zero leakage following any test	0	0
b. Valves (larger than 2 inches)	10 drops/min/gauge used	2,400	0
c. Control Valves	(30cc/hr). Due to leak tight flanges on pumps, no leakage	480	0
d. Heat Exchangers	is assumed to atmosphere.	240	0
5. Valves - Stem Leakoffs	Back seated double packing with leakoff - 1 cc/hr/in stem diameter used (see Table 6.3-1).	0	50
6. Miscellaneous Small Valves	Flanged Boyd packed stems - 1 drop/min used (3cc/hr).	600	0
7. Miscellaneous Large Valves (Larger than 2 inches)	Double packing 1cc/hr/in stem diameter used.	40	0

* Infrequent minor ECCS pump seal leakage that may occur during normal operation is bounded by the existing offsite dose analysis. The total ECCS recirculation loop leakage evaluated in the offsite dose analysis is 3760 cc/hr, as shown in this table. The total realistic ECCS recirculation loop leakage from all flanged connections and valves is significantly less than the conservative value used in the offsite dose analysis.

TABLE 6.3-7

DELETED

Table 6.3-8 (Sheet 1 of 58)
UNIT 1

FAILURE MODES AND EFFECTS ANALYSIS FOR ACTIVE FAILURES FOR THE SAFETY INJECTION SYSTEM

Item No.	Component	Function	Failure Mode	Potential Cause	Method of Detection	Effect on System	Effect on Plant	Remarks
1	LCV-62-135 Train A	Opens to connect RWST to CCPs' suction (parallel to LCV-62-136) (injection mode)	Fails to open, stuck closed or spurious closing after opening	Mechanical failure; Train A power failure; Train A SI Signal failure; operator error (HS in wrong position)	Indicating light in MCR; HS position	No redundancy in RWST suction path to CCPs	None. Train B RWST suction valve LCV-62-136 allows suction flow to both CCPs	Normally closed valve opens automatically on SI Signal to align CCP suction to RWST and is manually closed along with parallel valve LCV-62-136 for sump recirculation mode after CCP suction is transferred to RHR pump discharge. Automatic operation of 62-135 on SI Signal, is completely independent of valve 62-136. (This valve has a VCT LO-LO level automatic function, which is not within the scope of this SIS FMEA.)
		Closes to isolate RWST (recirc. mode)	Fails to close or stuck open	Mechanical failure; Train A power failure; operator error	Ind. light in MCR; HS position	RWST remains connected to CCPs' suction after switchover to recirculation, however, pump discharge head from RHR is greater than head from RWST and backflow to RWST is prevented by check valve 62-504.	None	
			Spuriously opens	Operator error (HS in wrong position)	Indicating light			
2	LCV-62-136 Train B	Opens to connect RWST to	Fails to open, stuck closed or spurious	Mechanical failure; Train B power failure;	Indicating light in MCR; HS position	No redundancy in RWST suction path to CCPs	None. Train B suction valve LCV-62-	Normally closed valve opens automatically on SI Signal to align CCP

FAILURE MODES AND EFFECTS ANALYSIS FOR ACTIVE FAILURES FOR THE SAFETY INJECTION SYSTEM

Item No.	Component	Function	Failure Mode	Potential Cause	Method of Detection	Effect on System	Effect on Plant	Remarks
		CCPs' suction (parallel to LCV-62-135) (injection mode)	closing after opening	Train B SI Signal failure			136 allows RWST suction flow to both CCPs	suction to RWST and is manually closed along with parallel valve LCV-62-135 for sump recirculation mode after CCP suction is transferred to RHR pump discharge. Automatic operation of 62-136, on SI Signal, is completely independent of valve 62-135. (This valve has a VCT LO-LO level automatic function, which is not within the scope of the SIS FMEA.)
	LCV-62-136 Train B	Closes to isolate RWST (Recirc. mode)	Fails to close or stuck open	Mechanical failure; Train B power failure; operator error	Indicating light in MCR; HS position	RWST remains connected to CCPs' suction after switchover to recirculation, however, head from RHR is greater than head from RWST and backflow to RWST is	None	
			Spuriously opens	Operator error (HS in wrong position)	Indicating light			

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FAILURE MODES AND EFFECTS ANALYSIS FOR ACTIVE FAILURES FOR THE SAFETY INJECTION SYSTEM

Item No.	Component	Function	Failure Mode	Potential Cause	Method of Detection	Effect on System	Effect on Plant	Remarks
						prevented by check valve 62-504.		
3	Check Valve 62-504	Provides suction flow path from RWST to CCPs (injection mode)	Stuck closed (See 'Remarks' Column)	Mechanical failure	Erratic readings on motor ammeter and pump discharge flow (FI-63-170) in control room. Suction pressure indication on PI-62-105 and 109 (local); Discharge pressure indication on PI-62-106 and 110 (local)	Pumps start on SI signal but suction flow not established. Pump damage unless operator secures pumps based on motor amp and flow readings	See 'Remarks' Column	Per IEEE Std. 500-1984, check valves at PWRs have a failure rate (fail to open) of 60 per million demands, which makes this a credible failure mode. Relative to the failure mode, the Design Criteria Document does not require consideration of this check valve as an active component for the opening function (and therefore its failure to open) during the injection mode. However, during the recirculation mode, it is an active component and its failure to close is analyzed below. The effect on the plant, if 'stuck closed' failure mode were to be

FAILURE MODES AND EFFECTS ANALYSIS FOR ACTIVE FAILURES FOR THE SAFETY INJECTION SYSTEM

Item No.	Component	Function	Failure Mode	Potential Cause	Method of Detection	Effect on System	Effect on Plant	Remarks
								considered, is no flow from charging pumps and no injection into cold legs until RCS pressure drops below discharge pressure of SI pumps.
	Check Valve 62-504	Prevents backflow from CCP suction header to RWST (Recirc. mode)	Stuck open or fails to backseat	Mechanical failure	Change in RWST level and temperature; Radiation Protection surveys of RWST area; loss of containment sump inventory	None. MOVs 62-135 and 62-136 are closed to complete switchover to recirc. mode	None.	Per IEEE Std. 500-1984, failure mode is internal leakage rather than gross failure to close
4	LCV-62-132 Train A	Isolates CCPs' normal suction source (volume control tank) when RWST suction is aligned (in series with	Fails to close, stuck open or spurious opening after closing	Mechanical failure, Train A power failure, Train A SI Signal failure; operator error	Indicating light in MCR	Loss of redundancy in VCT isolation	None. Train B isolation valve LCV-62-133 provides isolation of VCT suction path	Both LCV-62-132 and LCV-62-133 are interlocked with LCVs 62-135 and 62-136 in such a way that even on a SI Signal, neither 62-132 nor 62-133 will begin to close unless either 62-135 or 62-136 is fully open.

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FAILURE MODES AND EFFECTS ANALYSIS FOR ACTIVE FAILURES FOR THE SAFETY INJECTION SYSTEM

Item No.	Component	Function	Failure Mode	Potential Cause	Method of Detection	Effect on System	Effect on Plant	Remarks
		LCV-62-133)						The schematics were reviewed and it was determined that the use of non-divisional power and stem-mounted limit switches for the cross-division interlock did not introduce a common failure mode of LCVs 62-132 and 62-133. Failure to open not listed since it has no impact on safety function of System 63.
5	LCV-62-133 Train B	Isolates CCPs' normal suction source (VCT) when RWST is aligned (in series with LCV-62-132)	Fails to close, stuck open or spurious opening after closing	Mechanical failure; Train B power failure; Train B SI signal failure; operator error	Indicating light in MCR	Loss of redundancy in VCT isolation	None. Train A isolation valve LCV-62-132 provides isolation of VCT suction source	Both LCV-62-132 and LCV-62-133 are interlocked with LCVs 62-135 and 62-136 in such a way that even on a SI Signal, neither 62-132 nor 62-133 will begin to close unless either 62-135 or 62-136 is fully open. The schematics were reviewed and it was determined that the use

FAILURE MODES AND EFFECTS ANALYSIS FOR ACTIVE FAILURES FOR THE SAFETY INJECTION SYSTEM

Item No.	Component	Function	Failure Mode	Potential Cause	Method of Detection	Effect on System	Effect on Plant	Remarks
								of non-divisional power and stem-mounted limit switches for the cross-division interlock did not introduce a common failure mode of LCVs 62-132 and 62-133. Failure to open not listed since it has no impact on safety function of System 63.
6	Centrifugal Charging Pump A-A	Provides RCP seal injection and emergency core cooling by pumping into RCS cold legs, borated water from RWST during injection mode and contents of containment sump via RHR pumps during recirc. mode.	Fails to start; fails while running	Mechanical failure; Train A power failure; Train A SI Signal failure; motor overload; electrical fault Operator error (HS in wrong position)	Motor trip or overload alarm, indicating lights in main control room, no motor amps, no pump discharge pressure on PI-62-110 (local)	Loss of redundancy in high head injection portion of SI System.	None. CC Pump B-B can provide required high head injection flow for design basis range of break sizes	Pump starts automatically on Train A SI Signal. Automatic operation of each CCP in ECCS injection mode is completely independent of other CCP.

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FAILURE MODES AND EFFECTS ANALYSIS FOR ACTIVE FAILURES FOR THE SAFETY INJECTION SYSTEM

Item No.	Component	Function	Failure Mode	Potential Cause	Method of Detection	Effect on System	Effect on Plant	Remarks
7	Centrifugal Charging Pump B-B	Provides RCP seal injection and emergency core cooling by pumping into RCS cold legs, borated water from RWST during injection mode and contents of containment sump via RHR pumps during recirc. mode.	Fails to start; fails while running	Mechanical failure; Train B power failure, Train B SI Signal failure; motor overload; electrical fault Operator error (HS in wrong position)	Motor trip or overload alarm, indicating lights in main control room, no motor amps, no pump discharge pressure on PI-62-106 (local)	Loss of redundancy in high head injection portion of SI System	None. CC Pump A-A can provide required high head injection flow for design basis range of break sizes.	Pump starts automatically on Train B SI Signal. Automatic operation of each CCP in ECCS injection mode is completely independent of other CCP.
8	Check Valve 62-525	Provides discharge flow path for CCP A-A (injection and recirc modes)	Stuck closed	Mechanical failure	Pump motor amps less than full load; CCP total flow low as indicated on FI-63-170 in MCR	Loss of redundancy in high head injection portion of SI System	None. CCP B-B can provide required high pressure injection flow.	
		Prevents reverse flow	Stuck open or fails to	Mechanical failure	Pump motor amps above	See 'Remarks' column	None. See 'Remarks'	Since the failure of the check valve is the

FAILURE MODES AND EFFECTS ANALYSIS FOR ACTIVE FAILURES FOR THE SAFETY INJECTION SYSTEM

Item No.	Component	Function	Failure Mode	Potential Cause	Method of Detection	Effect on System	Effect on Plant	Remarks
		of CCP B-B discharge through CCP A-A	backseat		full load, low flow on FI-63-170 in MCR		column	single failure postulated, both CC Pumps A-A & B-B can be assumed to operate.
9	Check Valve 62-532	Provides discharge flow path for CCP B-B (injection and recirc. modes)	Stuck closed	Mechanical failure	Pump motor amps less than full load; CCP total flow low as indicated on FI-63-170 in MCR	Loss of redundancy in high head injection portion of SI System	None. CCP A-A can provide required high pressure injection flow.	
		Prevents reverse flow of CCP A-A discharge through CCP B-B	Stuck open or fails to backseat	Mechanical failure	Pump motor amps above full load, low flow on FI-63-170 in MCR	See 'Remarks' column	None. See 'Remarks' column	Since the failure of the check valve is the single failure postulated, both CC Pumps A-A & B-B can be assumed to operate
10	Check Valve 62-523	Opens to connect CCP A-A discharge to min. flow recirc.	Stuck closed	Mechanical failure	Possible abnormal readings on pump motor ammeter; CCP A-A damage.	Min flow circuit for CCP A-A protection unavailable, with damage to CCP A-A possible.	None. CCP B-B still available	Failure mode is credible; Per IEEE Std. 500-1984, check valves at PWRs have a failure rate (fail to open) of 60 per million demands
		Prevents reverse flow of CCP B-B	Stuck open or fails to	Mechanical failure	Pump motor amps above full load;	See 'Remarks' column	See 'Remarks'	Since the failure of the check valve is the single failure

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FAILURE MODES AND EFFECTS ANALYSIS FOR ACTIVE FAILURES FOR THE SAFETY INJECTION SYSTEM

Item No.	Component	Function	Failure Mode	Potential Cause	Method of Detection	Effect on System	Effect on Plant	Remarks
		discharge through CCP A-A	backseat		discharge press. high on idle pump		column	postulated, both CC pumps A-A & B-B can be assumed to operate.
11	Check Valve 62-530	Opens to connect CCP B-B discharge min. flow recirc.	Stuck closed	Mechanical failure	Possible abnormal readings on pump motor ammeter; CCP B-B damage	Min. flow circuit for CCP B-B protection unavailable, with damage to CCP B-B possible	None. CCP A-A still available	Failure mode is credible; Per IEEE Std. 500-1984, check valves at PWRs have a failure rate (fail to open) of 60 per million demands
		Prevents reverse flow of CCP A-A discharge through CCP B-B	Stuck open or fails to backseat	Mechanical failure	Pump motor amps above full load; Discharge press. high on idle pump	See 'Remarks' column	See 'Remarks' column	Since the failure of the check valve is the single failure postulated, both CC pumps A-A & B-B can be assumed to operate.
12	FCV-62-90 Train A	Isolates CCPs' normal charging path from CCP discharge header; in series with FCV-62-91	Fails to close or stuck open	Mechanical failure, Train A power failure, Train A SI Signal failure	Indicating light in MCR	Loss of redundancy in normal charging path isolation	None. Train B isolation valve FCV-62-91 provides isolation of normal charging path.	Failure to open will prevent realignment of charging after SI termination and is not listed since it has no impact on safety function of System 63.
13	FCV-62-91 Train B	Isolates CCPs' normal	Fails to close or stuck open	Mechanical failure, Train B	Indicating light in MCR	Loss of redundancy in	None. Train A isolation	Failure to open will prevent realignment of

FAILURE MODES AND EFFECTS ANALYSIS FOR ACTIVE FAILURES FOR THE SAFETY INJECTION SYSTEM

Item No.	Component	Function	Failure Mode	Potential Cause	Method of Detection	Effect on System	Effect on Plant	Remarks
		charging path from CCP discharge header; in series with FCV-62-90		power failure; Train B SI Signal failure		normal charging path isolation	valve FCV-62-90 provides isolation of normal charging path	charging after SI termination and is not listed since it has no impact on safety function of System 63.
14	FCV-63-1 Train A	Provide suction flow path from RWST to RHR pumps A-A & B-B (injection mode)	Spurious closing (not a credible failure mode). See 'Remarks' column	See 'Remarks' column	Alarm and indicating light in MCR	See 'Remarks' column	See 'Remarks' column	Valve is normally open administratively controlled (power off) to avoid or minimize possibility of spurious closing due to operator error. Probability of spurious operation due to hot shorts is reduced by wiring HS & XS contact on both sides of contactors. This failure mode, therefore, is extremely improbable (potential cause). However, if it did occur, RHR pumps will not be available until suction from containment sump can be established. (Effect on system) RHR pumps, if undamaged, can be

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FAILURE MODES AND EFFECTS ANALYSIS FOR ACTIVE FAILURES FOR THE SAFETY INJECTION SYSTEM

Item No.	Component	Function	Failure Mode	Potential Cause	Method of Detection	Effect on System	Effect on Plant	Remarks
								realigned for recirc. mode (Effect on plant)
		Closed to prevent backflow from RHR suction to RWST and for isolation of passive failure (recirc. mode)	Stuck open.	Mechanical failure; Train A power failure; operator error	Indicating light in MCR	RHR suction line from RWST pressurized up to RHRP A-A and B-B suction valves 74-3 and 74-21, respectively.	None. Check Valve 63-502 prevents backflow. Additional isolation is provided by FCV's 74-3 and 74-21 which automatically close when RHRP suction valves from sump, FCV-63-72 and FCV-63-73 start opening.	Normal RHR cooldown function not included in SI System FMEA
15	Check Valve 63-502	Provides suction flow path from RWST to RHRPs (Injection mode)	Stuck closed. See 'Remarks' column.	Mechanical failure	Erratic readings on RHRP motor ammeters and RHRP discharge pressure indicators PI-74-13 and	RHR pumps start on SI Signal but suction flow not established. Pump damage unless operator secures pumps based on motor amp and	See 'Remarks' column	This is a credible failure mode since, per IEEE Std. 500-1984, check valves at PWR's have a failure rate (fail to open) of 60 per million demands. Relative to the failure

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FAILURE MODES AND EFFECTS ANALYSIS FOR ACTIVE FAILURES FOR THE SAFETY INJECTION SYSTEM

Item No.	Component	Function	Failure Mode	Potential Cause	Method of Detection	Effect on System	Effect on Plant	Remarks
					PI-74-26 and flow indicators FI-63-91A/B and FI-63-92A/B in MCR. Local indication of suction pressure (PI-74-4 and PI-74-22) and discharge flow (FIS-74-12 and FIS-74-24) and discharge pressure (PI-74-6 and PI-74-18)	discharge pressure readings		mode, the Design Criteria Document does not require consideration of this check valve as an active component for the opening function (and therefore its failure to open) during the injection mode. However, during the recirculation mode, it is an active component and its failure to close is analyzed below. The effect on the plant, if 'Stuck Closed' failure mode were to be considered, is no injection flow from RHR pumps
		Prevents backflow from SIP suction header to RWST (recirc. mode)	Stuck open or fails to backseat	Mechanical failure	Loss of sump inventory, RWST level/temperature increase, Radiation Protection	None. Closure of FCV-74-3/21 or FCV-63-1 isolates RWST.	None.	Normal RHR cooldown function not included in SIS FMEA. Per IEEE Std. 500-1984, failure mode is internal leakage rather than gross failure to close.

FAILURE MODES AND EFFECTS ANALYSIS FOR ACTIVE FAILURES FOR THE SAFETY INJECTION SYSTEM

Item No.	Component	Function	Failure Mode	Potential Cause	Method of Detection	Effect on System	Effect on Plant	Remarks
					survey.			
16	FCV-74-3 Train A	Provides flow path to RHRP A-A from RWST (injection mode)	Stuck closed or spuriously closed	Mechanical failure; operator error	Alarm and valve position indicating light in MCR	Loss of redundancy in RHRS position of SI	None. RHRP B-B starts independently and automatically and can provide adequate injection flow.	Normally open valve; closes automatically at switchover to recirc. mode when FCV-63-72 starts to open.
		Isolates RHRP A-A suction from RWST and provides passive failure isolation (recirc. mode)	Fails to close or stuck open No failure assumed if closure is required for isolation of a passive failure.	Mechanical failure; Train A power failure; failure of close signal from FCV-63-72 limit switch	Valve position indication light in MCR	Operator stops RHRP A-A to terminate RWST outflow. Otherwise RWST inventory may be diverted to the sump via FCVs 63-1, 74-3 and 63-72.	None. RHRP B-B aligned automatically to sump, provides adequate recirc. flow.	If RWST and sump become connected, FCV-63-1 can be remote manually closed to isolate RWST (unless failure is due to Train A power failure)
17	FCV-74-21 Train B	Provides flow path to RHRP B-B from RWST (injection mode)	Stuck closed or spuriously closed	Mechanical failure; operator error	Alarm and valve position ind. light in MCR	Loss of redundancy in RHRS portion of SI	None. RHRP A-A starts independently and automatically and can provide adequate	Normally open valve; closes automatically at switchover to recirc. mode when FCV-63-73 starts to open.

FAILURE MODES AND EFFECTS ANALYSIS FOR ACTIVE FAILURES FOR THE SAFETY INJECTION SYSTEM

Item No.	Component	Function	Failure Mode	Potential Cause	Method of Detection	Effect on System	Effect on Plant	Remarks
							injection flow.	
		Isolates RHRP B-B suction from RWST and provides passive failure isolation. (recirc. mode)	Fails to close or stuck open No failure assumed if closure required for isolation of a passive failure.	Mechanical failure; Train B power failure; failure of close signal from FCV-63-73 limit switch	Valve position indication light in MCR	Operator stops RHRP B-B to terminate RWST outflow. Otherwise RWST inventory may be diverted to the sump via FCV's 74-21 and 63-73.	None. RHRP A-A aligned automatically to sump, provides adequate recirc. flow.	If RWST and sump become connected, FCV-63-1 can be remote manually closed to isolate RWST.
18	FCV-63-72 Train A	Provides suction flow path from containment sump to RHRP A-A to initiate recirc. mode of ECCS	Fails to open or stuck closed	Mechanical failure; Train A power failure; Train A SI (latched) signal failure; Train A RWST LVL Lo/CNTMT sump LVL Hi signal failure	Alarm and valve position indicating light in MCR	RHRP A-A suction cannot be switched over from RWST to containment sump. To prevent depletion of RWST, operator may secure RHRP A-A, resulting in loss of redundancy in recirc.	None. RHRP B-B, independently and automatically aligned to sump, can provide adequate recirc. flow	With RHRP A-A secured, the FCV-63-8 suction path to CCPs will be unavailable. Alternate suction path for CCPs can be established by opening Train B valves 63-11 & 63-6 and through normally open valve 63-177. Suction path for both SIP's (B-B directly and A-A through normally open valves 63-48 and 63-47) can also be established.
			Spuriously	Hot short in	Ind. light in	Loss of RWST	None. RHR	

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FAILURE MODES AND EFFECTS ANALYSIS FOR ACTIVE FAILURES FOR THE SAFETY INJECTION SYSTEM

Item No.	Component	Function	Failure Mode	Potential Cause	Method of Detection	Effect on System	Effect on Plant	Remarks
			opens during injection mode.	control wiring; operator error	MCR; decrease in RWST level; increase in sump level	inventory until operator responds. Unintentional flooding of containment. Inadvertent premature switchover of RHRP A-A suction to sump. Loss of RHRP A-A (NPSH/ Vortexing)	pump B-B available	
		Closes for isolation of passive failure	No additional failures assumed					
19	FCV-63-73 Train B	Provides suction flow path from containment sump to RHRP B-B to initiate recirc. mode of ECCS	Fails to open or stuck closed	Mechanical failure; Train B power failure; Train B SI (latched) signal failure; Train B RWST LVL Lo/CNTMT sump LVL Hi signal failure	Alarm and valve position indication light in MCR	RHRP B-B suction cannot be switched over from RWST to containment sump. To prevent depletion of RWST, operator may secure RHRP B-B resulting in loss of	None. RHRP A-A, independently and automatically aligned to sump, can provide adequate recirc. flow	With RHRP B-B secured, the FCV-63-11 suction path to SIPs will be unavailable. However, alternate paths for SIP suction can be established.

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FAILURE MODES AND EFFECTS ANALYSIS FOR ACTIVE FAILURES FOR THE SAFETY INJECTION SYSTEM

Item No.	Component	Function	Failure Mode	Potential Cause	Method of Detection	Effect on System	Effect on Plant	Remarks
						redundancy in recirc.		
			Spuriously opens during injection mode.	Hot short in control wiring; Operator error	Indicating light in MCR; decrease in RWST level; increase in sump level.	Loss of RWST inventory until operator responds. Unintentional flooding of containment. Inadvertent switchover of RHRP B-B suction to sump. Loss of RHRP B-B (NPSH/ Vortexing).	None RHRP A-A available	
		Closes for isolation of passive failures	No additional failures assumed					
20	FCV-63-8 Train A	Opens to provide primary flow path for CCPs' suction and alternate flow path for SIPs' suction from RHRP	Fails to open, stuck closed or spurious opening	Mechanical failure; Train A power failure; operator error	Valve position indicating lights in MCR	Loss of redundancy in flow paths from RHRS to suction of CCPs & SIPs.	None. Alternate suction flow path for CCPs' suction can be established from independent	In the alternate flow path, Train A valves 63-47 and 63-177 are normally open and do not close for recirc. alignment. Train A valve 63-7 is normally closed, but is in parallel with Train B valve 63-6.

FAILURE MODES AND EFFECTS ANALYSIS FOR ACTIVE FAILURES FOR THE SAFETY INJECTION SYSTEM

Item No.	Component	Function	Failure Mode	Potential Cause	Method of Detection	Effect on System	Effect on Plant	Remarks
		A-A discharge (recirc. mode)					RHR Train B via FCV's 63-11, 63-48, 63-47, 63-6/63-7 and 63-177	The alternate flow path, therefore, can be established even if due to Train A power failure.
		Remains closed during injection mode and closes for isolation of passive failures	Spuriously opens during injection mode No additional failure assumed beyond passive failure	Hot short in control wiring; operator error	Indicating light in MCR	Suction pressure boost to CCPs	None. See 'Remarks'	Spurious operation is very unlikely due to interlocks with valves 63-72, 63-3, 63-4 & 63-175, protective covers on HS and use of double contacts of HS & XS.
21	FCV-63-11 Train B	Opens to provide primary flow path for SIPs' suction and alternate flow path for CCPs' suction from RHRP B-B discharge (recirc. mode)	Fails to open, stuck closed or spurious closing	Mechanical failure; Train B power failure; operator error	Valve position indicating lights in MCR	Loss of redundancy in flow paths from RHRS to suction of CCPs and SIPs.	None. Alternate flow path for SIPs' suction can be established from independent RHR Train A via FCV's 63-8, 63-177, 63-6/63-7, 63-48 and 63-47	In the alternate flow path, Train B valve 63-48 is normally open. Train B valve 63-6 is normally closed, but in parallel with Train A valve 63-7. The alternate flow path, therefore, can be established even if valve 63-11 fails due to train B power failure.
		Remains closed during injection mode and	Spuriously opens during the injection mode	Hot short in control wiring; operator error	Indicating light in MCR	Suction pressure boost to SIPs.	None. See 'Remarks'	Spurious operation is very unlikely due to interlocks with valves 63-73, 63-3, 63-4 & 63-

FAILURE MODES AND EFFECTS ANALYSIS FOR ACTIVE FAILURES FOR THE SAFETY INJECTION SYSTEM

Item No.	Component	Function	Failure Mode	Potential Cause	Method of Detection	Effect on System	Effect on Plant	Remarks
		closes for isolation of passive failures	No additional failure assumed beyond passive failure					175 protective covers on HS and use of double contacts of HS & XS
22	Permissive interlock common to FCVs 63-8 and 63-11	Permit remote manual opening of MOVs 63-8 and 63-11 only if SIP min. flow circuit valve 63-3 (Train A) or valves 63-4 and 63-175 (Train B) are closed, to prevent contamination of RWST with sump water during recirc. mode.	No credible failure mode. See 'Remarks' column	See 'Remarks' column	See 'Remarks' column	None. See 'Remarks' column	None. See 'Remarks' column	The interlock for each valve is implemented using internal limit switches from the same division valve (i.e., 63-3 in the 63-8 circuit and 63-4 and 63-175 in the 63-11 circuit), and auxiliary (separation) relay contacts from stem-mounted limit switches of opposite division valves. A review of the schematics shows that no single failure of the stem-mounted LSs or power supplies in L-10, L-11A or L-11B can prevent the opening of both valves 63-8 and 63-11.
23	FCV-74-33 Train A	Closes to provide train	Fails to close or stuck open	Mechanical failure, Train A	Alarm and indicating	RHRP A-A remains	None. Train separation	Valve kept open during reactor operation and

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FAILURE MODES AND EFFECTS ANALYSIS FOR ACTIVE FAILURES FOR THE SAFETY INJECTION SYSTEM

Item No.	Component	Function	Failure Mode	Potential Cause	Method of Detection	Effect on System	Effect on Plant	Remarks
		separation in RHRS for 1) passive failure protection (cold leg recirc. mode) 2) for increased RHRS resistance to accommodate RHR A-A cold leg injection and containment spray, and 3) for increased RHRS resistance for RHR B-B HL injection	No additional failures assumed following a passive failure	power failure; operator error.	light in MCR	connected to cross-tie line up to Train B valves 74-35 and 63-172	can be achieved by closing Train B valves 74-35 and 63-172. SIP can be used for HL recirc.	injection mode. Spurious closing during injection as a failure mode is not a problem since both RHRPs would be available.
	FCV-74-33 Train A	Opens to provide flow path for RHRP A-A discharge to hot legs 1 and 3 (HL recirc. mode)	Fails to open, stuck closed or spurious closing	Mechanical failure; Train A power failure; operator error	Indicating light in MCR	RHRP A-A unavailable for hot leg recirc.	None. RHRP B-B can provide recirc. flow to HLs 1 and 3 through valves 74-35 and 63-172. SIPs can also	

FAILURE MODES AND EFFECTS ANALYSIS FOR ACTIVE FAILURES FOR THE SAFETY INJECTION SYSTEM

Item No.	Component	Function	Failure Mode	Potential Cause	Method of Detection	Effect on System	Effect on Plant	Remarks
							provide HL recirc.	
24	FCV-74-35 Train B	Closes to provide train separation in RHRS for 1) passive failure protection (cold leg recirc. mode) 2) for increased RHRS resistance to accommodate RHR B-B cold leg injection and containment spray, and 3) for increased RHRS resistance for RHR A-A HL injection	Fails to close or stuck open No additional failure assumed following a passive failure	Mechanical failure; Train B power failure; operator error	Alarm and indicating light in MCR	RHRP B-B remains connected to cross-tie line up to train A valve 74-33	None. Train separation can be achieved by closing Train A valve 74-33. SIP can be used for HL recirc.	Valve kept open during reactor operation and injection mode. Spurious closing during injection as a failure mode is not a problem since both RHRPs would be available
		Opens to provide flow path for RHRP B-B	Fails to open, stuck closed or spurious closing	Mechanical failure; Train B power failure; operator error	Indicating light in MCR	RHRP B-B unavailable for hot leg recirc.	None. HL 1 and 3 recirc. flow can be provided by	

FAILURE MODES AND EFFECTS ANALYSIS FOR ACTIVE FAILURES FOR THE SAFETY INJECTION SYSTEM

Item No.	Component	Function	Failure Mode	Potential Cause	Method of Detection	Effect on System	Effect on Plant	Remarks
		discharge to hot legs 1 and 3 (HL recirc. mode)					opening Train A valve 74-33 & Train B 63-172. SIPs can also provide HL recirc.	
25	FCV-63-172 Train B	Opens to provide flow path for RHRP flow to hot legs 1 and 3 (HL recirc. mode)	Fails to open, stuck closed or spurious closing	Mechanical failure; Train B power failure; operator error	Indicating light in MCR	No RHRP flow to hot legs 1 and 3	None. SIP A-A can supply HLs 1 and 3, with suction flow from RHRP A-A by opening Train A valves 63-8 and 63-7	FCV-63-172 is closed except during HL recirc. mode.
		Remains closed during CL injection mode	Spuriously opens	Hot short in control wiring; operator error	Indicating light in MCR	Inadvertent flow to HLs 1 & 3 and reduced flow to RCS CLs	None. Both RHR pumps are available and provide sufficient flow to CLs (4) & HLs 1 & 3.	Spurious operation very unlikely due to protective cover on HS and the use of double contacts on HS & XS.
		Closes to provide passive failure isolation	No additional failures assumed					

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FAILURE MODES AND EFFECTS ANALYSIS FOR ACTIVE FAILURES FOR THE SAFETY INJECTION SYSTEM

Item No.	Component	Function	Failure Mode	Potential Cause	Method of Detection	Effect on System	Effect on Plant	Remarks
26	FCV-63-5 Train B	Provide suction flow path from RWST to suction of SIPs (injection mode)	Spurious closing (not a credible failure mode). See 'Remarks' column	See 'Remarks' column	Alarm and indicating light in MCR	See 'Remarks' column	See 'Remarks' column	Procedures do not require the normally open valve to be closed until switchover to recirc. mode. HS provided with protective cover. Very unlikely error of action. Probability of spurious closing is reduced by protective cover over HS and by wiring HS & XS contacts on both sides of contactors. The failure mode is, therefore, not credible (potential cause). However, if it did occur, SIPs will not be available during injection mode. (Effect on system) Both CCPs, both RHRPs and all four accumulators provide injection flow. SIPs if undamaged, can be realigned for recirc. mode (Effect on plant)

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FAILURE MODES AND EFFECTS ANALYSIS FOR ACTIVE FAILURES FOR THE SAFETY INJECTION SYSTEM

Item No.	Component	Function	Failure Mode	Potential Cause	Method of Detection	Effect on System	Effect on Plant	Remarks
		Isolate RWST from suction of SIPs during recirc. mode. Passive failure (recirc. mode) isolation	Fails to close or stuck open No additional failure assumed beyond the passive failure	Mechanical failure; Train B power failure	Indicating light in MCR	None. Check valve 63-510 prevents backflow from sump to RWST	None.	
27	Check Valve 63-510	Opens to provide suction flow path from RWST to SIPs (injection mode)	Stuck closed. See 'Remarks' column	Mechanical failure	Erratic readings on SIP motor ammeters, SIP discharge pressure indicators PI-63-19 and PI-63-150 and SIP discharge flow indicators FI-63-20 and FI-63-151, all in MCR. Local indication of	SIPs unavailable	See 'Remarks' column	Per IEEE std. 500-1984, check valves at PWRs have a failure rate (fail to open) of 60 per million demands, which makes this a credible failure mode. Relative to the failure mode, the Design Criteria document does not require consideration of this check valve as an active component for the opening function (and therefore, its failure to open) during

FAILURE MODES AND EFFECTS ANALYSIS FOR ACTIVE FAILURES FOR THE SAFETY INJECTION SYSTEM

Item No.	Component	Function	Failure Mode	Potential Cause	Method of Detection	Effect on System	Effect on Plant	Remarks
					SIP suction pressure on PI-63-9 and PI-63-14			the injection mode. However, during the recirculation mode it is an active component and its failure to close is analyzed below. The effect on the plant if "stuck closed" failure mode were to be considered, is SIPs unavailable for injection mode. Both CCPs, both RHRPs and all four accumulators provide injection flow.
		Backseats to prevent flow from sump to RWST (Recirc. mode)	Stuck open or fails to backseat	Mechanical failure	Change in RWST level and temperature. Radiation Protection surveys of RWST area	None. FCV-63-5 is closed to complete switchover to recirc. mode.	None.	Per IEEE Std. 500-1984, failure mode is internal leakage rather than gross failure to close.
28	FCV-63-47 Train A	Provide suction flow path for SIP A-A (injection mode)	Spurious closing	Operator error, hot short in control circuit	Alarm and indicating light in MCR	SIP A-A unavailable (injection mode)	None. SIP B-B, both CCPs, both RHRPs and all four accumulators remain	Failure unlikely in both (injection and recirc.) modes since the valve is not required to be closed except for isolation of passive

FAILURE MODES AND EFFECTS ANALYSIS FOR ACTIVE FAILURES FOR THE SAFETY INJECTION SYSTEM

Item No.	Component	Function	Failure Mode	Potential Cause	Method of Detection	Effect on System	Effect on Plant	Remarks
							available to provide adequate injection flow for all break sizes.	failures. Failure of valve to close is not listed as a failure mode since the passive failure requiring its operation is the single failure.
		Provides connection to recirc. flow path between suction of SIPs and CCPs (Recirc. mode) Closes for isolation of passive failures (see 'Remarks' column)	Spurious closing	Operator error, hot short in control circuit	Alarm and indicating light in MCR	No redundant suction flow path from RHRP B-B to SIP A-A and CCPs or from RHRP A-A to SIP B-B (recirc. mode)	None. All pumps remain available, due to flow path from RHRP A-A to SIP A-A & CCPs (63-8, 63-177, and 63-6 or 63-7) and from RHRP B-B to SIP B-B (63-11)	
29	FCV-63-48 Train B	Provide suction flow path for SIP B-B (injection mode)	Spurious closing	Operator error; hot short in control circuit	Alarm and indicating light in MCR	SIP B-B unavailable (injection mode)	None. SIP A-A, both CCPs, both RHRPs, and all four accumulators remain available to provide	Failure unlikely in both (injection and recirc.) modes since the valve is not required to be closed except for isolation of passive failures. Failure of valve to close is not

Table 6.3-8 (Sheet 26 of 58)
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FAILURE MODES AND EFFECTS ANALYSIS FOR ACTIVE FAILURES FOR THE SAFETY INJECTION SYSTEM

Item No.	Component	Function	Failure Mode	Potential Cause	Method of Detection	Effect on System	Effect on Plant	Remarks
							adequate injection flow for all break sizes.	listed as a failure mode since the passive failure requiring its operation is the single failure.
		Provides connection for recirc. flow path between suction of SIPs and CCPs (Recirc. mode) Closes for isolation of passive failures (See 'Remarks' column)	Spurious closing	Operator error; hot short in control circuit	Alarm and indicating light in MCR	No redundant suction flow path to RHRP A-A to SIP B-B or from RHRP B-B to SIP A-A and CCPs	None. All pumps remain available, due to flow path from RHRP A-A to SIP A-A and CCPs (63-8, 63-177, 63-6/63-7) and from RHRP B-B to SIP B-B (63-11)	
30	Safety Injection Pump A-A	Provides emergency core cooling by pumping into RCS cold legs, borated water from RWST during injection mode and	Fails to start; fails while running	Mechanical failure; Train A power failure, Train A SI signal failure; motor overload; electrical fault; operator error (HS in wrong position)	Annunciation, and indicating lights in main control room, no motor amps, no header flow, no pump discharge	Loss of redundancy in intermediate head portion of SI System	None. SI Pump B-B can provide required intermediate head injection flow for design basis range of break sizes.	SIPs start automatically on SI signal. Automatic operation of each SIP in ECCS injection mode is completely independent of the SIP.

FAILURE MODES AND EFFECTS ANALYSIS FOR ACTIVE FAILURES FOR THE SAFETY INJECTION SYSTEM

Item No.	Component	Function	Failure Mode	Potential Cause	Method of Detection	Effect on System	Effect on Plant	Remarks
		contents of containment sump via RHR pumps during recirc. mode.			pressure			
31	Safety Injection Pump B-B	Provides emergency core cooling by pumping into RCS cold legs, borated water from RWST during injection mode and contents of containment sump via RHR pumps during recirc. mode.	Fails to start; fails while running	Mechanical failure; Train B power failure, Train B SI signal failure; motor overload; electrical fault; operator error (HS in wrong position)	Annunciation and indicating lights in main control room, no motor amps, no header flow, no pump discharge pressure	Loss of redundancy in intermediate head portion of SI System	None. SI Pump A-A can provide required intermediate head injection flow for design basis range of break sizes.	SI pumps start automatically on SI signal. Automatic operation of each SIP in ECCS injection mode is completely independent of other SIP.
32	FCV-63-3 Train A	Provide min. flow recirc. path from SIP A-A & B-B to RWST for pump protection	Spurious closing (not a credible failure mode). See 'Remarks' column	See 'Remarks' column	Alarm and indicating light in MCR	See 'Remarks' column	See 'Remarks' column	Procedures do not require closing this valve until switchover to CL recirc. mode. Probability of spurious closing is reduced by protective cover over

FAILURE MODES AND EFFECTS ANALYSIS FOR ACTIVE FAILURES FOR THE SAFETY INJECTION SYSTEM

Item No.	Component	Function	Failure Mode	Potential Cause	Method of Detection	Effect on System	Effect on Plant	Remarks
		(injection mode)						HS and by wiring HS & XS contacts on both sides of contactors. This failure mode is, therefore, not credible (potential cause). However, if it did occur, the min. flow circuit for both SIPs will be unavailable, with damage to SIPs possible for small break LOCA and slow RCS depressurization (effect on system). Potential loss of intermediate head SI (effect on plant).
	FCV-63-3 Train A	Isolate SIP A-A & B-B miniflow to RWST when SIP suction is from containment sump via RHRPs, to prevent contamination	Fails to close, stuck open or spurious reopening	Mechanical failure; Train A power failure; operator error	Indicating light in MCR	None. Operator can isolate RWST by closing Train B valves 63-4 and 63-175 and then open valves 63-8 or 63-11 to complete switchover of CCP and SIP	None	Schematics for valves 63-8 and 63-11 were reviewed and it was determined that at least one suction path to both SIPs and both CCPs can be established with a Train A power failure and failure of non-divisional power supply to panel

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FAILURE MODES AND EFFECTS ANALYSIS FOR ACTIVE FAILURES FOR THE SAFETY INJECTION SYSTEM

Item No.	Component	Function	Failure Mode	Potential Cause	Method of Detection	Effect on System	Effect on Plant	Remarks
		of RWST (recirc. mode)				suction.		L-10 or failure of any stem mounted-limit switch.
33	FCV-63-4 Train B	Connect SIP A-A discharge to min. flow recirc. line (injection mode)	Spurious closing	Operator error; hot short in control wiring	Alarm and indicating light in MCR	Min. flow circuit for SIP A-A protection unavailable, with damage to SIP A-A possible for small break LOCA and slow RCS depressurization	None. SIP B-B still available	Failure mode very unlikely since procedures do not require closing this valve until switchover to CL recirc. mode.
		Isolate SIP A-A miniflow to RWST when SIP suction is from containment sump via RHRPs, to prevent contamination of RWST (recirc. mode)	Fails to close, stuck open or spuriously reopens	Mechanical failure; Train B power failure; operator error	Indicating light in MCR	None. Operator can isolate RWST by closing Train A valve 63-3 and then open 63-8 or 63-11 to complete switchover of CCP and SIP suction to sump	None	Schematics for valves 63-8 and 63-11 were reviewed to determine that at least one suction path to both SIPs and both CCPs can be established with a Train B power failure and failure of non-divisional power to panel L-10 or failure of any stem-mounted limit switch.
34	FCV-63-175 Train B	Connect SIP B-B discharge to min. flow	Spurious closing	Operator error; hot short in control wiring	Alarm and indicating light in MCR	Min. flow circuit for SIP B-B protection	None. SIP A-A still available	Failure mode very unlikely since procedures do not

FAILURE MODES AND EFFECTS ANALYSIS FOR ACTIVE FAILURES FOR THE SAFETY INJECTION SYSTEM

Item No.	Component	Function	Failure Mode	Potential Cause	Method of Detection	Effect on System	Effect on Plant	Remarks
		recirc. line (injection mode)				unavailable, with damage to SIP B-B possible for small break LOCA and slow RCS depressurization		require closing this valve until switchover to CL recirc. mode.
		Isolate SIP B-B discharge from min flow recirc. line when SIP B-B suction is from containment sump via RHRPs, to prevent contamination of RWST (recirc. mode)	Fails to close or stuck open or spuriously reopens	Mechanical failure; Train B power failure; operator error	Indicating light in MCR	None. Operator can isolate RWST by closing train A valve 63-3 and then open 63-8 or -11 to complete switchover of CCP and SIP suction to sump.	None.	Schematics for valves 63-8 and 63-11 were reviewed to determine that at least one suction path to both SIPs and both CCPs can be established with a Train B power failure and failure of non-divisional power to panel L-10 or failure of any stem-mounted limit switch.
35	Check valve 63-524	Provides discharge flow path for SIP A-A	Stuck closed	Mechanical failure	Pump motor amps less than full load; no SIP A-A flow as indicated on FI-63-151 in MCR	Loss of redundancy in intermediate head injection portion of SI System	None. SIP B-B can provide required injection flow.	Failure mode is credible; Per IEEE Std. 500-1984, check valves at PWRs have a failure rate (fail to open) of 60 per million demands.

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FAILURE MODES AND EFFECTS ANALYSIS FOR ACTIVE FAILURES FOR THE SAFETY INJECTION SYSTEM

Item No.	Component	Function	Failure Mode	Potential Cause	Method of Detection	Effect on System	Effect on Plant	Remarks
		Prevents reverse flow of SIP B-B discharge through SIP A-A (injection mode)	Stuck open or fails to backseat	Mechanical Failure	Pump motor amps above full load, discharge pressure high on idle pump	See 'Remarks' column	See 'Remarks' column	Since the failure of the check valve is the single failure postulated, both SI Pumps A-A & B-B can be assumed to operate.
36	Check Valve 63-526	Provides discharge flow path for SIP B-B	Stuck closed	Mechanical failure	Pump motor amps less than full load; No SIP B-B flow as indicated on FI-63-20 in MCR	Loss of redundancy in intermediate head injection portion of SI system	None. SIP A-A can provide required injection flow.	Failure mode is credible; Per IEEE Std. 500-1984, check valves at PWRs have a failure rate (fail to open) of 60 per million demands.
		Prevents reverse flow of SIP A-A discharge through SIP B-B (injection mode)	Stuck open or fails to backseat	Mechanical failure	Pump motor amps above full load. Discharge pressure high on idle pump	See 'Remarks' column	See 'Remarks' column	Since the failure of the check valve is the single failure postulated, both SI Pumps A-A & B-B can be assumed to operate.
37	Check Valve 63-528	Opens to connect SIP A-A discharge to min. flow recirc. (injection	Stuck closed	Mechanical failure	Possible abnormal readings on pump motor ammeter; low flow on FI-	Min. flow circuit for SIP A-A protection unavailable, with damage to SIP A-A possible for	None. SIP B-B still available	Failure mode is credible; Per IEEE Std. 500-1984, check valves at PWRs have a failure rate (fail to open) of 60 per million demands

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FAILURE MODES AND EFFECTS ANALYSIS FOR ACTIVE FAILURES FOR THE SAFETY INJECTION SYSTEM

Item No.	Component	Function	Failure Mode	Potential Cause	Method of Detection	Effect on System	Effect on Plant	Remarks
		mode)			63-2 local. SIP A-A damage.	small break LOCA and slow RCS depressuirzation		
		Prevents reverse flow of SIP B-B discharge through SIP A-A (injection mode)	Stuck open or fails to backseat	Mechanical failure	Pump motor amps above full load. Discharge pressure high on idle pump	See 'Remarks' column	See 'Remarks' column	Since the failure of the check valve is the single failure postulated, both SI Pumps A-A & B-B can be assumed to operate.
38	Check Valve 63-530	Opens to connect SIP B-B discharge min. flow recirc. (injection mode)	Stuck closed	Mechanical failure	Possible abnormal readings on pump motor ammeter; low flow on FI-63-2 local. SIP B-B damage.	Min. flow circuit for SIP B-B protection unavailable, with damage to SIP B-B possible for small break LOCA and slow RCS depressurization	None. SIP A-A still available	Failure mode is credible; Per IEEE Std. 500-1984, check valves at PWRs have a failure rate (fail to open) of 60 per million demands
		Prevents reverse flow of SIP A-A discharge through SIP B-B (injection mode)	Stuck open or fails to backseat	Mechanical failure	Pump motor amps above full load; discharge pressure high on idle pump	See 'Remarks' column	See 'Remarks' column	Since the failure of the check valve is the single failure postulated, both SI pumps A-A & B-B can be assumed to operate.

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FAILURE MODES AND EFFECTS ANALYSIS FOR ACTIVE FAILURES FOR THE SAFETY INJECTION SYSTEM

Item No.	Component	Function	Failure Mode	Potential Cause	Method of Detection	Effect on System	Effect on Plant	Remarks
39	FCV-63-39 Train A	Normally locked open valve provides flow path (parallel with FCV-63-40) for discharge flow of CCPs to cold legs (injection and recirc. modes)	Spurious closing; See ‘Remarks’ column.	Operator error; See ‘Remarks’ column.	None	None. Train B valve 63-40 maintains flow path	None	Operator error is not credible since valve is locked open and procedures do not require closing this Valve under any operating scenario. Failure to close is not listed since closing is not a SIS safety function Breaker is locked opened to prevent spurious operation.
40	FCV-63-40 Train B	Normally locked open valve provides flow path (in parallel with FCV-63-39) for discharge flow of CCPs to cold legs (injection and	Spurious closing; See ‘Remarks’ column.	Operator errors; See ‘Remarks’ column.	None	None. Train A valve 63-39 maintains flow path	None	Operator error is not credible since valve is locked open and procedures do not require closing this valve under any operating scenario.

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FAILURE MODES AND EFFECTS ANALYSIS FOR ACTIVE FAILURES FOR THE SAFETY INJECTION SYSTEM

Item No.	Component	Function	Failure Mode	Potential Cause	Method of Detection	Effect on System	Effect on Plant	Remarks
		recirc. modes)						Failure to close is not listed since closing is not a SIS safety function. Breaker is locked open to prevent spurious operation.
41	FCV-63-25 Train B	Provide flow path (in parallel with FCV-63-26) for discharge flow of CCPs to cold legs (injection and recirc. modes) Closes to isolate passive failures or terminate CL injection.	Fails to open, stuck closed or spuriously recloses after opening	Mechanical failure; Train B power failure; Train B SI signal failure; operator error	Alarm and indicating light in MCR	None. FCV-63-26 independently and automatically opens to establish flow path.	None See 'Remarks column	Failure to close not listed as a failure mode since the passive failure requiring its operation is the single failure. SI termination of CCP cold leg injection can be accomplished by local valve operations if necessary
42	FCV-63-26 Train A	Provide flow path (in parallel with FCV-63-25)	Fails to open, stuck closed or spuriously recloses after	Mechanical failure; Train A power failure; Train A SI signal	Alarm and indicating light in MCR	None. FCV-63-25 independently and automatically	None	Failure to close not listed as a failure mode since the passive failure requiring its operation is

FAILURE MODES AND EFFECTS ANALYSIS FOR ACTIVE FAILURES FOR THE SAFETY INJECTION SYSTEM

Item No.	Component	Function	Failure Mode	Potential Cause	Method of Detection	Effect on System	Effect on Plant	Remarks
		for discharge flow of CCPs to cold legs (injection and recirc. modes) Closes to isolate passive failures or terminate CL injection.	opening	failure; operator error		opens to establish flow path.	See 'Remarks column	the single failure. SI termination of CCP cold leg injection can be accomplished by local valve operations if necessary
43	FCV-63-152 Train A	Provide flow path from SIP A-A discharge to cold legs (injection and CL recirc. modes)	Spurious closing	Operator error; hot short in control wiring	Alarm and indicating light in MCR, motor amperes less than full load, low flow indication on FI-63-151	No redundancy in SIP portion of cold leg SIS	None. SIP B-B through Train B valve 63-153 provides adequate CL flow.	Normally open valve; procedures do not require closing until switchover to HL recirc. for train isolation
		Closes to provide train isolation for HL recirc mode	Fails to close; stuck open or spuriously reopens after closing	Mechanical failure; Train A power failure; hot short in control wiring; operator error	Indicating light in MCR. Abnormal flow from SIP A-A	Inability to switch SIP A-A to HL recirc.	None. SIP B-B with Train B valve 63-153 isolated provides adequate recirc. flow.	RHR can also provide HL recirc.

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FAILURE MODES AND EFFECTS ANALYSIS FOR ACTIVE FAILURES FOR THE SAFETY INJECTION SYSTEM

Item No.	Component	Function	Failure Mode	Potential Cause	Method of Detection	Effect on System	Effect on Plant	Remarks
44	FCV-63-153 Train B	Provide flow path from SIP B-B discharge to cold legs (injection and CL recirc. modes)	Spurious closing	Operator error; hot short in control wiring	Alarm and indicating light in MCR, motor amperes less than full load; low flow ind. on FI-63-20	No redundancy in SIP portion of cold leg SIS	None. SIP A-A through Train A valve 63-152 provides adequate CL flow.	Normally open valve, procedures do not require closing until switchover to HL recirc., for train isolation
		Provide train isolation for HL recirc. mode	Fails to close, stuck open or spuriously reopens after closing	Mechanical failure; Train B power failure; hot short in control wiring; operator error	Indicating light in MCR. Abnormal flow from SIP B-B	Inability to switch SIP B-B to HL recirc.	None. SIP A-A with Train A valve 63-152 isolated provides adequate recirc. flow.	RHR can also provide HL recirc.
45	FCV-63-22 Train B	Provide flow path from discharge of SIPs A-A & B-B to cold legs (injection and CL recirc. modes)	Spurious closing (not a credible failure mode). See 'Remarks' column	See 'Remarks' column	Alarm and indicating light in MCR; no flow indication on FI-63-151 and FI-63-20	See 'Remarks' column	See 'Remarks' column	Valve is normally open, administratively controlled (power off) to minimize possibility of spurious closing by operator. Also, HS has a protective cover. Hot short in control wiring unlikely to cause spurious operation since control and selector switch contacts are wired on both sides of contactors (potential cause). If failure occurs

FAILURE MODES AND EFFECTS ANALYSIS FOR ACTIVE FAILURES FOR THE SAFETY INJECTION SYSTEM

Item No.	Component	Function	Failure Mode	Potential Cause	Method of Detection	Effect on System	Effect on Plant	Remarks
								in spite of these precautions, CL injection from SIPs will be unavailable (effect on system). Both CCPs and both RHRPs will be available and will provide injection and CL recirc. flow. For injection mode, all four accumulators are also available (effect on plant)
	FCV-63-22 Train B							Failure to close or stuck open failure mode is not listed since this valve, even though it will be closed for HL recirc. mode, does not have a safety related isolation function. SIP train isolation can be achieved by closing valves 63-152 and 63-153.
46	FCV-63-156 Train A	Provide flow path from SIP A-A discharge to hot legs 1	Fails to open, stuck closed or spuriously recloses	Mechanical failure, Train A power failure; operator error;	Indicating light in MCR; no flow indication of	No redundancy in SIP flow to hot legs	None. SIP B-B or one RHRP can provide	Normally closed valve, opened only for HL recirc. mode. Spurious operation very unlikely

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FAILURE MODES AND EFFECTS ANALYSIS FOR ACTIVE FAILURES FOR THE SAFETY INJECTION SYSTEM

Item No.	Component	Function	Failure Mode	Potential Cause	Method of Detection	Effect on System	Effect on Plant	Remarks
		and 3 (HL recirc. mode)		hot short in control wiring	FI-63-151; low motor amperes on SIP A-A		adequate HL recirc. flow.	since the HS has a protective cover and contacts of HS & XS are wired on both sides of contactor.
		Closes to isolate HL path to allow adequate CL injection flow, and for passive failures	Spuriously opens	Operator error; hot short in control wiring	Alarm and indicating light in MCR; high flow indication on FI-63-151; high motor amps on SIP A-A	Simultaneous SIP flow to CL & HL	None. SIP A-A & B-B operating will provide adequate CL injection flow	In the event of a passive failure, any further failures are not assumed.
47	FCV-63-157 Train B	Provide flow path from SIP B-B discharge to hot legs 2 and 4 (HL recirc. mode)	Fails to open, stuck closed or spuriously recloses	Mechanical failure, Train B power failure; operator error; hot short in control wiring	Indicating light in MCR; no flow indication on FI-63-20; low motor amperes on SIP B-B	No redundancy in SIP flow to hot legs	None. SIP A-A or one RHRP can provide adequate HL recirc. flow.	Normally closed valve, opened only for HL recirc. mode. Spurious operation very unlikely since HS has a protective cover and contacts of HX & XS are wired on both sides of contactor.
		Closes to isolate HL path to allow adequate CL injection flow,	Spuriously opens	Operator error; hot short in control wiring	Alarm and indicating light in MCR; high flow indicating on	Simultaneous SIP flow to CL & HL	None. SIP A-A & B-B operating will provide adequate CL	In the event of a passive failure, any further failures are not assumed.

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FAILURE MODES AND EFFECTS ANALYSIS FOR ACTIVE FAILURES FOR THE SAFETY INJECTION SYSTEM

Item No.	Component	Function	Failure Mode	Potential Cause	Method of Detection	Effect on System	Effect on Plant	Remarks
		and for passive failures			FI-63-151; high motor amps on SIP B-B		injection flow	
48	RHR Pump A-A	Provide low pressure, high flow emergency core cooling by pumping into RCS cold legs, borated water from RWST during injection mode and into RCS cold and/or hot legs, contents of containment sump during recirc. mode. Also, provide suction to CCPs and SIPs during recirc. mode.	Fails to start; fails while running	Mechanical failure; Train A power failure; Train A SI signal failure; motor overload electrical fault; operator error (HS in wrong position)	Alarm and indicating light in MCR; HS position	Loss of redundancy in low pressure injection portion of SI system	None. RHRP B-B can provide required low pressure injection flow for large breaks. During recirc. mode, even if failure is due to Train A power failure, suction path from RHRP B-B to both CCPs and both SIPs can be established by opening Train B valves, and through normally open Train A	Automatic operation of RHRP B-B in injection and recirc. modes is completely independent of RHRP A-A. RHR spray mode and normal cooldown mode are not within the scope of this FMEA.

FAILURE MODES AND EFFECTS ANALYSIS FOR ACTIVE FAILURES FOR THE SAFETY INJECTION SYSTEM

Item No.	Component	Function	Failure Mode	Potential Cause	Method of Detection	Effect on System	Effect on Plant	Remarks
							MOV's.	
49	RHR Pump B-B	Provide low pressure, high flow emergency core cooling by pumping into RCS cold legs, borated water from RWST during injection mode and into RCS cold and/or hot legs, contents of containment sump during recirc. mode. Also, provide suction to CCPs and SIPs during recirc. mode.	Fails to start; fails while running	Mechanical failure; Train B power failure; Train B SI signal failure; motor overload electrical fault; operator error (HS in wrong position)	Alarm and indicating light in MCR; HS position	Loss of redundancy in low pressure injection portion of SI system	None. RHRP A-A can provide required low pressure injection flow for large breaks. During recirc. mode, even if failure is due to Train B power failure, suction path from RHRP A-A to both CCPs and both SIPs can be established by opening Train A valves, and through normally open Train B MOV's.	Automatic operation of RHRP A-A in injection and recirc. modes is completely independent of RHRP B-B. RHR spray mode and normal cooldown mode are not within the scope of this FMEA.
50	FCV-74-12 Train A	Opens to provide min.	Fails to open, stuck closed or	Mechanical failure; operator	Indicating light in MCR;	Min. flow circuit for RHRP A-A	None. RHRP B-B still	

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FAILURE MODES AND EFFECTS ANALYSIS FOR ACTIVE FAILURES FOR THE SAFETY INJECTION SYSTEM

Item No.	Component	Function	Failure Mode	Potential Cause	Method of Detection	Effect on System	Effect on Plant	Remarks
		flow path for RHRP A-A protection below low flow setpoint (injection & recirc. modes)	spuriously closed	error (HS in wrong position); flow switch FIS-74-12 failure; power failure	low flow alarm (for failure not due to flow switch failure) HS position. Local FIS flow indication	unavailable, with damage to RHRP A-A possible for small or medium LOCA and slow RCS depressurization	available	
		Closes to isolate min. flow path above low flow setpoint (injection & recirc. modes)	Fails to close, stuck open or spuriously opened	Mechanical failure; operator error (HS in wrong position). Flow switch FIS-74-12 failure; Power failure	Indicating light in MCR; HS position, local FIS flow indication	Reduced LP-SI flow from RHRP A-A	None. RHRP B-B still available	
51	FCV-74-24 Train B	Opens to provide min. flow path for RHRP B-B protection below low flow setpoint (injection & recirc. modes)	Fails to open, stuck closed or spuriously closed	Mechanical failure; operator error (HS in wrong position); flow switch FIS-74-24 failure; power failure.	Indicating light in MCR; low flow alarm (for failure not due to flow switch failure) HS position.	Min. flow circuit for RHRP B-B unavailable, with damage to RHRP B-B possible for small or medium LOCA and slow RCS depressurization	None. RHRP A-A still available	
		Closes to	Fails to close,	Mechanical	Indicating	Reduced LP-SI	None. RHRP	

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FAILURE MODES AND EFFECTS ANALYSIS FOR ACTIVE FAILURES FOR THE SAFETY INJECTION SYSTEM

Item No.	Component	Function	Failure Mode	Potential Cause	Method of Detection	Effect on System	Effect on Plant	Remarks
		isolate min. flow path above low flow setpoint (injection & recirc. modes)	stuck open or spuriously opened	failure; operator error (HS in wrong position); flow switch FIS-74-24 failure; power failure.	light in MCR; HS position; local FIS flow indication	flow from RHRP B-B	A-A still available	
52	Check Valve 74-514	Opens to provide flow path for RHRP A-A discharge (injection & recirc. modes)	Stuck closed	Mechanical failure	Pump motor amps less than full load; low RHRP A-A flow alarm from FIS-74-12	Loss of redundancy in LP-SI	None. RHRP B-B can provide required low head flow	Failure mode is credible; Per IEEE Std. 500-1984, check valves at PWRs have a failure rate (fail to open) of 60 per million demands.
		Prevents reverse flow of RHRP B-B discharge through RHRP A-A (injection & recirc. mode)	Stuck open or fails to backseat	Mechanical failure	Pump motor amps above full load; discharge pressure high on idle pump	See 'Remarks' column	None. RHRP B-B can provide required low head flow	Since the failure of the check valve is the single failure postulated, both RHR Pumps A-A & B-B can be assumed to operate. The new check valve 74-544 also prevents reverse flow.
53	Check Valve 74-515	Opens to provide flow path for RHRP B-B discharge (injection &	Stuck closed	Mechanical failure	Pump motor amps less than full load; low RHRP B-B flow, alarm from FIS-74-	loss of redundancy in LP-SI	None. RHRP A-A can provide required low head flow	Failure mode is credible; Per IEEE Std. 500-1984, check valves at PWRs have a failure rate (fail to open) of 60 per million demands.

FAILURE MODES AND EFFECTS ANALYSIS FOR ACTIVE FAILURES FOR THE SAFETY INJECTION SYSTEM

Item No.	Component	Function	Failure Mode	Potential Cause	Method of Detection	Effect on System	Effect on Plant	Remarks
		recirc. modes)			24			
		Prevents reverse flow of RHRP A-A discharge through RHRP B-B (injection & recirc. mode)	Stuck open or fails to backseat	Mechanical failure	Pump motor amps above full load; discharge pressure high on idle pump	See 'Remarks' column	None. RHRP A-A can provide required low head flow	Since the failure of the check valve is the single failure postulated, both RHR Pumps A-A & B-B can be assumed to operate. The new check valve 74-545 also prevents reverse flow.
54	Check Valve 74-544	Opens to provide flow path for RHRP A-A discharge (injection & recirc. modes)	Stuck closed	Mechanical failure	Pump motor amps less than full load; low RHRP A-A flow, alarm from FIS-74-12	loss of redundancy in LP-SI	None. RHRP B-B can provide required low head flow	Per IEEE Std. 500-1984, check valves at PWRs have a failure rate (fail to open) of 60 per million demands.
		Prevents miniflow flow of RHRP B-B discharge through RHRP A-A miniflow (injection mode) resulting in inadequate	Stuck open or fails to backseat	Mechanical failure	Pump motor amps above full load; damage to RHRP A-A	RHR pump A-A could have inadequate miniflow & possible damage to RHRP A-A	None. RHRP B-B available.	

FAILURE MODES AND EFFECTS ANALYSIS FOR ACTIVE FAILURES FOR THE SAFETY INJECTION SYSTEM

Item No.	Component	Function	Failure Mode	Potential Cause	Method of Detection	Effect on System	Effect on Plant	Remarks
		RHRP A-A miniflow.						
55	Check Valve 74-545	Opens to provide flow path for RHRP B-B discharge (injection & recirc. modes)	Stuck closed	Mechanical failure	Pump motor amps less than full load; low RHRP B-B flow, alarm from FIS-74-24	Loss of redundancy in LP-SI	None. RHRP A-A can provide required low head flow	Per IEEE Std. 500-1984, check valves at PWRs have a failure rate (fail to open) of 60 per million demands.
		Prevents miniflow of RHRP A-A discharge through RHRP B-B miniflow (injection mode) resulting in inadequate RHRP B-B miniflow	Stuck open of fails to backseat	Mechanical failure	Pump motor amps above full load; damage to RHRP B-B	RHR pump B-B could have inadequate miniflow & possible damage to RHRP B-B	None. RHRP A-A available.	
56	FCV-63-93 Train A	Provide flow path for RHRP discharge to	Spurious closing (not a credible failure mode). See	See 'Remarks' column	Alarm and indicating light in MCR; possible low	See 'Remarks' column	See 'Remarks' column	Procedures do not require this normally open valve to be closed until manual switchover

FAILURE MODES AND EFFECTS ANALYSIS FOR ACTIVE FAILURES FOR THE SAFETY INJECTION SYSTEM

Item No.	Component	Function	Failure Mode	Potential Cause	Method of Detection	Effect on System	Effect on Plant	Remarks
		cold legs 2 and 3 (injection and CL recirc. modes)	'Remarks' column		flow alarm from FIS-74-12; low flow indicating on FI-63-91 A/B			to HL recirc. mode. Probability of spurious closing is reduced by protective cover over HS and by wiring HS & XS contacts on both sides of contactors. This failure mode is, therefore, not credible (potential cause). However, if it did occur, ECCS LP injection to loops 2 & 3 would be lost (effect on system). LP injection flow may be inadequate (effect on plant).
		Isolate RHRP A-A discharge from cold legs 2 and 3 to direct flow directly to hot legs 1 and 3.	Fails to close, stuck open or spuriously reopens	Mechanical failure; Train A power failure; operator error	Indicating light in MCR	No redundancy in LP portion of ECCS if failure due to Train A power failure. RHRP A-A can continue to provide LP-SI flow to CL 2/3 and split flow with HL 1/3 if failure not due to	None. RHRS Train B can provide adequate flow to suction of SIPs and CCPs. The SIPs can provide HL recirc. flow	Suction path to both SIPs and both CCPs can be established from RHRP B-B discharge by opening Train B valves and through normally open train A valves.

FAILURE MODES AND EFFECTS ANALYSIS FOR ACTIVE FAILURES FOR THE SAFETY INJECTION SYSTEM

Item No.	Component	Function	Failure Mode	Potential Cause	Method of Detection	Effect on System	Effect on Plant	Remarks
						Train A power failure.		
		Isolated for passive failure.	No further failures assumed					
57	FCV-63-94 Train B	Provide flow path for RHRP discharge to cold legs 1 and 4 (injection and CL recirc. modes)	Spurious closing (not a credible failure mode). See 'Remarks' column	See 'Remarks' column	Alarm and indicating light in MCR; possible low flow alarm from FIS-74-24; low flow indicating. on FI-63-92 A/B	See 'Remarks' column	See 'Remarks' column	Procedures do not require this normally open valve to be closed until manual switchover to HL recirc. mode. Probability of spurious closing is reduced by protective cover over HS and by wiring HS & XS contacts on both sides of contactors. This failure mode is, therefore, not credible (potential cause). However, if it did occur, ECCS LP injection to loops 1 & 4 would be lost (effect on system). LP injection flow may be inadequate (effect on plant).
		Isolate RHRP B-B discharge	Fails to close, stuck open or	Mechanical failure; Train B	Indicating light in MCR	No redundancy in LP portion of	None. RHRS Train A can	Suction path to both SIPs and both CCPs

FAILURE MODES AND EFFECTS ANALYSIS FOR ACTIVE FAILURES FOR THE SAFETY INJECTION SYSTEM

Item No.	Component	Function	Failure Mode	Potential Cause	Method of Detection	Effect on System	Effect on Plant	Remarks
		from cold legs 1 and 4 to direct flow directly to hot legs 1 and 3. Isolated for passive failure.	spuriously reopens No further failures assumed	power failure; operator error		ECCS if failure due to Train B power failure. RHRP B-B can continue to provide LP-SI flow to CL 1/4 and split flow with HL 1/3 if failure not due to Train A power failure.	provide adequate flow to suction of SIPs and CCPs. The SIPs can provide HL recirc. flow	can be established from RHRP A-A discharge by opening Train A valves and through normally open train B valves.
58	FCV-63-6 Train B	Provide recirc. mode flow path interconnectin g suction of SIPs and CCPs when RHRP discharge is aligned to suction side of CCPs/SIPs (in parallel with Train A FCV-	Fails to open, stuck closed or spuriously recloses	Mechanical failure; Train B power failure; operator error	Indicating light in MCR	None. Independent Train A valve 63-7 can be opened to ensure at least one suction flow path to both SIPs and both CCPs	None	Valve opened by operator during switchover to recirc. If failure is due to Train B power failure, RHRP A-A can provide recirc. flow and suction to SIPs and CCPs by opening of Train A valves and normally open (fail-as-is) Train B valves.

FAILURE MODES AND EFFECTS ANALYSIS FOR ACTIVE FAILURES FOR THE SAFETY INJECTION SYSTEM

Item No.	Component	Function	Failure Mode	Potential Cause	Method of Detection	Effect on System	Effect on Plant	Remarks
		63-7) (recirc. mode)						
			Spurious opening	Operator error; hot short in control wiring	Indicating light in MCR	None	None	Valves 63-7 and 63-6 are closed during reactor operation and injection mode of SI; If valve is spuriously opened during injection mode, suction headers of CCPs and SIPs are connected through 63-6 and 63-177 (normally open). The headers are already both connected to RWST.
59	FCV-63-7 Train A	Provide recirc. mode flow path suction of SIPs and interconnecting CCPs when RHRP discharge is aligned to suction side of CCPs/SIPs (in parallel with Train B FCV-63-6) (recirc.	Fails to open, stuck closed or spuriously recloses	Mechanical failure; Train A power failure; operator error	Indicating light in MCR	None. Independent Train B valve 63-6 can be opened to ensure at least one suction flow path to both SIPs and both CCPs	None	Valve opened by operator during switchover to recirc. If failure is due to Train A power failure, RHRP B-B can provide recirc. flow and suction to SIPs and CCPs by opening of Train B valves and normally open (fail-as-is) Train A valves.

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FAILURE MODES AND EFFECTS ANALYSIS FOR ACTIVE FAILURES FOR THE SAFETY INJECTION SYSTEM

Item No.	Component	Function	Failure Mode	Potential Cause	Method of Detection	Effect on System	Effect on Plant	Remarks
		mode)						
	FCV-63-7 Train A		Spurious opening	Operator error, hot short in control wiring	Indicating light in MCR	None	None	Valves 63-7 and 63-6 are closed during reactor operation and injection mode of SI; If valve is spuriously opened during injection mode, suction headers of CCPs and SIPs are connected through 63-7 and 63-177 (normally open). The headers are already both connected to RWST.
60	FCV-63-177 Train A	Normally open valve providing flow path interconnecting suction of SIPs and CCPs from RHRP discharge (recirc. mode)	Spurious closing	Operator error, hot short in control wiring	Indicating light in MCR	No interconnecting flow path to suction of CCPs and SIPs	None	Suction flow path to both CCPs is established by opening valve 63-8, to SIPs by opening valve 63-11 (SIP B-B directly and SIP A-A through normally open valves 63-48 and 63-47)
61	FCV-63-118 Train A	Provide flow path from Accumulator	No credible failure mode. See 'Remarks'	See 'Remarks' column	See 'Remarks' column	See 'Remarks column	See 'Remarks' column	Normally open valve (open in reactor operation as well as

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FAILURE MODES AND EFFECTS ANALYSIS FOR ACTIVE FAILURES FOR THE SAFETY INJECTION SYSTEM

Item No.	Component	Function	Failure Mode	Potential Cause	Method of Detection	Effect on System	Effect on Plant	Remarks
		Tank #1 to cold leg 1	column					injection and recirc. modes of SI) has no credible failure mode because: 1) permissive interlock in close circuit opens on Train A SI signal, 2) Train A SI signal and RCS pressure >1970 psig signal are used in the opening circuit, 3) protection against spurious operation due to hot short in control wiring is provided by using selector and hand switch contacts on line and neutral side of contactors and 4) operator error unlikely due to protective cover over HS (failure mode and potential cause). Breaker is locked opened to prevent spurious operation.
62	FCV-63-98 Train B	Provide flow path from Accumulator	No credible failure mode. See ‘Remarks’	See ‘Remarks’ column	See ‘Remarks’ column	See ‘Remarks’ column	See ‘Remarks’ column	Normally open valve (open in reactor operation as well as

FAILURE MODES AND EFFECTS ANALYSIS FOR ACTIVE FAILURES FOR THE SAFETY INJECTION SYSTEM

Item No.	Component	Function	Failure Mode	Potential Cause	Method of Detection	Effect on System	Effect on Plant	Remarks
		Tank #2 to cold leg 2	column					injection and recirc. Modes of SI) has no credible failure mode because: 1) permissive interlock in close circuit opens on Train B SI signal, 2) Train B SI signal and RCS pressure >1970 psig signal are used in the opening circuit, 3) protection against spurious operation due to hot short in control wiring is provided by using selector and hand switch contacts on line and neutral side of contractors and 4) operator error unlikely due to protective cover over HS (failure mode and potential cause). Breaker is locked opened to prevent spurious operation.
63	FCV-63-80 Train A	Provide flow path from Accumulator	No credible failure mode. See ‘Remarks’	See ‘Remarks’ column	See ‘Remarks’ column	See ‘Remarks’ column	See ‘Remarks’ column	Normally open valve (open in reactor operation as well as

FAILURE MODES AND EFFECTS ANALYSIS FOR ACTIVE FAILURES FOR THE SAFETY INJECTION SYSTEM

Item No.	Component	Function	Failure Mode	Potential Cause	Method of Detection	Effect on System	Effect on Plant	Remarks
		Tank #3 to cold leg 3	column					injection and recirc. Modes of SI) has no credible failure mode because: 1) permissive interlock in close circuit opens on Train A SI signal, 2) Train A SI signal and RCS pressure >1970 psig signal are used in the opening circuit, 3) protection against spurious operation due to hot short in control wiring is provided by using selector and hand switch contacts on line and neutral side of contactors and 4) operator error unlikely due to protective cover over HS (failure mode and potential cause). Breaker is locked opened to prevent spurious operation.
64	FCV-63-67 Train B	Provide flow path from Accumulator	No credible failure mode. See ‘Remarks’	See ‘Remarks’ column	See ‘Remarks’ column	See ‘Remarks' column	See 'Remarks' column	Normally open valve (open in reactor operation as well as

FAILURE MODES AND EFFECTS ANALYSIS FOR ACTIVE FAILURES FOR THE SAFETY INJECTION SYSTEM

Item No.	Component	Function	Failure Mode	Potential Cause	Method of Detection	Effect on System	Effect on Plant	Remarks
		Tank #4 to cold leg 4	column					injection and recirc. modes of SI) has no credible failure mode because: 1) permissive interlock in close circuit opens on Train B SI signal, 2) Train B SI signal and RCS pressure >1970 psig signal are used in the opening circuit, 3) protection against spurious operation due to hot short in control wiring is provided by using selector and hand switch contacts on line and neutral side of contactors and 4) operator error unlikely due to protective cover over HS (failure mode and potential cause). Breaker is locked open to prevent spurious operation.
65	Train A Emergency Power	Provides Class 1E diesel-backed	Loss of, or inadequate, voltage	Diesel generator failure; bus fault (Train A),	Alarm and indicator in MCR	No redundancy in SI system	None. Four accumulators, CCP B-B, SIP	Compensating provisions/actions will occur/can be performed

FAILURE MODES AND EFFECTS ANALYSIS FOR ACTIVE FAILURES FOR THE SAFETY INJECTION SYSTEM

Item No.	Component	Function	Failure Mode	Potential Cause	Method of Detection	Effect on System	Effect on Plant	Remarks
		power supply to active components of Train A of SI system (injection and recirc. modes)		operator error.			B-B and RHRP B-B remain available and can provide adequate flow for postulated range of break sizes.	for Train “A” valves required to change position following a LOCA to ensure the required function is not disabled due to Train “A” power failure.
66	Train B Emergency Power	Provides Class 1E diesel-backed power supply to active components of Train B of SI system (injection and recirc. modes)	Loss of, or inadequate, voltage	Diesel generator failure; bus fault (Train B), operator error.	Alarm and indicator in MCR	No redundancy in SI system	None. Four accumulators, CCP A-A, SIP A-A and RHRP A-A remain available and can provide adequate flow for postulated range of break sizes.	Compensating provisions/actions will occur/can be performed for Train “B” valves required to change position following a LOCA to ensure the required function is not disabled due to Train “B” power failure.
67	RWST level loops connected in 2 out of 4 logic. L-63-50 L-63-51 L-63-52	Provides open signal to FCV-63-72 & FCV-63-73 in combination with containment sump level	One loop fails high	Transmitter failure; open/short circuit in loop wiring, calibrating error	One of four level indicators (LI-63-50 thru LI-63-53) reading higher than other three	Resulting logic of RWST level control for Trains A & B changes from 2 out of 4 to 2 out of 3.	None	Train A & Train B signals for ECCS switchover generated in SSPS if 2 out of 4 RWST level loops show a low level AND 2 out of 4 containment sump level loops show a high

FAILURE MODES AND EFFECTS ANALYSIS FOR ACTIVE FAILURES FOR THE SAFETY INJECTION SYSTEM

Item No.	Component	Function	Failure Mode	Potential Cause	Method of Detection	Effect on System	Effect on Plant	Remarks
	L-63-53	loops and SI signal (initiation of recirc. mode).	One loop fails low	Transmitter failure; open/short circuit in loop wiring, calibrating error	One of four level indicators (LI-63-50 thru LI-63-53) reading lower than other three, bistable energizes light on MCB.	Resulting logic of RWST level control for Trains A & B changes from 2 out of 4 to 1 out of 3 until failed channel is bypassed; then 2 out of 3.	None	level AND a SI signal is present. All components in level loops from transmitter through bistable are Class 1E. Of the four level indicators, two (LI-63-50 & 51) are Class 1E, used for PAM indication (Category 1). Single failure analysis for PAM indication is acceptable. Class 1E portions of the level loops are isolated from non-Class 1E portions (e.g., level loops III & IV indicators and level switches for Lo-Lo level alarm) by Class 1E signal conditioner modules.
68	Containment Sump level Loops connected in 2 out of 4 logic. L-63-180	Provides open signal to FCV-63-72 & FCV-63-73 in combination with RWST level loops	One loop fails high	Transmitter failure; open/short circuit in loop wiring, calibrating error	One of four level indicators (LI-63-180 thru LI-63-183) reading higher than	Resulting logic of Containment Sump level control for Trains A & B changes from 2 out of 4 to 1 out of 3 until	None	Train A & Train B signals for ECCS switchover generated in SSPS if 2 out of 4 RWST level loops show a low level AND 2 out of 4 containment sump

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FAILURE MODES AND EFFECTS ANALYSIS FOR ACTIVE FAILURES FOR THE SAFETY INJECTION SYSTEM

Item No.	Component	Function	Failure Mode	Potential Cause	Method of Detection	Effect on System	Effect on Plant	Remarks
	L-63-181 L-63-182 L-63-183	and SI signal (initiation of recirc. mode).			other three. Bistable energizes light on MCB.	failed channel is bypassed; then 2 out of 3.		level loops show a high level AND a SI signal is present. All components in level loops from transmitter through bistable are Class 1E. Of the four level indicators, two (LI-63-180 & 181) are Class 1E, used for PAM indication (Category 1). Single failure analysis for PAM indication is acceptable. Non-Class 1E indicators on level loops III & IV are isolated from Class 1E level switches by Class 1E signal conditioner modules.
			One loop fails low	Transmitter failure; open/short circuit in loop wiring, calibrating error	One of four level indicators (LI-63-180 thru LI-63-183) reading lower than other three.	Resulting logic of containment sump level control for Trains A & B changes from 2 out of 4 to 2 out of 3.	None	
69	FCV-62-1228 Train A	Isolates CCPs' normal suction hydrogen vent line (volume control tank) when RWST suction is aligned (in	Fails to close, stuck open or spurious opening after closing.	Mechanical failure, instrumentation & control failure, operator error.	Status indicating lights	Loss of redundancy in CCP normal suction hydrogen vent line	None. Train B isolation valve FCV-62-1229 provides isolation of CCP normal suction vent path	Both FCV-62-1228 and FCV-62-1229 are interlocked with LCV-62-132 and LCV-62-133, respectively, in such a way that on a SI signal the valves 62-1228 and 62-1229 will begin to close only

FAILURE MODES AND EFFECTS ANALYSIS FOR ACTIVE FAILURES FOR THE SAFETY INJECTION SYSTEM

Item No.	Component	Function	Failure Mode	Potential Cause	Method of Detection	Effect on System	Effect on Plant	Remarks
		series with LCV-62-132)						when valves 62-132 and 133 begin to close.
70	FCV-62-1229 Train B	Isolates CCPs' normal suction hydrogen vent line (volume control tank) when RWST suction is aligned (in series with LCV-62-133)	Fails to close, stuck open or spurious opening after closing.	Mechanical failure, instrumentation & control failure, operator error.	Status indicating lights	Loss of redundancy in CCP normal suction hydrogen vent line	None. Train A isolation valve FCV-62-1228 provides isolation of CCP normal suction vent path	Both FCV-62-1228 and FCV-62-1229 are interlocked with LCV-62-132 and LCV-62-133, respectively, in such a way that on a SI signal the valves 62-1228 and 62-1229 will begin to close only when valves 62-132 and 133 begin to close.
71	RFV-63-626, -627, -637, -534, -535, -536, -577	Prevent over pressure in ECCS injection lines.	Spurious lifting of relief valve at a pressure below its set point.	Mechanical failure	Alarm due to increase in PRT level and/or pressure, except for RFV-63-577. For RFV-63-577, leakage is detected as "unidentified leakage" during periodic RCS leakage	Reduce ECCS injection flow rate due to diversion of flow into relief line.	None. Flow diverted into relief line is less than worst case ECCS flow reduction due to a single failure (i.e., loss of one train of ECCS).	It is not credible to have a pre-existing condition at the time of accident where one or more relief valves have opened spuriously or are experiencing significant seat leakage. The noted "method of detection" provides an adequate means of timely identification of such a condition so that appropriate corrective maintenance can be

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FAILURE MODES AND EFFECTS ANALYSIS FOR ACTIVE FAILURES FOR THE SAFETY INJECTION SYSTEM

Item No.	Component	Function	Failure Mode	Potential Cause	Method of Detection	Effect on System	Effect on Plant	Remarks
					surveillance testing.			performed. It also noted that the RCS pressure will be less than the subject relief valve's setpoints with the exception of the small break LOCA during post-accident ECCS operation.

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UNIT 2

FAILURE MODES AND EFFECTS ANALYSIS FOR ACTIVE FAILURES FOR THE SAFETY INJECTION SYSTEM

Item No.	Component	Function	Failure Mode	Potential Cause	Method of Detection	Effect on System	Effect on Plant	Remarks
1	LCV-62-135 Train A	Opens to connect RWST to CCPs' suction (parallel to LCV-62-136) (injection mode)	Fails to open, stuck closed or spurious closing after opening	Mechanical failure; Train A power failure; Train A SI Signal failure; operator error (HS in wrong position)	Indicating light in MCR; HS position	No redundancy in RWST suction path to CCPs	None. Train B RWST suction valve LCV-62-136 allows suction flow to both CCPs	Normally closed valve opens automatically on SI Signal to align CCP suction to RWST and is manually closed along with parallel valve LCV-62-136 for sump recirculation mode after CCP suction is transferred to RHR pump discharge. Automatic operation of 62-135 on SI Signal, is completely independent of valve 62-136. (This valve has a VCT LO-LO level automatic function, which is not within the scope of this SIS FMEA.)
		Closes to isolate RWST (recirc. mode)	Fails to close or stuck open	Mechanical failure; Train A power failure; operator error	Ind. light in MCR; HS position	RWST remains connected to CCPs' suction after switchover to recirculation, however, pump discharge head from RHR is greater than head from RWST and backflow to RWST is prevented by check valve 62-504.	None	
			Spuriously opens	Operator error (HS in wrong position)	Indicating light			
2	LCV-62-136 Train B	Opens to connect RWST to	Fails to open, stuck closed or spurious	Mechanical failure; Train B power failure;	Indicating light in MCR; HS position	No redundancy in RWST suction path to CCPs	None. Train B suction valve LCV-62-	Normally closed valve opens automatically on SI Signal to align CCP

FAILURE MODES AND EFFECTS ANALYSIS FOR ACTIVE FAILURES FOR THE SAFETY INJECTION SYSTEM

Item No.	Component	Function	Failure Mode	Potential Cause	Method of Detection	Effect on System	Effect on Plant	Remarks
		CCPs' suction (parallel to LCV-62-135) (injection mode)	closing after opening	Train B SI Signal failure			136 allows RWST suction flow to both CCPs	suction to RWST and is manually closed along with parallel valve LCV-62-135 for sump recirculation mode after CCP suction is transferred to RHR pump discharge. Automatic operation of 62-136, on SI Signal, is completely independent of valve 62-135. (This valve has a VCT LO-LO level automatic function, which is not within the scope of the SIS FMEA.)
		Closes to isolate RWST (Recirc. mode)	Fails to close or stuck open	Mechanical failure; Train B power failure; operator error	Indicating light in MCR; HS position	RWST remains connected to CCPs' suction after switchover to recirculation, however, head from RHR is greater than head from RWST and backflow to RWST is	None	
			Spuriously opens	Operator error (HS in wrong position)	Indicating light			

FAILURE MODES AND EFFECTS ANALYSIS FOR ACTIVE FAILURES FOR THE SAFETY INJECTION SYSTEM

Item No.	Component	Function	Failure Mode	Potential Cause	Method of Detection	Effect on System	Effect on Plant	Remarks
						prevented by check valve 62-504.		
3	Check Valve 62-504	Provides suction flow path from RWST to CCPs (injection mode)	Stuck closed (See 'Remarks' Column)	Mechanical failure	Erratic readings on motor ammeter and pump discharge flow (FI-63-170) in control room. Suction pressure indication on PI-62-105 and 109 (local); Discharge pressure indication on PI-62-106 and 110 (local)	Pumps start on SI signal but suction flow not established. Pump damage unless operator secures pumps based on motor amp and flow readings	See 'Remarks' Column	Per IEEE Std. 500-1984, check valves at PWRs have a failure rate (fail to open) of 60 per million demands, which makes this a credible failure mode. Relative to the failure mode, the Design Criteria Document does not require consideration of this check valve as an active component for the opening function (and therefore its failure to open) during the injection mode. However, during the recirculation mode, it is an active component and its failure to close is analyzed below. The effect on the plant, if 'stuck closed' failure

FAILURE MODES AND EFFECTS ANALYSIS FOR ACTIVE FAILURES FOR THE SAFETY INJECTION SYSTEM

Item No.	Component	Function	Failure Mode	Potential Cause	Method of Detection	Effect on System	Effect on Plant	Remarks
								mode were to be considered, is no flow from charging pumps and no injection into cold legs until RCS pressure drops below discharge pressure of SI pumps.
	Check Valve 62-504	Prevents backflow from CCP suction header to RWST (Recirc. mode)	Stuck open or fails to backseat	Mechanical failure	Change in RWST level and temperature; Radiation Protection surveys of RWST area; loss of containment sump inventory	None. MOVs 62-135 and 62-136 are closed to complete switchover to recirc. mode	None.	Per IEEE Std. 500-1984, failure mode is internal leakage rather than gross failure to close
4	LCV-62-132 Train A	Isolates CCPs' normal suction source (volume control tank) when RWST suction is aligned (in	Fails to close, stuck open or spurious opening after closing	Mechanical failure, Train A power failure, Train A SI Signal failure; operator error	Indicating light in MCR	Loss of redundancy in VCT isolation	None. Train B isolation valve LCV-62-133 provides isolation of VCT suction path	Both LCV-62-132 and LCV-62-133 are interlocked with LCVs 62-135 and 62-136 in such a way that even on a SI Signal, neither 62-132 nor 62-133 will begin to close unless either 62-135 or 62-136

FAILURE MODES AND EFFECTS ANALYSIS FOR ACTIVE FAILURES FOR THE SAFETY INJECTION SYSTEM

Item No.	Component	Function	Failure Mode	Potential Cause	Method of Detection	Effect on System	Effect on Plant	Remarks
		series with LCV-62-133)						is fully open. The schematics were reviewed and it was determined that the use of non-divisional power and stem-mounted limit switches for the cross-division interlock did not introduce a common failure mode of LCVs 62-132 and 62-133. Failure to open not listed since it has no impact on safety function of System 63.
5	LCV-62-133 Train B	Isolates CCPs' normal suction source (VCT) when RWST is aligned (in series with LCV-62-132)	Fails to close, stuck open or spurious opening after closing	Mechanical failure; Train B power failure; Train B SI signal failure; operator error	Indicating light in MCR	Loss of redundancy in VCT isolation	None. Train A isolation valve LCV-62-132 provides isolation of VCT suction source	Both LCV-62-132 and LCV-62-133 are interlocked with LCVs 62-135 and 62-136 in such a way that even on a SI Signal, neither 62-132 nor 62-133 will begin to close unless either 62-135 or 62-136 is fully open. The schematics were reviewed and it was

FAILURE MODES AND EFFECTS ANALYSIS FOR ACTIVE FAILURES FOR THE SAFETY INJECTION SYSTEM

Item No.	Component	Function	Failure Mode	Potential Cause	Method of Detection	Effect on System	Effect on Plant	Remarks
								determined that the use of non-divisional power and stem-mounted limit switches for the cross-division interlock did not introduce a common failure mode of LCVs 62-132 and 62-133. Failure to open not listed since it has no impact on safety function of System 63.
6	Centrifugal Charging Pump A-A	Provides RCP seal injection and emergency core cooling by pumping into RCS cold legs, borated water from RWST during injection mode and contents of containment sump via RHR pumps during recirc.	Fails to start; fails while running	Mechanical failure; Train A power failure; Train A SI Signal failure; motor overload; electrical fault Operator error (HS in wrong position)	Motor trip or overload alarm, indicating lights in main control room, no motor amps, no pump discharge pressure on PI-62-110 (local)	Loss of redundancy in high head injection portion of SI System.	None. CC Pump B-B can provide required high head injection flow for design basis range of break sizes	Pump starts automatically on Train A SI Signal. Automatic operation of each CCP in ECCS injection mode is completely independent of other CCP.

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FAILURE MODES AND EFFECTS ANALYSIS FOR ACTIVE FAILURES FOR THE SAFETY INJECTION SYSTEM

Item No.	Component	Function	Failure Mode	Potential Cause	Method of Detection	Effect on System	Effect on Plant	Remarks
		mode.						
7	Centrifugal Charging Pump B-B	Provides RCP seal injection and emergency core cooling by pumping into RCS cold legs, borated water from RWST during injection mode and contents of containment sump via RHR pumps during recirc. mode.	Fails to start; fails while running	Mechanical failure; Train B power failure, Train B SI Signal failure; motor overload; electrical fault Operator error (HS in wrong position)	Motor trip or overload alarm, indicating lights in main control room, no motor amps, no pump discharge pressure on PI-62-106 (local)	Loss of redundancy in high head injection portion of SI System	None. CC Pump A-A can provide required high head injection flow for design basis range of break sizes.	Pump starts automatically on Train B SI Signal. Automatic operation of each CCP in ECCS injection mode is completely independent of other CCP.
8	Check Valve 62-525	Provides discharge flow path for CCP A-A (injection and recirc modes) Prevents	Stuck closed Stuck open or	Mechanical failure Mechanical	Pump motor amps less than full load; CCP total flow low as indicated on FI-63-170 in MCR Pump motor	Loss of redundancy in high head injection portion of SI System See 'Remarks'	None. CCP B-B can provide required high pressure injection flow. None. See	 Since the failure of the

FAILURE MODES AND EFFECTS ANALYSIS FOR ACTIVE FAILURES FOR THE SAFETY INJECTION SYSTEM

Item No.	Component	Function	Failure Mode	Potential Cause	Method of Detection	Effect on System	Effect on Plant	Remarks
		reverse flow of CCP B-B discharge through CCP A-A	fails to backseat	failure	amps above full load, low flow on FI-63-170 in MCR	column	'Remarks' column	check valve is the single failure postulated, both CC Pumps A-A & B-B can be assumed to operate.
9	Check Valve 62-532	Provides discharge flow path for CCP B-B (injection and recirc. modes)	Stuck closed	Mechanical failure	Pump motor amps less than full load; CCP total flow low as indicated on FI-63-170 in MCR	Loss of redundancy in high head injection portion of SI System	None. CCP A-A can provide required high pressure injection flow.	
		Prevents reverse flow of CCP A-A discharge through CCP B-B	Stuck open or fails to backseat	Mechanical failure	Pump motor amps above full load, low flow on FI-63-170 in MCR	See 'Remarks' column	None. See 'Remarks' column	Since the failure of the check valve is the single failure postulated, both CC Pumps A-A & B-B can be assumed to operate
10	Check Valve 62-523	Opens to connect CCP A-A discharge to min. flow recirc.	Stuck closed	Mechanical failure	Possible abnormal readings on pump motor ammeter; CCP A-A damage.	Min flow circuit for CCP A-A protection unavailable, with damage to CCP A-A possible.	None. CCP B-B still available	Failure mode is credible; Per IEEE Std. 500-1984, check valves at PWRs have a failure rate (fail to open) of 60 per million demands
		Prevents reverse flow	Stuck open or fails to	Mechanical failure	Pump motor amps above	See 'Remarks' column	See 'Remarks'	Since the failure of the check valve is the

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UNIT 2

FAILURE MODES AND EFFECTS ANALYSIS FOR ACTIVE FAILURES FOR THE SAFETY INJECTION SYSTEM

Item No.	Component	Function	Failure Mode	Potential Cause	Method of Detection	Effect on System	Effect on Plant	Remarks
		of CCP B-B discharge through CCP A-A	backseat		full load; discharge press. high on idle pump		column	single failure postulated, both CC pumps A-A & B-B can be assumed to operate.
11	Check Valve 62-530	Opens to connect CCP B-B discharge min. flow recirc.	Stuck closed	Mechanical failure	Possible abnormal readings on pump motor ammeter; CCP B-B damage	Min. flow circuit for CCP B-B protection unavailable, with damage to CCP B-B possible	None. CCP A-A still available	Failure mode is credible; Per IEEE Std. 500-1984, check valves at PWRs have a failure rate (fail to open) of 60 per million demands
		Prevents reverse flow of CCP A-A discharge through CCP B-B	Stuck open or fails to backseat	Mechanical failure	Pump motor amps above full load; Discharge press. high on idle pump	See 'Remarks' column	See 'Remarks' column	Since the failure of the check valve is the single failure postulated, both CC pumps A-A & B-B can be assumed to operate.
12	FCV-62-90 Train A	Isolates CCPs' normal charging path from CCP discharge header; in series with FCV-62-91	Fails to close or stuck open	Mechanical failure, Train A power failure, Train A SI Signal failure	Indicating light in MCR	Loss of redundancy in normal charging path isolation	None. Train B isolation valve FCV-62-91 provides isolation of normal charging path.	Failure to open will prevent realignment of charging after SI termination and is not listed since it has no impact on safety function of System 63.
13	FCV-62-91	Isolates	Fails to close	Mechanical	Indicating	Loss of	None. Train	Failure to open will

FAILURE MODES AND EFFECTS ANALYSIS FOR ACTIVE FAILURES FOR THE SAFETY INJECTION SYSTEM

Item No.	Component	Function	Failure Mode	Potential Cause	Method of Detection	Effect on System	Effect on Plant	Remarks
	Train B	CCPs' normal charging path from CCP discharge header; in series with FCV-62-90	or stuck open	failure, Train B power failure; Train B SI Signal failure	light in MCR	redundancy in normal charging path isolation	A isolation valve FCV-62-90 provides isolation of normal charging path	prevent realignment of charging after SI termination and is not listed since it has no impact on safety function of System 63.
14	FCV-63-1 Train A	Provide suction flow path from RWST to RHR pumps A-A & B-B (injection mode)	Spurious closing (not a credible failure mode). See 'Remarks' column	See 'Remarks' column	Alarm and indicating light in MCR	See 'Remarks' column	See 'Remarks' column	Valve is normally open administratively controlled (power off) to avoid or minimize possibility of spurious closing due to operator error. Probability of spurious operation due to hot shorts is reduced by wiring HS & XS contact on both sides of contactors. This failure mode, therefore, is extremely improbable (potential cause). However, if it did occur, RHR pumps will not be available until suction from containment sump can be established. (Effect on system) RHR pumps, if

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FAILURE MODES AND EFFECTS ANALYSIS FOR ACTIVE FAILURES FOR THE SAFETY INJECTION SYSTEM

Item No.	Component	Function	Failure Mode	Potential Cause	Method of Detection	Effect on System	Effect on Plant	Remarks
								undamaged, can be realigned for recirc. mode (Effect on plant)
		Closed to prevent backflow from RHR suction to RWST and for isolation of passive failure (recirc. mode)	Stuck open.	Mechanical failure; Train A power failure; operator error	Indicating light in MCR	RHR suction line from RWST pressurized up to RHRP A-A and B-B suction valves 74-3 and 74-21, respectively.	None. Check Valve 63-502 prevents backflow. Additional isolation is provided by FCV's 74-3 and 74-21 which automatically close when RHRP suction valves from sump, FCV-63-72 and FCV-63-73 start opening.	Normal RHR cooldown function not included in SI System FMEA
15	Check Valve 63-502	Provides suction flow path from RWST to RHRPs (Injection mode)	Stuck closed. See 'Remarks' column.	Mechanical failure	Erratic readings on RHRP motor ammeters and RHRP discharge pressure indicators PI-	RHR pumps start on SI Signal but suction flow not established. Pump damage unless operator secures pumps based on motor	See 'Remarks' column	This is a credible failure mode since, per IEEE Std. 500-1984, check valves at PWR's have a failure rate (fail to open) of 60 per million demands.

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FAILURE MODES AND EFFECTS ANALYSIS FOR ACTIVE FAILURES FOR THE SAFETY INJECTION SYSTEM

Item No.	Component	Function	Failure Mode	Potential Cause	Method of Detection	Effect on System	Effect on Plant	Remarks
					74-13 and PI-74-26 and flow indicators FI-63-91A/B and FI-63-92A/B in MCR. Local indication of suction pressure (PI-74-4 and PI-74-22) and discharge flow (FI-74-12 and FI-74-24) and discharge pressure (PI-74-6 and PI-74-18)	amp and discharge pressure readings		Relative to the failure mode, the Design Criteria Document does not require consideration of this check valve as an active component for the opening function (and therefore its failure to open) during the injection mode. However, during the recirculation mode, it is an active component and its failure to close is analyzed below. The effect on the plant, if 'Stuck Closed' failure mode were to be considered, is no injection flow from RHR pumps
		Prevents backflow from SIP suction header to RWST (recirc. mode)	Stuck open or fails to backseat	Mechanical failure	Loss of sump inventory, RWST level/temperature increase, Radiation	None. Closure of FCV-74-3/21 or FCV-63-1 isolates RWST.	None.	Normal RHR cooldown function not included in SIS FMEA. Per IEEE Std. 500-1984, failure mode is internal leakage rather than gross failure to close.

FAILURE MODES AND EFFECTS ANALYSIS FOR ACTIVE FAILURES FOR THE SAFETY INJECTION SYSTEM

Item No.	Component	Function	Failure Mode	Potential Cause	Method of Detection	Effect on System	Effect on Plant	Remarks
					Protection survey.			
16	FCV-74-3 Train A	Provides flow path to RHRP A-A from RWST (injection mode)	Stuck closed or spuriously closed	Mechanical failure; operator error	Alarm and valve position indicating light in MCR	Loss of redundancy in RHRS position of SI	None. RHRP B-B starts independently and automatically and can provide adequate injection flow.	Normally open valve; closes automatically at switchover to recirc. mode when FCV-63-72 starts to open.
		Isolates RHRP A-A suction from RWST and provides passive failure isolation (recirc. mode)	Fails to close or stuck open No failure assumed if closure is required for isolation of a passive failure.	Mechanical failure; Train A power failure; failure of close signal from FCV-63-72 limit switch	Valve position indication light in MCR	Operator stops RHRP A-A to terminate RWST outflow. Otherwise RWST inventory may be diverted to the sump via FCVs 63-1, 74-3 and 63-72.	None. RHRP B-B aligned automatically to sump, provides adequate recirc. flow.	If RWST and sump become connected, FCV-63-1 can be remote manually closed to isolate RWST (unless failure is due to Train A power failure)
17	FCV-74-21 Train B	Provides flow path to RHRP B-B from RWST (injection mode)	Stuck closed or spuriously closed	Mechanical failure; operator error	Alarm and valve position ind. light in MCR	Loss of redundancy in RHRS portion of SI	None. RHRP A-A starts independently and automatically and can provide	Normally open valve; closes automatically at switchover to recirc. mode when FCV-63-73 starts to open.

FAILURE MODES AND EFFECTS ANALYSIS FOR ACTIVE FAILURES FOR THE SAFETY INJECTION SYSTEM

Item No.	Component	Function	Failure Mode	Potential Cause	Method of Detection	Effect on System	Effect on Plant	Remarks
							adequate injection flow.	
		Isolates RHRP B-B suction from RWST and provides passive failure isolation. (recirc. mode)	Fails to close or stuck open No failure assumed if closure required for isolation of a passive failure.	Mechanical failure; Train B power failure; failure of close signal from FCV-63-73 limit switch	Valve position indication light in MCR	Operator stops RHRP B-B to terminate RWST outflow. Otherwise RWST inventory may be diverted to the sump via FCV's 74-21 and 63-73.	None. RHRP A-A aligned automatically to sump, provides adequate recirc. flow.	If RWST and sump become connected, FCV-63-1 can be remote manually closed to isolate RWST.
18	FCV-63-72 Train A	Provides suction flow path from containment sump to RHRP A-A to initiate recirc. mode of ECCS	Fails to open or stuck closed	Mechanical failure; Train A power failure; Train A SI (latched) signal failure; Train A RWST LVL Lo/CNTMT sump LVL Hi signal failure	Alarm and valve position indicating light in MCR	RHRP A-A suction cannot be switched over from RWST to containment sump. To prevent depletion of RWST, operator may secure RHRP A-A, resulting in loss of redundancy in recirc.	None. RHRP B-B, independently and automatically aligned to sump, can provide adequate recirc. flow	With RHRP A-A secured, the FCV-63-8 suction path to CCPs will be unavailable. Alternate suction path for CCPs can be established by opening Train B valves 63-11 & 63-6 and through normally open valve 63-177. Suction path for both SIP's (B-B directly and A-A through normally open valves 63-48 and 63-47) can also be established.

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FAILURE MODES AND EFFECTS ANALYSIS FOR ACTIVE FAILURES FOR THE SAFETY INJECTION SYSTEM

Item No.	Component	Function	Failure Mode	Potential Cause	Method of Detection	Effect on System	Effect on Plant	Remarks
			Spuriously opens during injection mode.	Hot short in control wiring; operator error	Ind. light in MCR; decrease in RWST level; increase in sump level	Loss of RWST inventory until operator responds. Unintentional flooding of containment. Inadvertent premature switchover of RHRP A-A suction to sump. Loss of RHRP A-A (NPSH/ Vortexing)	None. RHR pump B-B available	
		Closes for isolation of passive failure	No additional failures assumed					
19	FCV-63-73 Train B	Provides suction flow path from containment sump to RHRP B-B to initiate recirc. mode of ECCS	Fails to open or stuck closed	Mechanical failure; Train B power failure; Train B SI (latched) signal failure; Train B RWST LVL Lo/CNTMT sump LVL Hi signal failure	Alarm and valve position indication light in MCR	RHRP B-B suction cannot be switched over from RWST to containment sump. To prevent depletion of RWST, operator may secure RHRP B-B resulting in	None. RHRP A-A, independently and automatically aligned to sump, can provide adequate recirc. flow	With RHRP B-B secured, the FCV-63-11 suction path to SIPs will be unavailable. However, alternate paths for SIP suction can be established.

FAILURE MODES AND EFFECTS ANALYSIS FOR ACTIVE FAILURES FOR THE SAFETY INJECTION SYSTEM

Item No.	Component	Function	Failure Mode	Potential Cause	Method of Detection	Effect on System	Effect on Plant	Remarks
						loss of redundancy in recirc.		
			Spuriously opens during injection mode.	Hot short in control wiring; Operator error	Indicating light in MCR; decrease in RWST level; increase in sump level.	Loss of RWST inventory until operator responds. Unintentional flooding of containment. Inadvertent switchover of RHRP B-B suction to sump. Loss of RHRP B-B (NPSH/ Vortexing).	None RHRP A-A available	
		Closes for isolation of passive failures	No additional failures assumed					
20	FCV-63-8 Train A	Opens to provide primary flow path for CCPs' suction and alternate flow path for SIPs' suction	Fails to open, stuck closed or spurious opening	Mechanical failure; Train A power failure; operator error	Valve position indicating lights in MCR	Loss of redundancy in flow paths from RHRS to suction of CCPs & SIPs.	None. Alternate suction flow path for CCPs' suction can be established from	In the alternate flow path, Train A valves 63-47 and 63-177 are normally open and do not close for recirc. alignment. Train A valve 63-7 is normally closed, but is in parallel

FAILURE MODES AND EFFECTS ANALYSIS FOR ACTIVE FAILURES FOR THE SAFETY INJECTION SYSTEM

Item No.	Component	Function	Failure Mode	Potential Cause	Method of Detection	Effect on System	Effect on Plant	Remarks
		from RHRP A-A discharge (recirc. mode)					independent RHR Train B via FCV's 63-11, 63-48, 63-47, 63-6/63-7 and 63-177	with Train B valve 63-6. The alternate flow path, therefore, can be established even if due to Train A power failure.
		Remains closed during injection mode and closes for isolation of passive failures	Spuriously opens during injection mode No additional failure assumed beyond passive failure	Hot short in control wiring; operator error	Indicating light in MCR	Suction pressure boost to CCPs	None. See 'Remarks'	Spurious operation is very unlikely due to interlocks with valves 63-72, 63-3, 63-4 & 63-175, protective covers on HS and use of double contacts of HS & XS.
21	FCV-63-11 Train B	Opens to provide primary flow path for SIPs' suction and alternate flow path for CCPs' suction from RHRP B-B discharge (recirc. mode)	Fails to open, stuck closed or spurious closing	Mechanical failure; Train B power failure; operator error	Status monitor and valve position indicating lights in MCR	Loss of redundancy in flow paths from RHRS to suction of CCPs and SIPs.	None. Alternate flow path for SIPs' suction can be established from independent RHR Train A via FCV's 63-8, 63-177, 63-6/63-7, 63-48 and 63-47	In the alternate flow path, Train B valve 63-48 is normally open. Train B valve 63-6 is normally closed, but in parallel with Train A valve 63-7. The alternate flow path, therefore, can be established even if valve 63-11 fails due to train B power failure.
		Remains closed during	Spuriously opens during	Hot short in control wiring;	Indicating light in MCR	Suction pressure boost to SIPs.	None. See 'Remarks'	Spurious operation is very unlikely due to

FAILURE MODES AND EFFECTS ANALYSIS FOR ACTIVE FAILURES FOR THE SAFETY INJECTION SYSTEM

Item No.	Component	Function	Failure Mode	Potential Cause	Method of Detection	Effect on System	Effect on Plant	Remarks
		injection mode and closes for isolation of passive failures	the injection mode No additional failure assumed beyond passive failure	operator error				interlocks with valves 63-73, 63-3, 63-4 & 63-175 protective covers on HS and use of double contacts of HS & XS
22	Permissive interlock common to FCVs 63-8 and 63-11	Permit remote manual opening of MOVs 63-8 and 63-11 only if SIP min. flow circuit valve 63-3 (Train A) or valves 63-4 and 63-175 (Train B) are closed, to prevent contamination of RWST with sump water during recirc. mode.	No credible failure mode. See 'Remarks' column	See 'Remarks' column	See 'Remarks' column	None. See 'Remarks' column	None. See 'Remarks' column	The interlock for each valve is implemented using internal limit switches from the same division valve (i.e., 63-3 in the 63-8 circuit and 63-4 and 63-175 in the 63-11 circuit), and auxiliary (separation) relay contacts from stem-mounted limit switches of opposite division valves. A review of the schematics shows that no single failure of the stem-mounted LSs or power supplies in L-10, L-11A or L-11B can prevent the opening of both valves 63-8 and

FAILURE MODES AND EFFECTS ANALYSIS FOR ACTIVE FAILURES FOR THE SAFETY INJECTION SYSTEM

Item No.	Component	Function	Failure Mode	Potential Cause	Method of Detection	Effect on System	Effect on Plant	Remarks
								63-11.
23	FCV-74-33 Train A	Closes to provide train separation in RHRS for 1) passive failure protection (cold leg recirc. mode) 2) for increased RHRS resistance to accommodate RHR A-A cold leg injection and containment spray, and 3) for increased RHRS resistance for RHR B-B HL injection	Fails to close or stuck open No additional failures assumed following a passive failure	Mechanical failure, Train A power failure; operator error.	Alarm and indicating light in MCR	RHRP A-A remains connected to cross-tie line up to Train B valves 74-35 and 63-172	None. Train separation can be achieved by closing Train B valves 74-35 and 63-172. SIP can be used for HL recirc.	Valve kept open during reactor operation and injection mode. Spurious closing during injection as a failure mode is not a problem since both RHRPs would be available.
		Opens to provide flow path for RHRP A-A discharge to	Fails to open, stuck closed or spurious closing	Mechanical failure; Train A power failure; operator error	Indicating light in MCR	RHRP A-A unavailable for hot leg recirc.	None. RHRP B-B can provide recirc. flow to HLs 1 and 3 through	

FAILURE MODES AND EFFECTS ANALYSIS FOR ACTIVE FAILURES FOR THE SAFETY INJECTION SYSTEM

Item No.	Component	Function	Failure Mode	Potential Cause	Method of Detection	Effect on System	Effect on Plant	Remarks
		hot legs 1 and 3 (HL recirc. mode)					valves 74-35 and 63-172. SIPs can also provide HL recirc.	
24	FCV-74-35 Train B	Closes to provide train separation in RHRS for 1) passive failure protection (cold leg recirc. mode) 2) for increased RHRS resistance to accommodate RHR B-B cold leg injection and containment spray, and 3) for increased RHRS resistance for RHR A-A HL injection	Fails to close or stuck open No additional failure assumed following a passive failure	Mechanical failure; Train B power failure; operator error	Alarm and indicating light in MCR	RHRP B-B remains connected to cross-tie line up to train A valve 74-33	None. Train separation can be achieved by closing Train A valve 74-33. SIP can be used for HL recirc.	Valve kept open during reactor operation and injection mode. Spurious closing during injection as a failure mode is not a problem since both RHRPs would be available
		Opens to	Fails to open,	Mechanical	Indicating	RHRP B-B	None. HL 1	

FAILURE MODES AND EFFECTS ANALYSIS FOR ACTIVE FAILURES FOR THE SAFETY INJECTION SYSTEM

Item No.	Component	Function	Failure Mode	Potential Cause	Method of Detection	Effect on System	Effect on Plant	Remarks
		provide flow path for RHRP B-B discharge to hot legs 1 and 3 (HL recirc. mode)	stuck closed or spurious closing	failure; Train B power failure; operator error	light in MCR	unavailable for hot leg recirc.	and 3 recirc. flow can be provided by opening Train A valve 74-33 & Train B 63-172. SIPs can also provide HL recirc.	
25	FCV-63-172 Train B	Opens to provide flow path for RHRP flow to hot legs 1 and 3 (HL recirc. mode)	Fails to open, stuck closed or spurious closing	Mechanical failure; Train B power failure; operator error	Indicating light in MCR	No RHRP flow to hot legs 1 and 3	None. SIP A-A can supply HLs 1 and 3, with suction flow from RHRP A-A by opening Train A valves 63-8 and 63-7	FCV-63-172 is closed except during HL recirc. mode.
		Remains closed during CL injection mode	Spuriously opens	Hot short in control wiring; operator error		Inadvertent flow to HLs 1 & 3 and reduced flow to RCS CLs	None. Both RHR pumps are available and provide sufficient flow to CLs (4) & HLs 1 & 3.	Spurious operation very unlikely due to protective cover on HS and the use of double contacts on HS & XS.
		Closes to provide	No additional failures					

FAILURE MODES AND EFFECTS ANALYSIS FOR ACTIVE FAILURES FOR THE SAFETY INJECTION SYSTEM

Item No.	Component	Function	Failure Mode	Potential Cause	Method of Detection	Effect on System	Effect on Plant	Remarks
		passive failure isolation	assumed					
26	FCV-63-5 Train B	Provide suction flow path from RWST to suction of SIPs (injection mode)	Spurious closing (not a credible failure mode). See 'Remarks' column	See 'Remarks' column	Alarm, indicating light in MCR	See 'Remarks' column	See 'Remarks' column	Procedures do not require the normally open valve to be closed until switchover to recirc. mode. HS provided with protective cover. Very unlikely error of action. Probability of spurious closing is reduced by protective cover over HS and by wiring HS & XS contacts on both sides of contactors. The failure mode is, therefore, not credible (potential cause). However, if it did occur, SIPs will not be available during injection mode. (Effect on system) Both CCPs, both RHRPs and all four accumulators provide injection flow. SIPs if undamaged, can be

FAILURE MODES AND EFFECTS ANALYSIS FOR ACTIVE FAILURES FOR THE SAFETY INJECTION SYSTEM

Item No.	Component	Function	Failure Mode	Potential Cause	Method of Detection	Effect on System	Effect on Plant	Remarks
		Isolate RWST from suction of SIPs during recirc. mode. Passive failure (recirc. mode) isolation	Fails to close or stuck open No additional failure assumed beyond the passive failure	Mechanical failure; Train B power failure	Indicating light in MCR	None. Check valve 63-510 prevents backflow from sump to RWST	None.	realigned for recirc. mode (Effect on plant)
27	Check Valve 63-510	Opens to provide suction flow path from RWST to SIPs (injection mode)	Stuck closed. See 'Remarks' column	Mechanical failure	Erratic readings on SIP motor ammeters, SIP discharge pressure indicators PI-63-19 and PI-63-150 and SIP discharge flow indicators FI-63-20 and FI-63-151, all in	SIPs unavailable	See 'Remarks' column	Per IEEE std. 500-1984, check valves at PWRs have a failure rate (fail to open) of 60 per million demands, which makes this a credible failure mode. Relative to the failure mode, the Design Criteria document does not require consideration of this check valve as an active component for the opening function

FAILURE MODES AND EFFECTS ANALYSIS FOR ACTIVE FAILURES FOR THE SAFETY INJECTION SYSTEM

Item No.	Component	Function	Failure Mode	Potential Cause	Method of Detection	Effect on System	Effect on Plant	Remarks
					MCR. Local indication of SIP suction pressure on PI-63-9 and PI-63-14			(and therefore, its failure to open) during the injection mode. However, during the recirculation mode it is an active component and its failure to close is analyzed below. The effect on the plant if "stuck closed" failure mode were to be considered, is SIPs unavailable for injection mode. Both CCPs, both RHRPs and all four accumulators provide injection flow.
		Backseats to prevent flow from sump to RWST (Recirc. mode)	Stuck open or fails to backseat	Mechanical failure	Change in RWST level and temperature. Radiation Protection surveys of RWST area	None. FCV-63-5 is closed to complete switchover to recirc. mode.	None.	Per IEEE Std. 500-1984, failure mode is internal leakage rather than gross failure to close.
28	FCV-63-47 Train A	Provide suction flow path for SIP A-A (injection	Spurious closing	Operator error, hot short in control circuit	Alarm and indicating light in MCR	SIP A-A unavailable (injection mode)	None. SIP B-B, both CCPs, both RHRPs and all four	Failure unlikely in both (injection and recirc.) modes since the valve is not required to be

FAILURE MODES AND EFFECTS ANALYSIS FOR ACTIVE FAILURES FOR THE SAFETY INJECTION SYSTEM

Item No.	Component	Function	Failure Mode	Potential Cause	Method of Detection	Effect on System	Effect on Plant	Remarks
		mode)					accumulators remain available to provide adequate injection flow for all break sizes.	closed except for isolation of passive failures. Failure of valve to close is not listed as a failure mode since the passive failure requiring its operation is the single failure.
		Provides connection to recirc. flow path between suction of SIPs and CCPs (Recirc. mode) Closes for isolation of passive failures (see 'Remarks' column)	Spurious closing	Operator error, hot short in control circuit	Alarm and indicating light in MCR		None. All pumps remain available, due to flow path from RHRP A-A to SIP A-A & CCPs (63-8, 63-177, and 63-6 or 63-7) and from RHRP B-B to SIP B-B (63-11)	
29	FCV-63-48 Train B	Provide suction flow path for SIP B-B (injection mode)	Spurious closing	Operator error; hot short in control circuit	Alarm and indicating light in MCR	SIP B-B unavailable (injection mode)	None. SIP A-A, both CCPs, both RHRPs, and all four accumulators remain	Failure unlikely in both (injection and recirc.) modes since the valve is not required to be closed except for isolation of passive

FAILURE MODES AND EFFECTS ANALYSIS FOR ACTIVE FAILURES FOR THE SAFETY INJECTION SYSTEM

Item No.	Component	Function	Failure Mode	Potential Cause	Method of Detection	Effect on System	Effect on Plant	Remarks
							available to provide adequate injection flow for all break sizes.	failures. Failure of valve to close is not listed as a failure mode since the passive failure requiring its operation is the single failure.
		Provides connection for recirc. flow path between suction of SIPs and CCPs (Recirc. mode) Closes for isolation of passive failures (See 'Remarks' column)	Spurious closing	Operator error; hot short in control circuit	Alarm and indicating light in MCR	No redundant suction flow path to RHRP A-A to SIP B-B or from RHRP B-B to SIP A-A and CCPs	None. All pumps remain available, due to flow path from RHRP A-A to SIP A-A and CCPs (63-8, 63-177, 63-6/63-7) and from RHRP B-B to SIP B-B (63-11)	
30	Safety Injection Pump A-A	Provides emergency core cooling by pumping into RCS cold legs, borated water from RWST during	Fails to start; fails while running	Mechanical failure; Train A power failure, Train A SI signal failure; motor overload; electrical fault; operator error	Annunciation, and indicating lights in main control room, no motor amps, no header flow,	Loss of redundancy in intermediate head portion of SI System	None. SI Pump B-B can provide required intermediate head injection flow for design basis	SIPs start automatically on SI signal. Automatic operation of each SIP in ECCS injection mode is completely independent of the SIP.

FAILURE MODES AND EFFECTS ANALYSIS FOR ACTIVE FAILURES FOR THE SAFETY INJECTION SYSTEM

Item No.	Component	Function	Failure Mode	Potential Cause	Method of Detection	Effect on System	Effect on Plant	Remarks
		injection mode and contents of containment sump via RHR pumps during recirc. mode.		(HS in wrong position)	no pump discharge pressure		range of break sizes.	
31	Safety Injection Pump B-B	Provides emergency core cooling by pumping into RCS cold legs, borated water from RWST during injection mode and contents of containment sump via RHR pumps during recirc. mode.	Fails to start; fails while running	Mechanical failure; Train B power failure, Train B SI signal failure; motor overload; electrical fault; operator error (HS in wrong position)	Annunciation and indicating lights in main control room, no motor amps, no header flow, no pump discharge pressure	Loss of redundancy in intermediate head portion of SI System	None. SI Pump A-A can provide required intermediate head injection flow for design basis range of break sizes.	SI pumps start automatically on SI signal. Automatic operation of each SIP in ECCS injection mode is completely independent of other SIP.
32	FCV-63-3 Train A	Provide min. flow recirc. path from SIP A-A & B-B to RWST for	Spurious closing (not a credible failure mode). See 'Remarks'	See 'Remarks' column	Alarm and indicating light in MCR	See 'Remarks' column	See 'Remarks' column	Procedures do not require closing this valve until switchover to CL recirc. mode. Probability of spurious

FAILURE MODES AND EFFECTS ANALYSIS FOR ACTIVE FAILURES FOR THE SAFETY INJECTION SYSTEM

Item No.	Component	Function	Failure Mode	Potential Cause	Method of Detection	Effect on System	Effect on Plant	Remarks
		pump protection (injection mode)	column					closing is reduced by protective cover over HS and by wiring HS & XS contacts on both sides of contactors. This failure mode is, therefore, not credible (potential cause). However, if it did occur, the min. flow circuit for both SIPs will be unavailable, with damage to SIPs possible for small break LOCA and slow RCS depressurization (effect on system). Potential loss of intermediate head SI (effect on plant).
		Isolate SIP A-A & B-B miniflow to RWST when SIP suction is from containment sump via RHRPs, to	Fails to close, stuck open or spurious reopening	Mechanical failure; Train A power failure; operator error	Indicating light in MCR	None. Operator can isolate RWST by closing Train B valves 63-4 and 63-175 and then open valves 63-8 or 63-11 to complete	None	Schematics for valves 63-8 and 63-11 were reviewed and it was determined that at least one suction path to both SIPs and both CCPs can be established with a Train A power failure and

FAILURE MODES AND EFFECTS ANALYSIS FOR ACTIVE FAILURES FOR THE SAFETY INJECTION SYSTEM

Item No.	Component	Function	Failure Mode	Potential Cause	Method of Detection	Effect on System	Effect on Plant	Remarks
		prevent contamination of RWST (recirc. mode)				switchover of CCP and SIP suction.		failure of non-divisional power supply to panel L-10 or failure of any stem mounted-limit switch.
33	FCV-63-4 Train B	Connect SIP A-A discharge to min. flow recirc. line (injection mode)	Spurious closing	Operator error; hot short in control wiring	Alarm and indicating light in MCR	Min. flow circuit for SIP A-A protection unavailable, with damage to SIP A-A possible for small break LOCA and slow RCS depressurization	None. SIP B-B still available	Failure mode very unlikely since procedures do not require closing this valve until switchover to CL recirc. mode.
		Isolate SIP A-A miniflow to RWST when SIP suction is from containment sump via RHRPs, to prevent contamination of RWST (recirc. mode)	Fails to close, stuck open or spuriously reopens	Mechanical failure; Train B power failure; operator error	Indicating light in MCR	None. Operator can isolate RWST by closing Train A valve 63-3 and then open 63-8 or 63-11 to complete switchover of CCP and SIP suction to sump	None	Schematics for valves 63-8 and 63-11 were reviewed to determine that at least one suction path to both SIPs and both CCPs can be established with a Train B power failure and failure of non-divisional power to panel L-10 or failure of any stem-mounted limit switch.
34	FCV-63-175	Connect SIP	Spurious	Operator error;	Alarm and	Min. flow circuit	None. SIP A-	Failure mode very

FAILURE MODES AND EFFECTS ANALYSIS FOR ACTIVE FAILURES FOR THE SAFETY INJECTION SYSTEM

Item No.	Component	Function	Failure Mode	Potential Cause	Method of Detection	Effect on System	Effect on Plant	Remarks
	Train B	B-B discharge to min. flow recirc. line (injection mode)	closing	hot short in control wiring	indicating light in MCR	for SIP B-B protection unavailable, with damage to SIP B-B possible for small break LOCA and slow RCS depressurization	A still available	unlikely since procedures do not require closing this valve until switchover to CL recirc. mode.
		Isolate SIP B-B discharge from min flow recirc. line when SIP B-B suction is from containment sump via RHRPs, to prevent contamination of RWST (recirc. mode)	Fails to close or stuck open or spuriously reopens	Mechanical failure; Train B power failure; operator error	Indicating light in MCR	None. Operator can isolate RWST by closing train A valve 63-3 and then open 63-8 or -11 to complete switchover of CCP and SIP suction to sump.	None.	Schematics for valves 63-8 and 63-11 were reviewed to determine that at least one suction path to both SIPs and both CCPs can be established with a Train B power failure and failure of non-divisional power to panel L-10 or failure of any stem-mounted limit switch.
35	Check valve 63-524	Provides discharge flow path for SIP A-A	Stuck closed	Mechanical failure	Pump motor amps less than full load; no SIP A-A flow as indicated on	Loss of redundancy in intermediate head injection portion of SI System	None. SIP B-B can provide required injection flow.	Failure mode is credible; Per IEEE Std. 500-1984, check valves at PWRs have a failure rate (fail to open) of 60 per million demands.

FAILURE MODES AND EFFECTS ANALYSIS FOR ACTIVE FAILURES FOR THE SAFETY INJECTION SYSTEM

Item No.	Component	Function	Failure Mode	Potential Cause	Method of Detection	Effect on System	Effect on Plant	Remarks
					FI-63-151 in MCR			
		Prevents reverse flow of SIP B-B discharge through SIP A-A (injection mode)	Stuck open or fails to backseat	Mechanical Failure	Pump motor amps above full load, discharge pressure high on idle pump	See 'Remarks' column	See 'Remarks' column	Since the failure of the check valve is the single failure postulated, both SI Pumps A-A & B-B can be assumed to operate.
36	Check Valve 63-526	Provides discharge flow path for SIP B-B	Stuck closed	Mechanical failure	Pump motor amps less than full load; No SIP B-B flow as indicated on FI-63-20 in MCR	Loss of redundancy in intermediate head injection portion of SI system	None. SIP A-A can provide required injection flow.	Failure mode is credible; Per IEEE Std. 500-1984, check valves at PWRs have a failure rate (fail to open) of 60 per million demands.
		Prevents reverse flow of SIP A-A discharge through SIP B-B (injection mode)	Stuck open or fails to backseat	Mechanical failure	Pump motor amps above full load. Discharge pressure high on idle pump	See 'Remarks' column	See 'Remarks' column	Since the failure of the check valve is the single failure postulated, both SI Pumps A-A & B-B can be assumed to operate.
37	Check Valve 63-528	Opens to connect SIP A-A discharge to min. flow	Stuck closed	Mechanical failure	Possible abnormal readings on pump motor	Min. flow circuit for SIP A-A protection unavailable, with	None. SIP B-B still available	Failure mode is credible; Per IEEE Std. 500-1984, check valves at PWRs have a failure

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FAILURE MODES AND EFFECTS ANALYSIS FOR ACTIVE FAILURES FOR THE SAFETY INJECTION SYSTEM

Item No.	Component	Function	Failure Mode	Potential Cause	Method of Detection	Effect on System	Effect on Plant	Remarks
		recirc. (injection mode)			ammeter; low flow on FI-63-2 local. SIP A-A damage.	damage to SIP A-A possible for small break LOCA and slow RCS depressuirzation		rate (fail to open) of 60 per million demands
		Prevents reverse flow of SIP B-B discharge through SIP A-A (injection mode)	Stuck open or fails to backseat	Mechanical failure	Pump motor amps above full load. Discharge pressure high on idle pump	See 'Remarks' column	See 'Remarks' column	Since the failure of the check valve is the single failure postulated, both SI Pumps A-A & B-B can be assumed to operate.
38	Check Valve 63-530	Opens to connect SIP B-B discharge min. flow recirc. (injection mode)	Stuck closed	Mechanical failure	Possible abnormal readings on pump motor ammeter; low flow on FI-63-2 local. SIP B-B damage.	Min. flow circuit for SIP B-B protection unavailable, with damage to SIP B-B possible for small break LOCA and slow RCS depressuirzation	None. SIP A-A still available	Failure mode is credible; Per IEEE Std. 500-1984, check valves at PWRs have a failure rate (fail to open) of 60 per million demands
		Prevents reverse flow of SIP A-A discharge through SIP	Stuck open or fails to backseat	Mechanical failure	Pump motor amps above full load; discharge pressure	See 'Remarks' column	See 'Remarks' column	Since the failure of the check valve is the single failure postulated, both SI pumps A-A & B-B can

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FAILURE MODES AND EFFECTS ANALYSIS FOR ACTIVE FAILURES FOR THE SAFETY INJECTION SYSTEM

Item No.	Component	Function	Failure Mode	Potential Cause	Method of Detection	Effect on System	Effect on Plant	Remarks
		B-B (injection mode)			high on idle pump			be assumed to operate.
39	FCV-63-39 Train A	Normally locked open valve provides flow path (parallel with FCV-63-40) for discharge flow of CCPs to cold legs (injection and recirc. modes)	Spurious closing; See 'Remarks' column.	Operator error; See 'Remarks' column.	None	None. Train B valve 63-40 maintains flow path	None	Operator error is not credible since valve is locked open and procedures do not require closing this Valve under any operating scenario. Failure to close is not listed since closing is not a SIS safety function. Breaker is locked opened to prevent spurious operation.
40	FCV-63-40 Train B	Normally locked open valve provides flow path (in parallel with FCV-63-39) for discharge flow of CCPs	Spurious closing; See 'Remarks' column.	Operator errors; See 'Remarks' column.	None	None. Train A valve 63-39 maintains flow path	None	Operator error is not credible since valve is locked open and procedures do not require closing this valve under any operating scenario.

FAILURE MODES AND EFFECTS ANALYSIS FOR ACTIVE FAILURES FOR THE SAFETY INJECTION SYSTEM

Item No.	Component	Function	Failure Mode	Potential Cause	Method of Detection	Effect on System	Effect on Plant	Remarks
		to cold legs (injection and recirc. modes)						Failure to close is not listed since closing is not a SIS safety function. Breaker is locked open to prevent spurious operation.
41	FCV-63-25 Train B	Provide flow path (in parallel with FCV-63-26) for discharge flow of CCPs to cold legs (injection and recirc. modes) Closes to isolate passive failures or terminate CL injection.	Fails to open, stuck closed or spuriously recloses after opening	Mechanical failure; Train B power failure; Train B SI signal failure; operator error	Alarm and indicating light in MCR	None. FCV-63-26 independently and automatically opens to establish flow path.	None See 'Remarks column	Failure to close not listed as a failure mode since the passive failure requiring its operation is the single failure. SI termination of CCP cold leg injection can be accomplished by local valve operations if necessary
42	FCV-63-26	Provide flow path (in	Fails to open, stuck closed or	Mechanical failure; Train A	Alarm and indicating	None. FCV-63-25 independently	None	Failure to close not listed as a failure mode

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FAILURE MODES AND EFFECTS ANALYSIS FOR ACTIVE FAILURES FOR THE SAFETY INJECTION SYSTEM

Item No.	Component	Function	Failure Mode	Potential Cause	Method of Detection	Effect on System	Effect on Plant	Remarks
	Train A	parallel with FCV-63-25) for discharge flow of CCPs to cold legs (injection and recirc. modes) Closes to isolate passive failures or terminate CL injection.	spuriously recloses after opening	power failure; Train A SI signal failure; operator error	light in MCR	and automatically opens to establish flow path.	See 'Remarks column	since the passive failure requiring its operation is the single failure. SI termination of CCP cold leg injection can be accomplished by local valve operations if necessary
43	FCV-63-152 Train A	Provide flow path from SIP A-A discharge to cold legs (injection and CL recirc. modes)	Spurious closing	Operator error; hot short in control wiring	Alarm and indicating light in MCR, motor amperes less than full load, low flow indication on FI-63-151	No redundancy in SIP portion of cold leg SIS	None. SIP B-B through Train B valve 63-153 provides adequate CL flow.	Normally open valve; procedures do not require closing until switchover to HL recirc. for train isolation
		Closes to provide train isolation for HL recirc mode	Fails to close; stuck open or spuriously reopens after closing	Mechanical failure; Train A power failure; hot short in control wiring; operator error	Indicating light in MCR. Abnormal flow from SIP A-A	Inability to switch SIP A-A to HL recirc.	None. SIP B-B with Train B valve 63-153 isolated provides adequate	RHR can also provide HL recirc.

FAILURE MODES AND EFFECTS ANALYSIS FOR ACTIVE FAILURES FOR THE SAFETY INJECTION SYSTEM

Item No.	Component	Function	Failure Mode	Potential Cause	Method of Detection	Effect on System	Effect on Plant	Remarks
							recirc. flow.	
44	FCV-63-153 Train B	Provide flow path from SIP B-B discharge to cold legs (injection and CL recirc. modes)	Spurious closing	Operator error; hot short in control wiring	Alarm and indicating light in MCR, motor amperes less than full load; low flow ind. on FI-63-20	No redundancy in SIP portion of cold leg SIS	None. SIP A-A through Train A valve 63-152 provides adequate CL flow.	Normally open valve, procedures do not require closing until switchover to HL recirc., for train isolation
		Provide train isolation for HL recirc. mode	Fails to close, stuck open or spuriously reopens after closing	Mechanical failure; Train B power failure; hot short in control wiring; operator error	Indicating light in MCR. Abnormal flow from SIP B-B	Inability to switch SIP B-B to HL recirc.	None. SIP A-A with Train A valve 63-152 isolated provides adequate recirc. flow.	RHR can also provide HL recirc.
45	FCV-63-22 Train B	Provide flow path from discharge of SIPs A-A & B-B to cold legs (injection and CL recirc. modes)	Spurious closing (not a credible failure mode). See 'Remarks' column	See 'Remarks' column	Alarm and indicating light in MCR; no flow indication on FI-63-151 and FI-63-20	See 'Remarks' column	See 'Remarks' column	Valve is normally open, administratively controlled (power off) to minimize possibility of spurious closing by operator. Also, HS has a protective cover. Hot short in control wiring unlikely to cause spurious operation since control and selector switch contacts are wired on both sides

FAILURE MODES AND EFFECTS ANALYSIS FOR ACTIVE FAILURES FOR THE SAFETY INJECTION SYSTEM

Item No.	Component	Function	Failure Mode	Potential Cause	Method of Detection	Effect on System	Effect on Plant	Remarks
								of contactors (potential cause). If failure occurs in spite of these precautions, CL injection from SIPs will be unavailable (effect on system). Both CCPs and both RHRPs will be available and will provide injection and CL recirc. flow. For injection mode, all four accumulators are also available (effect on plant) Failure to close or stuck open failure mode is not listed since this valve, even though it will be closed for HL recirc. mode, does not have a safety related isolation function. SIP train isolation can be achieved by closing valves 63-152 and 63-153.
46	FCV-63-156 Train A	Provide flow path from SIP	Fails to open, stuck closed or	Mechanical failure, Train A	Indicating light in MCR;	No redundancy in SIP flow to hot	None. SIP B-B or one	Normally closed valve, opened only for HL

FAILURE MODES AND EFFECTS ANALYSIS FOR ACTIVE FAILURES FOR THE SAFETY INJECTION SYSTEM

Item No.	Component	Function	Failure Mode	Potential Cause	Method of Detection	Effect on System	Effect on Plant	Remarks
		A-A discharge to hot legs 1 and 3 (HL recirc. mode)	spuriously recloses	power failure; operator error; hot short in control wiring	no flow indication of FI-63-151; low motor amperes on SIP A-A	legs	RHRP can provide adequate HL recirc. flow.	recirc. mode. Spurious operation very unlikely since the HS has a protective cover and contacts of HS & XS are wired on both sides of contactor.
		Closes to isolate HL path to allow adequate CL injection flow, and for passive failures	Spuriously opens	Operator error; hot short in control wiring	Alarm and indicating light in MCR; high flow indication on FI-63-151; high motor amps on SIP A-A	Simultaneous SIP flow to CL & HL	None. SIP A-A & B-B operating will provide adequate CL injection flow	In the event of a passive failure, any further failures are not assumed.
47	FCV-63-157 Train B	Provide flow path from SIP B-B discharge to hot legs 2 and 4 (HL recirc. mode)	Fails to open, stuck closed or spuriously recloses	Mechanical failure, Train B power failure; operator error; hot short in control wiring	Indicating light in MCR; no flow indication on FI-63-20; low motor amperes on SIP B-B	No redundancy in SIP flow to hot legs	None. SIP A-A or one RHRP can provide adequate HL recirc. flow.	Normally closed valve, opened only for HL recirc. mode. Spurious operation very unlikely since HS has a protective cover and contacts of HX & XS are wired on both sides of contactor.
		Closes to isolate HL path to allow	Spuriously opens	Operator error; hot short in control wiring	Alarm and indicating light in MCR;	Simultaneous SIP flow to CL & HL	None. SIP A-A & B-B operating will	In the event of a passive failure, any further failures are not

FAILURE MODES AND EFFECTS ANALYSIS FOR ACTIVE FAILURES FOR THE SAFETY INJECTION SYSTEM

Item No.	Component	Function	Failure Mode	Potential Cause	Method of Detection	Effect on System	Effect on Plant	Remarks
		adequate CL injection flow, and for passive failures			high flow indicating on FI-63-151; high motor amps on SIP B-B		provide adequate CL injection flow	assumed.
48	RHR Pump A-A	Provide low pressure, high flow emergency core cooling by pumping into RCS cold legs, borated water from RWST during injection mode and into RCS cold and/or hot legs, contents of containment sump during recirc. mode. Also, provide suction to CCPs and	Fails to start; fails while running	Mechanical failure; Train A power failure; Train A SI signal failure; motor overload electrical fault; operator error (HS in wrong position)	Alarm and indicating light in MCR; HS position	Loss of redundancy in low pressure injection portion of SI system	None. RHRP B-B can provide required low pressure injection flow for large breaks. During recirc. mode, even if failure is due to Train A power failure, suction path from RHRP B-B to both CCPs and both SIPs can be established by opening Train B valves, and through	Automatic operation of RHRP B-B in injection and recirc. modes is completely independent of RHRP A-A. RHR spray mode and normal cooldown mode are not within the scope of this FMEA.

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Item No.	Component	Function	Failure Mode	Potential Cause	Method of Detection	Effect on System	Effect on Plant	Remarks
		SIPs during recirc. mode.					normally open Train A MOV's.	
49	RHR Pump B-B	Provide low pressure, high flow emergency core cooling by pumping into RCS cold legs, borated water from RWST during injection mode and into RCS cold and/or hot legs, contents of containment sump during recirc. mode. Also, provide suction to CCPs and SIPs during recirc. mode.	Fails to start; fails while running	Mechanical failure; Train B power failure; Train B SI signal failure; motor overload electrical fault; operator error (HS in wrong position)	Alarm and indicating light in MCR; HS position	Loss of redundancy in low pressure injection portion of SI system	None. RHRP A-A can provide required low pressure injection flow for large breaks. During recirc. mode, even if failure is due to Train B power failure, suction path from RHRP A-A to both CCPs and both SIPs can be established by opening Train A valves, and through normally open Train B MOV's.	Automatic operation of RHRP A-A in injection and recirc. modes is completely independent of RHRP B-B. RHR spray mode and normal cooldown mode are not within the scope of this FMEA.

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FAILURE MODES AND EFFECTS ANALYSIS FOR ACTIVE FAILURES FOR THE SAFETY INJECTION SYSTEM

Item No.	Component	Function	Failure Mode	Potential Cause	Method of Detection	Effect on System	Effect on Plant	Remarks
50	FCV-74-12 Train A	Opens to provide min. flow path for RHRP A-A protection below low flow setpoint (injection & recirc. modes)	Fails to open, stuck closed or spuriously closed	Mechanical failure; operator error (HS in wrong position); flow switch FS-74-12A, FS-74-12B failure; power failure	Indicating light in MCR; low flow alarm (for failure not due to flow switch failure) HS position. Local FI flow indication	Min. flow circuit for RHRP A-A unavailable, with damage to RHRP A-A possible for small or medium LOCA and slow RCS depressurization	None. RHRP B-B still available	
		Closes to isolate min. flow path above low flow setpoint (injection & recirc. modes)	Fails to close, stuck open or spuriously opened	Mechanical failure; operator error (HS in wrong position). Flow switch FS-74-12A, FS-74-12B failure; Power failure	Indicating light in MCR; HS position, local FI flow indication	Reduced LP-SI flow from RHRP A-A	None. RHRP B-B still available	
51	FCV-74-24 Train B	Opens to provide min. flow path for RHRP B-B protection below low flow setpoint (injection & recirc. modes)	Fails to open, stuck closed or spuriously closed	Mechanical failure; operator error (HS in wrong position); flow switch FS-74-24A, FS-74-24B failure; power failure.	Indicating light in MCR; low flow alarm (for failure not due to flow switch failure) HS position.	Min. flow circuit for RHRP B-B unavailable, with damage to RHRP B-B possible for small or medium LOCA and slow RCS	None. RHRP A-A still available	

FAILURE MODES AND EFFECTS ANALYSIS FOR ACTIVE FAILURES FOR THE SAFETY INJECTION SYSTEM

Item No.	Component	Function	Failure Mode	Potential Cause	Method of Detection	Effect on System	Effect on Plant	Remarks
						depressurization		
		Closes to isolate min. flow path above low flow setpoint (injection & recirc. modes)	Fails to close, stuck open or spuriously opened	Mechanical failure; operator error (HS in wrong position); flow switch FS-74-24A, FS-74-24B failure; power failure.	Indicating light in MCR; HS position; local FI flow indication	Reduced LP-SI flow from RHRP B-B	None. RHRP A-A still available	
52	Check Valve 74-514	Opens to provide flow path for RHRP A-A discharge (injection & recirc. modes)	Stuck closed	Mechanical failure	Pump motor amps less than full load; low RHRP A-A flow alarm from FS-74-12A	Loss of redundancy in LP-SI	None. RHRP B-B can provide required low head flow	Failure mode is credible; Per IEEE Std. 500-1984, check valves at PWRs have a failure rate (fail to open) of 60 per million demands.
		Prevents reverse flow of RHRP B-B discharge through RHRP A-A (injection & recirc. mode)	Stuck open or fails to backseat	Mechanical failure	Pump motor amps above full load; discharge pressure high on idle pump	See 'Remarks' column	None. RHRP B-B can provide required low head flow	Since the failure of the check valve is the single failure postulated, both RHR Pumps A-A & B-B can be assumed to operate. The new check valve 74-544 also prevents reverse flow.
53	Check Valve 74-515	Opens to provide flow path for	Stuck closed	Mechanical failure	Pump motor amps less than full load;	loss of redundancy in	None. RHRP A-A can provide	Failure mode is credible; Per IEEE Std. 500-1984, check valves

FAILURE MODES AND EFFECTS ANALYSIS FOR ACTIVE FAILURES FOR THE SAFETY INJECTION SYSTEM

Item No.	Component	Function	Failure Mode	Potential Cause	Method of Detection	Effect on System	Effect on Plant	Remarks
		RHRP B-B discharge (injection & recirc. modes)			low RHRP B-B flow, alarm from FS-74-24A	LP-SI	required low head flow	at PWRs have a failure rate (fail to open) of 60 per million demands.
		Prevents reverse flow of RHRP A-A discharge through RHRP B-B (injection & recirc. mode)	Stuck open or fails to backseat	Mechanical failure	Pump motor amps above full load; discharge pressure high on idle pump	See 'Remarks' column	None. RHRP A-A can provide required low head flow	Since the failure of the check valve is the single failure postulated, both RHR Pumps A-A & B-B can be assumed to operate. The new check valve 74-545 also prevents reverse flow.
54	Check Valve 74-544	Opens to provide flow path for RHRP A-A discharge (injection & recirc. modes)	Stuck closed	Mechanical failure	Pump motor amps less than full load; low RHRP A-A flow, alarm from FS-74-12A	loss of redundancy in LP-SI	None. RHRP B-B can provide required low head flow	Per IEEE Std. 500-1984, check valves at PWRs have a failure rate (fail to open) of 60 per million demands.
		Prevents miniflow flow of RHRP B-B discharge through RHRP A-A miniflow (injection	Stuck open or fails to backseat	Mechanical failure	Pump motor amps above full load; damage to RHRP A-A	RHR pump A-A could have inadequate miniflow & possible damage to RHRP A-A	None. RHRP B-B available.	

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UNIT 2

FAILURE MODES AND EFFECTS ANALYSIS FOR ACTIVE FAILURES FOR THE SAFETY INJECTION SYSTEM

Item No.	Component	Function	Failure Mode	Potential Cause	Method of Detection	Effect on System	Effect on Plant	Remarks
		mode) resulting in inadequate RHRP A-A miniflow.						
55	Check Valve 74-545	Opens to provide flow path for RHRP B-B discharge (injection & recirc. modes)	Stuck closed	Mechanical failure	Pump motor amps less than full load; low RHRP B-B flow, alarm from FS-74-24A	Loss of redundancy in LP-SI	None. RHRP A-A can provide required low head flow	Per IEEE Std. 500-1984, check valves at PWRs have a failure rate (fail to open) of 60 per million demands.
		Prevents miniflow of RHRP A-A discharge through RHRP B-B miniflow (injection mode) resulting in inadequate RHRP B-B miniflow	Stuck open of fails to backseat	Mechanical failure	Pump motor amps above full load; damage to RHRP B-B	RHR pump B-B could have inadequate miniflow & possible damage to RHRP B-B	None. RHRP A-A available.	
56	FCV-63-93	Provide flow	Spurious	See 'Remarks'	Alarm and	See 'Remarks'	See	Procedures do not

FAILURE MODES AND EFFECTS ANALYSIS FOR ACTIVE FAILURES FOR THE SAFETY INJECTION SYSTEM

Item No.	Component	Function	Failure Mode	Potential Cause	Method of Detection	Effect on System	Effect on Plant	Remarks
	Train A	path for RHRP discharge to cold legs 2 and 3 (injection and CL recirc. modes)	closing (not a credible failure mode). See 'Remarks' column	column	indicating light in MCR; possible low flow alarm from FS-74-12A; low flow indicating on FI-63-91 A/B	column	'Remarks' column	require this normally open valve to be closed until manual switchover to HL recirc. mode. Probability of spurious closing is reduced by protective cover over HS and by wiring HS & XS contacts on both sides of contactors. This failure mode is, therefore, not credible (potential cause). However, if it did occur, ECCS LP injection to loops 2 & 3 would be lost (effect on system). LP injection flow may be inadequate (effect on plant).
		Isolate RHRP A-A discharge from cold legs 2 and 3 to direct flow directly to hot legs 1 and 3.	Fails to close, stuck open or spuriously reopens	Mechanical failure; Train A power failure; operator error	Indicating light in MCR	No redundancy in LP portion of ECCS if failure due to Train A power failure. RHRP A-A can continue to provide LP-SI flow to CL 2/3	None. RHRS Train B can provide adequate flow to suction of SIPs and CCPs. The SIPs can provide HL	Suction path to both SIPs and both CCPs can be established from RHRP B-B discharge by opening Train B valves and through normally open train A valves.

FAILURE MODES AND EFFECTS ANALYSIS FOR ACTIVE FAILURES FOR THE SAFETY INJECTION SYSTEM

Item No.	Component	Function	Failure Mode	Potential Cause	Method of Detection	Effect on System	Effect on Plant	Remarks
		Isolated for passive failure.	No further failures assumed			and split flow with HL 1/3 if failure not due to Train A power failure.	recirc. flow	
57	FCV-63-94 Train B	Provide flow path for RHRP discharge to cold legs 1 and 4 (injection and CL recirc. modes)	Spurious closing (not a credible failure mode). See 'Remarks' column	See 'Remarks' column	Alarm and indicating light in MCR; possible low flow alarm from FS-74-24A; low flow indicating. on FI-63-92 A/B	See 'Remarks' column	See 'Remarks' column	Procedures do not require this normally open valve to be closed until manual switchover to HL recirc. mode. Probability of spurious closing is reduced by protective cover over HS and by wiring HS & XS contacts on both sides of contactors. This failure mode is, therefore, not credible (potential cause). However, if it did occur, ECCS LP injection to loops 1 & 4 would be lost (effect on system). LP injection flow may be inadequate (effect on plant).

FAILURE MODES AND EFFECTS ANALYSIS FOR ACTIVE FAILURES FOR THE SAFETY INJECTION SYSTEM

Item No.	Component	Function	Failure Mode	Potential Cause	Method of Detection	Effect on System	Effect on Plant	Remarks
		Isolate RHRP B-B discharge from cold legs 1 and 4 to direct flow directly to hot legs 1 and 3.	Fails to close, stuck open or spuriously reopens	Mechanical failure; Train B power failure; operator error	Indicating light in MCR	No redundancy in LP portion of ECCS if failure due to Train B power failure. RHRP B-B can continue to provide LP-SI flow to CL 1/4 and split flow with HL 1/3 if failure not due to Train A power failure.	None. RHRS Train A can provide adequate flow to suction of SIPs and CCPs. The SIPs can provide HL recirc. flow	Suction path to both SIPs and both CCPs can be established from RHRP A-A discharge by opening Train A valves and through normally open train B valves.
		Isolated for passive failure.	No further failures assumed					
58	FCV-63-6 Train B	Provide recirc. mode flow path interconnectin g suction of SIPs and CCPs when RHRP discharge is aligned to suction side of	Fails to open, stuck closed or spuriously recloses	Mechanical failure; Train B power failure; operator error	Indicating light in MCR	None. Independent Train A valve 63-7 can be opened to ensure at least one suction flow path to both SIPs and both CCPs	None	Valve opened by operator during switchover to recirc. If failure is due to Train B power failure, RHRP A-A can provide recirc. flow and suction to SIPs and CCPs by opening of Train A valves and normally open (fail-as-is) Train B

FAILURE MODES AND EFFECTS ANALYSIS FOR ACTIVE FAILURES FOR THE SAFETY INJECTION SYSTEM

Item No.	Component	Function	Failure Mode	Potential Cause	Method of Detection	Effect on System	Effect on Plant	Remarks
		CCPs/SIPs (in parallel with Train A FCV-63-7) (recirc. mode)						valves.
			Spurious opening	Operator error; hot short in control wiring	Indicating light in MCR	None	None	Valves 63-7 and 63-6 are closed during reactor operation and injection mode of SI; If valve is spuriously opened during injection mode, suction headers of CCPs and SIPs are connected through 63-6 and 63-177 (normally open). The headers are already both connected to RWST.
59	FCV-63-7 Train A	Provide recirc. mode flow path suction of SIPs and interconnecting CCPs when RHRP discharge is aligned to suction side of CCPs/SIPs (in	Fails to open, stuck closed or spuriously recloses	Mechanical failure; Train A power failure; operator error	Indicating light in MCR	None. Independent Train B valve 63-6 can be opened to ensure at least one suction flow path to both SIPs and both CCPs	None	Valve opened by operator during switchover to recirc. If failure is due to Train A power failure, RHRP B-B can provide recirc. flow and suction to SIPs and CCPs by opening of Train B valves and normally open (fail-as-is) Train A

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UNIT 2

FAILURE MODES AND EFFECTS ANALYSIS FOR ACTIVE FAILURES FOR THE SAFETY INJECTION SYSTEM

Item No.	Component	Function	Failure Mode	Potential Cause	Method of Detection	Effect on System	Effect on Plant	Remarks
		parallel with Train B FCV-63-6) (recirc. mode)						valves.
			Spurious opening	Operator error, hot short in control wiring	Indicating light in MCR	None	None	Valves 63-7 and 63-6 are closed during reactor operation and injection mode of SI; If valve is spuriously opened during injection mode, suction headers of CCPs and SIPs are connected through 63-7 and 63-177 (normally open). The headers are already both connected to RWST.
60	FCV-63-177 Train A	Normally open valve providing flow path interconnecting suction of SIPs and CCPs from RHRP discharge (recirc. mode)	Spurious closing	Operator error, hot short in control wiring	Indicating light in MCR	No interconnecting flow path to suction of CCPs and SIPs	None	Suction flow path to both CCPs is established by opening valve 63-8, to SIPs by opening valve 63-11 (SIP B-B directly and SIP A-A through normally open valves 63-48 and 63-47)

Table 6.3-8 (Sheet 50 of 59)
UNIT 2

FAILURE MODES AND EFFECTS ANALYSIS FOR ACTIVE FAILURES FOR THE SAFETY INJECTION SYSTEM

Item No.	Component	Function	Failure Mode	Potential Cause	Method of Detection	Effect on System	Effect on Plant	Remarks
61	FCV-63-118 Train A	Provide flow path from Accumulator Tank #1 to cold leg 1	No credible failure mode. See 'Remarks' column	See 'Remarks' column	See 'Remarks' column	See 'Remarks column	See 'Remarks' column	Normally open valve (open in reactor operation as well as injection and recirc. modes of SI) has no credible failure mode because: 1) permissive interlock in close circuit opens on Train A SI signal, 2) Train A SI signal and RCS pressure >1970 psig signal are used in the opening circuit, 3) protection against spurious operation due to hot short in control wiring is provided by using selector and hand switch contacts on line and neutral side of contactors and 4) operator error unlikely due to protective cover over HS (failure mode and potential cause). Breaker is locked opened to prevent spurious operation.

FAILURE MODES AND EFFECTS ANALYSIS FOR ACTIVE FAILURES FOR THE SAFETY INJECTION SYSTEM

Item No.	Component	Function	Failure Mode	Potential Cause	Method of Detection	Effect on System	Effect on Plant	Remarks
62	FCV-63-98 Train B	Provide flow path from Accumulator Tank #2 to cold leg 2	No credible failure mode. See 'Remarks' column	See 'Remarks' column	See 'Remarks' column	See 'Remarks' column	See 'Remarks' column	Normally open valve (open in reactor operation as well as injection and recirc. Modes of SI) has no credible failure mode because: 1) permissive interlock in close circuit opens on Train B SI signal, 2) Train B SI signal and RCS pressure >1970 psig signal are used in the opening circuit, 3) protection against spurious operation due to hot short in control wiring is provided by using selector and hand switch contacts on line and neutral side of contractors and 4) operator error unlikely due to protective cover over HS (failure mode and potential cause). Breaker is locked opened to prevent spurious operation.

FAILURE MODES AND EFFECTS ANALYSIS FOR ACTIVE FAILURES FOR THE SAFETY INJECTION SYSTEM

Item No.	Component	Function	Failure Mode	Potential Cause	Method of Detection	Effect on System	Effect on Plant	Remarks
63	FCV-63-80 Train A	Provide flow path from Accumulator Tank #3 to cold leg 3	No credible failure mode. See 'Remarks' column	See 'Remarks' column	See 'Remarks' column	See 'Remarks' column	See 'Remarks' column	Normally open valve (open in reactor operation as well as injection and recirc. Modes of SI) has no credible failure mode because: 1) permissive interlock in close circuit opens on Train A SI signal, 2) Train A SI signal and RCS pressure >1970 psig signal are used in the opening circuit, 3) protection against spurious operation due to hot short in control wiring is provided by using selector and hand switch contacts on line and neutral side of contactors and 4) operator error unlikely due to protective cover over HS (failure mode and potential cause). Breaker is locked opened to prevent spurious operation.

Table 6.3-8 (Sheet 53 of 59)
UNIT 2

FAILURE MODES AND EFFECTS ANALYSIS FOR ACTIVE FAILURES FOR THE SAFETY INJECTION SYSTEM

Item No.	Component	Function	Failure Mode	Potential Cause	Method of Detection	Effect on System	Effect on Plant	Remarks
64	FCV-63-67 Train B	Provide flow path from Accumulator Tank #4 to cold leg 4	No credible failure mode. See ‘Remarks’ column	See ‘Remarks’ column	See ‘Remarks’ column	See ‘Remarks’ column	See 'Remarks' column	Normally open valve (open in reactor operation as well as injection and recirc. modes of SI) has no credible failure mode because: 1) permissive interlock in close circuit opens on Train B SI signal, 2) Train B SI signal and RCS pressure >1970 psig signal are used in the opening circuit, 3) protection against spurious operation due to hot short in control wiring is provided by using selector and hand switch contacts on line and neutral side of contactors and 4) operator error unlikely due to protective cover over HS (failure mode and potential cause). Breaker is locked open to prevent spurious operation.

FAILURE MODES AND EFFECTS ANALYSIS FOR ACTIVE FAILURES FOR THE SAFETY INJECTION SYSTEM

Item No.	Component	Function	Failure Mode	Potential Cause	Method of Detection	Effect on System	Effect on Plant	Remarks
65	Train A Emergency Power	Provides Class 1E diesel-backed power supply to active components of Train A of SI system (injection and recirc. modes)	Loss of, or inadequate, voltage	Diesel generator failure; bus fault (Train A), operator error.	Alarm and indicator in MCR	No redundancy in SI system	None. Four accumulators, CCP B-B, SIP B-B and RHRP B-B remain available and can provide adequate flow for postulated range of break sizes.	Compensating provisions/actions will occur/can be performed for Train “A” valves required to change position following a LOCA to ensure the required function is not disabled due to Train “A” power failure.
66	Train B Emergency Power	Provides Class 1E diesel-backed power supply to active components of Train B of SI system (injection and recirc. modes)	Loss of, or inadequate, voltage	Diesel generator failure; bus fault (Train B), operator error.	Alarm and indicator in MCR	No redundancy in SI system	None. Four accumulators, CCP A-A, SIP A-A and RHRP A-A remain available and can provide adequate flow for postulated range of break sizes.	Compensating provisions/actions will occur/can be performed for Train “B” valves required to change position following a LOCA to ensure the required function is not disabled due to Train “B” power failure.
67	RWST level loops connected in 2 out of 4	Provides open signal to FCV-63-72 & FCV-63-73 in	One loop fails high	Transmitter failure; open/short circuit in loop wiring,	One of four level indicators (LI-63-50	Resulting logic of RWST level control for Trains A & B changes	None	Train A & Train B signals for ECCS switchover generated in SSPS if 2 out of 4

FAILURE MODES AND EFFECTS ANALYSIS FOR ACTIVE FAILURES FOR THE SAFETY INJECTION SYSTEM

Item No.	Component	Function	Failure Mode	Potential Cause	Method of Detection	Effect on System	Effect on Plant	Remarks
	logic. L-63-50 L-63-51 L-63-52 L-63-53	combination with containment sump level loops and SI signal (initiation of recirc. mode).		calibrating error	thru LI-63-53) reading higher than other three	from 2 out of 4 to 2 out of 3.		RWST level loops show a low level AND 2 out of 4 containment sump level loops show a high level AND a SI signal is present. All components in level loops from transmitter through bistable are Class 1E. Of the four level indicators, two (LI-63-50 & 51) are Class 1E, used for PAM indication (Category 1). Single failure analysis for PAM indication is acceptable. Class 1E portions of the level loops are isolated from non-Class 1E portions (e.g., level loops III & IV indicators and level switches for Lo-Lo level alarm) by Class 1E signal conditioner modules.
			One loop fails low	Transmitter failure; open/short circuit in loop wiring, calibrating error	One of four level indicators (LI-63-50 thru LI-63-53) reading lower than other three. Open circuit in trip output wiring would not energize the trip status light on the MCB. Trip output is not fail-safe for these channels. The Tech Spec	Resulting logic of RWST level control for Trains A & B changes from 2 out of 4 to 1 out of 3 until failed channel is bypassed; then 2 out of 3.	None	

FAILURE MODES AND EFFECTS ANALYSIS FOR ACTIVE FAILURES FOR THE SAFETY INJECTION SYSTEM

Item No.	Component	Function	Failure Mode	Potential Cause	Method of Detection	Effect on System	Effect on Plant	Remarks
					surveillance test would be the only way to detect this failure.			
68	Containment Sump level Loops connected in 2 out of 4 logic. L-63-180 L-63-181 L-63-182 L-63-183	Provides open signal to FCV-63-72 & FCV-63-73 in combination with RWST level loops and SI signal (initiation of recirc. mode).	One loop fails high	Transmitter failure; open/short circuit in loop wiring, calibrating error	One of four level indicators (LI-63-180 thru LI-63-183) reading higher than other three. Bistable energizes light on MCB.	Resulting logic of Containment Sump level control for Trains A & B changes from 2 out of 4 to 1 out of 3 until failed channel is bypassed; then 2 out of 3.	None	Train A & Train B signals for ECCS switchover generated in SSPS if 2 out of 4 RWST level loops show a low level AND 2 out of 4 containment sump level loops show a high level AND a SI signal is present. All components in level loops from transmitter through bistable are
			One loop fails low	Transmitter failure; open/short circuit in loop wiring, calibrating error	Normal level is 0%. THis would be detecteable by level indicator only if containment level is rising due to HELB inside	Resulting logic of containment sump level control for Trains A & B changes from 2 out of 4 to 2 out of 3.	None	Class 1E. Of the four level indicators, two (LI-63-180 & 181) are Class 1E, used for PAM indication (Category 1). Single failure analysis for PAM indication is acceptable. Non-Class 1E indicators on level loops III & IV are isolated from Class 1E

Table 6.3-8 (Sheet 57 of 59)
UNIT 2

FAILURE MODES AND EFFECTS ANALYSIS FOR ACTIVE FAILURES FOR THE SAFETY INJECTION SYSTEM

Item No.	Component	Function	Failure Mode	Potential Cause	Method of Detection	Effect on System	Effect on Plant	Remarks
					containment. Eagle 21 may generate a trouble alarm, but we don't take credit for this. Trip channels are not fail-safe. Tech Spec surveillance test would be the only way to detect this failure. For this transmitter this is an 18 month interval.			level switches by Class 1E signal conditioner modules.
69	FCV-62-1228 Train A	Isolates CCPs' normal suction hydrogen vent line (volume control tank) when RWST	Fails to close, stuck open or spurious opening after closing.	Mechanical failure, instrumentation & control failure, operator error.	Status indicating lights	Loss of redundancy in CCP normal suction hydrogen vent line	None. Train B isolation valve FCV-62-1229 provides isolation of CCP normal suction vent	Both FCV-62-1228 and FCV-62-1229 are interlocked with LCV-62-132 and LCV-62-133, respectively, in such a way that on a SI signal the valves 62-

Table 6.3-8 (Sheet 58 of 59)
UNIT 2

FAILURE MODES AND EFFECTS ANALYSIS FOR ACTIVE FAILURES FOR THE SAFETY INJECTION SYSTEM

Item No.	Component	Function	Failure Mode	Potential Cause	Method of Detection	Effect on System	Effect on Plant	Remarks
		suction is aligned (in series with LCV-62-132)					path	1228 and 62-1229 will begin to close only when valves 62-132 and 133 begin to close.
70	FCV-62-1229 Train B	Isolates CCPs' normal suction hydrogen vent line (volume control tank) when RWST suction is aligned (in series with LCV-62-133)	Fails to close, stuck open or spurious opening after closing.	Mechanical failure, instrumentation & control failure, operator error.	Status indicating lights	Loss of redundancy in CCP normal suction hydrogen vent line	None. Train A isolation valve FCV-62-1228 provides isolation of CCP normal suction vent path	Both FCV-62-1228 and FCV-62-1229 are interlocked with LCV-62-132 and LCV-62-133, respectively, in such a way that on a SI signal the valves 62-1228 and 62-1229 will begin to close only when valves 62-132 and 133 begin to close.
71	RFV-63-626, -627, -637, -534, -535, -536, -577	Prevent over pressure in ECCS injection lines.	Spurious lifting of relief valve at a pressure below its set point.	Mechanical failure	Alarm due to increase in PRT level and/or pressure, except for RFV-63-577. For RFV-63-577, leakage is detected as "unidentified leakage" during	Reduce ECCS injection flow rate due to diversion of flow into relief line.	None. Flow diverted into relief line is less than worst case ECCS flow reduction due to a single failure (i.e., loss of one train of ECCS).	It is not credible to have a pre-existing condition at the time of accident where one or more relief valves have opened spuriously or are experiencing significant seat leakage. The noted "method of detection" provides an adequate means of timely identification of such a condition so that

Table 6.3-8 (Sheet 59 of 59)
UNIT 2

FAILURE MODES AND EFFECTS ANALYSIS FOR ACTIVE FAILURES FOR THE SAFETY INJECTION SYSTEM

Item No.	Component	Function	Failure Mode	Potential Cause	Method of Detection	Effect on System	Effect on Plant	Remarks
					periodic RCS leakage surveillance testing.			appropriate corrective maintenance can be performed. It also noted that the RCS pressure will be less than the subject relief valve's setpoints with the exception of the small break LOCA during post-accident ECCS operation.

TABLE 6.3-9 (Sheet 1 of 4)
UNIT 1

FAILURE MODES AND EFFECTS ANALYSIS FOR THE SAFETY INJECTION SYSTEM (PASSIVE FAILURES RECIRC. MODE)

Item No.	Component	Function	Failure Mode	Potential Cause	Method of Detection	Effect on System	Effect on Plant	Remarks
1	SIP suction header and valves.	Interconnects SIP A-A & B-B suction (through valves 63-47 & 63-48) with the RWST and RHR sources	Rupture or leak	Mechanical failure; (gasket, flange)	Low SIP discharge flow on FI-63-151, FI-63-20; low SIP discharge pressure on PI-63-150, PI-63-19, all in MCR. Low suction pr. ind. on PI-63-9, PI-63-14 (local); Flooding of SIP room/s and/or pipe chase. Area radiation alarms in pump room/s. RB sump level goes down.	Loss of flow from SIPs and loss of sump inventory until operator isolates the leak; potential damage to SIPs. Contamination of pump rooms and pipe chase from sump water. Leak can be isolated via FCV-63-5, -6, -7, -11, -22, -156, -157.	None. RHRS can provide adequate HL recirc. flow. CCPs provide CL flow.	The passive failure assumed is 50 gpm for 30 minutes in the Aux. Bldg., which will not deplete the RB sump.
2	Piping and valves in SIP pump discharge to	Provides flow path from SIP discharge to HLs or CLs	Rupture or leak	Mechanical failure (gasket or flange)	Low/high flow on affected train	Reduced flow to one train of HLs and both trains of CLs. Loss of	None. RHRS can provide adequate HL recirc. flow.	The passive failure assumed is 50 gpm for 30 minutes in the Aux. Bldg., which will not

TABLE 6.3-9 (Sheet 2 of 4)
UNIT 1

FAILURE MODES AND EFFECTS ANALYSIS FOR THE SAFETY INJECTION SYSTEM (PASSIVE FAILURES RECIRC. MODE)

	HLs or CLs				depending on location of break relative to flow element. Flooding & area radiation alarms in Aux. Bldg. RB sump level goes down.	sump inventory until operator secures affected train by isolating FCV-63-5, -6, -7, -11, -22, -156, -157.	CCPs provide CL flow.	deplete the RB sump. The passive failure assumed inside containment is analyzed and found acceptable.
3	Piping and valves on the suction side of one train of RHRS	Provides flow path from containment sump to RHRP A-A or RHRP B-B	Rupture or leak	Mechanical failure (gasket, flange)	Low flow from affected pump; alarm from FIS-74-12 or FIS-74-24; Flooding of pump room and/or pipe chase; Area radiation alarm in pump room. RB sump level goes down.	Loss of redundancy in suction flow to CCPs, SIPs and one RHRP unavailable for recirc. flow. Loss of sump inventory until operator isolates the leak by closing FCV-63-72, 74-3, 63-93, 74-33, 63-8 if Train A and 63-73, 74-21, 63-94, 74-35, 63-11 if Train B. Contamination of pump room and pipe chase from	None. One RHRP remains available to supply recirc. flow and suction for SIPs and CCPs.	The passive failure assumed is 50 gpm for 30 minutes in the Aux. Bldg, which will not deplete the sump.

TABLE 6.3-9 (Sheet 3 of 4)
UNIT 1

FAILURE MODES AND EFFECTS ANALYSIS FOR THE SAFETY INJECTION SYSTEM (PASSIVE FAILURES RECIRC. MODE)

						sump water.		
4	Piping and valves in one train of RHR pump discharge to suction of SIPs & CCPs	Provides suction flow path from RHRP discharge to suction of SIPs & CCPs	Rupture or leak	Mechanical failure (gasket, flange)	Low/high flow depending on location of break relative to flow element. Flooding and area radiation alarms in Aux. Bldg. RB sump level goes down.	Reduced recirc. flow. Loss of sump inventory until operator secures affected train by closing FCV-63-72, 74-3, 63-93, 74-33, 63-8 if Train A and 63-73, 74-21, 63-94, 74-35, 63-11 if Train B.	None. RHRP remains available to supply recirc. flow and suction for SIPs and CCPs.	The passive failure assumed is 50 gpm for 30 minutes in the Aux. Bldg., which will not deplete the sump.
5	Piping and valves between RHRS & HLs 1/3 or CLs	Provides flow path from RHRS to HLs 1/3 or CLs	Rupture or leak (outside containment)	Mechanical failure (gasket, flange)	Low/high flow depending on location of break relative to flow element. Flooding & area radiation alarms in Aux. Bldg. RB sump level goes down.	Loss of recirc. flow from RHRS. Loss of sump inventory until operator isolates the leak by closing FCV-63-72, 74-3, 63-93, 74-33, 63-8 if Train A and 63-73, 74-21, 63-94, 74-35, 63-11 if Train B and FCV-63-172 as required. One train of RHR remains	None. One train of RHR and both SIPs and CCPs still available to provide flow to HLs and CLs.	The passive failure assumed is 50 gpm for 30 minutes in the Aux. Bldg., which will not deplete the sump. The passive failure assumed inside containment is analyzed and found acceptable.

TABLE 6.3-9 (Sheet 4 of 4)
UNIT 1

FAILURE MODES AND EFFECTS ANALYSIS FOR THE SAFETY INJECTION SYSTEM (PASSIVE FAILURES RECIRC. MODE)

						available and SIS provides HL flow; CCPs provide cold leg flow.		
6	Piping & valves in CCP suction and discharge lines.	Provides flow path to recirc. flow to CLs in recirculation.	Rupture or leak (outside containment)	Mechanical failure (gasket, flange)	Low/high flow depending on location of break relative to flow element. Flooding & area radiation alarms in Aux. Bldg. RB sump level goes down.	Loss of CCP flow to CLs. Loss of sump inventory until operator isolates the leak by closing FCV-63-6, -7, -8, 62-132, -133, -135, -136, 63-25, -26.	None. SIPs and one RHRP remain available for recirc.	The passive failure assumed is 50 gpm for 30 minutes in the Aux. Bldg., which will not deplete the sump. The passive failure assumed inside containment is analyzed and found acceptable.

TABLE 6.3-9 (Sheet 1 of 4)
UNIT 2

FAILURE MODES AND EFFECTS ANALYSIS FOR THE SAFETY INJECTION SYSTEM (PASSIVE FAILURES RECIRC. MODE)

Item No.	Component	Function	Failure Mode	Potential Cause	Method of Detection	Effect on System	Effect on Plant	Remarks
1	SIP suction header and valves.	Interconnects SIP A-A & B-B suction (through valves 63-47 & 63-48) with the RWST and RHR sources	Rupture or leak	Mechanical failure; (gasket, flange)	Low SIP discharge flow on FI-63-151, FI-63-20; low SIP discharge pressure on PI-63-150, PI-63-19, all in MCR. Low suction pr. ind. on PI-63-9, PI-63-14 (local); Flooding of SIP room/s and/or pipe chase. Area radiation alarms in pump room/s. RB sump level goes down.	Loss of flow from SIPs and loss of sump inventory until operator isolates the leak; potential damage to SIPs. Contamination of pump rooms and pipe chase from sump water. Leak can be isolated via FCV-63-5, -6, -7, -11, -22, -156, -157.	None. RHRS can provide adequate HL recirc. flow. CCPs provide CL flow.	The passive failure assumed is 50 gpm for 30 minutes in the Aux. Bldg., which will not deplete the RB sump.
2	Piping and valves in SIP pump discharge to	Provides flow path from SIP discharge to HLs or CLs	Rupture or leak	Mechanical failure (gasket or flange)	Low/high flow on affected train	Reduced flow to one train of HLs and both trains of CLs. Loss of	None. RHRS can provide adequate HL recirc. flow.	The passive failure assumed is 50 gpm for 30 minutes in the Aux. Bldg., which will not

TABLE 6.3-9 (Sheet 2 of 4)
UNIT 2

FAILURE MODES AND EFFECTS ANALYSIS FOR THE SAFETY INJECTION SYSTEM (PASSIVE FAILURES RECIRC. MODE)

	HLs or CLs				depending on location of break relative to flow element. Flooding & area radiation alarms in Aux. Bldg. RB sump level goes down.	sump inventory until operator secures affected train by isolating FCV-63-5, -6, -7, -11, -22, -156, -157.	CCPs provide CL flow.	deplete the RB sump. The passive failure assumed inside containment is analyzed and found acceptable.
3	Piping and valves on the suction side of one train of RHRS	Provides flow path from containment sump to RHRP A-A or RHRP B-B	Rupture or leak	Mechanical failure (gasket, flange)	Low flow from affected pump; alarm from FS-74-12A or FS-74-24A; Flooding of pump room and/or pipe chase; Area radiation alarm in pump room. RB sump level goes down.	Loss of redundancy in suction flow to CCPs, SIPs and one RHRP unavailable for recirc. flow. Loss of sump inventory until operator isolates the leak by closing FCV-63-72, 74-3, 63-93, 74-33, 63-8 if Train A and 63-73, 74-21, 63-94, 74-35, 63-11 if Train B. Contamination of pump room and pipe chase from	None. One RHRP remains available to supply recirc. flow and suction for SIPs and CCPs.	The passive failure assumed is 50 gpm for 30 minutes in the Aux. Bldg, which will not deplete the sump.

TABLE 6.3-9 (Sheet 3 of 4)
UNIT 2

FAILURE MODES AND EFFECTS ANALYSIS FOR THE SAFETY INJECTION SYSTEM (PASSIVE FAILURES RECIRC. MODE)

						sump water.		
4	Piping and valves in one train of RHR pump discharge to suction of SIPs & CCPs	Provides suction flow path from RHRP discharge to suction of SIPs & CCPs	Rupture or leak	Mechanical failure (gasket, flange)	Low/high flow depending on location of break relative to flow element. Flooding and area radiation alarms in Aux. Bldg. RB sump level goes down.	Reduced recirc. flow. Loss of sump inventory until operator secures affected train by closing FCV-63-72, 74-3, 63-93, 74-33, 63-8 if Train A and 63-73, 74-21, 63-94, 74-35, 63-11 if Train B.	None. RHRP remains available to supply recirc. flow and suction for SIPs and CCPs.	The passive failure assumed is 50 gpm for 30 minutes in the Aux. Bldg., which will not deplete the sump.
5	Piping and valves between RHRS & HLs 1/3 or CLs	Provides flow path from RHRS to HLs 1/3 or CLs	Rupture or leak (outside containment)	Mechanical failure (gasket, flange)	Low/high flow depending on location of break relative to flow element. Flooding & area radiation alarms in Aux. Bldg. RB sump level goes down.	Loss of recirc. flow from RHRS. Loss of sump inventory until operator isolates the leak by closing FCV-63-72, 74-3, 63-93, 74-33, 63-8 if Train A and 63-73, 74-21, 63-94, 74-35, 63-11 if Train B and FCV-63-172 as required. One train of RHR remains	None. One train of RHR and both SIPs and CCPs still available to provide flow to HLs and CLs.	The passive failure assumed is 50 gpm for 30 minutes in the Aux. Bldg., which will not deplete the sump. The passive failure assumed inside containment is analyzed and found acceptable.

TABLE 6.3-9 (Sheet 4 of 4)
UNIT 2

FAILURE MODES AND EFFECTS ANALYSIS FOR THE SAFETY INJECTION SYSTEM (PASSIVE FAILURES RECIRC. MODE)

						available and SIS provides HL flow; CCPs provide cold leg flow.		
6	Piping & valves in CCP suction and discharge lines.	Provides flow path to recirc. flow to CLs in recirculation.	Rupture or leak (outside containment)	Mechanical failure (gasket, flange)	Low/high flow depending on location of break relative to flow element. Flooding & area radiation alarms in Aux. Bldg. RB sump level goes down.	Loss of CCP flow to CLs. Loss of sump inventory until operator isolates the leak by closing FCV-63-6, -7, -8, 62-132, -133, -135, -136, 63-25, -26.	None. SIPs and one RHRP remain available for recirc.	The passive failure assumed is 50 gpm for 30 minutes in the Aux. Bldg., which will not deplete the sump. The passive failure assumed inside containment is analyzed and found acceptable.

TABLE 6.3-10

PRINCIPAL ECCS VALVE POSITIONS

VALVE ID	Normal Position	Cold Leg Inject	Recirculation Cold Leg/ Hot Leg
FCV-63-26,25	Closed	Open	Open/Open
FCV-63-39,40	Open	Open	Open/Open
FCV-63-172	Closed	Closed	Closed/Open
FCV-63-156,157	Closed	Closed	Closed/Open
FCV-63-93,94	Open	Open	Open(1)/Closed(2)
FCV-63-152,153	Open	Open	Open/Closed
FCV-74-3,21	Open	Open	Closed/Closed
FCV-74-33	Open	Open	Closed/Open(3)
FCV-74-35	Open	Open	Closed/Open(4)
FCV-63-8,11	Closed	Closed	Open/Open
FCV-63-5	Open	Open	Closed/Closed
FCV-63-72,73	Closed	Closed	Open/Open
FCV-63-1	Open	Open	Closed/Closed
FCV-63-3,4	Open	Open	Closed/Closed
FCV-63-175	Open	Open	Closed/Closed
FCV-63-6,7	Closed	Closed	Open/Open
LCV-62-135,136	Closed	Open	Closed/Closed
LCV-62-132,133	Open	Closed	Closed/Closed
FCV-62-1228,1229	Open	Closed	Closed/Closed
FCV-74-16,28	Open	Open	Open/Open
FCV-63-118,98,80,67	Open	Open	Open/Open
FCV-63-71	Closed	Closed	Closed/Closed
FCV-63-84	Closed	Closed	Closed/Closed
FCV-63-47,48	Open	Open	Open/Open
FCV-63-23	Closed	Closed	Closed/Closed
FCV-62-90,91	Open	Closed	Closed/Closed
FCV-62-98,99	Open	Open	Open/Open
FCV-63-22	Open	Open	Open/Open or Closed (5)

(1) Valve closed if RHR spray is required.
(2) Position shown for RHRP HL recirc.
(3) Position shown for RHRP A-A HL recirc.
(4) Position shown for RHRP B-B HL recirc.
(5) Passive valve - closure not required.

WBN

TABLE 6.3-11

NORMALIZED DECAY HEAT

Time (Seconds)	Decay Heat Fraction (Btu/Btu)
1.0000E+02	4.2815E-02
2.0000E+02	3.6520E-02
4.0000E+02	3.1101E-02
6.0000E+02	2.8237E-02
1.0000E+03	2.4937E-02
2.0000E+03	2.1006E-02
4.0000E+03	1.7195E-02
6.0000E+03	1.5237E-02
1.0000E+04	1.3168E-02
2.0000E+04	1.0825E-02
4.0000E+04	8.9280E-03
6.0000E+04	7.9480E-03
1.0000E+05	6.8570E-03
2.0000E+05	5.5870E-03
4.0000E+05	4.5050E-03
6.0000E+05	3.8960E-03
1.0000E+06	3.2860E-03
2.0000E+06	2.5940E-03
4.0000E+06	2.0160E-03
6.0000E+06	1.7310E-03
1.0000E+07	1.4880E-03

TABLE 6.3-12
UNIT 1

AVAILABLE NPSH DURING ECCS OPERATION

Pump	Flow (gpm)	Supply	$NPSH_R$ (ft)	$NPSH_A$(ft)	$NPSH_M$ (ft)
Injection - both trains operating					
CCP A-A	450	RWST	14.5	46.2	31.7
CCP B-B	450	RWST	14.5	46.7	32.2
SIP A-A	450	RWST	16.5	38.3	21.8
SIP B-B	450	RWST	15.5	37.8	22.3
RHRP A-A	4000	RWST	16.5	30.6	14.1
RHRP B-B	4000	RWST	16.5	31.6	15.1
CSP A-A	5000	RWST	13.5	37.3	23.8
CSP B-B	5000	RWST	13.5	27.9	14.4

Pump	Flow (gpm)	Supply	$NPSH_R$ (ft)	$NPSH_A$(ft)	$NPSH_M$ (ft)
Injection - A train operating					
CCP A-A	550	RWST	28.0	55.4	27.4
SIP A-A	660	RWST	25.0	54.5	29.5
RHRP A-A	5000	RWST	21.0	54.1	33.1
CSP A-A	5000	RWST	13.5	53.5	40.0

Pump	Flow (gpm)	Supply	$NPSH_R$ (ft)	$NPSH_A$(ft)	$NPSH_M$ (ft)
Injection - B train operating					
CCP B-B	550	RWST	28.0	55.7	27.7
SIP B-B	660	RWST	25.0	54.1	29.1
RHRP B-B	5000	RWST	21.0	55.7	34.7
CSP B-B	5000	RWST	13.5	50.2	36.7

Pump	Flow (gpm)	Supply	$NPSH_R$ (ft)	$NPSH_A$(ft)	$NPSH_M$ (ft)
Recirculation - both trains operating					
RHRP A-A	4550	Sump	19.0	30.0	11.0
RHRP B-B	4550	Sump	19.0	30.9	11.9
CSP A-A	4700	Sump	12.5	19.0	6.5
CSP B-B	4700	Sump	12.5	17.2	4.7

TABLE 6.3-12
UNIT 2

AVAILABLE NPSH DURING ECCS OPERATION

Pump	Flow (gpm)	Supply	$NPSH_R$ (ft)	$NPSH_A$(ft)	$NPSH_M$ (ft)
		Injection - both trains operating			
CCP A-A	450	RWST	14.5	46.1	31.6
CCP B-B	450	RWST	14.5	46.4	31.9
SIP A-A	450	RWST	17.5	37.9	20.4
SIP B-B	450	RWST	17.5	37.6	20.1
RHRP A-A	4000	RWST	14.1	30.4	16.3
RHRP B-B	4000	RWST	14.1	31.1	17.0
CSP A-A	5000	RWST	13.5	37.5	24.0
CSP B-B	5000	RWST	13.5	28.2	14.7

Pump	Flow (gpm)	Supply	$NPSH_R$ (ft)	$NPSH_A$(ft)	$NPSH_M$ (ft)
		Injection - A train operating			
CCP A-A	550	RWST	28.0	55.2	27.2
SIP A-A	660	RWST	32.0	56.7	24.7
RHRP A-A	5000	RWST	19.0	54.1	35.1
CSP A-A	5000	RWST	13.5	53.4	39.9

Pump	Flow (gpm)	Supply	$NPSH_R$ (ft)	$NPSH_A$(ft)	$NPSH_M$ (ft)
		Injection - B train operating			
CCP B-B	550	RWST	28.0	55.4	27.4
SIP B-B	660	RWST	32.0	53.6	21.6
RHRP B-B	5000	RWST	19.0	55.1	36.1
CSP B-B	5000	RWST	13.5	50.2	36.7

Pump	Flow (gpm)	Supply	$NPSH_R$ (ft)	$NPSH_A$(ft)	$NPSH_M$ (ft)
		Recirculation - both trains operating			
RHRP A-A	4550	Sump	17.0	31.1	14.1
RHRP B-B	4550	Sump	17.0	31.8	14.8
CSP A-A	4630	Sump	12.2	21.1	8.9
CSP B-B	4630	Sump	12.2	18.6	6.4

CAD MAINTAINED DRAWING

PRESS. TEMP DESIGN DATA		
LINE MK NO.	PRESSURE (PSIG)	TEMPERATURE (°F)
1	150	200
2	100	400
3	600	400
4	1750	200
5	2735	200
6	2485	650
7	150	150
8	700	300
9		
10		
11	150	100
12	50	190
13	220	200
14	1550	130

PIPING

FIRST NO. USED
500
OPEN NO.S
531, 566, 567, 579, 580, 645, 648, 523, 522, 516, 521, 667
LAST NO. USED
709

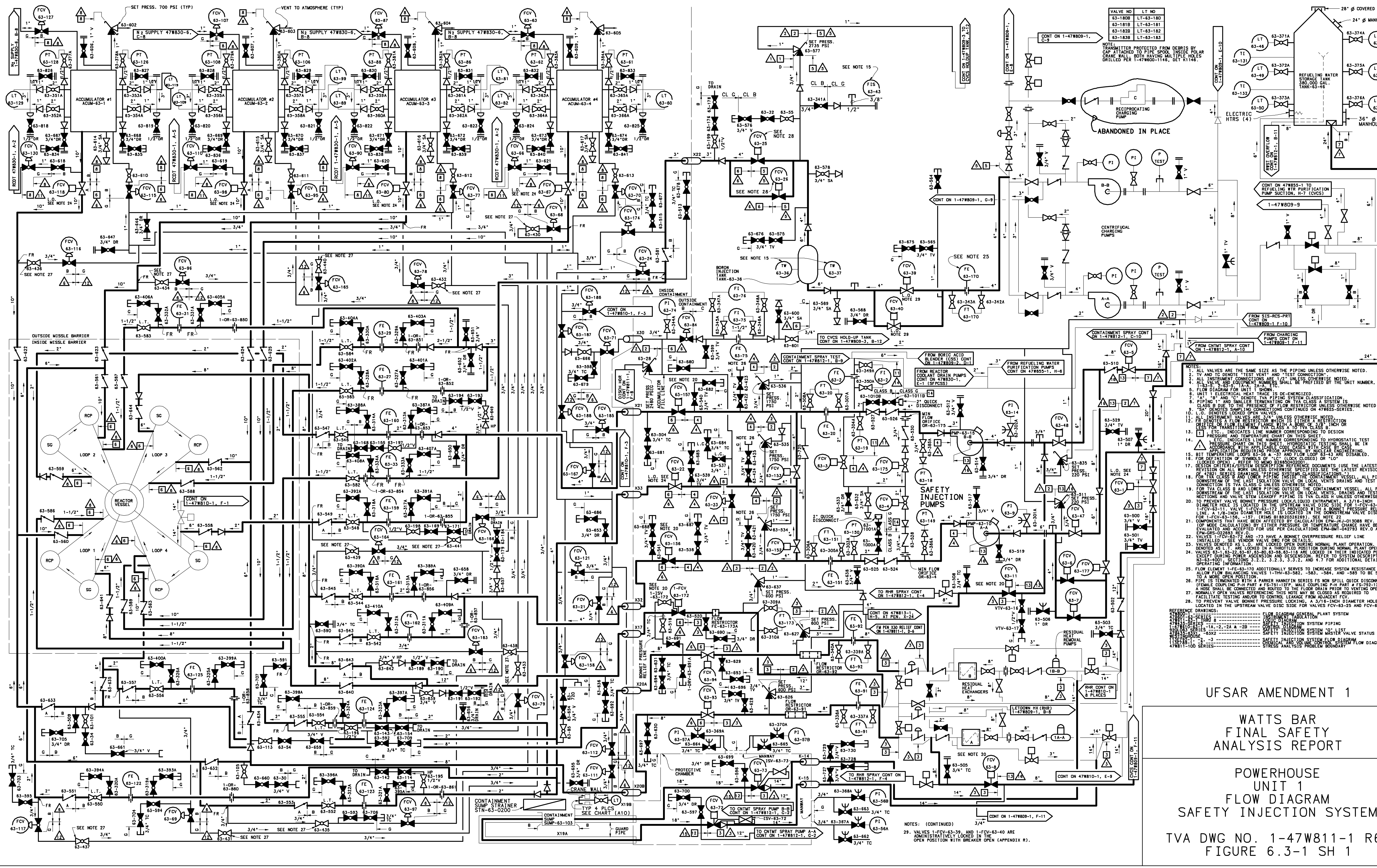
INSTRUMENT ROOT VALVES

FIRST NO. USED
301A
OPEN NO.S
LAST NO. USED
410A

LINE MK NO.	HYDROSTATIC TEST PRESS. (PSIG)
1	188
2	125
3	750
4	2188
5	3419
6	3107
7	188
8	875
9	
10	
11	1313
12	3870
13	3728
14	188
15	63
16	275

FOR PIPING ONLY, LIMITING COMPONENTS TO BE DETERMINED BY FIELD.

HYDROSTATIC TEST PRESSURE DATA IS HISTORICAL INFORMATION AND NO LONGER MAINTAINED AS DESIGN OUTPUT.



NOTES:
1. ALL VALVES ARE THE SAME SIZE AS THE PIPING UNLESS OTHERWISE NOTED.
2. TV AND TC DENOTE "TEST VENT" AND "TEST CONNECTION".
3. ALL INSTRUMENT CONNECTIONS ARE 1/2" UNLESS OTHERWISE NOTED.
4. ALL VALVE AND EQUIPMENT NUMBERS SHALL BE PREFIXED BY THE UNIT NUMBER, 1-63-5, 2-63-5, 1A-A, 2A-A, ETC.
5. FLOW DIAGRAM FOR UNIT 1 SHOWN.
6. UNIT 1 ELECTRICAL HEAT TRACE IS DE-ENERGIZED.
7. "A", "B" AND "C" DENOTE TVA PIPING SYSTEM CLASSIFICATION.
8. PIPING 1" AND SMALLER TERMINATING ON TVA CLASS A SYSTEM IS CLASS B DUE TO THE PRESENCE OF FLOW RESTRICTOR UNLESS OTHERWISE NOTED.
9. "SA" DENOTES SAMPLING CONNECTIONS CONTINUED ON 47W625-SERIES.
10. L.O. DENOTES LOCKED OPEN VALVES.
11. ALL INSTRUMENT VALVES ARE 3/4" UNLESS OTHERWISE NOTED.
12. FR DENOTES A FLOW RESTRICTOR WHICH IS EITHER A RESTRICTION ORIFICE OR FLOW ELEMENT FLANGE WITH A BORE OF 3/8" INCH OR LESS FOR TRANSITION FROM TVA CLASS A TO TVA CLASS B.
13. $\boxed{1}$, ETC., INDICATES LINE NUMBER CORRESPONDING TO DESIGN PRESSURE AND TEMPERATURE CHART ON THIS SHEET.
14. $\triangle 1$, ETC, INDICATES LINE NUMBER CORRESPONDING TO HYDROSTATIC TEST PRESSURE CHART ON THIS SHEET. HYDROSTATIC TESTING SHALL BE IN ACCORDANCE WITH APPLICABLE CODE CASES WITH CASE BY CASE APPLICATION REQUIRING PRIOR APPROVAL BY NUCLEAR ENGINEERING.
15. BIT TEMPERATURE LOOPS 63-36 & -37 AND FLOW LOOP 63-43 ARE DISABLED.
16. FOR DEFINITION OF SYMBOLS OF "LC" (LOCK CLOSED) OR "LO" (LOCKED OPEN), REFER TO MEMO B26850402001.
17. DESIGN CRITERIA/SYSTEM DESCRIPTION REFERENCE DOCUMENTS (USE THE LATEST REVISION ON ALL WORK UNLESS OTHERWISE SPECIFIED.SEE THE LATEST REVISION OF 47B21 SERIES DRAWINGS "PIPING SYSTEMS CLASSIFICATIONS.")
18. FOR TVA CLASS B AND LOWER PIPING INSIDE THE CONTAINMENT VESSEL, ALL PIPING DOWNSTREAM OF THE LAST ISOLATION VALVE ON LOCAL VENTS DRAINS AND TEST CONNECTION IS TVA CLASS G UNLESS OTHERWISE NOTED.
19. FOR TVA CLASS B AND LOWER PIPING OUTSIDE THE CONTAINMENT VESSEL, ALL PIPING DOWNSTREAM OF THE LAST ISOLATION VALVE ON LOCAL VENTS, DRAINS AND TEST CONNECTIONS AND VALVE STEM LEAKOFF PIPING IS TVA CLASS H UNLESS OTHERWISE NOTED.
20. TO PREVENT VALVE BONNET PRESSURE LOCK/LIQUID ENTRAPMENT, A 3/16-INCH DIAMETER HOLE IS LOCATED IN THE UPSTREAM VALVE DISC SIDE FOR 1-FCV-63-8 AND 1-FCV-63-11. VALVE 1-FCV-63-172 IS PROVIDED WITH A BONNET PRESSURE RELIEF LINE. A 1/8-INCH DIAMETER HOLE IS LOCATED IN THE DOWNSTREAM VALVE DISC SIDE FOR 1-FCV-63-156, -157. (RIMS B18920326269). SEE NOTE 22
21. COMPONENTS THAT HAVE BEEN AFFECTED BY CALCULATION EPM-JKJ-013088 REV. 4 (OP MODE CALCULATION) BY EITHER PRESSURE OR TEMPERATURE CHANGE HAVE BEEN EVALUATED AND ACCEPTED FOR USE PER CALCULATIONS EPM-BMT-042793 REV.0 AND EPM-GRS-022693 REV.0.
22. VALVES 1-FCV-63-72 AND -73 HAVE A BONNET OVERPRESSURE RELIEF LINE INSTALLED . SEE VENDOR VALVE DWG FOR DETAILS.
23. VALVES DENOTED AS L.O. ARE LOCKED OPEN DURING NORMAL PLANT OPERATION. VALVES DENOTED AS L.T. ARE LOCKED IN A THROTTLED POSITION DURING NORMAL PLANT OPERATION.
24. VALVES 63-1,63-22,63-67,63-80,63-98,63-118 ARE LOCKED IN THEIR INDICATED POSITION EXCEPT DURING POWER ASCENSION AND DESCENSION. REFER TO SYSTEM DESCRIPTION N3-63-4001, SECTIONS 3.2.2, 3.2.3, 3.3.2, AND 4.7 FOR ADDITIONAL DETAILED OPERATING INFORMATION.
25. FLOW ELEMENT 1-FE-63-170 ADDITIONALLY SERVES TO INCREASE SYSTEM RESISTANCE TO ALLOW FLOW BALANCING VALVES 1-THV-63-582, -583, -584, AND -585 TO BE SET TO A MORE OPEN POSITION.
26. PIPE IS TERMINATED WITH A PARKER HANNIFIN SERIES FS NON SPILL QUICK DISCONNECT. (FEMALE COUPLING P-H PART # FS-751-12FP, MALE COUPLING P-H PART # FS-752-12FP). A HOSE SHALL BE CONNECTED AND ROUTED TO THE FLOOR DRAIN PRIOR TO VENTING OPERATIONS.
27. NORMALLY OPEN VALVES REFERENCING THIS NOTE MAY BE CLOSED AS REQUIRED TO FACILITATE TESTING AND/OR TO CONTROL LEAKAGE FROM ADJACENT FCV.
28. TO PREVENT VALVE BONNET PRESSURE LOCKING, A 3/16-INCH DIAMETER HOLE IS LOCATED IN THE UPSTREAM VALVE DISC SIDE FOR VALVES FCV-63-25 AND FCV-63-26.

REFERENCE DRAWINGS:
47W800-1 ------------------------- FLOW DIAGRAM GENERAL PLANT SYSTEM
47B601-63-SERIES ----------------- INSTRUMENT TABULATION
47W611-63-1 THRU 8 --------------- LOGIC DIAGRAM
47W435-SERIES -------------------- SAFETY INJECTION SYSTEM PIPING
1-47W610-63-1,-1A,-2,-2A & -2B ---- CONTROL DIAGRAM
92QL063 SERIES ------------------- SAFETY INJECTION SYSTEM "Q" LIST
47B435-63X1, -63X2 --------------- SAFETY INJECTION SYSTEM MASTER VALVE STATUS REPORT
WESTINGHOUSE
113E792-1, -2, -3 ---------------- SAFETY INJECTION SYSTEM FLOW DIAGRAM
113E789-1, -2 -------------------- CHEMICAL AND VOLUME CONTROL SYSTEM FLOW DIAGRAM
47W811-100 SERIES----------------- STRESS ANALYSIS PROBLEM BOUNDARY

NOTES: (CONTINUED)
29. VALVES 1-FCV-63-39, AND 1-FCV-63-40 ARE ADMINISTRATIVELY LOCKED IN THE OPEN POSITION WITH BREAKER OPEN (APPENDIX R).

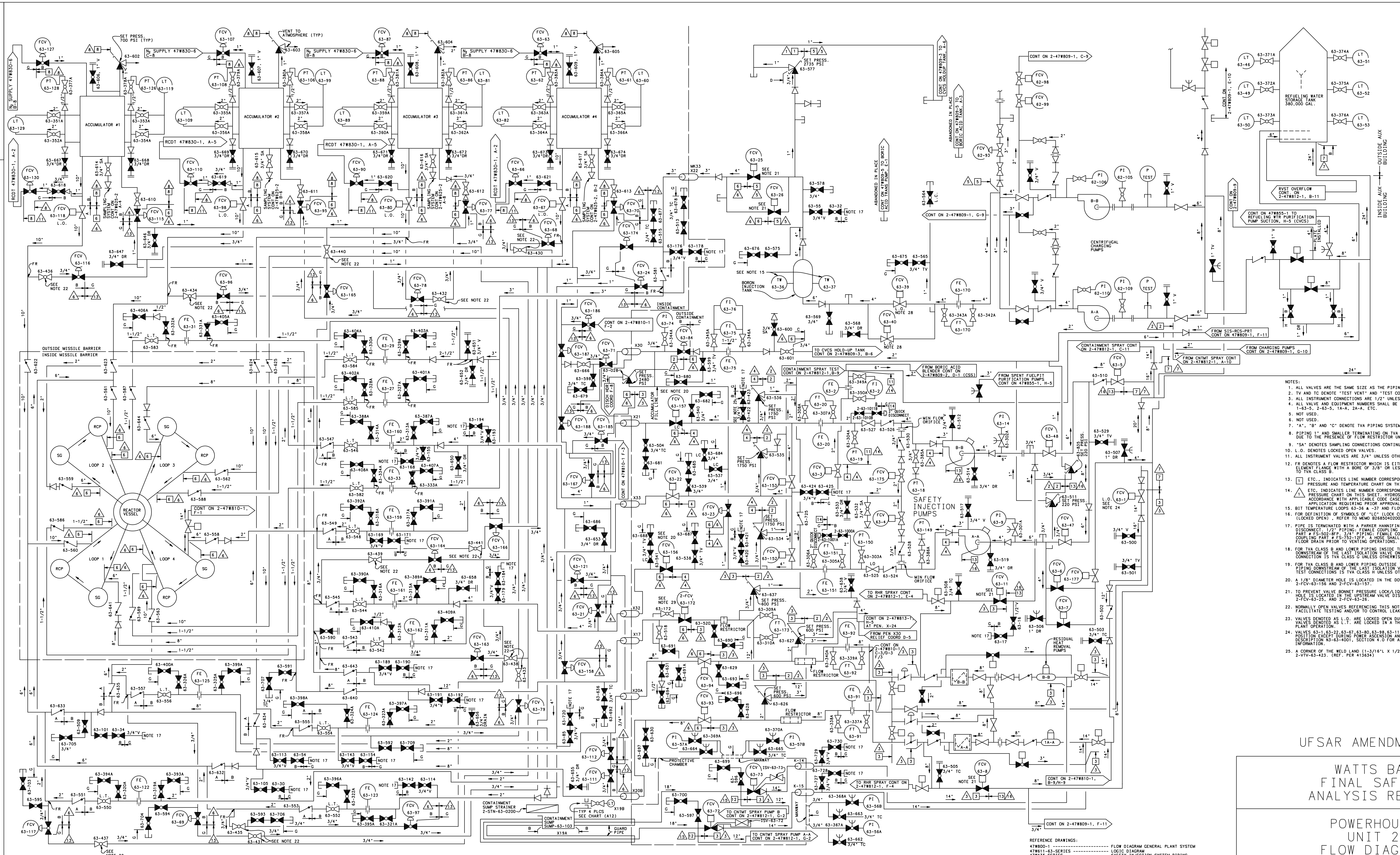
UFSAR AMENDMENT 1

WATTS BAR
FINAL SAFETY
ANALYSIS REPORT

POWERHOUSE
UNIT 1
FLOW DIAGRAM
SAFETY INJECTION SYSTEM

TVA DWG NO. 1-47W811-1 R64
FIGURE 6.3-1 SH 1

CAD MAINTAINED DRAWING



VALVE NO.	LT NO.
2-63-180	2-LT-63-180
2-63-181	2-LT-63-181
2-63-182	2-LT-63-182
2-63-183	2-LT-63-183

PRESS. TEMP DESIGN DATA		
LINE MK NO.	PRESSURE (PSIG)	TEMPERATURE (F°)
1	150	200
2	100	400
3	600	400
4	1750	200
5	2735	200
6	2485	650
7	150	150
8	700	300
9		
10		
11	150	100
12	50	190
13	220	200
14	1550	130

LINE MK NO.	HYDROSTATIC TEST PRESS. (PSIG)
1	188
2	125
3	750
4	2188
5	3419
6	3107
7	188
8	875
9	
10	
11	1050
12	3870
13	3728
14	188
15	63
16	275

NOTES:

1. ALL VALVES ARE THE SAME SIZE AS THE PIPING UNLESS OTHERWISE NOTED.
2. TV AND TC DENOTE "TEST VENT" AND "TEST CONNECTION".
3. ALL INSTRUMENT CONNECTIONS ARE 1/2" UNLESS OTHERWISE NOTED.
4. ALL VALVE AND EQUIPMENT NUMBERS SHALL BE PREFIXED BY THE UNIT NUMBER, 1-63-5, 2-63-5, 1A-A, 2A-A, ETC.
5. NOT USED.
6. NOT USED.
7. "A", "B" AND "C" DENOTE TVA PIPING SYSTEM CLASSIFICATION.
8. PIPING 1" AND SMALLER TERMINATING ON TVA CLASS A SYSTEM IS CLASS B DUE TO THE PRESENCE OF FLOW RESTRICTOR UNLESS OTHERWISE NOTED.
9. "SA" DENOTES SAMPLING CONNECTIONS CONTINUED ON 47W625-SERIES.
10. L.O. DENOTES LOCKED OPEN VALVES.
11. ALL INSTRUMENT VALVES ARE 3/4" UNLESS OTHERWISE NOTED.
12. FR DENOTES A FLOW RESTRICTOR WHICH IS EITHER A RESTRICTION ORIFICE OR FLOW ELEMENT FLANGE WITH A BORE OF 3/8" OR LESS FOR TRANSITION FROM TVA CLASS A TO TVA CLASS B.
13. $\boxed{1}$ ETC., INDICATES LINE NUMBER CORRESPONDING TO DESIGN PRESSURE AND TEMPERATURE CHART ON THIS SHEET.
14. △1 ETC, INDICATES LINE NUMBER CORRESPONDING TO HYDROSTATIC TEST PRESSURE CHART ON THIS SHEET. HYDROSTATIC TESTING SHALL BE IN ACCORDANCE WITH APPLICABLE CODE CASES WITH CASE BY CASE APPLICATION REQUIRING PRIOR APPROVAL BY NUCLEAR ENGINEERING DESIGN.
15. BIT TEMPERATURE LOOPS 63-36 & -37 AND FLOW LOOP 63-43 ARE DISABLED.
16. FOR DEFINITION OF SYMBOLS OF "LC" (LOCK CLOSED) OR "LO" (LOCKED OPEN) , REFER TO MEMO B26850402001.
17. PIPE IS TERMINATED WITH A PARKER HANNIFIN SERIES FS NON SPILL QUICK DISCONNECT. 1/2" PIPING: FEMALE COUPLING PART # FS-501-8FP: MALE COUPLING PART # FS-502-8FP. 3/4" PIPING: FEMALE COUPLING PART # FS-751-12FP: MALE COUPLING PART # FS-752-12FP. A HOSE SHALL BE CONNECTED AND ROUTED TO THE FLOOR DRAIN PRIOR TO VENTING OPERATIONS.
18. FOR TVA CLASS B AND LOWER PIPING INSIDE THE CONTAINMENT VESSEL, ALL PIPING DOWNSTREAM OF THE LAST ISOLATION VALVE ON LOCAL VENTS, DRAINS AND TEST CONNECTION IS TVA CLASS G UNLESS OTHERWISE NOTED.
19. FOR TVA CLASS B AND LOWER PIPING OUTSIDE THE CONTAINMENT VESSEL, ALL PIPING DOWNSTREAM OF THE LAST ISOLATION VALVE ON LOCAL VENTS, DRAINS AND TEST CONNECTIONS IS TVA CLASS H UNLESS OTHERWISE NOTED.
20. A 1/8" DIAMETER HOLE IS LOCATED IN THE DOWNSTREAM VALVE DISC SIDE FOR 2-FCV-63-156 AND 2-FCV-63-157.
21. TO PREVENT VALVE BONNET PRESSURE LOCK/LIQUID ENTRAPMENT, A 3/16" DIAMETER HOLE IS LOCATED IN THE UPSTREAM VALVE DISC SIDE FOR 2-FCV-63-8, 2-FCV-63-11, 2-FCV-63-25, AND 2-FCV-63-26.
22. NORMALLY OPEN VALVES REFERENCING THIS NOTE MAY BE CLOSED AS REQUIRED TO FACILITATE TESTING AND/OR TO CONTROL LEAKAGE FROM ADJACENT FCV.
23. VALVES DENOTED AS L.O. ARE LOCKED OPEN DURING NORMAL PLANT OPERATION. VALVES DENOTED AS L.T. ARE LOCKED IN A THROTTLED POSITION DURING NORMAL PLANT OPERATION.
24. VALVES 63-1,63-22,63-67,63-80,63-98,63-118 ARE LOCKED IN THEIR INDICATED POSITION EXCEPT DURING POWER ASCENSION AND DESCENSION. REFER TO SYSTEM DESCRIPTION N3-63-4001, SECTION 4.0 FOR ADDITIONAL DETAILED OPERATING INFORMATION.
25. A CORNER OF THE WELD LAND (1-3/16"L X 1/2"W X 5/32"D) WAS REMOVED ON VALVE 2-VTV-63-423. (REF. PER 413634)

NOTES: (CONTINUED)

26. ALL VALVE STEM LEAKOFF PIPING INSIDE CONTAINMENT SHALL BE TVA CLASS G.
27. ALL VALVE STEM LEAKOFF PIPING OUTSIDE CONTAINMENT SHALL BE TVA CLASS H.
28. VALVES 2-FCV-63-39, AND 2-FCV-63-40 ARE ADMINISTRATIVELY LOCKED IN THE OPEN POSITION WITH BREAKER OPEN (APPENDIX R).
29. VALVE 2-FCV-63-172 IS ADMINISTRATIVELY LOCKED IN THE CLOSED POSITION WITH BREAKER OPEN (APPENDIX R).

REFERENCE DRAWINGS:

47W800-1 ---------------------- FLOW DIAGRAM GENERAL PLANT SYSTEM
47W611-63-SERIES -------------- LOGIC DIAGRAM
47W435-SERIES ----------------- SAFETY INJECTION SYSTEM PIPING
47W610-63-SERIES -------------- CONTROL DIAGRAM
47B21-1------------------------ PIPING SYSTEM CLASSIFICATION.
N3-63-4001--------------------- SAFETY INJECTION SYSTEM DESCRIPTION
113E792-1, -2, -3 ------------- SAFETY INJECTION SYSTEM FLOW DIAGRAM
113E789-1, -2 ----------------- CHEMICAL AND VOLUME CONTROL SYSTEM FLOW DIAGRAM
47W811-100 SERIES ------------- STRESS ANALYSIS PROBLEM BOUNDARY

UFSAR AMENDMENT 2

WATTS BAR
FINAL SAFETY
ANALYSIS REPORT

POWERHOUSE
UNIT 2
FLOW DIAGRAM
SAFETY INJECTION SYSTEM

TVA DWG NO. 2-47W811-1 R54
FIGURE 6.3-1 SH 1(U2)

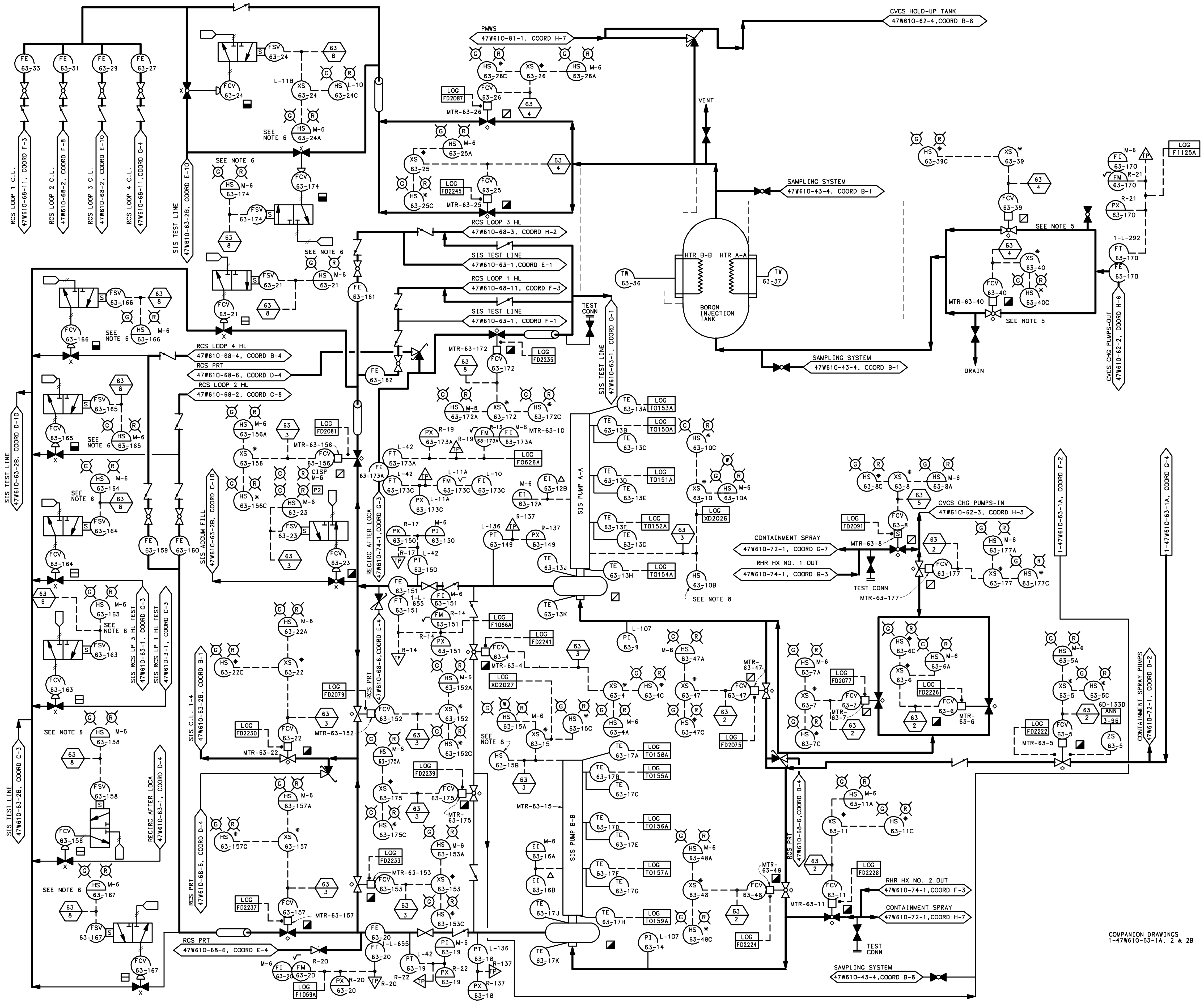
CAD MAINTAINED DRAWING

NOS ALLOCATED:

1 THRU 55 & 131
THRU 179

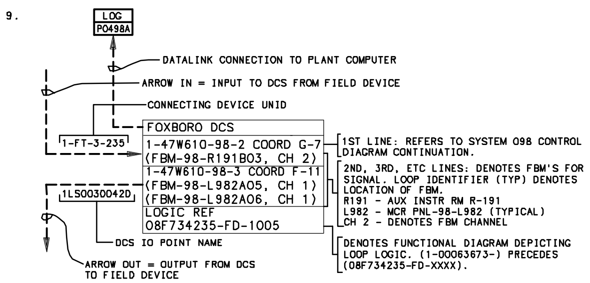
NOS NOT USED:

18, 28, 30, 32, 34,
44, 45, 54, 55, 142,
143, 149, 154, 155,
168, 169, 171, 176,
178, 179



NOTES:

1. THE SAFETY INJECTION SYSTEM SUPPLIES BORATED WATER TO THE REACTOR COOLANT SYSTEM IN THE EVENT OF A LOSS-OF-COOLANT ACCIDENT (LOCA). FOR A DETAILED DESCRIPTION OF THIS SYSTEM, REFER TO WESTINGHOUSE DOCUMENT SD-TVA/TEN-200/D.
2. INSTRUMENTS ASSOCIATED WITH BACKUP CONTROL AND MARKED WITH AN ASTRISK (*) WILL BE LOCATED ON THE APPROPRIATE 480V MOV OR 6.9KV SHUTDOWN BOARDS. OTHER INSTRUMENTS ASSOCIATED WITH BACKUP CONTROL WILL BE LOCATED IN THE AUXILIARY CONTROL ROOM ON PANELS INDICATED ON THE DIAGRAM.
3. AN ANNUCIATION POINT MAY BE SHOWN MORE THAN ONCE FOR MULTIPLE INPUTS.
4. LS-63-44, -45 ARE MOUNTED IN RWST MOAT, ON PANEL L-352.
5. FLOW LOOP 43 AND TEMPERATURE LOOPS 36 & 37 ARE DISABLED, FCV'S 39 & 40 ARE LOCKED OPEN.
6. CONTROL SWITCH IS PART OF MATRIX ASSEMBLY 1-XS-63-100 IN PANEL 1-M-6.
7. REFER TO TABLE A2 FOR ADDITIONAL DIGITAL COMPUTER POINTS (POWER AVAILABLE, PUMP RUNNING, ETC.).
8. 1-HS-63-10B AND 1-HS-63-15B HAS BEEN DISCONNECTED DUE TO APPENDIX R INTERACTION.
9.



REFERENCES:

TVA
47B601-63-SERIES ---------------- INSTRUMENT TABULATIONS
47W611-63-2,-3,-5,-6,-7,-8 ----- LOGIC
47W435-1-SERIES ----------------- PIPING
47W811-1 ------------------------- FLOW DIAG
1-47W611-63-1,-4 ---------------- LOGIC
47B601-99-SERIES ------------ REACTOR PROTECTION SYS TABULATIONS
1-47W610-99-SERIES ---------- REACTOR PROTECTION SYS CONTROL DIAG

108D408-SERIES -------------- PROCESS CONTROL BLOCK DIAGRAMS
WESTINGHOUSE
113E792-1, -2, -3 -------------- SIS FLOW DIAG

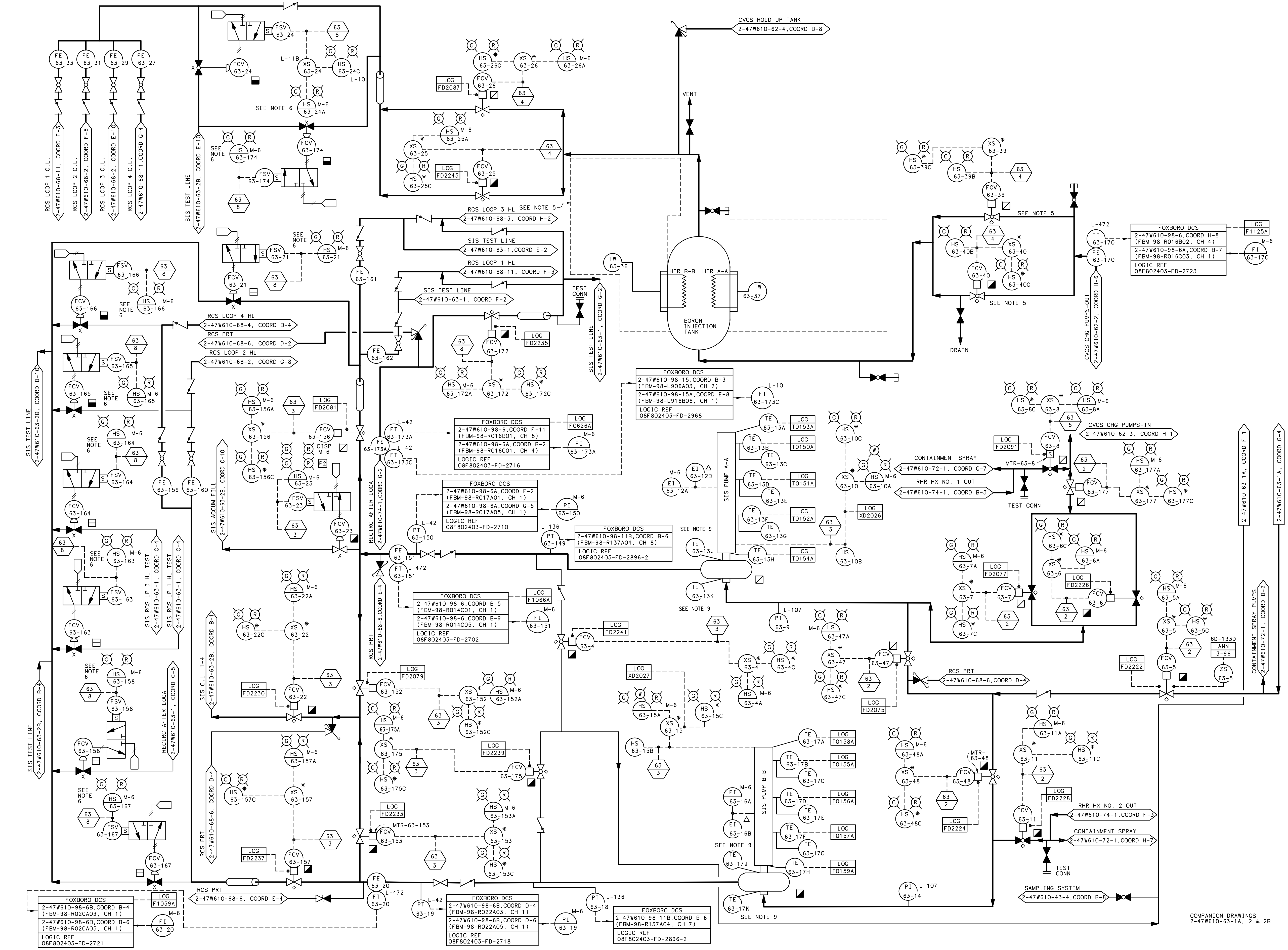
SYMBOLS:

△ - LOCATED ON OR NEAR ASSOCIATED EQUIPMENT
* - CONTROL OR TRANSFER SWITCH LOCATED ON SWITCHGEAR.
TJ - - - TEST SIGNAL INJECTION POINT.
TP - - - SIGNAL TEST POINT
PRESS. REG SET PRESS. TO BE DETERMINED BY MFRR REQUIREMENTS FOR THE DEVICE BEING SUPPLIED.
FOR EAGLE 21 SYMBOLS, SEE 1-47W610-99-1.

WATTS BAR
FINAL SAFETY
ANALYSIS REPORT

POWERHOUSE
UNIT 1
ELECTRICAL
CONTROL DIAGRAM
SAFETY INJECTION SYSTEM
TVA DWG NO. 1-47W610-63-1 R17
FIGURE 6.3-1 SH 2

CAD MAINTAINED DRAWING



NOTES:

1. THE SAFETY INJECTION SYSTEM SUPPLIES BORATED WATER TO THE REACTOR COOLANT SYSTEM IN THE EVENT OF A LOSS-OF-COOLANT ACCIDENT (LOCA). FOR A DETAILED DESCRIPTION OF THIS SYSTEM, REFER TO WESTINGHOUSE DOCUMENT SD-TVA/TEN-200/D.
2. INSTRUMENTS ASSOCIATED WITH BACKUP CONTROL AND MARKED WITH AN ASTERISK (✱) WILL BE LOCATED ON THE APPROPRIATE 480V MOV OR 6.9KV SHUTDOWN BOARDS. OTHER INSTRUMENTS ASSOCIATED WITH BACKUP CONTROL WILL BE LOCATED IN THE AUXILIARY CONTROL ROOM ON PANELS INDICATED ON THE DIAGRAM.
3. AN ANNUNCIATION POINT MAY BE SHOWN MORE THAN ONCE FOR MULTIPLE INPUTS
4. LS-63-44, -45 ARE MOUNTED IN RWST MOAT, ON PANEL L-352.
5. FCV'S 39 & 40 ARE LOCKED OPEN.
6. CONTROL SWITCH IS PART OF MATRIX ASSEMBLY 2-XS-63-100 IN PANEL 2-M-6.
7. REFER TO TABLE A2 ON DWG. 2-47W610-63-2 FOR ADDITIONAL DIGITAL COMPUTER POINTS (POWER AVAILABLE, PUMP RUNNING, ETC.).
8. UNIT INTERFACE SEPARATION BOUNDARIES ARE CONTROLLED BY INSTRUCTION TI-12.08, CONTROL OF UNIT INTERFACES.
9. THERMOCOUPLE CONNECTIONS ARE CURRENTLY PLUGGED.

REFERENCES:

TVA

2-47W611-63-2,-3,-5,-6,-7,-8 ------ LOGIC
2-47W435-1-SERIES ---------------- PIPING
2-47W811-1 ---------------------- FLOW DIAG
2-47W611-63-1,-4 ----------------- LOGIC
2-47W610-99-SERIES --------- REACTOR PROTECTION SYS CONTROL DIAG

108D408-SERIES ------------ PROCESS CONTROL BLOCK DIAGRAMS

WESTINGHOUSE

113E792-1, -2, -3 -------------- SIS FLOW DIAG

SYMBOLS:

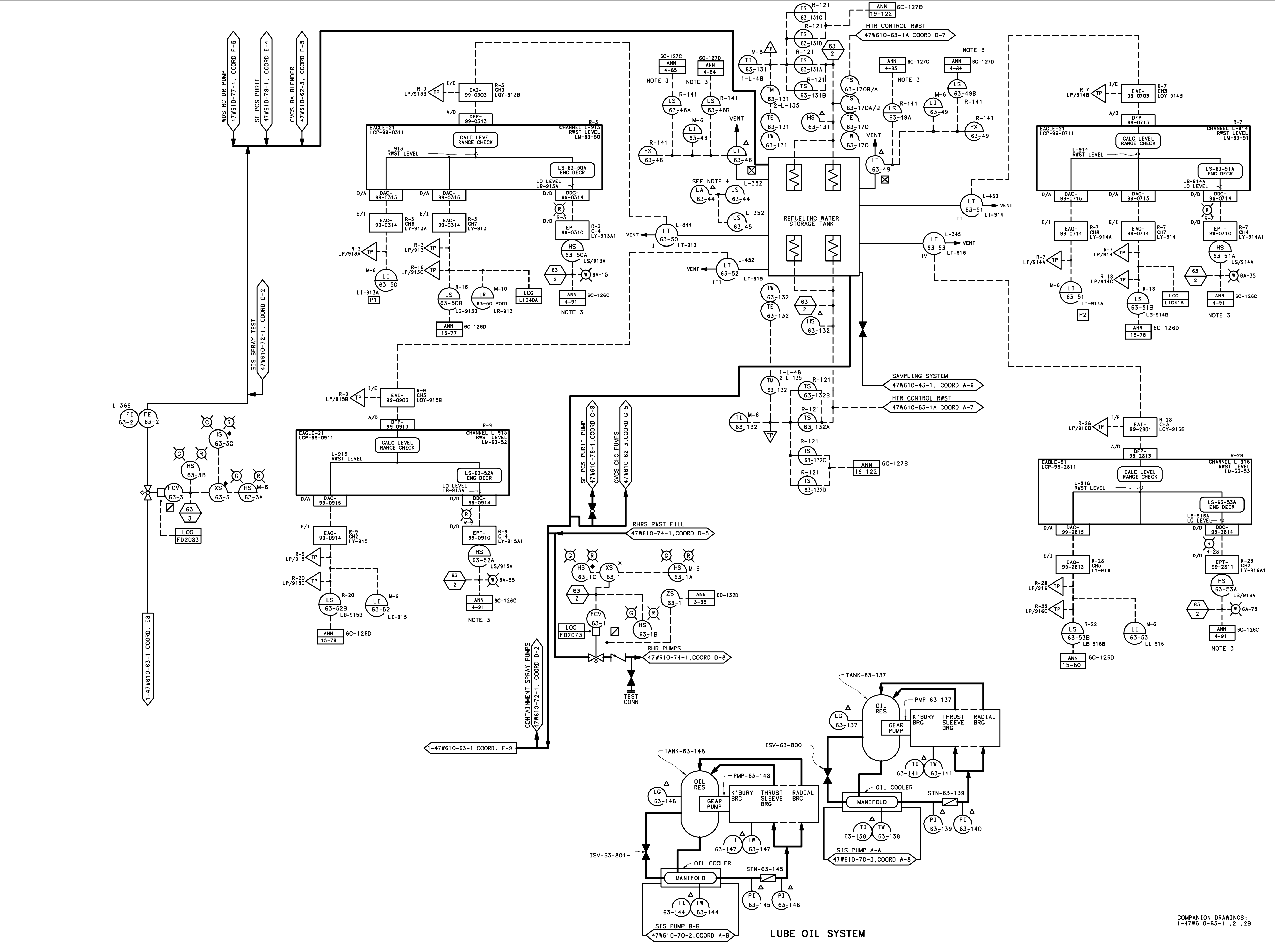
△ - LOCATED ON OR NEAR ASSOCIATED EQUIPMENT
✱ - CONTROL OR TRANSFER SWITCH LOCATED ON SWITCHGEAR.
TJ - TEST SIGNAL INJECTION POINT.
TP - SIGNAL TEST POINT
PRESS. REG SET PRESS. TO BE DETERMINED BY MFRR REQUIREMENTS FOR THE DEVICE BEING SUPPLIED.
FOR EAGLE 21 SYMBOLS, SEE 2-47W610-99-1.

COMPANION DRAWINGS
2-47W610-63-1A, 2 & 2B

WATTS BAR
FINAL SAFETY
ANALYSIS REPORT

POWERHOUSE
UNIT 2
ELECTRICAL
CONTROL DIAGRAM
SAFETY INJECTION SYSTEM
TVA DWG NO. 2-47W610-63-1 R26
FIGURE 6.3-1 SH 2(U2)

CAD MAINTAINED DRAWING



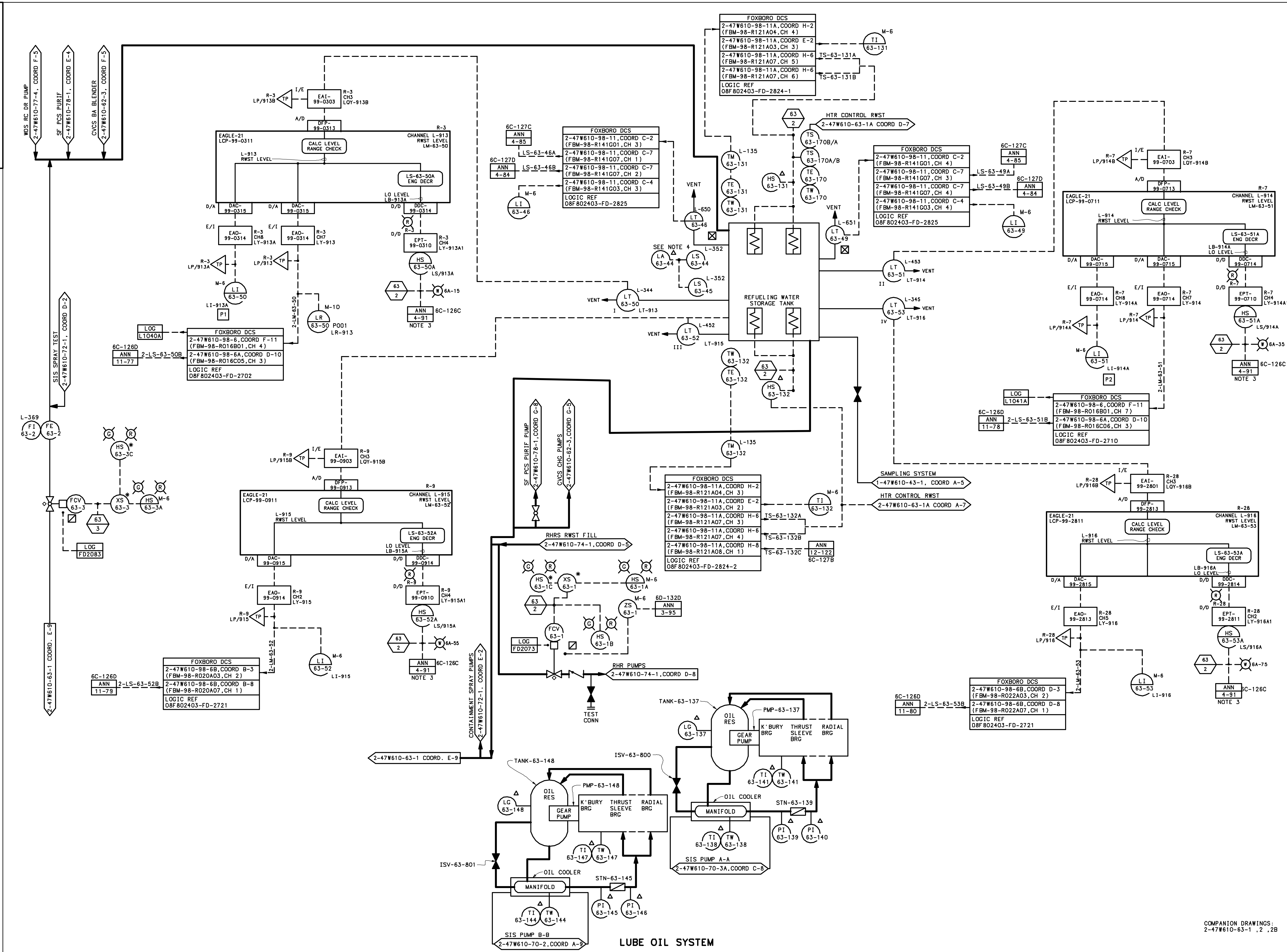
NOTES:

1. FOR GENERAL NOTES AND REFERENCE DRAWINGS SEE 1-47W610-63-1.
2. REFER TO TABLE A2 FOR ADDITIONAL DIGITAL COMPUTER POINTS (POWER AVAILABLE, PUMP RUNNING, ETC.).
3. TO OBTAIN THE COMPUTER POINT IDENTIFIER FOR A CORRESPONDING EAGLE-21 TEST POINT OR TVA UNID SEE TABLE 17 ON 1-47W610-99-6.

WATTS BAR
FINAL SAFETY
ANALYSIS REPORT

POWERHOUSE
UNIT 1
ELECTRICAL
CONTROL DIAGRAM
SAFETY INJECTION SYSTEM
TVA DWG NO. 1-47W610-63-1A R5
FIGURE 6.3-1-2 SH A

CAD MAINTAINED DRAWING



NOTES:

1. FOR GENERAL NOTES AND REFERENCE DRAWINGS SEE 2-47W610-63-1.
2. ALARM LIGHT 127B, IN WINDOW BOX 6C, WILL LIGHT FOR 2-TS-63-132C (RWST TEMP HI & RWST TEMP LO).

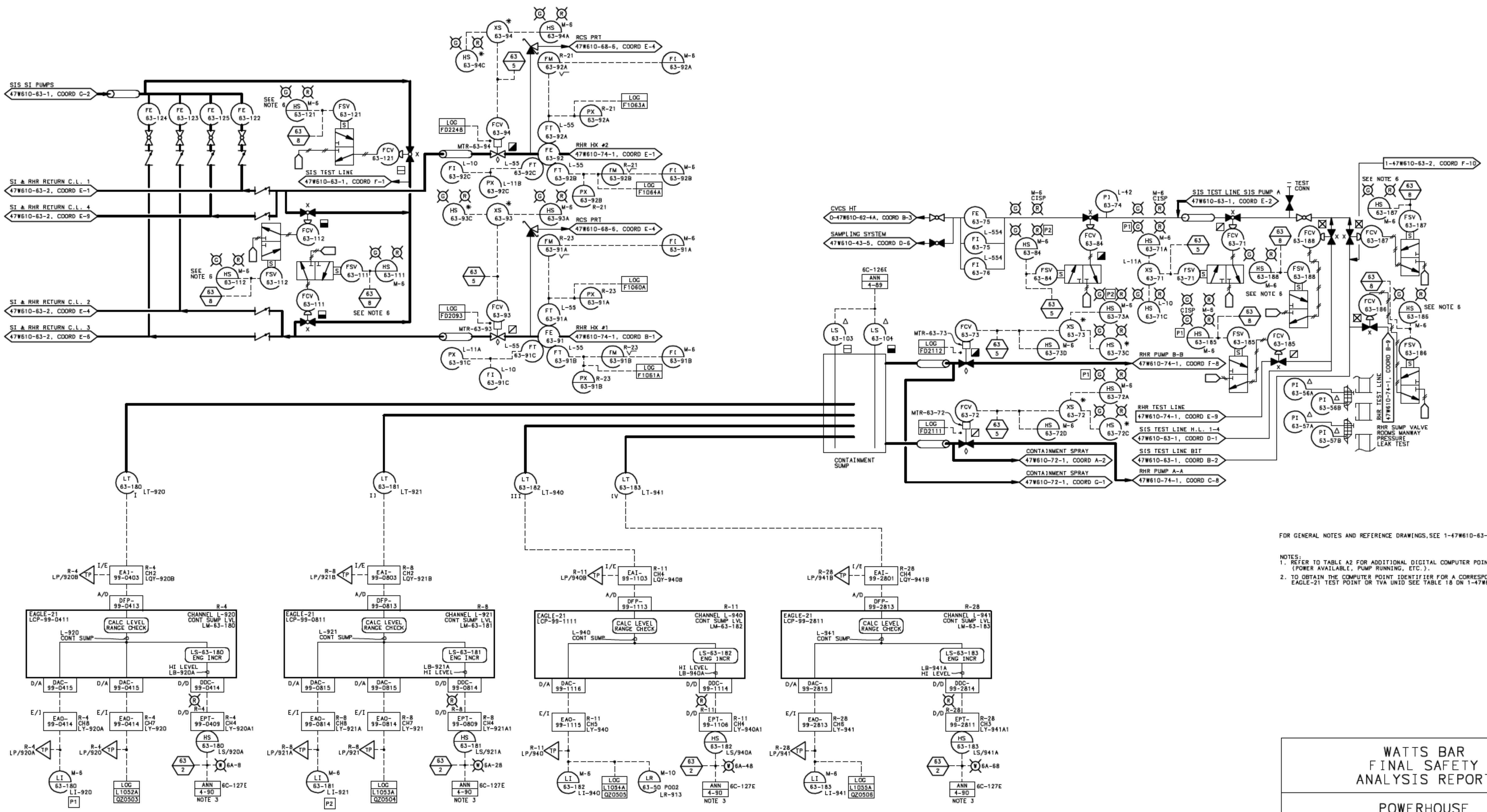
COMPANION DRAWINGS:
2-47W610-63-1 ,2 ,2B

UFSAR AMENDMENT 1

WATTS BAR
FINAL SAFETY
ANALYSIS REPORT

POWERHOUSE
UNIT 2
ELECTRICAL
CONTROL DIAGRAM
SAFETY INJECTION SYSTEM
TVA DWG NO. 2-47W610-63-1A R22
FIGURE 6.3-1-2 SH A(U2)

CAD MAINTAINED DRAWING



CONTAINMENT SUMP LEVEL MEASUREMENT FOR RHR RECIRCULATION

FOR GENERAL NOTES AND REFERENCE DRAWINGS, SEE 1-47W610-63-1.

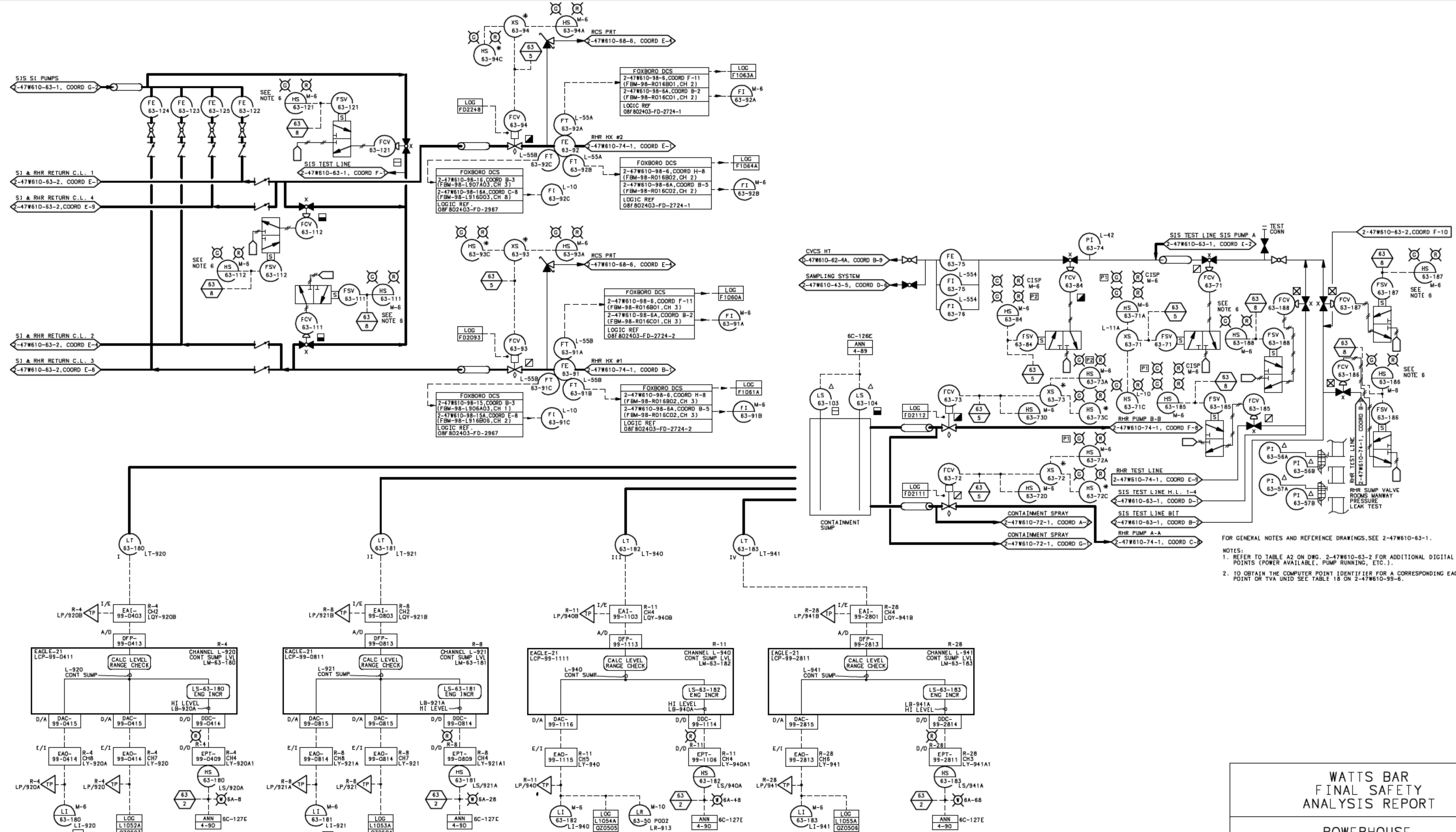
NOTES:
1. REFER TO TABLE A2 FOR ADDITIONAL DIGITAL COMPUTER POINTS (POWER AVAILABLE, PUMP RUNNING, ETC.).
2. TO OBTAIN THE COMPUTER POINT IDENTIFIER FOR A CORRESPONDING EAGLE-21 TEST POINT OR TVA UNID SEE TABLE 18 ON 1-47W610-99-6.

COMPANION DWGS:
1-47W610-63-1, 1A, 2

WATTS BAR
FINAL SAFETY
ANALYSIS REPORT

POWERHOUSE
UNIT 1
ELECTRICAL
CONTROL DIAGRAM
SAFETY INJECTION
TVA DWG NO. 1-47W610-63-2B R8
FIGURE 6.3-1-3 SH B

CAD MAINTAINED DRAWING



FOR GENERAL NOTES AND REFERENCE DRAWINGS, SEE 2-47W610-63-1.

NOTES:
1. REFER TO TABLE A2 ON DWG. 2-47W610-63-2 FOR ADDITIONAL DIGITAL COMPUTER POINTS (POWER AVAILABLE, PUMP RUNNING, ETC.).
2. TO OBTAIN THE COMPUTER POINT IDENTIFIER FOR A CORRESPONDING EAGLE-21 TEST POINT OR TVA UNID SEE TABLE 18 ON 2-47W610-99-6.

CONTAINMENT SUMP LEVEL MEASUREMENT FOR RHR RECIRCULATION

COMPANION DWGS:
2-47W610-63-1, 1A ,2

WATTS BAR
FINAL SAFETY
ANALYSIS REPORT

POWERHOUSE
UNIT 2
ELECTRICAL
LOGIC DIAGRAM
EMERGENCY GAS TREATMENT SYSTEM
TVA DWG NO. 2-47W610-63-2B R25
FIGURE 6.3-1-3 SH B(U2)

CAD MAINTAINED DRAWING

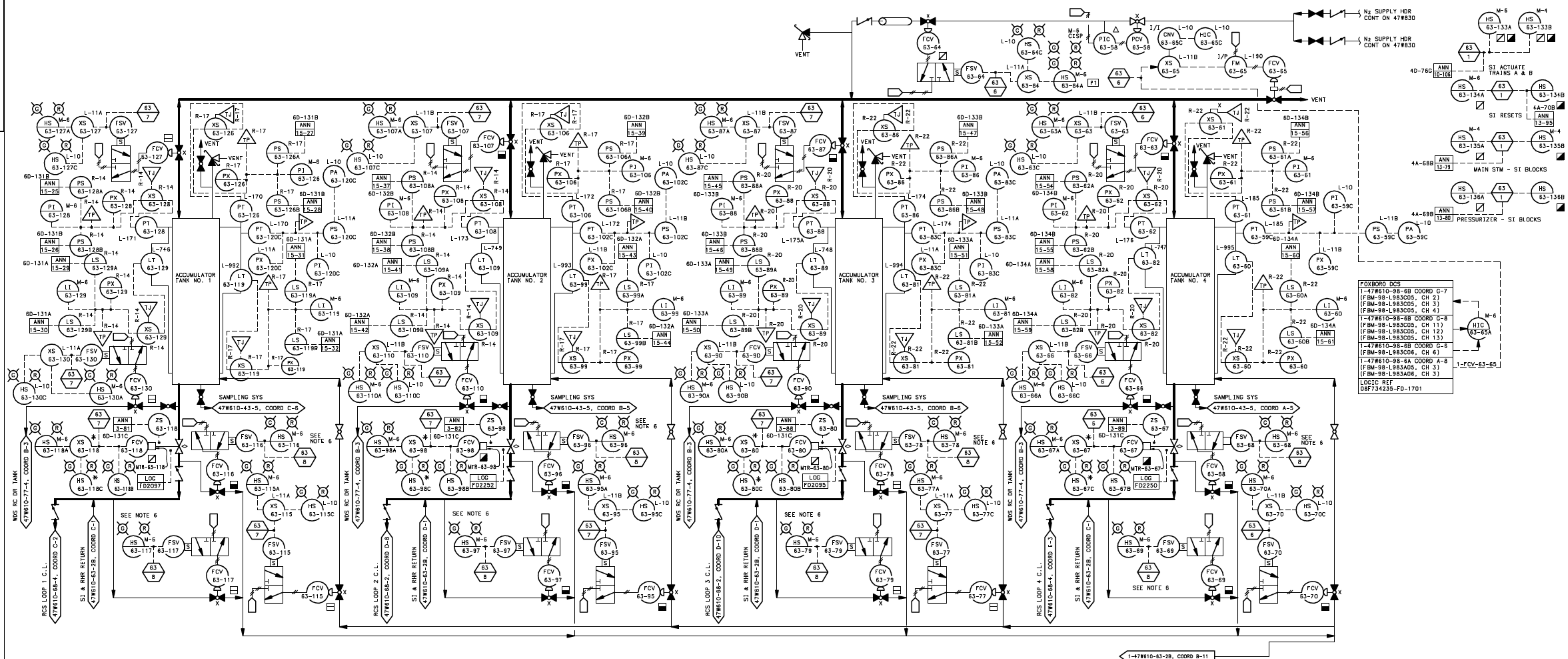


TABLE A2
DIGITAL COMPUTER POINTS

INSTRUMENT ID	POINT ID	DESCRIPTION	INSTRUMENT ID	POINT ID	DESCRIPTION
1-FCV-063-0001	FD2072	RWST TO RHR PUMP FCV	1-FCV-063-0080	FD2094	SIS ACCUM TANK 3 FLOW ISO VLV
1-FCV-063-0003	FD2082	SIS PUMPS RECIRC TO RWST ISV	1-FCV-063-0093	FD2092	RHR TO RCS 2 & 3 FCV
1-FCV-063-0004	FD2240	SIS PMP 1A-A MINIFLOW VALVE	1-FCV-063-0094	FD2247	RHR TO RCS 1 & 4 FCV
1-FCV-063-0005	FD2221	RWST TO SIS PMP FCV	1-FCV-063-0098	FD2251	SIS ACCUM TANK 2 FLOW ISO VLV
1-FCV-063-0006	FD2225	SIS PMP INLET TO CVCS CHG PMP	1-FCV-063-0118	FD2096	SIS ACCUM TANK 1 FLOW ISO VLV
1-FCV-063-0007	FD2076	SIS PMP INLET TO CVCS CHG PMP	1-FCV-063-0152	FD2078	SIS PUMP A-A OUTLET FCV
1-FCV-063-0008	FD2090	RHR HTX A TO CVCS CHARGING PUMP	1-FCV-063-0153	FD2232	SIS PUMP B-B OUTLET FCV
1-FCV-063-0011	FD2227	RHR HTX B TO SIS PUMP	1-FCV-063-0156	FD2080	SIS PUMP OUTLET RCS LOOP 1 & 3 HL
1-FCV-063-0022	FD2229	SIS PUMP COLD LEG INJECTION	1-FCV-063-0157	FD2236	SIS PUMP OUTLET RCS LOOP 2 & 4 HL
1-FCV-063-0025	FD2244	SIS BORON INJ TANK SHUTOFF VLV	1-FCV-063-0172	FD2234	RHR TO RCS HL 1 & 3 FLOW ISO VLV
1-FCV-063-0026	FD2086	SIS BORON INJ TANK SHUTOFF VLV	1-FCV-063-0175	FD2238	SIS PMP 1B-B MINIFLOW VALVE
1-FCV-063-0047	FD2074	SIS PUMP 1A-A INLET VALVE	1-HS-063-0010A	HD2004	SIS PUMP A-A MOTOR HS
1-FCV-063-0048	FD2223	SIS PUMP 1B-B INLET VALVE	1-HS-063-0015A	HD2042	SIS PUMP B-B MOTOR HS
1-FCV-063-0067	FD2249	SIS ACCUM TK 4 FLOW ISO VALVE	1-PMP-063-0010	XD2028	SIS PUMP A-A MOTOR
1-FCV-063-0072	FD2089	CNTMT SUMP TO RHR PUMP 1A-A	1-PMP-063-0015	XD2078	SIS PUMP B-B MOTOR
1-FCV-063-0073	FD2246	CNTMT SUMP TO RHR PUMP 1B-B			

SEE SHEETS 1, 1A & 2B FOR ASSOCIATED INSTRUMENTATION

NOTES:
1. REFER TO TABLE A2 FOR ADDITIONAL DIGITAL COMPUTER POINTS (POWER AVAILABLE, PUMP RUNNING, ETC.).

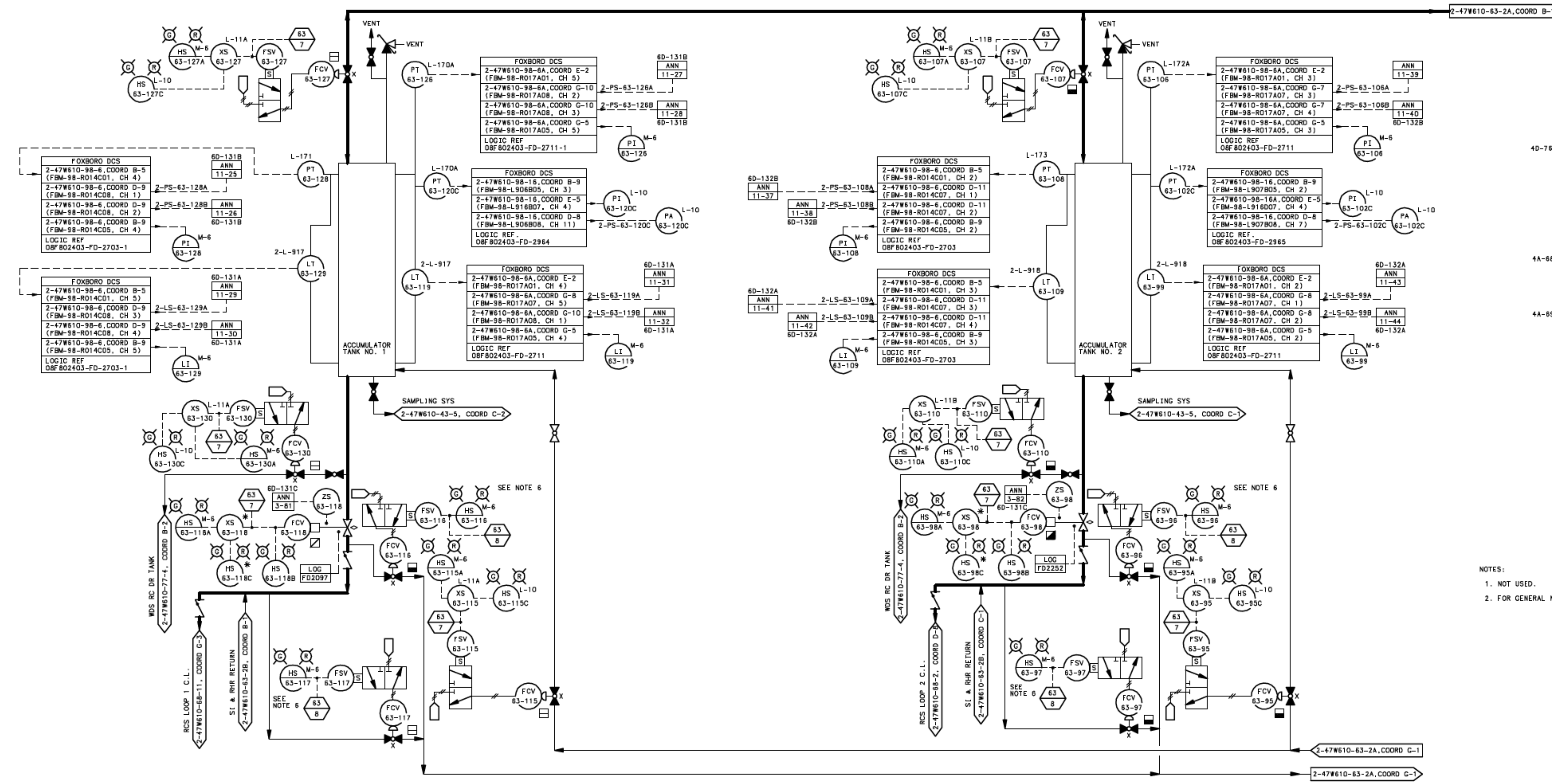
UFSAR AMENDMENT 1

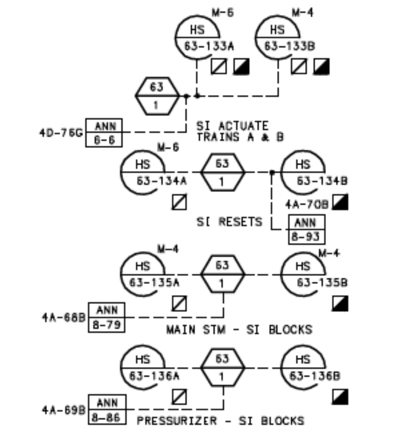
COMPANION DRAWINGS:
1-47W610-63-1, 1A, 2B

WATTS BAR
FINAL SAFETY
ANALYSIS REPORT

POWERHOUSE
UNIT 1
ELECTRICAL
CONTROL DIAGRAM
SAFETY INJECTION
TVA DWG NO. 1-47W610-63-2 R18
FIGURE 6.3-1-3

CAD MAINTAINED DRAWING





NOTES:

1. NOT USED.
2. FOR GENERAL NOTES AND REFERENCE DRAWINGS SEE 2-47W610-63-1.

TABLE A2
DIGITAL COMPUTER POINTS

INSTRUMENT ID	POINT ID	DESCRIPTION	SIGNAL
2-FCV-063-0001	FD2072	RWST TO RHR PUMP FCV	POWER AVAILABLE
2-FCV-063-0003	FD2082	SIS PUMPS RECIRC TO RWST ISV	POWER AVAILABLE
2-FCV-063-0004	FD2240	SIS PMP 2A-A MINIFLOW VALVE	POWER AVAILABLE
2-FCV-063-0005	FD2221	RWST TO SIS PMP FCV	POWER AVAILABLE
2-FCV-063-0006	FD2225	SIS PMP INLET TO CVCS CHG PMP	POWER AVAILABLE
2-FCV-063-0007	FD2076	SIS PMP INLET TO CVCS CHG PMP	POWER AVAILABLE
2-FCV-063-0008	FD2090	RHR HTX A TO CVCS CHARGING PUMP	POWER AVAILABLE
2-FCV-063-0011	FD2227	RHR HTX B TO SIS PUMP	POWER AVAILABLE
2-FCV-063-0022	FD2229	SIS PUMP COLD LEG INJECTION	POWER AVAILABLE
2-FCV-063-0025	FD2244	SIS BORON INJ TANK SHUTOFF VLV	POWER AVAILABLE
2-FCV-063-0026	FD2086	SIS BORON INJ TANK SHUTOFF VLV	POWER AVAILABLE
2-FCV-063-0047	FD2074	SIS PUMP 2A-A INLET VALVE	POWER AVAILABLE
2-FCV-063-0048	FD2223	SIS PUMP 2B-B INLET VALVE	POWER AVAILABLE
2-FCV-063-0067	FD2249	SIS ACCUM TK 4 FLOW ISO VALVE	POWER AVAILABLE
2-FCV-063-0072	FD2089	CNTMT SUMP TO RHR PUMP 2A-A	POWER AVAILABLE
2-FCV-063-0073	FD2246	CNTMT SUMP TO RHR PUMP 2B-B	POWER AVAILABLE

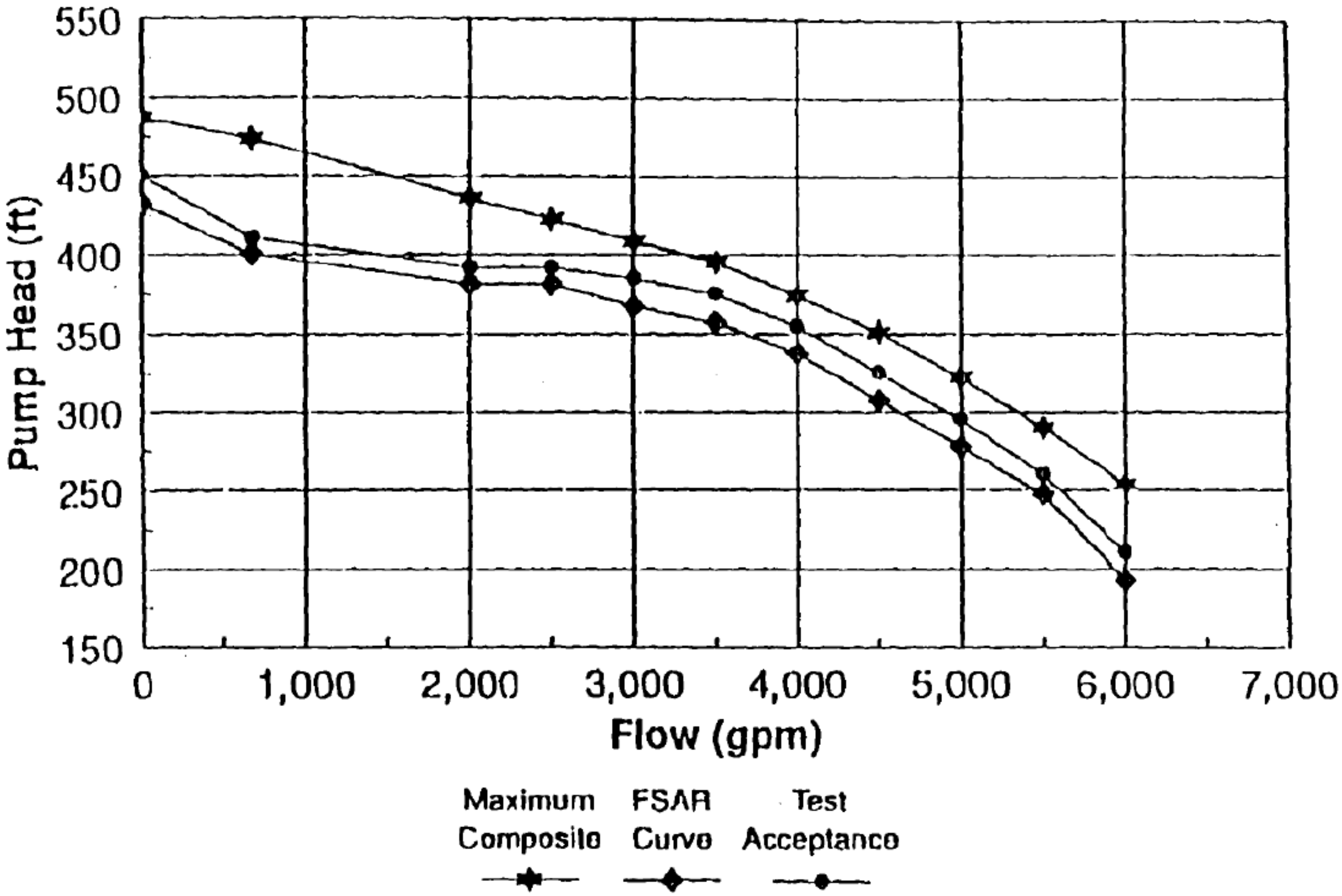
SEE COMPANION DRAWINGS FOR ASSOCIATED INSTRUMENTATION

INSTRUMENT ID	POINT ID	DESCRIPTION	SIGNAL
2-FCV-063-0080	FD2094	SIS ACCUM TANK 3 FLOW ISO VLV	POWER AVAILABLE
2-FCV-063-0093	FD2092	RHR TO RCS 2 & 3 FCV	POWER AVAILABLE
2-FCV-063-0094	FD2247	RHR TO RCS 1 & 4 FCV	POWER AVAILABLE
2-FCV-063-0098	FD2251	SIS ACCUM TANK 2 FLOW ISO VLV	POWER AVAILABLE
2-FCV-063-0118	FD2096	SIS ACCUM TANK 1 FLOW ISO VLV	POWER AVAILABLE
2-FCV-063-0152	FD2078	SIS PUMP A-A OUTLET FCV	POWER AVAILABLE
2-FCV-063-0153	FD2232	SIS PUMP B-B OUTLET FCV	POWER AVAILABLE
2-FCV-063-0156	FD2080	SIS PUMP OUTLET RCS LOOP 1 & 3 HL	POWER AVAILABLE
2-FCV-063-0157	FD2236	SIS PUMP OUTLET RCS LOOP 2 & 4 HL	POWER AVAILABLE
2-FCV-063-0172	FD2234	RHR TO RCS HL 1 & 3 FLOW ISO VLV	POWER AVAILABLE
2-FCV-063-0175	FD2238	SIS PMP 2B-B MINIFLOW VALVE	POWER AVAILABLE
2-HS-063-0010A	HD2004	SIS PUMP A-A MOTOR HS	HS IN/OUT
2-HS-063-0015A	HD2042	SIS PUMP B-B MOTOR HS	HS IN/OUT
2-PMP-063-0010	XD2028	SIS PUMP A-A MOTOR	POWER AVAILABLE
2-PMP-063-0010	XD2026	SIS PUMP A-A MOTOR	PUMP RUNNING
2-PMP-063-0015	XD2078	SIS PUMP B-B MOTOR	POWER AVAILABLE
2-PMP-063-0015	XD2077	SIS PUMP B-B MOTOR	PUMP RUNNING

COMPANION DRAWINGS:
2-47W610-63-SERIES

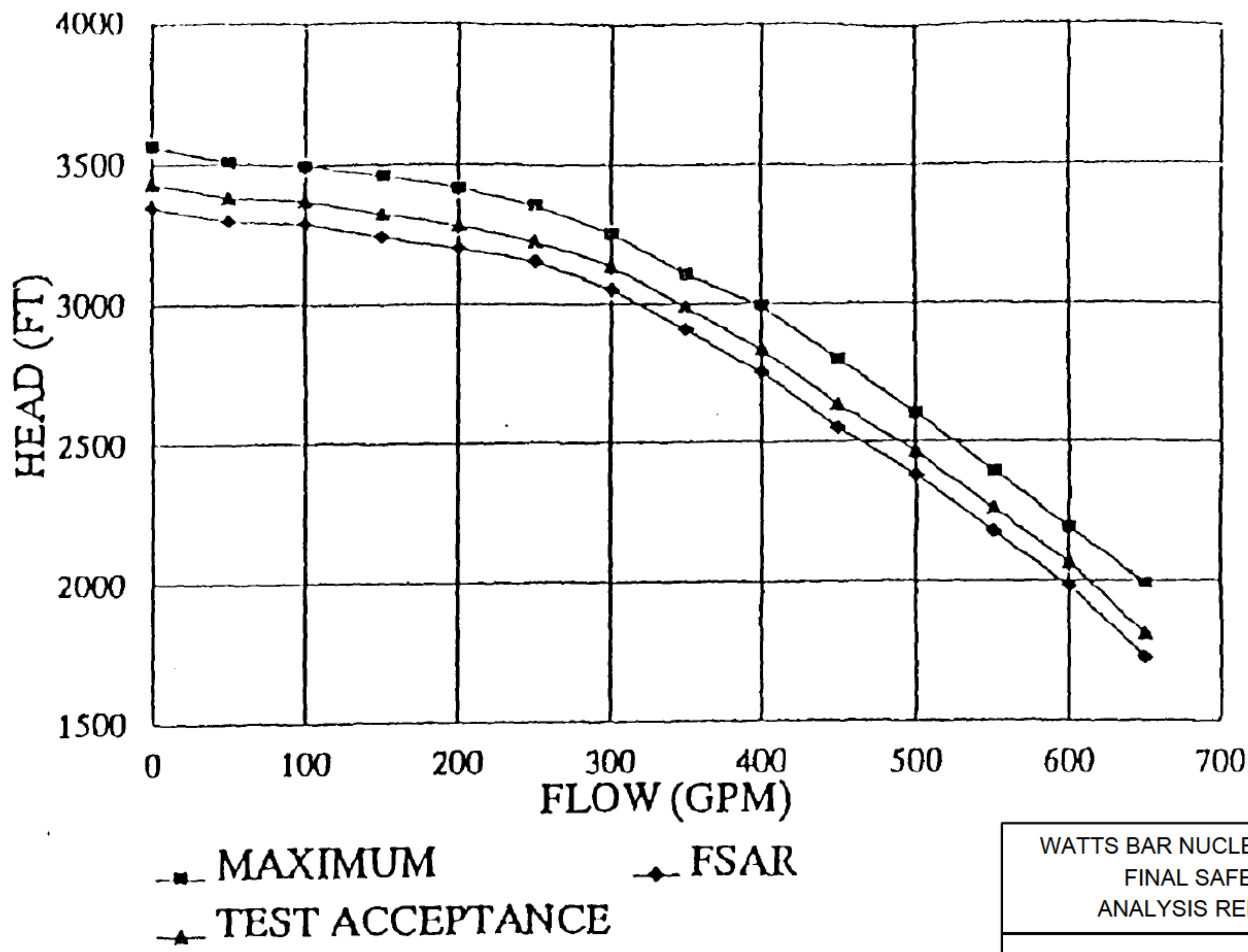
WATTS BAR
FINAL SAFETY
ANALYSIS REPORT

POWERHOUSE
UNIT 2
ELECTRICAL
CONTROL DIAGRAM
WASTE DISPOSAL SYSTEM
TVA DWG NO. 2-47W610-63-2 R14
FIGURE 6.3-1-3(U2)



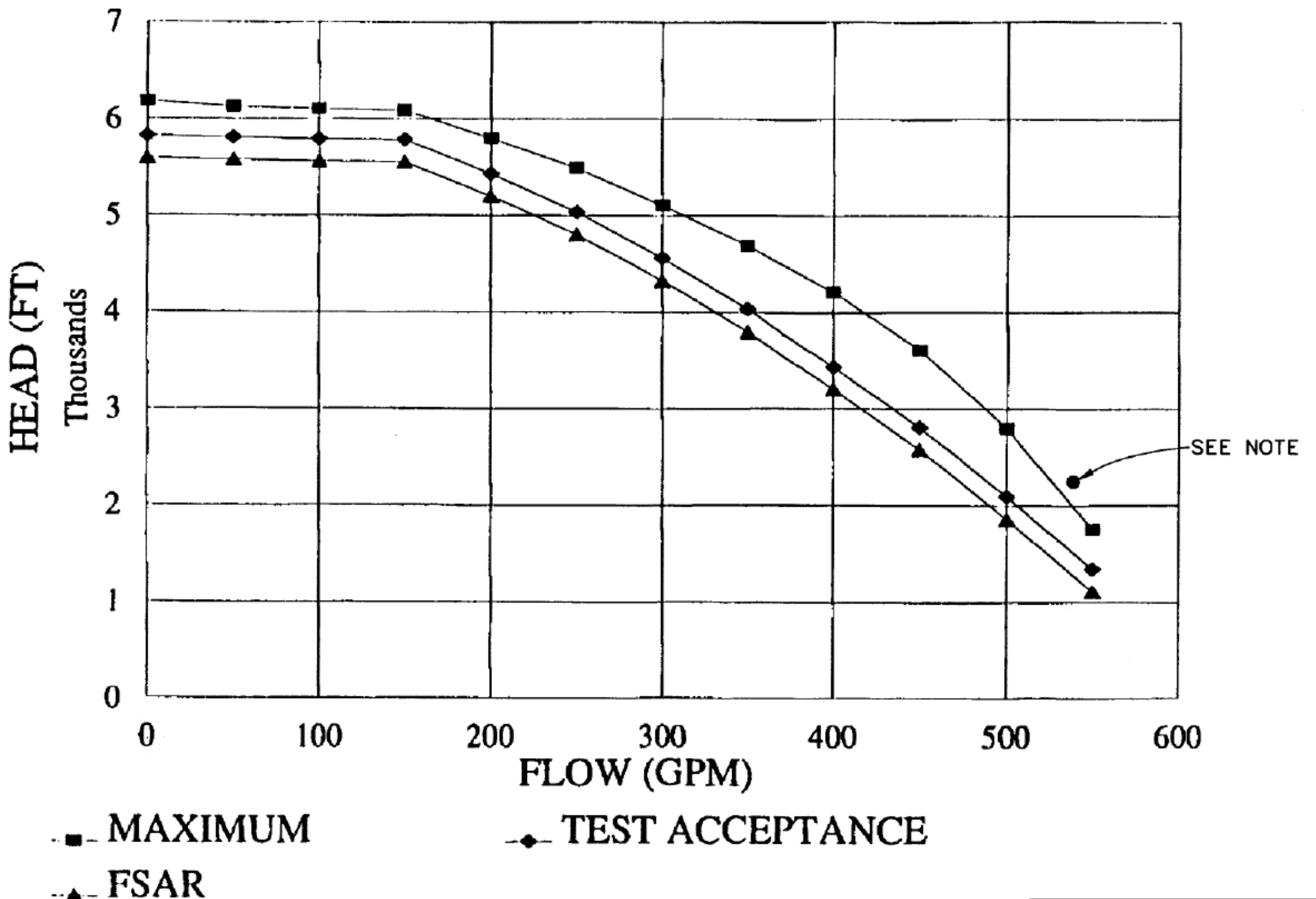
WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

Performance Curves for the
Residual Heat Removal Pumps
FIGURE 6.3-2



WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

Performance Curves for the
Safety Injection Pumps
FIGURE 6.3.3



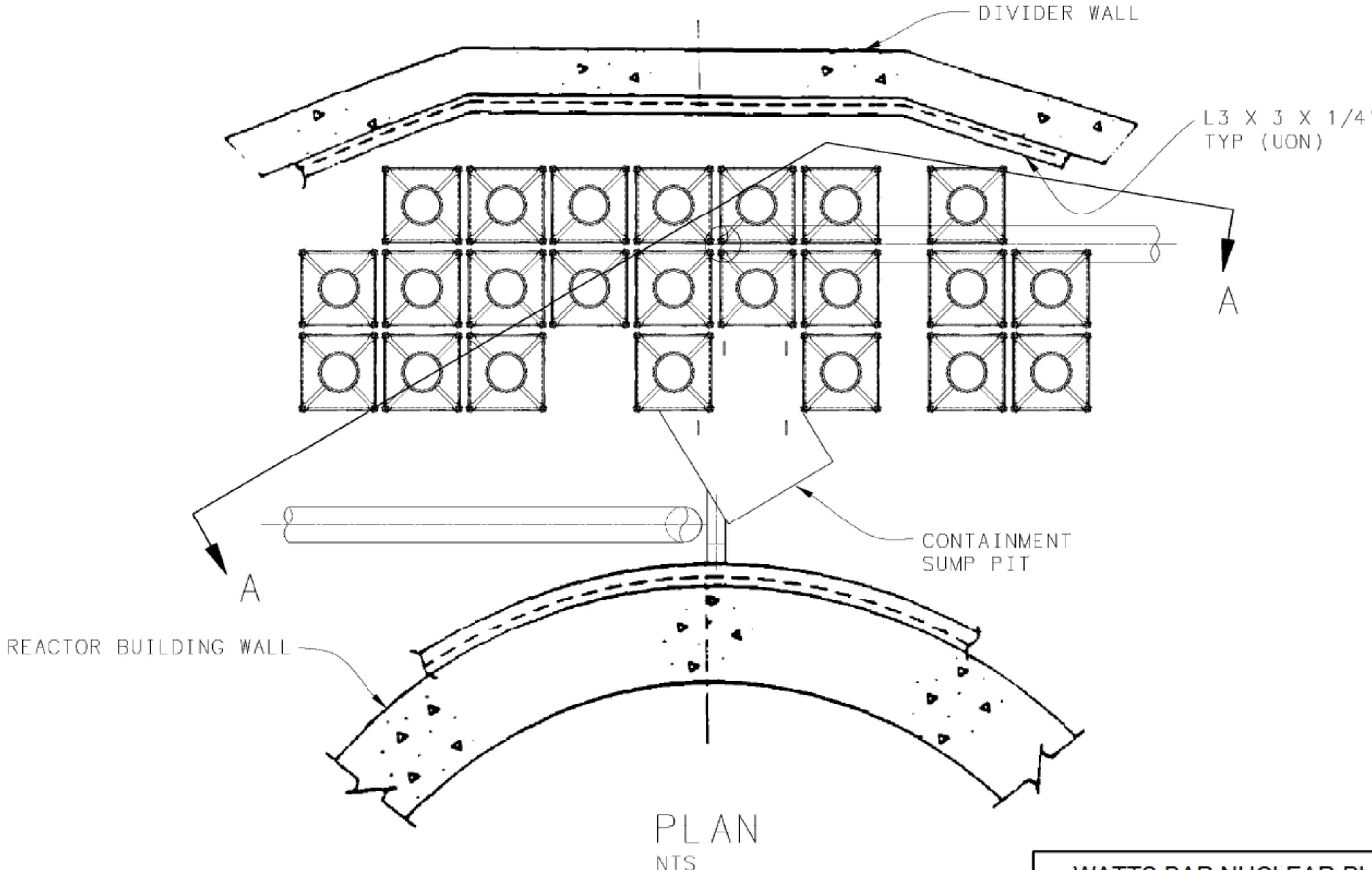
NOTES:
1. AS-FOUND TEST DATA REFLECTED SLIGHTLY HIGHER HEAD/FLOW VALUES THAN THE MAXIMUM CURVE. A BOUNDING AS-FOUND TEST POINT HAS BEEN EVALUATED AND DETERMINED TO BE ACCEPTABLE.

WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

Performance Curves for the
Charging Pumps
FIGURE 6.3-4

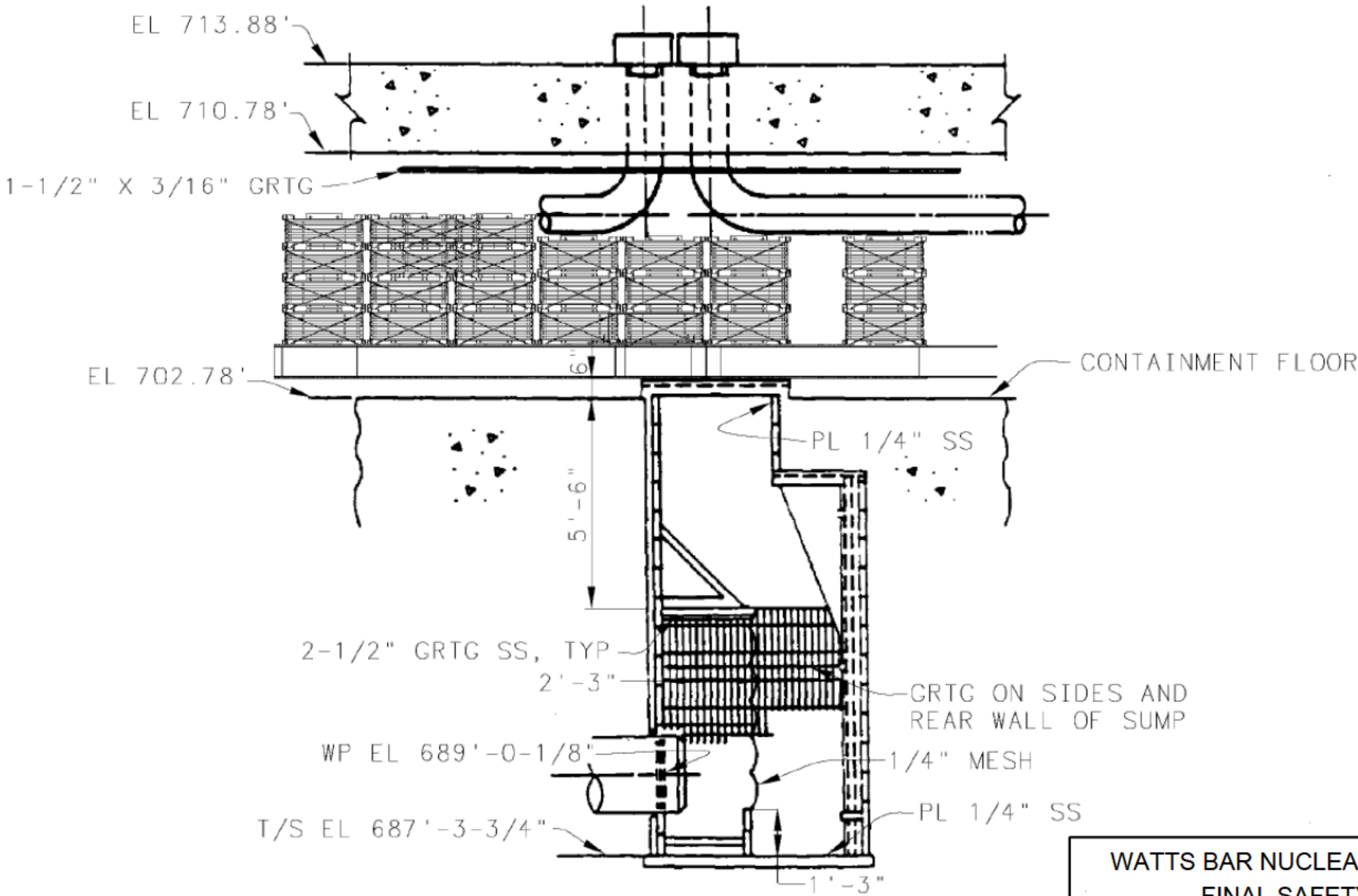
FIGURE 6.3-5 Sheets 1 and 2

DELETED



WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

Containment
Sump
FIGURE 6.3-6



SECTION A-A

NTS

WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

Containment
Sump
FIGURE 6.3-6A

6.4 HABITABILITY SYSTEMS

The Main Control Room Habitability System (MCRHS) is the set of equipment, components, supplies, and other features, including the building enclosure, provided to ensure that a suitable environment is maintained for personnel and equipment in the MCRHS area for safe, long-term occupancy during normal and emergency operations of the plant.

The Main Control Room Habitability Zone (MCRHZ) is the envelope of spaces which are maintained habitable by pressurization to 1/8 inch water gage minimum above atmospheric to minimize infiltration of airborne contaminants which may be present outside the pressure boundary. It is also called the MCRHS area.

6.4.1 Design Bases

Design bases of the system include:

1. The capability to withstand the safe shutdown earthquake.

2. The capability to continue to function properly following any single active failure.

3. The capability to continue to function during such outside environmental conditions as the maximum possible flood or the design basis tornado.

4. The capability to detect presence of smoke in the air intake and isolate the MCRHZ.

5. The capability to shield MCR personnel from radiation sources and detect and limit the introduction of airborne radioactive contamination such that exposure of MCR personnel will not exceed limits specified in Appendix A to 10 CFR 50, General Design Criterion 19.

6. The capability to permit safe shutdown of the plant from within the MCRHS area following an accident, including the design basis loss-of-coolant accident (LOCA).

6.4.2 System Design

6.4.2.1 Definition of MCRHS Area

The MCRHS area includes all rooms on plan Elevation 755 of the Control Building (refer to the equipment plans presented in Section 1.2). All rooms to which MCR personnel may require access during emergency operations are included within this envelope. The MCR requires continuous occupancy. Other rooms in the MCRHS area which may require less frequent access include the kitchen, toilet facilities, technical support center (TSC), NRC office, mechanical equipment room, offices, conference rooms, locker room, relay room, and DPSO shop.

All controls and displays necessary to bring the plant to a safe shutdown condition are included within the MCRHS area. Emergency food and water are provided as necessary during emergencies. Medical supplies are housed within the MCR. Toilet and kitchen facilities which may be required by MCR personnel are included also. Heating, ventilating, air conditioning, and air cleanup components to which access may be necessary are enclosed within the MCRHS area.

6.4.2.2 Ventilation System Design

The Control Building Heating, Ventilating, Air Conditioning, and Air Cleanup (HVACAC) System design is described in detail in Section 9.4.1. Flow diagrams, logic diagrams, control diagrams, and component data are also included in that section.

6.4.2.3 Leak Tightness

The flow rate necessary to maintain the MCRHS area at the required positive pressure is determined by the leakage characteristics of the MCRHS enclosure. The pressurization flow rate in emergency modes of operation is limited by the permissible dose set forth in 10 CFR 50, Appendix A, Criterion 19. Analyses indicate that if a pressurization flow rate in excess of 711 cfm is utilized, the dose to MCR personnel increases. Thus, a low leakage MCRHS enclosure is required.

Although no infiltration is expected from interfacing areas, an infiltration flow rate is calculated to conservatively determine the dose in the MCRHS area. The infiltration flow rate is limited by the permissible dose set forth in 10 CFR 50, Appendix A, Criterion 19. Analysis indicates that the calculated infiltration rate is acceptable.

The enclosure is formed by the:

1. Monolithic reinforced concrete floor, walls and roof described in Section 3.8.4.

2. Metal pressure barrier beneath each control room console.

3 Low leakage seals for all electrical lines penetrating the enclosure.

4. Low leakage doors and door seals.

5. Low leakage ventilation system isolation dampers.

This enclosure is virtually insensitive to wind effects since only a small part of each end of the Control Building and the roof are exposed to the outside. Practically no Control Building penetrations exist on the building interfaces to the outside.

The walls, floors, and roof of the Control Building are of monolithic concrete construction. Few leakage paths exist in this type of construction.

Penetrations of the enclosure are provided with low leakage seals. Beneath each console in the MCR, a welded steel pressure barrier is provided. Electrical lines penetrating this barrier or any other portion of the MCRHS enclosure are provided with low leakage seals to restrict exfiltration and infiltration. Doors and weather stripping with low leakage characteristics are installed in doorways which penetrate the MCRHS enclosure. In addition, dampers in ducts which interface areas adjacent to the MCRHS enclosure are provided with operators and low leakage seals to provide a positive barrier to exfiltration and infiltration.

A survey of potential leakage paths was conducted to ensure that the amount of exfiltration from the MCRHS area is small enough that the required emergency pressurization flow rate does not exceed the limiting value of 711 cfm. The potential leakage paths and the expected exfiltration via each path at a minimum MCRHS positive pressure of 1/8-inch w.g. (water gage) are summarized in Table 6.4-1 for each mode of MCRHS operation. Refer to Section 6.4.3 for a discussion of the operating modes of the MCRHS.

A survey of the infiltration leakage was taken to ensure that the MCRHS area dose would be within allowable limits. The potential and expected infiltration leakage for each path at 1/8-inch w.g. during the emergency mode is summarized in Table 6.4-2.

6.4.2.4 Interaction with Other Zones and Pressure-Containing Equipment

6.4.2.4.1 Other Ventilation Zones

Portions of the Auxiliary Building and Turbine Building are adjacent to the MCRHS area on the north and south sides respectively. In addition, the MCRHS area interfaces with other areas of the Control Building. There are few penetrations of the MCRHS enclosure except those entering the spreading room which is located directly below the MCRHS area. No adverse interaction that may enhance the transfer of toxic or radioactive gases into the MCRHS area is expected with any of these zones.

The north wall, i.e., q-line wall, of the MCRHS area separates the MCRHZ from the shutdown board rooms, the Elevation 757.0 floor of the Auxiliary Building. Elevation 757.0 of the Auxiliary Building is maintained at a slightly positive pressure during normal operation of the plant. This positive pressure does not exceed the positive pressure level maintained in the MCRHS area. During emergency operation initiated from a control room isolation (CRI) signal, the shutdown board room pressurizing air supply fans are automatically de-energized by the CRI. Therefore, no significant pressure differential will ever exist between this part of the Auxiliary Building and the MCRHS area which could promote migration of airborne radioactive contamination or toxic gases into the MCRHS area.

The south wall, i.e., the n-line wall, of the MCRHS area is adjacent to the Turbine Building. The Turbine Building general ventilation system is not safety-related and is not designed to operate in an emergency. The Turbine Building is maintained at atmospheric pressure during normal operation with a slight negative pressure being provided by the roof ventilators to induce outdoor air through louvers and dampers. Thus, no significant pressure differentials are expected which could overcome the outward-acting positive pressure maintained in the MCRHS areas.

The spreading room, at Elevation 729, is directly below the central portion of the MCRHS area. This room is normally maintained at a slightly negative pressure with respect to atmospheric pressure. Upon MCRHS area isolation, both the air supply and exhaust to this room are stopped. Isolation dampers are used to isolate the room from the outside. Therefore, the spreading room is at approximately atmospheric pressure, or slightly negative, so any leakage between the MCRHS area and the spreading room is exfiltration from the MCRHS area.

The areas at the east and west ends of the Control Building which are immediately below the MCRHS area are open to the Turbine Building and, therefore, are at the same pressure as the rest of the Turbine Building. As discussed previously, no adverse pressure differentials are expected in the Turbine Building.

6.4.2.4.2 Pressure-Containing Equipment

In general, pressure-containing equipment or piping is not permitted in the MCRHS area; except for several small hand-held fire extinguishers and self-contained breathing air apparatuses which are stored in the MCRHS area to provide for habitability during emergencies.

Zones interfacing with the MCRHS and which contain high-pressure equipment are portions of the Turbine Building and the areas at the east and west ends of the Control Building directly below the MCRHS area. These areas contain steam piping and feedwater lines and occasional transient compressed gas cylinders which may be brought in for maintenance activities. No adverse pressure differentials are expected from failure of these lines since any significant differential pressure would result in rupture of the glass sections of the Turbine Building walls and all common walls and floor between the two buildings are seismic Category I with sealed penetrations. Areas of the Auxiliary Building which contain high-pressure equipment have no direct interface with the MCRHS area.

6.4.2.5 Shielding Design

Refer to Section 12.3.2.

6.4.2.6 Control Room Emergency Provisions

The MCRHS Area is designed for long-term occupation by personnel required during emergency operation. Supplies and emergency equipment are stored in the habitability area, except that operator protective clothing is stored in the operations support center, medical supplies are available from the medical emergency response team, and food is made available from off-site sources by the emergency control center.

6.4.2.7 MCRHS Fire Protection

The Fire Protection System is described in Section 9.5.1.

6.4.3 System Operational Procedures

The MCRHS operates in one of three modes to maintain the internal environmental conditions commensurate with outside conditions. The three operating modes are the normal mode, the emergency mode, and the extreme emergency mode.

Normal Mode

In the normal operations mode, all doors into the MCRHS area are normally closed and are used only for necessary ingress and egress during which the air handling unit fans provide outside air to the MCRHS area. Since airflow is balanced in conjunction with air outflow from the MCR and adjacent rooms, the pressure in this area remains positive. Balancing dampers are provided to keep the MCRHZ pressure at a minimum of 1/8 inch water gage above outside atmosphere and adjacent areas. The positive pressure in the MCRHZ with respect to the surrounding areas is monitored and alarmed in the MCR. Upon receipt of an abnormal indication, the MCR operator will take corrective action to reestablish the required differential pressure.

Emergency Mode

The emergency operations mode is utilized for any condition requiring MCRHS isolation. Isolation of the MCRHS area occurs automatically upon the actuation of a safety injection signal from either reactor unit or upon indication of high radiation, or smoke concentrations in the outside air supply stream to the building. Isolation of the MCRHS area may also be accomplished manually at any time by the control room operators.

Upon receipt of a signal for MCRHS isolation, the following conditions directly affecting the MCRHS are implemented automatically:

1. Both Control Building emergency air cleanup fans operate to recirculate a portion of the control room air conditioning system return air through the cleanup trains composed of HEPA filters and charcoal adsorbers. One of the emergency air cleanup fans is subsequently placed in the standby mode by the operator.

2. Both Control Building emergency pressurizing air supply fans operate to supply a reduced stream of outside air to the MCR air conditioning system to keep the MCRHS area pressurized, relative to the outdoors and adjacent areas, thereby minimizing the inleakage of unprocessed or contaminated air. This fresh air is routed through the emergency air cleanup trains. One of the two emergency pressurizing fans (and its associated emergency air intake) is subsequently placed in the standby mode by the operator.

3. The exhaust fan in the toilet rooms is stopped and double isolation dampers are closed to prevent the inflow of unfiltered outside air to the MCRHS area.

4. The shutdown board rooms pressurizing air supply fans in the Auxiliary Building Elevation. 757.0 are automatically de-energized.

In addition, the following conditions which normally can indirectly affect the MCRHS are automatically implemented:

1. The spreading room supply and exhaust fans are stopped and the operating battery room exhaust fan continues to run.

2 Double isolation dampers in the spreading room supply duct and a single isolation damper in the exhaust duct will close to prevent infiltration of outside air to the spreading room.

3. The normal operating electric board room air handling units continue to supply the same outside air quantity to the Control Building lower floors.

4. Automatic isolation valves close to stop the flow of unfiltered pressurizing air to the MCRHS.

In the emergency mode, determination of the appropriate emergency pressurizing fan to place in standby is based on the operator's judgement.

The operator has the capability to compare radiation levels at the two emergency air intakes, as described under the extreme emergency operating mode below.

In the emergency operations mode, ingress and egress in the MCRHS area is administratively restricted to essential movement and takes place through one of the designated entryways on the Elevation 755 level. During this mode, a maximum of 711 cfm of outside air is drawn in and mixed with 3289 cfm of recirculated air, drawn through an air cleanup unit, and processed in the MCR air handling unit for proper humidity and temperature levels. In this mode, air leakage resistance from the MCRHS area will assure the maintenance of a minimum 1/8-inch w.g. positive pressure in the MCR habitability zone with the doors closed. Such a capability is demonstrated during preoperational test and periodically thereafter.

<u>Extreme Emergency Mode</u>

The Control Building outside air intakes are provided with radiation monitors that indicate and annunciate in the MCR. This instrumentation allows the operator to compare radiation levels at the two emergency air intakes and select the less contaminated intake for operation during emergency conditions. If the intake monitors indicate that extremely high air contamination levels exist outside (e.g., post-LOCA conditions approaching Regulatory Guide 1.4 releases which prohibit outdoor movement), the air intake having the lower contamination level is chosen and the extreme emergency operations mode is utilized. It is not required, however, from a dose standpoint, that the less contaminated air intake be chosen initially (see Section 15.5).

During the extreme emergency operations mode, necessary ingress and egress is restricted to just one entryway on Elevation 755. All other doors from the MCRHS area are sealed with heavy tape to reduce the outleakage from the MCRHS area. Such a practice reduces air leakage through the doorjamb seals. This procedure provides a greater leakage margin during critical periods of the emergency and maintains the entire MCRHS area above the minimum 1/8 inch w.g. positive pressure.

The restricted ingress or egress under control room operator surveillance for all emergency modes minimizes the unfiltered airflow into the MCRHZ to approximately 10 cfm.

The basis for this position is that during this brief period when the door is open the air flow will be from inside the MCRHS area to the outside. Since the pressure will never be less than atmospheric in the MCRHS area during this interval, little contamination is expected to leak into the MCRHS area. In such circumstances the makeup air input of 711 cfm to the MCRHS area is considered sufficient to prevent unfiltered air infiltration into the MCRHS area.

6.4.4 Design Evaluations

6.4.4.1 Radiological Protection

Refer to Section 12.3.

6.4.4.2 Toxic Gas Protection

The evaluation of MCR habitability included consideration of possible hazards created by accidental release of potentially toxic chemicals. The evaluation considered chemicals stored both onsite and offsite within a 5-mile radius. Possible shipments of toxic chemicals by barge, rail, or road routes within a 5-mile radius were also considered.

Watts Bar Steam Plant is an offsite storage location for potentially hazardous chemicals within the 5-mile radius considered. Chemicals stored at the steam plant include acetone, anhydrous ammonia, carbon dioxide, methanol, nitrogen, sulfuric acid, isopropyl alcohol, calcium oxide, bentonite, soda ash, salt (NaCl), sodium sulfite, dichlorodifluoromethane, freon, acetylene, and sodium hypochlorite. However, only very small quantities of the chemicals, excluding carbon dioxide and nitrogen, are stored at the steam plant. Since nitrogen and carbon dioxide are asphyxiants and large concentrations of these chemicals are required to create a hazard, and since only small quantities are stored, which are bounded by the on-site quantities, no hazard to MCR personnel at Watts Bar Nuclear Plant is foreseen.

The potable water supply is obtained from the Watts Bar Utility District located on State Route 68 approximately two miles from Watts Bar Nuclear Plant. The utility maintains a relatively small inventory of chlorine for use in the treatment process. However, this quantity is less than the quantity requiring analysis per Regulatory Guide 1.78 and is not a hazard to MCR operators.

The only known shipments of potentially toxic chemicals transported past the site by road route are the small quantities of chemicals shipped to Watts Bar Steam Plant as discussed above. These are transported via State Route 68 which passes within 1 mile of Watts Bar Nuclear Plant. The frequency of shipment is less than the guideline value given in NRC Regulatory Guide 1.78 for all of the chemicals except carbon dioxide and nitrogen. The quantity of each shipment is small for all of the chemicals. Therefore, no hazard to MCR personnel is expected.

The only rail line within a five-mile radius is the spur track which serves the plant itself. Any chemicals transported to the site were evaluated as stored on site. Barge traffic passing the plant site is discussed in Section 2.2.2.2. Release of these commodities does not result in introduction of toxic gases to the MCRHS area. The shipments are not considered to pose a hazard to MCR personnel unless smoke generated by a barge fire should be blown toward the Control Building air intake. If this should occur, however, ionization-type smoke detectors in the intakes initiate MCRHS isolation and preclude entrance of combustion products into the MCRHS area. The small amount of smoke which could possibly enter the area prior to isolation is quickly removed by the air cleanup units. Therefore, MCR habitability is not degraded by accidents involving these products.

Chemicals stored on site which may be potentially hazardous to MCR personnel include, but are not limited to, argon, carbon dioxide, ammonium hydroxide, hydrazine, glutaraldehyde, freons, hydrogen, nitrogen, sodium hypochlorite, ethanolamine and commercially compounded chemicals used to treat the water systems. It was determined that the remaining chemicals do not constitute a hazard to control room personnel since they are stored in small quantities, are liquids with low vapor pressures at normal temperatures or are stored as solids.

Analysis was performed for the potentially hazardous chemicals utilizing the approach outlined in NRC Regulatory Guide 1.78, "Assumptions for Evaluating the Habitability of a Nuclear Power Plant Control Room During A Postulated Hazardous Chemical Release." Major assumptions included Pasquill stability Class G and adverse wind direction. Wind speed was selected as 1 meter per second based on Regulatory Guide 1.4, "Assumptions Used for Evaluating the Radiological Consequences of a Loss of Coolant Accident for Pressurized Water Reactors."

A 24-ton capacity carbon dioxide tank is located in the yard approximately 40 feet from the east end of the Control Building. Analysis indicated that upon a carbon dioxide release, the maximum concentration in the control room would be less than the 1% maximum per Regulatory Guide 1.78.

Ammonium hydroxide and hydrazine are stored in the Turbine Building in 625 gallon and 250 gallon tanks, respectively. Upon a spill of either of these tanks, most of the liquid would drain into the Turbine Building sump and any vapors given off would be dispersed by the Turbine Building ventilation system. Analysis for the ammonium hydroxide, which has a significantly higher vapor pressure than hydrazine, shows that the control room would not become uninhabitable due to ammonia vapors drawn in from outside.

Potential releases of ethanolamine, used for steam generator corrosion control and glutaraldehyde, used as a biocide in the component cooling water system, were also analyzed and determined to have no affect on the MCR.

Sodium hypochlorite may be stored in the Sodium Hypochlorite Building. The solution has a pH of 11-13 at this concentration. In order for chlorine to form upon a spill, the pH would have to be lowered to about a pH of 4. Since no acidic solutions are present to cause this reduction in pH, no chlorine would be given off, any liquid would be contained within the Sodium Hypochlorite Building, and any vapors would be dispersed by the building ventilation system.

Chemical compounds injected near the Intake Pump Station and used to treat the raw water systems on site were similarly evaluated for potential affects on control room habitability. Analysis confirmed that subsequent to a release of the chemical tank contents, potential control room concentrations of these chemicals were within acceptable limits.

Hydrogen is stored in 54,800 scf tanks at the hydrogen trailers south of the switchyard, and nitrogen is stored in a 856,612 scf tank in the yard east of the Control Building. Analysis has shown that gases drawn into the control room from these tanks would not prevent maintaining an oxygen level above 20%, which meets a 19.5% safe oxygen level. Likewise, analysis has shown that this safe oxygen level would not be affected by a release of refrigerant R-11, R-12, or R-22 used in air-conditioning systems, or freon 1301, as used in some fire extinguishing systems on site.

It was therefore concluded that no hazard to control room habitability is posed by any of the chemicals stored on site, offsite within a 5-mile radius, or transported by the site by barge, rail, or road within a 5-mile radius.

6.4.5 Testing and Inspection

Tests and inspections conducted on the MCR habitability system are mainly concerned with the HVACAC system, the capability to keep a positive pressure within the MCRHS area, and the operation of the airborne hazards monitors. The scope includes preoperational and periodic tests. The preoperational tests objectives are identified in Chapter 14.

6.4.6 Instrumentation Requirements

Several kinds of instrumentation are utilized in the MCRHS. Beta radiation sensors and smoke monitors are installed in the makeup air intake duct to detect harmful concentrations of these airborne hazards. Thermostats and humidistats are positioned in the MCR to control HVACAC system operations. Static pressure differential sensors are installed in the air cleanup units to measure the pressure change across each air purification element bank.

Temperature sensors are utilized for duct heater element control to keep the incoming air above specified limits. Flow sensors are installed downstream from each MCR air handling unit to sense the presence of substandard air flows and initiate startup of the standby redundant HVACAC train. Differential pressure transmitters sense the pressure in the MCRHZ with respect to the adjacent areas and differential pressure switches in the transmitter instrument loop alarm on low pressure. During control room isolation, these switches also start the standby air cleanup unit and associated emergency pressurization system on low differential pressure.

Instrumentation details of the control room HVAC system are provided in Section 9.4.1. General descriptions of safety related plant instrumentation are provided in Section 7.1. The detailed instrumentation drawings of the control room HVAC system are listed in Table 1.7-1.

REFERENCES

None

WBN

TABLE 6.4-1

AIR LEAKAGE (EXFILTRATION) PATHS IN THE WATTS BAR MCRHS AREA CONTROL ROOM

Leakage Path	Flow Rate[4] (cfm) Normal Operation Mode	Emergency Mode	Extreme Emergency Mode
Doors	215.1	215.1	215.1[6]
Toilet Damper	825[1]	9.7	9.7
Spreading Room Dampers	1200[1]	16.5	16.5
Other Dampers	3.1	3.1	3.1
Penetrations (electrical, piping, and ducts)	0.1	0.1	0.1
Concrete Walls, Floor, and Roof	0.2	0.2	0.2
Duct Leakage (to outside of MCRHS)	24.4	24.4	24.4
Total[2]	2267.9	269.1	269.1[6]
Air Intake,[3] maximum	3200[5]	711	711
Net Excess Capacity	932.1	441.9	441.9

NOTES:

1 During normal operation, this flow path is normally open.

2 If the toilet exhaust fan or the spreading room supply fan fails to shut down during emergency mode concurrent with isolation damper failing open, a maximum of 24 cfm additional out-leakage may occur.

3 During both emergency modes, the ventilation supply is isolated with butterfly valves.

4 All numbers rounded to the nearest tenth.

5 Flowrate is conservative since the values, recorded during preoperational tests while maintaining +1/8" w.g. pressure in the MCR, were smaller.

6 Doors will be taped; therefore, the leakage is conservatively stated.

WBN

TABLE 6.4-2

AIR LEAKAGE (INFILTRATION) PATHS IN THE WATTS BAR MCRHS AREA CONTROL ROOM

Leakage Path	Flow Rate (cfm)
Door into Turbine Building (for egress/ingress)	10.0[(1)]
Emergency Pressurizing System Discharge Duct	0[(3)]
Control Air for Fire Protection	2.0
Pneumatically Operated Dampers and Valves	24.0[(2)]
Pneumatically Operated Instruments	1.0
Normal Pressurizing Duct	0[(3)]
Battery Room Exhaust	1.8
Safety Margin	36.2
Total	75.0
Initial Use of Pneumatic Valves and Dampers	24.0
Steady-State Total	51.0

NOTES:

1 To account for the possible increase in air exchange due to ingress or egress, an additional 10 cfm was added.

2 Initially, the pneumatic dampers and valves release air into the MCRHS area; after dampers and valves are set, they are no longer used.

3 These ducts are under negative pressure; therefore, leakage will be out of the MCRHS.

6.5 FISSION PRODUCT REMOVAL AND CONTROL SYSTEMS

6.5.1 Engineered Safety Feature (ESF) Filter Systems

Three Engineered Safety Feature (ESF) air cleanup systems units are provided for fission product removal in post-accident environments. These are:

1. The emergency gas treatment system (EGTS) air cleanup units.

2. The Auxiliary Building gas treatment system (ABGTS) air cleanup units.

3. The Main Control Room emergency air cleanup units.

6.5.1.1 Design Bases

6.5.1.1.1 Emergency Gas Treatment System Air Cleanup Units

The design bases are:

1. To provide fission product removal capabilities sufficient to keep radioactivity levels in the Shield Building annulus air released to the environs during a DBA LOCA sufficiently low to assure compliance with 10 CFR 100 guidelines.

2. These air cleanup units are a part of the EGTS. See Section 6.2.3.1.2 for the design bases for other portions of this system.

6.5.1.1.2 Auxiliary Building Gas Treatment System Air Cleanup Units

The design bases are:

1. To provide fission product removal capabilities sufficient to keep radioactivity levels in the Auxiliary Building secondary containment enclosure (ABSCE) air released to the environs during a postulated accident sufficiently low to assure compliance with 10 CFR 100 guidelines.

2. These air cleanup units are a part of the ABGTS. See Section 6.2.3.1.3 for the design basis for other portions of this system.

6.5.1.1.3 DELETED

6.5.1.1.4 Main Control Room Emergency Air Cleanup Units

The design bases are:

1. To provide air purification capabilities sufficient to keep air purity levels in the main control room and adjoining areas defined in Section 6.4 within limits needed to satisfy Criterion 19 of 10 CFR 50, Appendix A.

2. These air cleanup units are a part of the Main Control Room Habitability System (MCRHS) area HVAC system. See Section 9.4.1.1 for the design bases for other portions of this system.

6.5.1.2 System Design

6.5.1.2.1 Emergency Gas Treatment System Air Cleanup Units

The air cleanup units are a part of the air cleaning subsystems of the EGTS. See Section 6.2.3.2.2 for a description of the system design of the air cleanup subsystem, and the function, operation and control of the air cleanup units within that system.

The rated capacity of each redundant air cleanup unit in the subsystem is 4000 cfm. Both units are located in the EGTS room on Elevation 757. They are adjacent to each other, but separated by a concrete barrier wall.

The air cleanup units are steel housings containing air treatment equipment, samples, heaters, a drain, test fittings, and access facilities for maintenance. The air treatment equipment within the housing includes a demister, relative humidity heater, prefilter bank, HEPA filter bank, two banks of carbon adsorbers in series and another HEPA filter bank. These components are installed in the order listed.

The housing incorporates a quench-type water supply and drain system for flooding the carbon in case of fire. A drain is also incorporated into the housing adjacent to the demister installation to allow moisture separated from the air stream to flow by gravity to a water collection tank in the Auxiliary Building. Integral to this housing are test fittings properly sized and positioned to permit orderly and efficient testing of the HEPA filter and carbon adsorber banks.

The relative humidity heater installed in the air cleanup units is an electric heater designed to heat the incoming air sufficiently to reduce the relative humidity of saturated air to 70%. Included in this installation is a temperature limiting controller that will shut the heater off if excessive temperatures are detected.

The HEPA filters are 1000 cfm capacity units designed to remove at least 99.97% of the particulates greater than 0.3 micron in diameter, and meet the requirements of military specification MIL-F-51068. The carbon adsorbers are Type II unit trays, fabricated in accordance with AACC Standard CS-8T requirements. AACC-CS-8T has been superseded; and, ANSI/ASME-N509-989 specifies ASME AG-1-1988 to be used. Therefore, all new charcoal Type II cells shall meet AG-1, Section FD, with the exception that the 1991 version of the code be used. Existing Type II cells do not have to be replaced to meet the AG-1 code if being refilled. New replacement charcoal adsorbent (for use in new and refilled Type II cells) shall be procured to meet the ASME AG-1-1991 requirements in lieu of the 1988 version (or later version, provided proper evaluation justifies adequacy), with the exception that laboratory testing of adsorbent be in accordance with ASTM D3803-1989. These trays contain two-inch-thick impregnated carbon beds. Each bank of carbon adsorber trays typically contains one test type tray to facilitate periodic sampling of the carbon.

The total numbers of filters and adsorber unit trays provided in each air cleanup unit are listed in Table 6.5-5. Compliance of the design, testing and maintenance features of the EGTS air cleanup units with Regulatory Guide 1.52 is tabulated in Table 6.5-1.

6.5.1.2.2 Auxiliary Building Gas Treatment System Air Cleanup Units

See Section 6.2.3.2.3 for a description of the system design of the ABGTS and the function, operation and control of the air cleanup units within that system.

The rated capacity of each redundant air cleanup unit in this gas treatment system is 9000 cfm. Each unit is located in a separate room, one adjacent to each reactor unit on Elevation 737.

Each of these air cleanup units is a steel housing equipped with air treatment components, samples, heaters, test fittings and access facilities for maintenance. The air treatment components within the housing include a demister, a relative humidity heater, prefilter bank, HEPA filter bank, two banks of carbon adsorbers in series, and another HEPA filter bank. This equipment is installed in the order listed. The housing incorporates a quench-type water supply and drain system for flooding the carbon in case of fire. A drain is also incorporated into the housing adjacent to the demister section to allow moisture separated from the air stream to flow by gravity to a water collection tank in the Auxiliary Building. Integral to the housing are test fittings properly sized and positioned to permit orderly and efficient testing of the HEPA filter and carbon adsorber banks.

The relative humidity heater installed in the air cleaning units is an electric heater designed to heat the incoming air sufficiently to reduce the relative humidity of saturated air to 70%. Included in this installation is a temperature limiting controller that shuts off the heater if excessive temperatures are detected.

The HEPA filters installed in the air cleanup units are 1000 cfm units designed to remove at least 99.97% of the particulates greater than 0.3 micron in diameter, and meet the requirements of military specification MIL-F-51068. The carbon adsorbers installed in the air cleanup units are Type II unit trays, fabricated in accordance with AACC Standard CS-8T requirements. AACC-CS-8T has been superseded; and, ANSI/ASME-N509-989 specifies ASME AG-1-1988 to be used. Therefore, all new charcoal Type II cells shall meet AG-1, Section FD, with the exception that the 1991 version of the code be used. Existing Type II cells do not have to be replaced to meet the AG-1 code if being refilled. New replacement charcoal adsorbent (for use in new and refilled Type II cells) shall be procured to meet the ASME AG-1-1991 requirements in lieu of the 1988 version (or later version, provided proper evaluation justifies adequacy), with the exception that laboratory testing of adsorbent be in accordance with ASTM D3803-1989. The total numbers of filters and adsorber unit trays provided in each air cleanup unit are listed in Table 6.5-5.

Compliance of the design, testing and maintenance features of the ABGTS air cleanup units with Regulatory Guide 1.52 is tabulated in Table 6.5-2.

6.5.1.2.3 DELETED

6.5.1.2.4 Main Control Room Emergency Air Cleanup Units

See Section 9.4.1.2 for a description of the system design of the main control room emergency ventilation system and the function, operation and control of the emergency air cleanup units within that system.

Two 100% capacity air cleanup units, each rated at 4000 cfm, are provided for the control room. Both units are located in the mechanical-equipment room on Elevation 755.

Each of the air cleanup units has a stainless steel housing equipped with air treatment components, samples, test fittings and access facilities for maintenance. The air treatment components within the housing include a HEPA filter bank and a carbon adsorber bank, installed in the order listed. Integral to the housing are test fittings properly sized and proportioned to permit orderly and efficient testing of the HEPA filter and carbon adsorber banks. The HEPA filters utilized are 1000 cfm units designed to remove at least 99.97% of the particulates greater than 0.3 microns in diameter, and meet the requirements of military

specification MIL-F-51068.

The carbon adsorbers installed in the housing are Type II unit trays fabricated in accordance with AACC Standard CS-8T requirements. AACC-CS-8T has been superseded; and, ANSI/ASME-N509-989 specifies ASME AG-1-1988 to be used. Therefore, all new charcoal Type II cells shall meet AG-1, Section FD, with the exception that the 1991 version of the code be used. Existing Type II cells do not have to be replaced to meet the AG-1 code if being refilled. New replacement charcoal adsorbent (for use in new and refilled Type II cells) shall be procured to meet the ASME AG-1-1991 requirements in lieu of the 1988 version (or later version, provided proper evaluation justifies adequacy), with the exception that laboratory testing of adsorbent be in accordance with ASTM D3803-1989. The total numbers of filters and adsorber unit trays provided in each air cleanup unit are listed in Table 6.5-5.

Compliance of the design, testing, and maintenance features of the main control room emergency air cleanup units with Regulatory Guide 1.52 is tabulated in Table 6.5-4.

6.5.1.3 Design Evaluation

6.5.1.3.1 Emergency Gas Treatment System Air Cleanup Units

See Section 6.2.3.3.2.

6.5.1.3.2 Auxiliary Building Gas Treatment System Air Cleanup Units

See Section 6.2.3.3.3.

6.5.1.3.3 DELETED

6.5.1.3.4 Main Control Room Emergency Air Cleanup Units

See Section 6.4.4.

6.5.1.4 Tests and Inspections

6.5.1.4.1 Emergency Gas Treatment System Air Cleanup Units

Preoperational testing of the EGTS air cleanup units to applicable Regulatory Guide 1.52 requirements, as listed in Table 6.5-1, is conducted to verify the units leak tightness, HEPA and carbon adsorber bank efficiencies, and heater operation. Included in the testing scope are functional tests on all cleanup unit instrumentation, alarms, and data displays. Preoperational test requirements and acceptance criteria are addressed in Chapter 14 (historical information).

Periodic testing in accordance with the Technical Specifications assures continued satisfactory performance of the units. See Section 6.2.3.4.1 for testing and inspection procedures for other portions of the EGTS.

6.5.1.4.2 Auxiliary Building Gas Treatment System Air Cleanup Units

Preoperational testing of the ABGTS air cleanup units to applicable Regulatory Guide 1.52 requirements, as listed in Table 6.5-2, is conducted to verify the units leak tightness, HEPA and carbon adsorber bank efficiencies and heater performances. Included in the testing scope are functional tests on all cleanup units instrumentation, alarm, and data displays. Preoperational test requirements and acceptance criteria are addressed in Chapter 14.

Periodic testing in accordance with the Technical Specifications assures continued satisfactory performance of the units. See Section 6.2.3.4.2 for testing and inspection of other portions of the ABGTS.

6.5.1.4.3 DELETED

6.5.1.4.4 Main Control Room Emergency Air Cleanup Units

Preoperational testing of the main control room emergency air cleanup units to applicable Regulatory Guide 1.52 requirements, as listed in Table 6.5-4, is conducted to verify the units leaktightness, and HEPA and carbon adsorber bank efficiencies. Included in the testing scope are functional tests on all cleanup units instrumentation, alarm, and data displays. Preoperational test requirements and acceptance criteria are addressed in Chapter 14.

Periodic testing in accordance with the Technical Specification assures continued satisfactory performance of the units. See Section 9.4.1.4 for testing and inspection of other portions of the Control Building HVAC system.

6.5.1.5 Instrumentation Requirements

6.5.1.5.1 Emergency Gas Treatment System Air Cleanup Units

Permanently installed pressure differential gauges across the prefilter, both HEPA filter banks, and both carbon adsorbers allow periodic surveillance of dust loadings and pressure drops on individual components in the filter trains. Temperature instrumentation indicates air temperatures both upstream and downstream of the relative humidity heaters. The heaters are equipped with high temperature cutoffs. Instrumentation requirements for the operation and control of the safety-related functions of the EGTS are covered in Section 6.2.3.5.1.

6.5.1.5.2 Auxiliary Building Gas Treatment System Air Cleanup Units

Permanently installed pressure differential gauges across the prefilter, both HEPA filter banks, and both carbon adsorbers allow periodic surveillance of dust loadings and pressure drops on individual components in the filter trains. Temperature instrumentation indicates air temperatures downstream of the relative humidity heaters. The heaters are equipped with high temperature cutoffs.

Instrumentation requirements for the operation and control of the safety-related functions of the ABGTS are covered in Section 6.2.3.5.2.

6.5.1.5.3 DELETED

6.5.1.5.4 Main Control Room Emergency Air Cleanup Units

Permanently installed pressure differential gauges across the HEPA filter and carbon adsorber allow periodic surveillance of dust loadings and pressure drops on individual components in the filter trains. Temperature instrumentation indicates air temperature downstream of the carbon adsorber. Instrumentation for operation and control of the safety-related functions of the main control room emergency air cleanup system are discussed in Section 6.4.6.

6.5.1.6 Materials

6.5.1.6.1 Emergency Gas Treatment System Air Cleanup Units

Materials for HEPA filters and carbon adsorbers in the EGTS are designed for a stable and dependable operation in the accident environments discussed above. The carbon adsorbers are individually encased, flat-bed, tray-type units. Each tray contains new, commercially pure, activated carbon treated with iodine or an iodine compound to facilitate removal of organic and inorganic iodine compounds. The carbon ignition temperature after impregnation is greater than 620°F. Adsorber material and gaskets can withstand gamma doses of 1×10^8 rads accumulated in a 1-month period.

6.5.1.6.2 Auxiliary Building Gas Treatment System Air Cleanup Units

Same as in Section 6.5.1.6.1.

6.5.1.6.3 DELETED

6.5.1.6.4 Main Control Room Emergency Air Cleanup Units

Same as in Section 6.5.1.6.1.

6.5.2 Containment Spray System for Fission Product Cleanup

6.5.2.1 Design Bases

There are no formal design bases established for air cleanup by the containment spray system. This was done with the knowledge that water from the containment spray system will remove halogens and particulates from the containment atmosphere following a LOCA. No credit, however, was taken for this removal process in accident analyses presented in Section 15.5.3. In such circumstances, no design bases are needed for this air purification action.

6.5.2.2 System Design

See Section 6.2.2.2.

6.5.2.3 Design Evaluation

See Section 6.2.2.3.

6.5.2.4 Tests and Inspections

See Section 6.2.2.4.

6.5.2.5 Instrumentation Requirements

See Section 6.2.2.5.

6.5.2.6 Materials

See Section 6.2.2.6.

6.5.3 Fission Product Control Systems

6.5.3.1 Primary Containment

The primary containment is designed to assure that an acceptable upper limit leakage of radioactive material is not exceeded under design basis accident conditions. For purposes of integrity, the primary containment is composed of both the free-standing steel shell containment vessel and the containment isolation system. This structure and system are directly relied upon to maintain containment integrity. The primary containment functional design is described in Section 6.2.1.

Containment isolation can be initiated by either of two signals:

Phase A signal is generated by either of the following:

1. Manual - either of two momentary controls.

2. Safety injection signal generated by one or more of the following:

    a. Low steamline pressure in any steamline.
    b. Low pressurizer pressure.
    c. High containment pressure.
    d. Manual - either of two momentary controls.

Phase B signal is generated by either of the following:

1. Manual - two sets (two switches per set) - actuation of both switches in either set is necessary for spray initiation.

2. High-high containment pressure signals.

Containment isolation Phase A exists if containment isolation Phase B exists; i.e., when the Phase B signal is initiated by automatic instrumentation. Phase A containment isolation does not occur when the Phase B signal is initiated manually. The instrumentation circuits that generate both Phase A and Phase B signals are described in Section 7.1.2.1.2.

Containment purge system isolation (containment purge lines only) can be initiated by either of two signals:

1. Manual - Phase A or B manual initiate
    - SIS manual initiate

2. Automatic - SIS auto-initiate
    - Purge exhaust high radiation (Train A or B sensor)

An analysis was performed to determine the offsite radiological consequences of a LOCA during a containment purge. A DBA-LOCA was considered. The purge system will isolate 4 seconds after the detection of high radiation in the purge exhaust. The containment valve isolation signal is also generated by the safety injection (SI) signal from the reactor protection system (RPS) which is allocated a maximum response time of 2.0 seconds. The dose evaluation uses a 5 second step purge release based on the purge lines remaining wide open for 5 seconds. Subsequent plant specific analyses issued in support of this 2.0 second response time document that the 5.0 second step function closure characteristic assumed in this dose evaluation for the containment purge contribution remains bounding and conservative when compared to the actual valve closure characteristic with purge discharge continuing at a progressively diminishing rate until 6.0 seconds. In accordance with Branch Technical Position CSB 6-4 and later NRC guidance (Regulatory Guide 1.183), only reactor coolant normal activity was used as the source term prior to purge isolation because the purge valves will be closed prior to significant fuel damage and subsequent gap activity release. Regulatory Guide 1.4 assumptions were used (See Chapter 15) except for the following: 1) reactor coolant source terms, 2) an iodine spiking factor of 10, 3) 100% noble gasses and 100% iodines become airborne (which is conservative since this means no iodine partitioning), 4) the mass release of containment air is based on two 24-inch purge lines, and 5) purge flow goes directly to the environment without filtration via the containment purge exhaust filters.

The results found in Table 6.5-7, which when added to the offsite doses from the Regulatory Guide 1.4 LOCA results are less than the 10 CFR 100 limits of 25 rem gamma, 300 rem beta, and 300 rem inhalation.

The primary containment design bases and layout are further discussed in Section 6.2.1. The design and operation of the containment purge ventilation system is discussed in Section 9.4.6.

6.5.3.2 Secondary Containments

Two secondary containment barriers are provided at the Watts Bar Nuclear Plant. One of these is formed by the Shield Building that surrounds the steel primary containment vessel. The other secondary containment barrier is the Auxiliary Building structure that encloses all equipment in the building that may handle, collect, or store radioactive materials during normal operation or during accident conditions.

Because the Shield Building completely encloses the free-standing primary containment, all airborne leakage from primary containment passes into the annular region provided by this arrangement. See Part I of Table 6.5-8 and Section 6.2.3 for additional information on the operation of the air cleanup system that processes annulus air following a DBA.

Table 6.5-8 and Section 6.2.3 provide expected performance parameters for the Annulus Air Cleanup System subsequent to a DBA. The LOCA accident dose analysis as described in Section 15.5.3 employs more conservative assumptions relative to this system. See Section 15.5.3.

The Auxiliary Building is a conventional reinforced concrete structure located between the Reactor Buildings and the Control Building. Certain of the building's interior and exterior walls, floor slabs, and a part of its roof form the isolation barrier as outlined in Figures 6.2.3-4 through 6.2.3-9. The accident conditions for which the Auxiliary Building isolation barrier serves as the containment barrier are these involving irradiated fuel within the confines of the building and spills or leaks of radioactive materials from tanks and process lines inside the building. During a LOCA, any through-the-line leakage from primary containment into the Auxiliary Building will bypass the Shield Building annulus. In this case, the Auxiliary Building isolation barrier will serve as a secondary containment enclosure. See Part II of Table 6.5-8 and Sections 6.2.3, 9.4.3, and 9.4.5 for additional information on the Auxiliary Building secondary containment functions.

6.5.4 Ice Condenser as a Fission Product Cleanup System

The ice condenser system is an engineered safety feature designed to serve as a containment air purification and cleanup system. The ice condenser serves primarily as a large heat sink to readily reduce the containment temperature and pressure and condense the steam. For this purpose, ice is stored in a closed compartment between the lower and upper compartments of the containment. The containment is designed such that the only significant flow path from the lower to the upper compartment is through the ice bed. Immediately following a LOCA, a large pressure differential exists between the lower and upper compartment thereby providing flow through the ice bed. Later in the transient, flow is provided by two 40,000 cfm fans which circulate upper containment air into the lower compartment. Since all flow between the lower and upper compartments must pass through the ice bed, the ice bed 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 fission product transport and removal following a LOCA. The major benefit of the ice bed is its capacity to condense steam and thus remove molecular iodine from the containment atmosphere. To assure that the iodine remains in solution, the ice contains sodium tetraborate so that the combined condensate and ice melt is at an alkaline pH which promotes iodine hydrolysis to non-volatile forms.

The physical characteristics of the ice condenser system are discussed in Section 6.7. The ice bed fission product removal capability is discussed in this section.

6.5.4.1 Ice Condenser Design Basis (Fission Product Cleanup Function)

The design basis of the ice condenser as an iodine removal system is to use the chemical and physical properties of ice to reduce the fission product iodine concentration in the post LOCA containment atmosphere.

6.5.4.2 Ice Condenser System Design

The function of the post LOCA iodine removal served by the ice condenser is accomplished by chemically controlling the alkaline ice to a pH range of 9.0 to 9.5. This is accomplished by adding sodium tetraborate to the demineralized water in the solution of $Na_2B_4O_7 \bullet 10H_2O$ for a

boron concentration of 1900 ± 100 ppm prior to ice basket loading. During the accident, the melting ice provides a medium for removal of iodine from the containment atmosphere and fixation of the iodine in solution.

6.5.4.2.1 Component Description

The component description of the ice condenser system is given in Section 6.7.

6.5.4.2.2 System Operation

The operation of the ice condenser system is described in Section 6.2.1.3.2 and Section 6.2.1.3.3.

6.5.4.3 Ice Condenser System Design Evaluation (Fission Product Cleanup Function)

As a result of experimental and analytical efforts by Westinghouse, the ice condenser system has been proved to be an effective passive system for removing elemental iodine from the containment atmosphere and thereby reducing the offsite doses following a loss of coolant accident.

The experimental program and results of the ice condenser system effectiveness in removal of elemental iodine is reported in WCAP-7426, a non-proprietary topical report. The results of these extensive bench scale tests clearly indicated that an ice condenser system containing sodium tetraborate ice could effectively remove elemental iodine from the containment atmosphere.

In order to apply the results of the bench scale experimental program, an analytical model applicable to the plant ice condenser system was developed from the data of the experimental program.

The purpose of this section is to describe the analytical model and present the results of the ice condenser iodine removal effectiveness analysis.

Analytical Model

Following a LOCA a large volume of steam discharges into the containment lower compartment. Containment pressure and temperature rise immediately. At first, the increased pressure in the lower compartment forces steam through the ice condenser sections. Later, recirculation fans circulate the iodine-air-steam mixture through the ice condenser.

In addition to steam, iodine may be liberated into the containment as gaseous elemental iodine. It is also assumed that a fraction of the iodine in the containment atmosphere exists as methyl iodine which is not removed by the ice condenser. Elemental iodine is readily soluble in aqueous solutions and is removed from the air-steam mixture by the ice condenser.

The ice in the ice condenser contains sodium tetraborate normally referred to as alkaline ice by virtue of the alkalinity of the ice melt.

Data obtained from the experimental program as reported in WCAP-7426 can be classified as (1) alkaline ice and (2) acid ice. Since alkaline ice is used in the ice condenser, the iodine removal efficiency from those tests results were correlated.

The theoretical analysis for iodine removal by alkaline ice treats the ice condenser as consisting of two distinct compartments, an ice section and a rain section. Melt, falling from the ice into the sump, comprises the rain section (see Figure 6.5-1). Steam condenses from the air-steam mixture in both sections. In the ice section, $(1 + \lambda_v/\lambda_f)$ grams of melt mixture are formed per gram of steam condensed, where λ_v is latent heat of vaporization of water and λ_f is latent heat of fusion of water. In the rain section, however, only 1 gram of melt mixture is formed per gram of steam condensed. Melt temperature rises above 32°F as steam condenses in the rain. As ice continues melting, the rain section plays a more significant role in iodine removal.

An equation for iodine removal efficiency is obtained by solving the multi-component diffusion equations for steam-air-iodine mixtures in both ice condenser sections.

In the rain section, iodine is treated as a trace component with air and steam as the bulk constituents. Iodine from the bulk vapor diffuses through a gaseous boundary layer into the spherical drop as it falls through the rain section.

Condensation of water vapor and absorption of iodine in the ice section were treated in a similar manner. Ice is modeled as a flat plate surrounded by an essentially stagnant air-steam-iodine boundary layer through which steam and iodine diffuse.

The solution of the diffusion equations based on the above assumptions results in the following relationship:

$$\eta_I = Y_S \, \eta_S$$

where:

η_I = the iodine removal efficiency $\left(\dfrac{gm \;\; iodine \;\; removed}{gm \;\; iodine \;\; fed \;\; to \;\; condenser} \right)$ -

Y_S = the mole fraction steam in inlet gas stream

η_S = the steam condensation efficiency $\left(\dfrac{gm \;\; steam \;\; condensed}{gm \;\; steam \;\; fed \;\; to \;\; condenser} \right)$

Since the steam condensation efficiency in an ice condenser is nearly 100%, the iodine removal efficiency is directly related to the mole fraction of steam in the inlet gas stream.

Application of Ice Condenser Iodine Removal Model

The ice condenser iodine removal model has been applied to an ice condenser containment.

This model assumes iodine is released from the reactor system after blowdown and mixed with steam from boil off and is swept to the ice condenser by the recirculation fans.

The vapor composition of the lower compartment is a homogeneous mixture of iodine, steam from core boil off, and air.

The ice bed iodine removal efficiency, η_I has been computed on a time dependent basis and is shown in Table 15.5-7.

6.5.4.4 Condenser System Tests and Inspections

During initial ice loading, periodic tests are conducted to verify that the boron concentration and pH of the ice is within acceptable limits. This is accomplished by measuring the pH and boron concentration of samples of the solution prior to freezing. At routine intervals during plant operation, samples of the ice are taken, melted, and measured for pH and boron concentration to verify that these values are still within acceptable limits. The initial concentration of boron can only increase due to dissipation of some H_2O by sublimation.

6.5.4.4.1 Ice Condenser System Instrumentation

The ice condenser is a passive system which requires no instrumentation for operation.

6.5.4.5 Ice Condenser Materials

See Section 6.7.18.

REFERENCES

None.

TABLE 6.5-1 (Sheet 1 of 2)

REGULATORY GUIDE 1.52, REV. 2, SECTION APPLICABILITY
FOR THE EMERGENCY GAS TREATMENT SYSTEM

Reg. Guide Section	Applicability To This System	Comment Index	Reg. Guide Section	Applicability To This System	Comment Index
C.1.a	yes	Note 1	C.3.i	yes	Note 7
C.1.b	yes	Note 2	C.3.j	yes	Note 7
C.1.c	yes	Note 2	C.3.k	yes	Note 9
C.1.d	yes	--	C.3.l	no	Note 7
C.1.e	yes	--	C.3.m	yes	--
			C.3.n	no	Notes 6, 7, & 13
C.2.a	yes	--	C.3.o	yes	--
C.2.b	yes	--	C.3.p	no	Notes 10 & 7
C.2.c	yes	--			
C.2.d	no	Note 3	C.4.a	no	Note 10
C.2.e	yes	--	C.4.b	no	Note 11
C.2.f	yes	--	C.4.c	no	Note 7
C.2.g	yes	Note 12	C.4.d	no	Note 14
C.2.h	yes	--	C.4.e	yes	
C.2.i	yes	--			
C.2.j	no	Note 4	C.5.a	yes	Note 8
C.2.k	no	Note 5	C.5.b	yes	Note 8
C.2.l	no	Note 6	C.5.c	yes	Note 8
			C.5.d	yes	Note 8
C.3.a	yes	Note 7			
C.3.b	yes	Note 7	C.6.a	yes	Notes 7 & 8
C.3.c	yes	Note 7	C.6.b	yes	Note 7
C.3.d	yes	Note 7			
C.3.e	yes	Note 7			
C.3.f	yes	--			
C.3.g	yes	Note 7			
C.3.h	yes	--			

NOTES

1. The emergency gas treatment system is designed to withstand conditions resulting from the design basis LOCA.

2. The design is consistent with assumptions found in Regulatory Guide 1.4. Regulatory Guides 1.3 and 1.183 are not applicable.

3. No significant pressure surges to this system are envisioned resulting from the design basis LOCA. Thus, the system needs no special protection features to offset pressure surges.

TABLE 6.5-1 (Sheet 2 of 2)

REGULATORY GUIDE 1.52, REV. 2, SECTION APPLICABILITY
FOR THE EMERGENCY GAS TREATMENT SYSTEM

4. Each unit is totally enclosed and structurally adequate to permit intact removal. However, the probability of the need to remove the unit intact is small and to do so is impractical.

5. There are no outdoor air intakes associated with the emergency gas treatment system.

6. No enhancement in safety is foreseen by utilizing low leakage ductwork in this system. Any leakage which occurs inside the Shield Building eventually reenters the EGTS and is processed. No leakage to the Auxiliary Building from the ductwork between the Shield Building and the filter housing is foreseen, since air inside the duct is at a lower pressure than the surroundings. Any contaminated leakage into the Auxiliary Building is processed by the ABGTS before release to the environs. System leakage is determined based on analysis of the impact on acceptable accident dose limits of 10 CFR 100. Leakage from ductwork on the downstream side of the filter housing causes no problem since this air is cleaned up by the emergency gas treatment and auxiliary building gas treatment systems. However, the air cleanup ductwork is leak-tested in accordance with ANSI N509-1976.

7. Compliance with ANSI/ASME N509 is not required since the system was designed and constructed before publication of the ANSI document. The system conformed to this section of Regulatory Guide 1.52 Rev. 0 at the time of design and construction, and leakage testing is performed in accordance with ANSI N509-1976. Whenever possible, parts or components used as replacements will comply fully with the latest issue of ANSI/ASME N509. For welding requirements for ductwork, see Note 13.

8. Compliance with ANSI/ASME N510 is not required since the system was designed and fabricated before publication of the ANSI document. However, the system is tested, when possible, using the procedures outlined in ASME N510-1989.

9. Crossover flow ducts provide the capability to cool an inactive unit loaded with radioactive material to limit the temperature rise from radioactively induced heat and thus prevent auto ignition of the charcoal.

10. Compliance with this section is not a licensing requirement.

11. Space constraints do not permit compliance with this section.

12. The system design provides for total pressure drop indication across the filter housing and low flow annunciation of the operating fan in the MCR.

13. Those portions of TVA Classes Q and S Category I duct which are of welded construction and are fabricated or repaired after January 12, 1987, meet the welding requirements of ANSI/ASME N509-1976. The workmanship samples are not required to have penetrant testing (PT) or magnetic particle testing (MT).

14. The ventilation system is operated for 15 minutes every 31 days with heaters on. This is consistent with the guidance in Reg. Guide 1.52, Revision 4.

TABLE 6.5-2 (Sheet 1 of 2)

REGULATORY GUIDE 1.52, REV. 2, SECTION APPLICABILITY
FOR THE AUXILIARY BUILDING GAS TREATMENT SYSTEM

Reg. Guide Section	Applicability To This System	Comment Index	Reg. Guide Section	Applicability To This System	Comment Index
C.1.a	yes	Note 1	C.3.i	yes	Note 7, 8
C.1.b	yes	Note 2	C.3.j	yes	Note 7
C.1.c	yes	Note 2	C.3.k	yes	Note 8
C.1.d	yes	--	C.3.l	no	Note 7
C.1.e	yes	--	C.3.m	yes	--
			C.3.n	no	Notes 5, 7, & 12
C.2.a	yes	--	C.3.o	yes	--
C.2.b	yes	--	C.3.p	no	Notes 7 & 9
C.2.c	yes	--			
C.2.d	no	Note 3	C.4.a	no	Note 9
C.2.e	yes	--	C.4.b	no	Note 6
C.2.f	yes	--	C.4.c	no	Note 7
C.2.g	yes	Note 11	C.4.d	no	Note 13
C.2.h	yes	--	C.4.e	yes	--
C.2.i	yes	--			
C.2.j	no	Note 4	C.5.a	yes	Note 10
C.2.k	yes	--	C.5.b	yes	Note 10
C.2.l	no	Note 5	C.5.c	yes	Note 10
			C.5.d	yes	Note 10
C.3.a	yes	Note 7			
C.3.b	yes	Note 7	C.6.a	yes	Notes 7 & 10
C.3.c	yes	Note 7	C.6.b	yes	Note 7
C.3.d	yes	Note 7			
C.3.e	yes	Note 7			
C.3.f	yes	--			
C.3.g	yes	Note 7			
C.3.h	yes	--			

NOTES

1. The postulated DBA for the auxiliary building gas treatment system is the design basis LOCA.

2. The design is consistent with assumptions found in Regulatory Guide 1.4.

3. No significant pressure surges to this system are envisioned resulting from the design basis LOCA. Thus the system needs no special protection features to mitigate pressure surges.

TABLE 6.5-2 (Sheet 2 of 2)

REGULATORY GUIDE 1.52, REV. 2, SECTION APPLICABILITY
FOR THE AUXILIARY BUILDING GAS TREATMENT SYSTEM (Cont'd)

4. It would be possible to remove the unit intact, but not practical. The probability of the need to do so is considered to be negligible.

5. The use of low leakage ductwork would not enhance the safety of the system since any leakage that occurs is eventually routed back to the ABGTS and processed before being released to the environs. Leakage from the Auxiliary Building secondary containment enclosure to the environs is negligible since it is maintained at a negative pressure with respect to the atmosphere. Final acceptable system leakage is determined based on analysis of the impact on acceptable accident dose limits of 10 CFR 100. However, the air cleanup ductwork is leak-tested in accordance with ANSI N509-1976.

6. Space constraints do not permit compliance with this section.

7. Compliance with ANSI/ASME N509 is not required since the system was designed and fabricated before publication of the ANSI document. The system conformed to this section of Regulatory Guide 1.52 Rev. 0 at the time of design and fabrication, and leakage testing is performed in accordance with ANSI N509-1976. Whenever possible, parts or components used as replacements comply with the latest issue of ANSI/ASME N509. For welding requirements for ductwork, see Note 12.

8. The amount of radioactive material collected during the postulated DBA is too small to raise the adsorber bank temperature near the carbon ignition temperature. However, water sprays are provided in the event of a charcoal fire.

9. Compliance with this section is not a licensing requirement.

10. Compliance with ANSI/ASME N510 is not required since the system was designed and fabricated before publication of the ANSI document. However, when possible, the system is tested using the procedures outlined in ASME N510-1989.

11. Low airflow in the operating ABGTS train is annunciated in the MCR.

12. Those portions of TVA Classes Q and S Category I duct which are of welded construction and are fabricated or repaired after January 12, 1987, meet the welding requirements of ANSI/ASME N509-1976. The workmanship samples are not required to have penetrant testing (PT) or magnetic particle testing (MT).

13. The ventilation system is operated for 15 minutes every 31 days with heaters on. This is consistent with the guidance in Reg. Guide 1.52, Revision 4.

TABLE 6.5-3

DELETED

TABLE 6.5-4 (Sheet 1 of 2)

REGULATORY GUIDE 1.52, REV.2, SECTION APPLICABILITY
FOR THE MAIN CONTROL ROOM AIR CLEANUP SUBSYSTEM

Reg. Guide Section	Applicability To This System	Comment Index	Reg. Guide Section	Applicability To This System	Comment Index
C.1.a	yes	Note 1	C.3.i	yes	Note 12
C.1.b	yes	--	C.3.j	yes	Note 12
C.1.c	yes	--	C.3.k	no	Note 10
C.1.d	yes	--	C.3.l	no	Note 12
C.1.e	yes	--	C.3.m	yes	----
			C.3.n	no	Notes 7, 12, & 14
C.2.a	no	Notes 3 & 9	C.3.o	yes	----
C.2.b	yes	Note 2	C.3.p	no	Note 12
C.2.c	yes	--			
C.2.d	no	Note 4	C.4.a	no	Note 11
C.2.e	yes	--	C.4.b	no	Note 11
C.2.f	yes	--	C.4.c	no	Note 12
C.2.g	yes	Note 5	C.4.d	yes	----
C.2.h	yes	--	C.4.e	yes	----
C.2.i	yes	--			
C.2.j	no	Note 6	C.5.a	yes	Note 13
C.2.k	yes	--	C.5.b	yes	Note 13
C.2.l	no	Note 7	C.5.c	yes	Note 13
			C.5.d	yes	Note 13
C.3.a	no	Notes 3 & 8			
C.3.b	no	Notes 3 & 8	C.6.a	yes	Notes 12 & 13
C.3.c	no	Notes 3 & 8	C.6.b	yes	Note 12
C.3.d	yes	Note 12			
C.3.e	yes	Note 12			
C.3.f	yes	--			
C.3.g	yes	Note 12			
C.3.h	yes	--			

NOTES

1. The postulated design basis accident (DBA) for the main control room air cleanup units is the DBA LOCA.

2. All equipment is protected from natural phenomena and no high pressure equipment exists in the area. Rotating equipment is suitably encased and therefore, no missiles are expected to be generated which could result in loss of redundancy.

3. Each redundant air cleanup subsystem contains a HEPA filter bank and a carbon absorber bank.

4. No pressure surges of any significance to this system are envisioned during the postulated DBA identified in Note 1.

5. Differential pressure sensors are used to sense failure of an air cleanup unit, switch to the backup unit, and annunciate in the main control room. Differential pressure sensors

TABLE 6.5-4 (Sheet 2 of 2)

REGULATORY GUIDE 1.52, REV.2, SECTION APPLICABILITY
FOR THE MAIN CONTROL ROOM AIR CLEANUP SUBSYSTEM

for the HEPA and absorber banks are located on the air cleanup unit housings in the mechanical equipment room located next to the main control room. This mechanical equipment room is readily accessible to main control room personnel.

6. The amount of radioactive material collected by the filter and absorber banks in the DBA LOCA is not sufficient to create a serious radiation hazard. Furthermore, adequate capacity for air cleanup is provided to protect the main control room personnel for the full 30 day duration of the postulated emergency. Therefore, there is no need for a filter or absorber bank replacement during the emergency.

7. No enhancement in safety is foreseen by utilizing low leakage ducting in this system. Leakage from commercial grade ducting within the main control room cannot jeopardize safety because all supply and exhaust air is clean. No safety hazard due to small duct leakage outside the enclosed space containing the main control room is envisioned. During emergencies, essentially all air in-leakage into ducting with air below atmospheric pressure is cleaned up in its passage through the air cleanup unit. The external ducting having air at a positive pressure and potentially entraining contaminants which can be introduced into the main control room due to the leakage and the air cleanup units are leak-tested in accordance with ANSI N509-1976.

8. No equipment of this kind is utilized in the system.

9. The small quantities of outside air brought inside do not contain sufficient moisture to cause the mixture of recirculated air and outside air to have a humidity level sufficiently high to degrade the absorber bank performance.

10. The amount of radioactive material collected during the entire 30 day emergency due to the postulated DBA is too small to raise the absorber bank temperature near the carbon ignition temperature. However, water sprays are provided in the event of a charcoal fire.

11. Compliance with this section is not a licensing requirement.

12. Compliance with ANSI/ASME N509 is not required since the system was designed and fabricated well before publication of the ANSI document. The system conformed to this section of Regulatory Guide 1.52 Rev. 0 at the time of design and fabrication, and leakage testing is performed in accordance with ANSI N509-1976. Whenever possible, parts or components used as replacements comply with the latest issue of ANSI/ASME N509. For welding requirements for ductwork, see Note 14.

13. Compliance with ANSI/ASME N510 is not required since the system was designed and fabricated before publication of the ANSI document. However, the system is tested, when possible, using the procedures outlined in ASME N510-1989.

14. Those portions of TVA classes Q and S Category I duct which are of welded construction and are fabricated or repaired after January 12, 1987, meet the welding requirements of ANSI/ASME N509-1976. The workmanship samples are not required to have penetrant testing (PT) or magnetic testing (MT).

TABLE 6.5-5

ESF AIR CLEANUP UNIT DATA

I. Emergency Gas Treatment System
Air Flow Rate: 4,000 ft^3/min each

Type	Banks/ Train	Cells/ Bank	Cells/ Train	Total Cells
Prefilter	1	4	4	8
HEPA	2	4	8	16
Carbon	2	12	24	48

II. Auxiliary Building Gas Treatment System
Air Flow Rate: 9,000 ft^3/min each

Type	Banks/ Train	Cells Bank	Cells/ Train	Total Cells
Prefilter	1	9	9	18
HEPA	2	9	18	36
Carbon	2	27	54	108

III. Main Control Room Emergency Air Cleanup Subsystem
Air Flow Rate: 4,000 ft^3/min each
(Makeup 711 ft^3/min, recirculate 3,289 ft^3/min)

Type	Banks/ Train	Cells/ Bank	Cells/ Train	Total Cells
HEPA	1	4	4	8
Carbon	1	12	12	24

TABLE 6.5-6

DELETED

TABLE 6.5-7

PRIMARY CONTAINMENT OPERATION FOLLOWING A DBA

General

A. Type of Structure: Free-Standing Steel Shell (See Sections 3.8.1, 3.8.2, and 3.8.5)

B. Internal Fission Product Removal Systems

1. Containment Spray System: See Section 6.5.2

2. Ice Condenser System: See Section 6.5.4

C. Free Volume: 1,270,000 cu. ft.

Offsite Radiological Consequences - LOCA During Purge

(Purge Contribution Only) With and Without Iodine Spike:

	2-Hour Exclusion Area Boundary No I Spike (rem)	30-Day Low Population Zone No I Spike (rem)	2-Hour Exclusion Area Boundary With I Spike (rem)	30-Day Low Population Zone With I Spike (rem)
Gamma	6.507E-04	1.819E-04	4.190E-03	1.171E-03
Beta	3.191E-04	8.920E-05	1.144E-03	3.198E-04
Inhalation (ICRP-30)	2.437E-02	6.812E-03	2.442E-01	6.826E-02

Mass Release of Containment Air During Purge: 1.233E6 grams

Time-Dependent Parameters
Leak Rate of Primary Containment: 0.25 %/day

Leakage Fractions - To Annulus: 0.75
To Auxiliary Building: 0.25
To Environment: 0

Effectiveness of Fission Product Removal Systems

Containment Spray System: No credit taken for post-LOCA cleanup capability

Ice Condenser System: See Section 6.5.4

TABLE 6.5-8 (Sheet 1 of 2)

SECONDARY CONTAINMENT OPERATION FOLLOWING A DBA

PART I - Shield Building Secondary Containment Enclosure

General

Type of Structure: Reinforced Concrete

Free Volume: 396,000 cubic feet

Annulus Width: Approximately 5 feet

Location of Fission Product Removal Systems:
See Sections 6.5.1 and 6.5.4

Time-Dependent Parameters

Steady State Inleakage Rate: 250 cfm for a postulated single failure of one EGTS train.

Steady State Inleakage Rate: 957 cfm - Unit 1; 832 cfm - Unit 2 for a postulated single failure scenario which results in one pressure control train in full exhaust to the shield building exhaust stack while the other train remains functional. This flow rate is associated with two EGTS fans operating. Following operator action to place one fan in standby, the inleakage flow rate reduces to a steady state value of 694 cfm - Unit 1; 604 cfm - Unit 2.

Pressure: -0.50 inch water gauge (nominal required value at the top of Auxiliary Building elevation)

Air Cleanup System Flow Rate: 4000 +/- 10% cfm for each train

Steady State Recirculation Flow Rate: 3350 cfm for a postulated single failure of one EGTS train

Steady State Recirculation Flow Rate: 6,131 cfm - Unit 1; 5,737 cfm - Unit 2 for a postulated single failure scenario which results in one pressure control train in full exhaust to the shield building exhaust stack while the other train remains functional. This flow rate is associated with two EGTS fans operating. Following operator action to place one fan in standby, the recirculation flow rate reduces to a steady state value of 4,278 cfm- Unit 1; 3,455 - Unit 2..

Steady State Exhaust Flow Rate: 250 cfm for a postulated single failure of one EGTS train

TABLE 6.5-8 (Sheet 2 of 2)

SECONDARY CONTAINMENT OPERATION FOLLOWING A DBA

Steady State Exhaust Flow Rate: 957 cfm - Unit 1; 832 cfm - Unit 2 for a postulated single failure scenario which results in one pressure control train in full exhaust to the shield building exhaust stack while the other train remains functional. This flow rate is associated with two EGTS fans operating. Following operator action to place one fan in standby, the inleakage flow rate reduces to a steady state value of 694 cfm - Unit 1; 604 cfm - Unit 2.

Effectiveness of Fission Product Removal System:
See Section 6.5.3

PART II - Auxiliary Building Secondary Containment Enclosure

General

Type of Structure: Reinforced Concrete

Free Volume: 6.9×10^6 cubic feet[+]

[+] This value is an applied maximum value in calculations demonstrating the drawdown capabilities of ABGTS. It is conservative in relation to an ABSCE configuration that includes the free volume of only a single Reactor Building.

Location of Fission Product Removal Systems:
See Section 6.5.1

Time-Dependent Parameters

Average Residence Time: 0.3 hr

Vacuum Relief Flow Rate: 1370 cfm (minimum)

Pressure: -0.25 inch water gauge

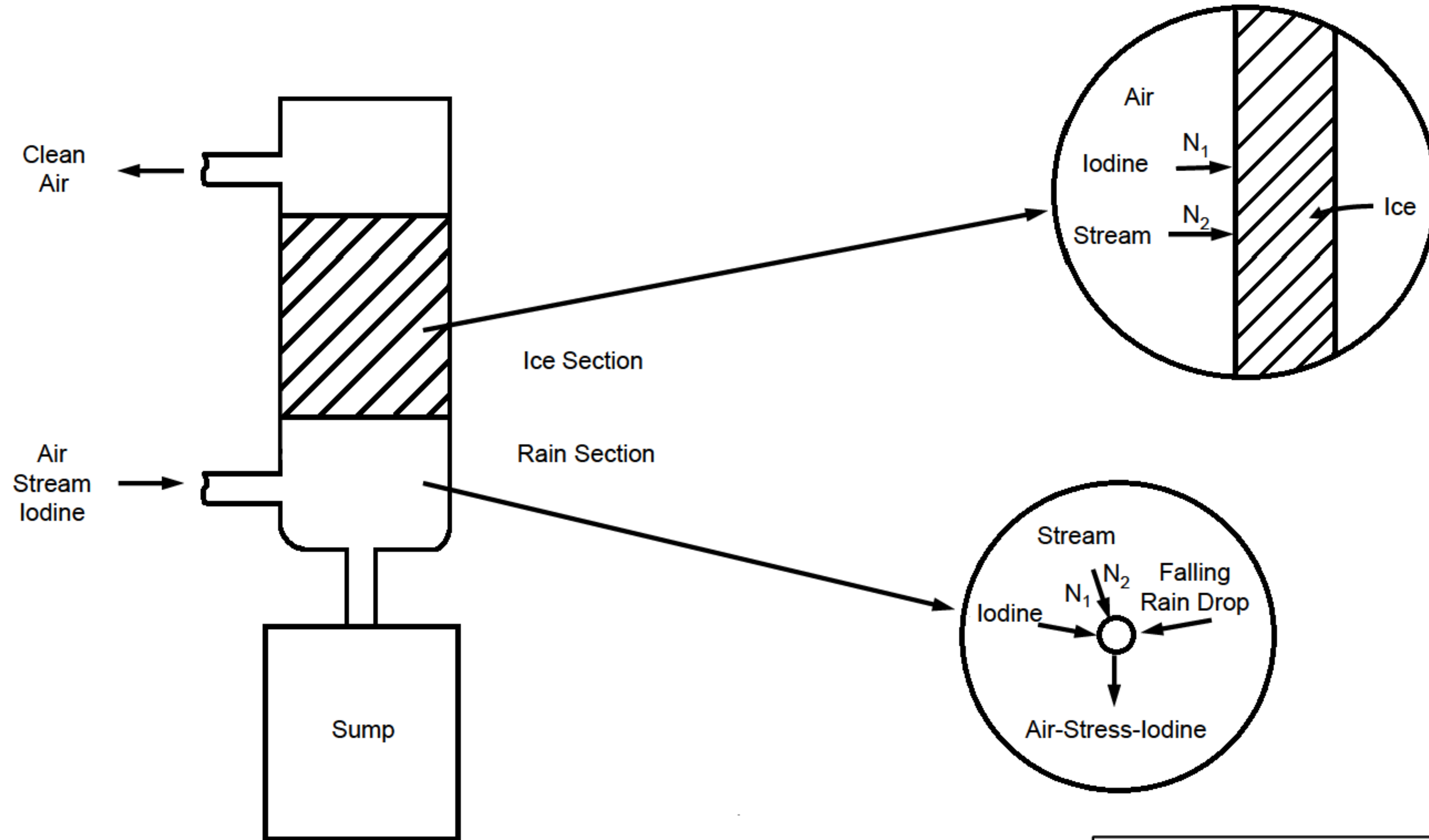
Air Cleanup System Flow Rate: 9,000* cfm

Recirculation Flow Rate: 0 cfm

Exhaust Flow Rate: 9,000* cfm

Effectiveness of Fission Product Removal System:
See Section 6.5.3

* A minimum airflow capability of 9300 cfm maintained by periodic surveillance and replacement of filters, as needed.



WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

Ice
Condenser
FIGURE 6.5-1

6.6 INSERVICE INSPECTION OF ASME CODE CLASS 2 AND 3 COMPONENTS

6.6.1 Components Subject to Examination and/or Test

All TVA Class A (ASME Code Class 1), B (ASME Code Class 2), and C and D (ASME Code Class 3), components containing water, steam, or radioactive waste shall be examined and tested in accordance with ASME Section XI of the ASME Boiler and Pressure Vessel Code as required by 10 CFR 50, Section 50.55 a(g), except where specific written relief has been requested. The in-service inspection requirements are contained in Section 5.2.8 for ASME Code Class 1 components and Section 3.8.2.7.9 for ASME Code Class MC and metallic liners of Code Class CC components. The inservice inspection requirements are contained in Section 3.8.5.1.1 for ASME Code Class CC concrete components. In addition, this program implements applicable portions of the WBN Technical Specifications.

6.6.2 Accessibility

Watts Bar design was established prior to the publication of Subsections IWC and IWD of Section XI, ASME Code; however, accessible Class 2 and 3 components will be inservice examined in accordance with the guidelines of Subsections IWC and IWD of ASME Section XI. Accessible Class 2 components were preservice examined in accordance with subsection IWC of ASME Section XI.

6.6.3 Examination Techniques and Procedures

The visual, surface, and volumetric examination procedures used by TVA are performed in accordance with the guidelines of subarticle IWA-2200, Section XI, ASME Code.

Code Cases to be used are identified in the Inservice Inspection Program in accordance with Subarticle IWA-2440 of ASME Section XI.

6.6.4 Inspection Intervals

An inspection schedule for Class 2 and Class 3 system components was developed in accordance with the guidelines of Subarticles IWA-2400, IWC-2400 and IWD-2400, Section XI, ASME Code.

6.6.5 Examination Categories and Requirements

The examination categories and requirements for Class 2 and 3 components are in accordance with subsections IWC and IWD of ASME Section XI to the extent practicable.

6.6.6 Evaluation of Examination Results

Evaluation of examination results shall be in accordance with Article IWA- 3000 of Section XI of the ASME Code.

Components with unacceptable indications will be repaired or replaced in accordance with the guidelines of Articles IWA-4000 and/or IWA-7000.

6.6.7 System Pressure Tests

The program for Class 2 and 3 system pressure tests shall be in accordance with articles IWA-5000, IWC-5000, and IWD-5000, ASME Code, Section XI, except where specific written relief has been requested and approved by NRC.

6.6.8 Protection against Postulated Piping Failures

Design measures have been taken to ensure that the containment vessel and all essential equipment within or outside of the containment including components of the reactor coolant pressure boundary, and other safety-related components have been adequately protected against the effects of blowdown jet and pipe whip.

REFERENCES

None.

6.7 ICE CONDENSER SYSTEM

Figure 6.7-1 shows the general layout of the ice condenser system.

6.7.1 Floor Structure and Cooling System

6.7.1.1 Design Bases

The ice condenser floor is a concrete structure containing embedded refrigeration system piping.

Figure 6.7-2 shows the general layout of the floor structure. The functional requirements for both normal and accident conditions can be separated into five groups: wear slab, floor cooling, insulation section, subfloor, and floor drain. Each group is described in detail below.

Wear Slab and Floor Cooling System

1. Function

   The wear slab is a concrete structure whose function is to provide a cooled surface as well as to provide personnel access support for maintenance and/or inspection. The wear slab also serves to contain the floor cooling piping.

   The floor cooling system intercepts approximately 90% of the heat flowing toward the ice condenser compartments from the lower crane wall and equipment room during normal operation. The floor cooling system is designed with defrost capability. During periods of wall panel defrosting, it is necessary to heat the floor above 32°F. During an accident, the floor cooling is terminated by the containment isolation valves which are closed automatically. The refrigeration system interface and cooling function is described in Section 6.7.6.2. The cavity below the wear slab is filled with an insulation material to resist the flow of heat into the ice bed during all operating conditions.

2. Design Criteria and Codes

   Refer to the discussion on ice condenser structural design in Section 6.7.16. The following codes are also used in the design:

   a. ANSI B31.5-66, including Addenda B31.5a-1968, Refrigeration Piping.

   b. American Welding Society (AWS) Structural Welding Code AWS D1.1-72 with Revisions 1-73 and 2-74 except later editions may be used for prequalified joint details, base materials, and qualification of welding procedures and welders.

Visual inspection of structural welds will meet the minimum requirements of Nuclear Construction Issues Group documents NCIG-01 and NCIG-02 as specified on the design drawings or other engineering design output. See Item d below.

c. AISC Manual of Steel Construction, Seventh Edition, 1970.

d. Nuclear Construction Issues Group (NCIG)
NCIG-01, Revision 2 - Visual Welding Acceptance Criteria (VWAC) for Structural Welding

NCIG-02, Revision 0 - Sampling Plan for Visual Reinspection of Welds

The referenced NCIG documents may be used after June 26, 1985, for weldments that were designed and fabricated to the requirements of AISC/AWS.

NCIG-02, Revision 0, was used as the original basis for the Department of Energy (DOE) Weld Evaluation Project (WEP) EG&G Idaho, Incorporated, statistical assessment of TVA performed welding at WBNP. Any further sampling reinspections of structural welds subsequent to issuance of NCIG-02, Revision 2, are performed in accordance with NCIG-02, Revision 2 requirements.

The applicability of the NCIG documents is specified in controlled design output documents such as drawings and construction specifications. Inspectors performing visual weld examination to the criteria of NCIG-01 are trained in the subject criteria.

3. <u>Design Conditions</u>

a. <u>Thermal Conditions</u>

1)	Initial Cooldown -	top of wear slab	70°F
		bottom of wear slab	12°F
2)	Defrost Cycle -	top of wear slab	33°F
		bottom of wear slab	70°F

b. <u>Seismic Loading</u>

1) Operating Basis Earthquake (OBE)

Loads	0.36 g
Vertical OBE	0.40 g radial
Horizontal OBE	0.52 g tangential

2) Safe Shutdown Earthquake (SSE)

Loads 0.53 g	
Vertical SSE	0.60 g radial
Horizontal SSE	0.78 g tangential

c. Design Basis Accident (DBA) Loads

1)	Pressure load on floor	9 psi
2)	Floor momentum load (due to deflectors)	36.4 kips

d. Ice Loading - assume 6 in. solid ice on floor 4300 lbs/bay

e. Live Loading 250 lbs/ft^2

f. Dead Loads

¼-inch plate	1410 lbs
½-inch pipe	164 lbs
Concrete wear slab	9700 lbs

g. Wall Panel - 121 lbs/in. over back 8 inches of slab

h. "Foam" Concrete Density “Nominal” 35 lbs/ft^3

During seismic and/or accident conditions the insulation is designed to support loads transferred by the wear slab.

Structural Subfloor

Refer to Section 3.8.

Floor Drain

1. Function

The floor drain is a passive structural component during normal operation. Its only function during normal operation is to minimize heated air inflow to the lower plenum.

During melt-out caused by a LOCA or HELB, the floor drain flapper is required to open to release water from the Ice Condenser to the Containment sump.

The section of floor drain pipe inserted vertically below the wear slab is designed and analyzed to the requirements of the ASME Code, Section III, Class 3. Under accident conditions, the floor drains must not fail in a mode which prevents outflow of water.

2. Design Criteria and Codes

Flapper gate welding complies with American Welding Society Structural Welding Code, AWS Dl.1-1972, as specified in Section 6.7.18. Piping complies with ASME standard code, Safety Related Piping, ASME Section III.

3. Design Conditions

Normal Operation	
Design temperature, maximum.	120°F
Nominal ΔP across valve	less than 1 psf

Accident Conditions	
ΔP across check valve	12-14 psi
Temperature pipe and valve	250°F

6.7.1.2 System Design

Wear Slab and Floor Cooling System

The wear slab is a 4-inch-thick layer of high strength concrete (3000 psi) having an exposed top surface area of 145 ft^2/bay. See Figure 6.7-3 for top surface typical geometry. The concrete has a density of 150 lbs/ft^3 and is prepared with air entrainment admixtures to minimize spalling from freeze/thaw cycles. Steel reinforcing is used in the wear slab to assure adequate and uniform strength. A protective coating is applied to the top of the wear slab which provides an additional water barrier for the wear slab. The floor cooling system consists of ½-inch schedule 80 carbon-steel ASTM A-333 Grade 6 piping which is embedded in the wear slab of each bay in a serpentine fashion (see Figure 6.7-3), thereby providing ample cooling of the wear slab surface.

The cooling pipes contained in each wear slab rest on a steel plate which extends across the full width of the floor for maximum effectiveness in intercepting heat passing up through the floor. Expansion joints are located at each bay and expansion material is located at the slab perimeter. The floor coolant flow rate per bay is adjusted by means of needle valves and is monitored by a temperature sensing element located at the downstream end of each of the bay floor piping. Should a leak develop each individual bay piping loop can be isolated by closing two valves. The coolant contained in the piping is a corrosion inhibited glycol/water solution.

For defrosting purposes, electric heating of the glycol is provided. In general, components requiring periodic maintenance such as pumps, heaters and control valves are located outside of the ice condenser.

The insulation cavity is filled with a low density, closed cell, foam concrete. The nominal density of the foam concrete is 35 lbs/ft^3; the compressive strength is 110 psi. The thermal conductivity per inch thickness is nominally 1.0 Btu/hr-°F-ft^2. The insulation cavity for the foam concrete is sealed by a vapor barrier to provide additional assurance that the insulation section resists infusion of water vapor and thus retains a high thermal resistance. The top surface of the foam concrete is covered with a course of grouting which provides seating surface for the floor plate and cooling coil assemblies.

Floor Drain

Special consideration has been given in the design to minimize gate leakage.

The floor drains employ a section of pipe 12 inches in diameter, inserted vertically below the wear slab. This pipe is insulated with foam glass insulation. The horizontal run is a 12-inch diameter steel pipe embedded in the subfloor, which is at a relatively warm temperature. The drain gate is a 12-inch diameter horizontal flapper gate fabricated from cadmium-plated carbon steel welded per AWS D1.1-1972. The gate is designed to remain closed against the cold air head in the ice condenser to minimize air outleakage during normal operation. It is designed to tolerate a 15 psi back pressure when closed. The gate is in a warm environment, and no freezing will occur.

6.7.1.3 Design Evaluation

Wear Slab

The wear slab, during normal operating conditions, is subject only to its dead weight consisting of concrete, steel reinforcing, steel plates and piping. Six inches of 100% density ice is assumed to be uniformly distributed over the entire floor. The dead weight amounts to 11,200 lbs per bay, the equivalent of 0.56 psi. The live load for maintenance purposes is assumed to be 250 lbs/ft^2. The vertical seismic input is 0.36 g for OBE and 0.53 for SSE. The dead load plus seismic loads are insignificant because the highest load on the floor is contributed by blowdown pressure during design accident conditions. The blowdown pressure is 9 psi (boundary value for analysis), and added to this value, for design purposes, is a 40% design margin, and a dynamic factor of 1.53. This results in a minimum value for design of 19.28 psi.

The most severe loading condition is the combination of the dead load, the SSE seismic acceleration of 0.53 g, the 19.28 psi pressure load and 8.1 psi locally near the deflectors due to flow impulse loadings. The wear slab is designed to accommodate the heatup and cooldown cycles and OBE without overstressing the concrete and coolant piping.

Floor Cooling System

The embedded piping for floor cooling is ½-inch schedule 80 pipe. ANSI B31.5-68 data shows that the pipe can tolerate internal pressures of 4812 psi.

In addition, the piping is tested to 200 psi. The pipe is sized to allow for at least 38 mils of corrosion. Nevertheless, the glycol coolant contains corrosion inhibitors, and as a result pipe corrosion is negligible. The ¼-inch floor plate is integrated with the concrete through ½-inch diameter anchors welded to the plate on 12-inch centers. These anchors prevent thermal loads from concentrating in the piping.

Insulation Section

The insulation section supports wear slab loads. For a conservative analysis the wear slab dead weight + seismic + DBA loads were assumed to be transferred to the foam concrete section. The compressive strength of the foam concrete is sufficient to accept these floor loads.

Floor Drain

Drains are provided at the bottom of the ice condenser compartment to allow the melt/condensate water to flow out of the compartment during a loss-of-coolant accident. These drains are provided with gates that are designed to seal the ice condenser during normal plant operation to prevent loss of cold air from the ice condenser. These gates remain closed against the cold-air head (1 psf) of the ice condenser and open before the water head reaches a value of 18 inches of water.

For a small pipe break, the water inventory in the ice condenser is produced in proportion to the energy added from the accident. The water collecting on the floor of the condenser compartment then flows out through the drains. For intermediate and large pipe breaks the ice condenser doors are open and water drains through both the doors and the drains.

For a large pipe break, a short time (on the order of seconds) is required for the water to fall from the ice condenser to the floor of the compartment. Results of fullscale section tests performed at Waltz Mill show that, for the design blowdown accident, a major fraction of the water drained from the ice condenser, and no increase in containment pressure was indicated even for the severe case with no drains.

A number of tests were performed with the reference flow proportional-type door installed at the inlet to the ice condenser and a representative hinged door installed at the top of the condenser. Tests were conducted with and without the reference water drain area, equivalent to 15 ft^2 for the plant, at the bottom of the condenser compartment.

These tests were performed with the maximum reference blowdown rate, with an initial low blowdown rate followed by the reference rate, with a low blowdown rate followed by the simulated core residual heat rate.

The results of all of these tests show satisfactory condenser performance with the reference type doors vent, and drain for a wide range of blowdown rates. Also, these tests demonstrate the insensitivity of the final peak pressure to the water drain area. In particular, the results of these full-scale section tests indicate that, even for the reference blowdown rate, and with no drain area provided, the drain water did not exert a significant back pressure on the ice condenser lower doors. A major fraction of the water drained from the ice condenser compartment by the end of the initial blowdown. The effect of this test result is that containment final peak pressure is not affected by drain performance.

Although drains are not necessary for the large break performance, 15 ft^2 of drain area was provided for small breaks.

For small breaks, water flows through the drains at the same rate that it is produced in the ice condenser. Therefore, the water on the floor of the compartment reaches a steady height which is dependent only on the energy input rate.

To determine that the 15 ft^2 drain area met these requirements, the water height is calculated for various small break sizes up to a 30,000 gpm break. Above 30,000 gpm, the ice condenser doors would be open to provide additional drainage. The maximum height of water required was calculated to be 2.2 ft above the drain gate. Since this height resulted in a water level which is more than 1 ft below the bottom elevation of the doors, it is concluded that water does not accumulate in the ice condenser for this condition and that a 15 ft^2 drain gives satisfactory performance.

During normal unit operation, the sole function of the gate is to remain in a closed position, minimizing air leakage across the seat. To avoid unnecessary unseating of the valve seat, a 12-inch drain header is connected to the 12-inch line immediately ahead of the valve. Any spillage or defrost water drains off without causing the gate to be opened.

The arrangement of the drain system for the lower inlet region of the ice compartment is shown in Figure 6.7-2.

Special consideration has been given in the design to prevent freezing of the gates and to minimize leakage.

To minimize the potential for gate freezing, a section of pipe is inserted vertically below the seal slab, while the horizontal run of pipe (steel) is embedded in a warm concrete wall before it reaches the valve. The gate itself is in the upper region of the lower compartment, where the ambient temperature is above freezing.

The gate is held in a closed position by virtue of its design as a vertical flapper with an offset hinge at the top.

In order to reduce gate leakage to an acceptable value, a sealant is applied to the seating surface after installation of the gates. Tests show that this reduces leakage to practically zero. Maximum allowable leakage rate would be approached as a limit only if all the sealant were to disappear completely from all the gates, which is unlikely. Sealant is replaced as necessary.

Conclusion

On the basis of the structural analysis performed on the floor structure, it is concluded that the floor is adequate for all anticipated loading conditions. In addition, the floor design is compatible with ice condenser wall panel defrosting. The water resulting from the wall panel defrosting produces no adverse effect on the structural integrity of the floor. The use of concrete with entrained air affords ample resistance to the effects of water. Additionally, the floor structure contains water vapor seals. The seals typically include a protective surface coating on the wear slab top surface, a vapor barrier between the foam concrete and the structural subfloor, a leveling course of grout on the top surface of the foam concrete, and a steel plate (in the wear slab) with lapping material in the plate to plate joints. As a result, the effects of water on the floor and insulation is negligible.

6.7.2 Wall Panels

6.7.2.1 Design Basis

Function

The wall panels are designed to thermally insulate the ice bed under normal operating conditions from the heat conducted through the crane wall, the containment wall, and the end walls. In addition, they are designed to provide a circulation path for cold air and a heat transfer surface next to the ice bed so that the ice is maintained at its design temperature range.

The supporting structure of the wall panel also provides for transfer of radial and tangential loads from the lattice-frame columns to the crane wall anchor embedments.

Criteria and Codes

The structural parts of the wall panels are designed to meet the requirements given in Section 6.7.16.

Design Conditions

The service temperature range is 10°F to 20°F and the DBA temperature is 250°F.

The design loads are presented in Table 6.7-1. The loading combinations considered in the design are those given in Section 6.7.16. For the SSE plus DBA combination, ten loading

cases are considered.

6.7.2.2 System Design

The wall panel design incorporates provisions for installation on the crane wall, containment wall, and end walls of the ice bed annulus. Containment and end wall panels are similar.

The crane wall panel design incorporates transverse beam sections which are fabricated from standard structural sections and to which the lattice frame column mounting lugs are attached. These sections are attached to the rear mounting angle assemblies by insulated bolts.

Wall panels are attached to the crane and end walls by studs welded to the anchor embedments and to the containment by studs welded to the shell. The crane wall panels extend from the bottom of the upper plenum to the lower support structure where they are supported on the inner circumferential beams of the horizontal platform. The containment wall panels extend from the bottom of the upper plenum to the top of the floor wear slab.

Cooling ducts are incorporated in the design to provide flow from the air handlers in the duct adjacent to the ice bed and return flow in the outer duct of the panel. This provides an even distribution of duct face temperature. Each bottom duct assembly provides a flow path between the inner and outer duct to allow return flow through the outer duct.

The ducts are fabricated as sandwich panels utilizing corrugated sheet sections enclosed in sheet metal enclosures. This type of sandwich construction provides resistance to differential pressure loads and results in minimal overall weight and flow restrictions. Flow sections of wall panels are seal welded to prevent air leakage.

Materials of construction of the wall panels conform to the design criteria discussed in Section 6.7.18.

Areas between air ducts and walls are insulated and areas between adjacent air ducts are insulated and covered with a lap strip to provide a seal between wall surface and ice bed. Elastomers and sealants are insignificantly affected by exposure to a 5 R/hr gamma radiation field over a period of forty years.

6.7.2.3 Design Evaluation

The wall panels have been analyzed for seismic and design basis accident (DBA) loading conditions as well as service loads.

Analysis for DBA Pressure Load

The wall panels are bolted to transverse beam sections with a maximum span of about 24 inches. In the analysis, the wall panels were taken as a 24 in. x 36 in. sandwich plate simply supported on all four sides.

It is noted that a DBA pressure of 19 psig was used in these analyses. The duct internal pressure was neglected in the analyses because it is negligible in relation to the 19 psig (internal design pressure 0.5 psig).

Analysis for Seismic and DBA Transverse Beam Loads

A transverse beam section was investigated for its ability to transmit the imposed seismic and DBA loads from the lattice frame column attachment to the crane wall. A two dimensional beam analysis utilizing the "STASYS" program was employed.

Various loading modes were used with values as shown in Table 6.7-1 (Parts B, C, D and E).

Overall Conclusion

Based on the analyses described in the foregoing, it is concluded that the wall panel assembly meets the design requirements given in Sections 6.7.16 and 6.7.18.

6.7.3 Lattice Frames and Support Columns

6.7.3.1 Design Basis

Function

The lattice frames and support columns assembly provide the following functions:

1. Positions the ice baskets in the ice bed and controls the hydraulic diameter.

2. Provides lateral support for the ice baskets under normal seismic and accident loads.

3. Allows passage of steam and air through the space around ice baskets.

4. Allows for basket installation and removal requirements.

Structural Requirements

Refer to Section 6.7.16.

Design Criteria

1. The lattice frames are designed to be compatible with the periodic weighing, procedure for the ice baskets.

2. The structure is designed to position the ice columns in the required array to maintain the performance of the ice condenser. In particular, the flow area around each ice column is maintained within the limits established by the general design criteria.

3. The lattice frame allows loading of the ice baskets in position, and permits lifting of complete basket columns for removal in sections.

Materials Requirements

Refer to the listing of acceptable materials in Section 6.7.18. All accessible steel components are covered by protective coating.

General Thermal and Hydraulic Performance

1. The lattice frames space the ice basket columns so that the hydraulic diameter around each ice column is maintained for all modes of operation.

2. Differential thermal expansion between crane wall and lattice frame structure, together with other applicable loads, do not stress the lattice frames or associated supporting structure beyond the design limits, or adversely affect the spacing between lattice frames.

3. Forces across the lattice frames in the horizontal and vertical direction due to seismic and blowdown loads do not overstress the lattice frame and supporting, structure beyond the design limits.

Interface Requirements

1. Lattice Frame to Ice Basket Columns

   The lattice frame locates and aligns the ice basket array. Sufficient clearance is provided to assure ease of ice basket installation, while limiting radial basket motion to a nominal amount. The lattice frame structure is also capable of withstanding design and operating seismic and accidental loading.

2. Lattice Frame to Lattice Frame Column

   The lattice frame is attached to the lattice frame columns. The column bases are adjustable so that matching of columns to lower support structure can accommodate the range of manufacturing and installation tolerances.

3. Lattice Frame, Columns to Crane Wall Air Duct Panels

   The lattice frame columns are bolted to the wall panel cradles. Lateral seismic loading from ice baskets and lattice frame is transmitted to the crane wall through the lattice frame columns and the wall panel cradles.

4. Lattice Frame Columns to Lower Support Structure

   Lattice frame columns interface with the lower support structures. The columns are designed to allow for accumulation of dimensional tolerances at interfaces.

5. Lattice Frame Columns to Intermediate Deck

   The top end of the lattice frame columns at each bay supports the intermediate deck and related supports.

6. Allowance is made for mounting the ice condenser temperature sensing system onto the lattice frames.

Design Load

The lattice frames and support columns are designed to withstand dead loads, live loads, seismic loads (including impact and accident loads) and remain within the allowable limits established in Section 6.7.16. Differential thermal expansion loads due to normal and accident

conditions are also considered. Structural loads are not transmitted through the lattice frames and columns to the containment structure. Figures 6.7-4 and 6.7-5 show the lattice frame loading orientation and distribution.

The lattice frame and column are designed to withstand the following load combinations in both the tangential and radial directions:

Dead Loads + Operating Basis Earthquake
Dead Loads + Safe Shutdown Earthquake
Dead Loads + Design Basis Accident
Dead Loads + Design Basis Accident + Safe Shutdown Earthquake

6.7.3.2 System Design

The lattice frames are structural steel grid work structures located in the ice condenser annulus and fitted between the lattice frame support columns and clearing the wall panel ducts.

The lattice frames are mounted radially across the ice condenser annulus for the full 300° of annulus circumference at each of eight levels between the lower support structure and the intermediate deck. The first level is located 15 feet above the wear slab or ice condenser floor and the next seven levels are vertically spaced at 6 feet intervals. A total of 576 lattice frames are required for the ice condenser assembly. Three lattice frames are required per level in each of the 24 bays and this configuration is repeated for the eight levels.

The lattice frames are mounted to rectangular steel columns which are placed at the crane wall side and at the containment side of the condenser annulus. The column bases are attached to the lower support structures. Columns at the crane walls are attached along the length of the wall panel cradles and to the lower support structure, while those at the containment side are free-standing, i.e., the bases are fastened to the lower support structure but there are no connections with the wall panels or the containment vessel wall. This arrangement prevents transmission of loads from ice baskets, lattice frames and columns to the containment vessel. The vertical columns and crane wall support and maintain the lattice frame geometry during normal and accident loading conditions.

The lattice frames are welded steel structures consisting of radial struts supported by welded cross bracing as shown in Figure 6.7-6. Basically the lattice frame is about 125 inches long, 48 inches at its widest point, and 7½ inches deep. The entire welded structure weighs about 1200 pounds. Individual free path penetrations are provided for each of 27 ice baskets. The lattice frame struts that form the ice basket restraints are all double fillet welded to the stringers. This assures a consistent weld design and ensures the integrity of the entire structure in operation.

Flexible radial members on the lattice frame are located at the containment side to accommodate differential thermal expansion in the tangential direction, and to allow for minor column misalignment at installation. The flexible radial members are attached to the vertical support columns.

The lattice frame attachment at the crane wall consists of horizontal ear-like tabs that accommodates the bolting. One tab is slotted in the tangential direction to allow for differential thermal expansion between the concrete crane wall and the steel structures. Lattice frame tabs are fastened to brackets on the vertical support columns. The columns, in turn, are bolted to the crane-side wall panel cradles. The wall panel cradles are fastened to the crane wall studs and transmit the lattice frame and ice basket horizontal loads to the crane wall, while the vertical loads are transmitted to the lower support structure.

The cross bracings and radial struts are arranged so that the ice baskets are positioned in the free path penetrations. The free path diameter controls the radial clearance between ice baskets and the lattice frames. The penetrations are spaced to assure the proper hydraulic diameter around each ice basket and to allow free passage of air and steam through the surrounding passages. Small pads on the radial struts control the tagential ice basket clearance.

All of the welding and inspection was done in accordance with the American Welding Standard Procedure, D1.1-72. The welds are inspected visually and then by magnetic particle inspection. The magnetic particle inspection is applied to selectively located welds throughout the structure.

All accessible exposed steel components are covered by a protective coating.

6.7.3.3 Design Evaluation

The lattice frames are analyzed using the ICES-STRUDLE II system of computer programs for frame analysis. STRUDLE is a general program operating as a subsystem of the Integrated Civil Engineering (ICES) program. The lattice frames are treated as three dimensional structures composed of joints, support joints, and structural members connecting the joints. Figure 6.7-7 illustrates the analytical model generated for the lattice frames. Each structural joint is assigned a circled number, and each structural member an uncircled number.

The lattice frame is treated as a cantilevered structure in the horizontal plane and restrained vertically at the four column connections. The model in Figure 6.7-7 shows flexible connections at the crane wall and no connection at the Containment wall. Variations in flexibility of the crane wall connections are considered in the analysis to simulate the behavior of the slotted tab connection and the connections to lattice frame columns and air duct wall panels.

The analysis of the loads for the individual maximums of D + OBE, D + SSE and D + DBA is determined. A survey is also conducted for the loading combinations of D + SSE + DBA for each lattice frame level at reference seismic orientation, 45°, and 90° from reference to determine the maximum loading condition on the lattice frame. The survey shows that the highest loads occur on the lattice frame at the 33 feet level, and that the combination of D + SSE + DBA, horizontally and vertically produces the maximum stresses.

Maximum stresses are calculated at each structural member at the edge of the fillet weld for all loading conditions.

Fatigue stresses due to OBE loading were calculated and are within the allowable limits defined in Section 6.7.16.

The vertical support columns and brackets which support the lattice frames are structurally analyzed to determine structural integrity. The worst load combinations of D + OBE, D + SSE, D + SSE + DBA are considered in the analysis. The stress analysis indicates that the stress for all loading conditions is below the allowable limits as defined in Section 6.7.16.

The vertical support numbers are also analyzed to determine buckling characteristics. Analysis using classical buckling methods indicates that this phenomena is not a concern.

6.7.4 Ice Baskets

Alternate/Optional Hardware

The initial plant construction installed ice columns utilizing four individual baskets approximately 12 foot long each coupled together. Upon completion of the ice condenser, removing and installing 12 foot (long) baskets is impossible in most row locations near walls due to interference from overhanging equipment in the upper plenum area.

As an alternative, two foot (short) replacement baskets can be coupled together in a grouping of six to make up equivalent 12 foot baskets that serve the same form, fit and function as the original long baskets. For a 48 foot ice basket column, a coupling with an internal cruciform insert, attached by welding, is located at each 6 foot elevation point which coincides with each lattice frame support location in the ice bed. The basket material and fabrication processes for the short basket are the same as for the long basket, except the sodium dichromate dip after galvanizing is no longer required for the short basket. Ice basket columns constructed from short baskets have an insignificant effect on the structural integrity and thermal performance of the ice condenser containment.

Also as an alternative, self-tapping screws with predrilled holes meeting all the design requirements for strength (140,000 psi min tensile for Unit 1 and 140,000 psi min. tensile or minimum 31 HRC for Unit 2), finish, head style, etc. may be used in all basket replacement activities. These alternate screws can also be used as replacements for the self-drilling, self-tapping screws originally furnished during initial plant construction.

6.7.4.1 Design Basis

Function

The function of the ice baskets is to contain borated ice in 12-inch diameter columns 48 feet high. The ice absorbs the thermal energy resulting from LOCA or steam line break in the Containment structure. The baskets are arranged to promote heat transfer from the steam to ice during and following these accidents. The function of the ice baskets is also to provide adequate structural support for the ice and maintain the geometry for heat transfer during or following the worst loading combinations.

Loading Modes

The following loading conditions are considered in the design of the ice baskets: dead weight, seismic loads, blowdown loads, and impact loads between the basket, ice and lattice frames. The baskets withstand these loads and remain within the allowable limits established in Section 6.7.16.

Design Consideration

1. The structural stability and deformation requirements are determined to ensure no loss of function under accident and safe shutdown earthquake loads.

2. The ice baskets are designed to facilitate maintenance and for a lifetime consistent with that of the unit.

3. The structure is designed to maintain the ice in the required array to maintain the integrity of performance of the ice condenser. In particular, the hydraulic diameter and heat transfer area are maintained within the limits established by test to be consistent with the containment design pressure.

4. Any section of the ice basket is capable of supporting the total weight of the ice above that section.

General Thermal and Hydraulic Performance Requirements

The ice baskets are fabricated from perforated sheet metal which has open area to provide sufficient ice heat transfer surface. The adequacy of the design and the performance were confirmed by test.

Interface Requirements

1. Lattice Frame

   The lattice frames at every 6-ft act as horizontal restraints along the length. The design provides a nominal ¼-inch. radial clearance between the ice baskets and the lattice frames. Lattice frame and basket coupling elevations coincide to prevent damage to the basket during impact.

2. Lower Support Structure

   Ice basket bottoms are designed to be supported by and held down by attachments to the lower support structure. The basket supports are designed for structural adequacy under accident and safe shutdown earthquake loads and permit weighing of selected ice baskets.

3. Basket Alignment

The ice condenser crane aligns with baskets to facilitate basket weighing and/or removal. The baskets are capable of accepting basket lifting and handling tools.

4. Basket Loading

The ice baskets are capable of being loaded by a pneumatic ice distribution system. The baskets contain a minimum of 2.26×10^6 pounds of ice.

5. External Basket Design

The baskets are designed to minimize any external protrusions which would interfere with lifting, weighing, removal and insertion.

6. Basket Coupling

Baskets are capable of being coupled together in 48-foot columns.

7. Basket Couplings and Stiffening Rings

Couplings or rings are located at 6 feet intervals along the basket and have internal inserts to support the ice from falling down to the bottom of the ice column during and after a DBA and/or SSE.

Design and Test Loads

The minimum test and basic design loads are given in Table 6.7-2.

6.7.4.2 System Design

The ice condenser is an insulated cold storage room in which ice is maintained in an array of vertical cylindrical columns. The columns are formed by perforated metal baskets with the space between columns forming the flow channels for steam and air. The ice condenser is contained in the annulus formed by the containment vessel wall and the crane wall circumferentially over a 300° arc.

The ice columns are composed of four baskets approximately 12 feet long each, filled with flake ice. The baskets are formed from a 14 gage (.075) perforated sheet metal, as shown in Figure 6.7-8. The perforations are 1.0 in. x 1.0 in. holes, spaced on a 1.25-inch center. The radius at the junction of the perforation is 1/16 inch. The ice basket material is made from ASTM-569 (Unit 1 or ASTM-569 and/or A1011 (Unit 2) which is a commercial quality, low carbon steel. The basket component parts are corrosion protected by a hot dip galvanized process. The perforated basket assembly has an open area of approximately 64% to provide the necessary surface area for heat transfer between the steam/air mixture and the ice to limit the containment pressure within design limits. The basket heat transfer performance was confirmed by the autoclave test.

Interconnection couplings and stiffening rings are located at the bottom and 6-feet. levels, respectively, of each basket section. The bottom coupling and stiffening ring are cylindrical in shape and approximately 3 inches high with a rolled internal lip (Unit 1/Unit 2 and or welded bottom ring. (Unit 2). The lip (Unit 1 or lip/ring (Unit 2) provides stiffening to the basket and a stop for the cruciforms at 6 feet intervals. These cruciforms prevent the ice in the basket from displacing axially in the event of loss of ice caused by sublimation or partial melt down due to accident conditions. These couplings are attached to the ice basket by locking sheet metal screws and basket detents.

The baskets are assembled into the lattice frames to form a continuous column of ice that is 48 feet high. The bottom wire mesh is designed to allow water to flow out of the basket and has attachments for mechanical connection to the lower support structure to prevent uplift of the ice baskets during SSE and DBA. The lattice frames provide only lateral ice basket support at intervals corresponding to the stiffened ice basket sections.

The vertical loads of the ice and ice basket is transmitted by the basket to the lower support structure. The attachment between the ice basket and the lower support structure is disengaged to permit weighing of the baskets. The columns of ice can be lifted and removed in sections, and provision is made for lifting and weighing the whole length of selected columns for surveillance purposes.

Fabrication

The fabrication steps are as follows:

1. The sheet metal is purchased in the hot-rolled and pickled condition.

2. The perforator oils and perforates the material and ships to the basket fabricator.

3. The basket fabricator rolls the perforated metal into a cylindrical shape 12 inches in diameter by 141.57 or 143.25 inches long (for bottom basket or upper basket respectively) and material is degreased.

4. The sides of the rolled cylinder are continuously welded using the gas metal arc process.

5. Following the welding the cylinder is pickled, washed, fluxed, hot dip galvanized, and dipped in a sodium dichromate bath.

6. The couplings and stiffening ring blanks are cut from sheets or coils of hot rolled, pickled and oiled material. These are formed by a rolling process and are 3 inches high with a roll-formed internal lip and are of a diameter to fit inside the perforated basket.

7. The cruciforms are die-formed from steel strip.

8. Following the forming operations, stiffeners and couplings with cruciforms in place are pickled, washed, fluxed, hot dip galvanized, and dipped in a sodium dichromate bath.

9. The column bottom is fabricated by a procedure similar to Item 6 above. The appurtenances are welded in place and the piece is galvanized per Item 8 above.

10. The remaining appurtenances are cut to size, machined, welded where required, followed by galvanizing as above, and plated where required.

11. The completed couplings, bottoms, appurtenances, stiffening rings and cylinders are next assembled. The stiffening rings are inserted inside the cylinder until the side is adjacent to the 2.5-inch upperforated area in the center of the cylinder and attached by a self-drilling, self-tapping, locking machine screw and four basket detents.

12. For the column bottom, two U-bolts and nuts and washers fasten the mounting bracket assembly to the plate of the basket end.

13. The bottom is inserted into the cylinder until the cylinder rests against the step of the bottom and is attached mechanically by 12 self-drilling, self-tapping, locking machine screws.

14. For the upper baskets, the couplings are inserted in the cylinders approximately 1½-inches and attached with 12 screws as above.

15. All welding and inspection is performed in accordance with AWS publication D1.1-72, including latest revisions.

Installation

The completed baskets are placed in the lattice frames from the top deck by first lowering a bottom basket into the lattice frames and locking in place, extending approximately 2 inches above the top lattice frame. The second upper basket is lifted with the crane and gripper fixture and placed on top of the bottom basket inserting the coupling into the top of the bottom basket and attaching with, self-drilling, self-tapping screws.

Next the locking or holding fixture is released and the two baskets lowered until the top is approximately 2 inches above the lattice frames as above. The third and fourth baskets are installed in the same manner as the second.

When the full column is assembled and ready to set on the lower support structure, the bolts and mounting bracket are loosened and the column lowered to facilitate alignment of the yoke with hole in the support structure. After alignment and insertion of the clevis pin, the 4 bolts are tightened. A hitch pin cotter is inserted to retain the clevis pin.

Materials

The listing of acceptable materials for the ice basket are presented in Section 6.7.18.

6.7.4.3 Design Evaluation

Basket Evaluation

The perforated metal baskets, manufactured from A-569 (Unit 1/Unit 2 and/or A1011 (Unit 2) low-carbon 14-gage sheet with 1.0-inch by 1.0-inches holes on 1.25-inches centers, are evaluated by analyses and tests and found to be within the allowable limits defined in Section 6.7.16. Three different methods are used in determining basket adequacy. The first method employs classical strength of materials techniques, the second uses limit analysis, and the third confirmed the basket integrity by tests.

Stress Analysis

This method considers the ice basket as being composed of a number of line (vertical basket element) and stay (circumferential basket element) elements and the collapse of the ice basket may be precipitated by the local yielding and/or buckling of the individual line elements.

When the basket is loaded both axially and laterally as a beam, the line elements are subjected to axial compression, a lateral shear and a bending load. This combined stress state can possibly lead to local yielding, plastic collapse, line element buckling and ultimately to structural failure. All these modes of possible failures are analyzed and the results are found to be within the allowable criteria. Analysis indicates that the critical line element buckling load is about 77,000 lbs. The maximum vertical load, D + SSE is 2753 lbs. Therefore the possibility of elastic buckling is remote. For a case with only lateral load, the analysis indicates that a factor of safety 3.15 exists between the allowable basket load and the maximum lateral load that exists. A summary of stresses is tabulated in Table 6.7-3. For the various design cases considered, it is seen that the design stress is always below the allowable stress.

Analysis was also made of the case where the ice melts out so that it occupies only one-half side of the basket. The eccentricity would be 3 inches, but the ice mass would be halved, giving a shear stress of 450 psi. This gives a combined maximum shear stress of 3850 psi, again, well below the allowable.

Limit Analysis

Limit analysis is performed on the ice basket in order to determine by analysis the lower bound collapse load when the basket is simultaneously loaded in the axial and lateral directions. The following modes of failure are considered:

1. Plastic collapse of the compression side,

2. Shear yield of the neutral plane.

A summary of the combinations of concentric axial load and distributed load that causes basket failure is presented in Figure 6.7-9. Also superimposed in this figure is the design and test load envelope.

Ice Basket Appurtenance Evaluation

The ice basket connections are analyzed to ensure structural integrity during all design load combinations of dead weight, operating basis earthquake, safe shutdown earthquake, and design bases accident. The primary area of concern is the ice basket to lower support structure connection. This area is shown in Figure 6.7-8.

The item, material and minimum yield stress are presented in Table 6.7-4. The allowable stress limits for D + OBE, D + SSE and D + DBA, and D + SSE + DBA are tabulated in Tables 6.7-5 through 6.7-7 respectively. The loads used in the analysis of these parts envelop minimum design loads plus load factors necessary for the Watts Bar analysis.

Clevis Pin

The clevis pin transmits the ice basket loads to the lower support structure through a 1-inch x 2-inch bar welded to the top of the structure. A minimum clearance of 1/16 inch is provided both vertically and horizontally to provide a pinned connection, thereby eliminating the transfer of any moment to the structure resulting, from basket deflection because of horizontal loads.

The stresses on the ½-inch diameter pin are tabulated in Table 6.7-8.

Column Bottom Mounting

The mounting bracket is attached to the basket bottom as shown in Figure 6.7-8. The design loads are transmitted through the mountings and clevis pin from the ice basket bottom.

The stresses in the mounting bracket, plates and bolt are tabulated in Tables 6.7-9 through 6.7-11, respectively.

Ice Basket End

The column bottom is shown in Figure 6.7-8. The loads that are transmitted through the clevis pin assembly are distributed to the ice basket through the rigid plate and the cylindrical ice basket end section. Wire mesh is used to contain the ice and to provide drainage for water. The stress summary for the ice basket end is shown in Table 6.7-12.

The intermediate ice basket coupling screws were also analyzed and the results of the analysis are given in Tables 6.7-13 through 6.7-16 and indicate that they are structurally adequate for maximum loading conditions defined in Section 6.7-16.

6.7.5 Crane and Rail Assembly

6.7.5.1 Design Basis

Function

The crane and rail assembly is designed to carry components and tools into, out of, and within the ice condenser area during erection, maintenance, and inspection periods.

Criteria and Codes

The crane is designed in accordance with the requirements of the Electric Overhead Crane Institute Specification 61. It is designed so that under all loadings it is not derailed.

The rail is designed according to Section 6.7.16. These criteria provide assurance that the rail maintains its structural integrity.

Design Conditions

The service temperature range is 15°F to 100°F.

During unit erection, two cranes can be used in the ice condenser region, each carrying up to 6,000 pounds. A separation of at least two bays is maintained between their centers. Prior to installation of air handling units, one crane is removed. The heaviest load actually expected after this time is less than 2,500 pounds. The crane remains normally parked (without load) outside the ice condenser while the reactor is at power. The crane and supporting structure are designed to withstand dynamic loading during operating modes specified above.

The design loads for the crane are presented in Table 6.7-17.

6.7.5.2 System Design

The design of the 3-ton capacity crane is shown in Figure 6.7-10. The bridge, boom, and hoist of the crane are all motor operated. The bridge speeds are approximately 38 and 110 feet per minute. The boom member is capable of rotating 360° in either direction at a speed of approximately 2 revolutions per minute. The electric hoist is mounted on the boom member with two stainless steel cables reeved over two sheaves mounted on the boom and around two sheaves on the hook block assembly. The hoist provides approximately 71 feet of lift at speeds of 7 and 20 feet per minute. It is equipped with an upper and lower limit switch to ensure that the cables will not completely unwind from the hoist drum. The hoist automatically switches to low speed approximately 2 feet below the highest point of travel.

The total crane weight is approximately 7200 pounds.

The predominant material of construction is A36 steel. The main structural members are painted to prevent corrosion.

The crane travels on two circular rails that run through the ice condenser area as shown in Figure 6.7-11. The circular diameters of the rails are 95 and 109 feet. The top flange plate and rail section are continuously welded to the web plate under controlled conditions. The top flange and web plates are A-441 steel heat treated and normalized, fine grain practice, and the lower rail section is special analysis steel with a hard non-peening rolling surface.

6.7.5.3 Design Evaluation

The crane rails and supporting structures are analyzed as a part of the top deck structure (see Section 6.7.10). All stresses were maintained within limits prescribed in the design criteria, Section 6.7.16, for all design conditions defined in Section 6.7.5.1.

6.7.6 Refrigeration System

6.7.6.1 Design Basis

Function

The refrigeration system serves to cool down the ice condenser from ambient conditions of the reactor containment and to maintain the desired equilibrium temperature in the ice compartment. It also provides the coolant supply for ice machines A, B, and C during ice loading. The refrigeration system additionally includes a defrost capability for critical surfaces within the ice compartment.

During a postulated loss-of-coolant accident the refrigeration system is not required to provide any heat removal function. However, the refrigeration system components which are physically located within the containment must be structurally secured (not become missiles) and the component materials must be compatible with the post-LOCA environment.

Design Conditions

1. Operating Conditions

   See individual component sections:

   A. Floor cooling - Section 6.7.1

   B. Air handling units (AHUs) - Section 6.7.7

2. Performance Requirements

   A. The mandatory design parameters that relate to refrigeration performance are:

      i. Maximum total weight of ice in columns 3.0×10^6 lbs.

      ii. Minimum total weight of ice in columns 2.26×10^6 lbs.

iii. Nominal ice condenser cooling air temperature 10°F - 15°F*.

* Technical Specifications limit the maximum ice bed temperature to less than or equal to 27°F.

B. The design must also provide a sufficiently well-insulated ice condenser annulus such that, with a complete loss of all refrigeration capacity, sufficient time exists for an orderly reactor shutdown prior to ice melting. A design objective is that the insulation of the cavity is adequate to prevent ice melting for approximately 7 days in the unlikely event of a complete loss of refrigeration capability.

C. The not-directly-safety-related design-objective parameters are:

i. <u>Ice Sublimation</u>

Ice sublimation and mass transfer is reduced to the lowest possible limits by maintaining essentially isothermal conditions within the ice bed and by minimizing local temperature gradients. A design objective is to limit the sublimation of the ice bed to less than 2% per year by weight. The normal steady-state sublimation appearing on the wall panels as frost is calculated to be significantly less than the total design objective. Calculations incorporating both radiative and convective modes of heat transfer result in a sublimation rate of less than 0.3% per year to the wall panels.

ii. An appropriate combination of refrigeration capacity and insulation capability is achieved to permit the following:

a. Maintain the average ice bed temperature in the range of 15° to 20°F under the most adverse non-accident conditions. Technical Specifications limit the maximum ice bed temperature to less than or equal to 27°F.

b. Cool the ice condenser down to 15°F in 14 days (initial cooldown prior to ice loading).

The ice condenser is structurally designed to withstand the various extreme loading parameters including DBA + SSE. The ice condenser design and the reactor containment supporting walls are analyzed for heat transfer through the boundaries of the ice condenser. The configuration and sizing of the cooling components is then determined to achieve the various design requirements.

One of the most important design criterion for the ice condenser is that the insulation must maintain the ice condenser chamber below 31°F for a significant period of time given that a malfunction or failure of any refrigeration component has occurred. Most system anomalies can be remedied during this period. For any repair which would require more time, a scheduled reactor shutdown can be completed in a safe and orderly fashion. Eliminating the "emergency factor" from the operation of the refrigeration system places the performance of the refrigeration components in an operational category without mandatory safety related design requirements.

6.7.6.2 System Design

The refrigeration system serves as a central heat sink for sensible heat and heat of fusion picked up, respectively, in the ice condensers and in the ice machines. A circulating system of ethylene glycol solution carries the heat from the various heat transfer surfaces to the chiller packages. Cooling of the ice condenser is achieved by a three stage system:

First stage - Refrigerant Loop
Second stage - Glycol Loop
Third stage - Air Cooling Loop

First Stage - Refrigerant Loop

Five 50 ton chiller packages are installed in the unit. Each package consists of two separate 25 ton compressor units, individually operable. See Figure 6.7-12 for refrigerant cycle diagram. Ethylene glycol solution is cooled during its passage through the evaporator, and heat is removed from the chiller unit by cooling water flowing through the condenser. The condenser cooling water is provided from the non-essential service water system. The chiller units operate individually to maintain outlet temperature of ethylene glycol at approximately -5°F, nominal.

Refer to Table 6.7-18 for chiller package parameters such as operating temperatures, flow rates, pressure drops, rating basis, etc.

Second Stage - Glycol Loop

The second cycle (Figure 6.7-13) carries the heat removed from the ice condenser air handling units, the floor cooling system and the ice machines (when operating) to the refrigerant cycle evaporator/cooler units. The liquid circulating, through this cycle is a corrosion inhibited 50% ethylene glycol solution. It is compatible with most common piping materials and standard gasket and packing materials. Piping and valve materials used in this loop are predominantly carbon steel with stainless or alloy trim. Diaphragm valves are provided with ethylene propylene diaphragms. Piping and equipment carrying chilled ethylene glycol solution are covered with low temperature thermal insulation.

Six glycol circulating pumps two operating and four on standby are provided to convey the cooled glycol from the ten refrigeration units four normally operating and six on standby to the air handling packages (30 dual units per containment) and to the ice compartment floor cooling system of each containment. The design includes provisions for interconnecting the chiller units and pumps as required. The heated glycol is then returned to the refrigeration units thereby completing the glycol loop. The heat is extracted from the air in its passage through the air handlers and from the floor cooling system. Two rows of air handlers located along inner and outer walls are served by respective glycol supply and return headers. The return headers are connected to a vented expansion tank located above the upper deck in each unit.

Pairs of containment isolation valves are installed on supply and return lines on both sides of containment penetration. Closure of these valves in response to a containment isolation signal, (Phase A, derived from safety injection or manually) isolates the ethylene glycol piping inside the containment vessel from the external refrigeration system. In the event of a LOCA, the glycol heats up from approximately -5°F or 0°F to the containment accident temperature and expands harmlessly into the expansion tank. The liquid trapped between a pair of isolation valves is relieved around the inner isolation valve through a bypass line via a small check valve. The bypass line also contains test connections for periodic leak testing of the isolation valves and the check valve.

The ice condenser floor is kept cold by chilled glycol solution circulating through pipe coils embedded in the concrete wear slab (see Section 6.7.1 for floor cooling diagrams). During normal operation, one floor cooling pump feeds a circular header, which distributes the coolant to individual coils located in each bay. A second circular header returns the flow to pump suction.

The glycol solution is maintained at the proper temperature by continuously bleeding solution out of the system and feeding cold solution into it at the same rate. The cold solution is taken from the glycol stream returning from the air handling units to the external refrigeration system. The bleed flow is sent back into the same line downstream of the feed connection. Feed and bleed flow is maintained by the same pump that drives solution through the coils. Bleed flow rate is regulated by a temperature control valve. A second pump is available for use while pump No. 1 is being serviced. A manual throttling valve bypassing the temperature control valve can perform the latter's function during brief maintenance periods.

Floor temperature is generally maintained between the temperatures of the ice bed and the wall panels. There should, therefore, be essentially no frosting on the floor surface. It is necessary, however, to heat the floor above 32°F any time the wall panels are being defrosted in order to keep the water melting off the wall panels from freezing to the floor. At this time, the floor is heated with warm glycol. After defrosting is completed, the system is restored to its normal cooling status. The defrost cycle is relatively brief and its effect on the ice bed is negligible.

Components requiring periodic maintenance (pumps, heater, control valve) are located in the upper compartment. The cooling coils in the concrete wear slab rest on a steel plate to effectively intercept heat passing up through the floor. The coils are made of heavy steel pipe to minimize chances of developing a leak by gradual corrosion of pipe material. Should a leak develop, any individual loop can be isolated by closing two valves inside the lower region of the ice condenser.

Table 6.7-18 has additional detailed parameters for the glycol cycle components.

Third Stage - Air Cooling Loop

The ice condenser compartment is designed to be kept below the freezing point throughout the life of the unit. It is cooled to 15°F prior to ice loading and kept between 10°F to 15°F, nominal, indefinitely, barring occurrence of a loss-of-coolant accident, extensive failure of the refrigeration system, or permissible excursion during ice loading. Ice bed temperature is maintained at the specified level by means of chilled air circulating through the boundary planes of the compartment. Starting in the upper plenum, which constitutes the top boundary, air enters one of 30 air handling packages located in the plenum. The air handler cools the air then blows it down through a series of insulated duct panels lining the inner, outer and end walls of the ice condenser. When the air reaches the lower support structure at the inner wall or end walls or the floor level at the outer wall, it turns back up to the plenum through a parallel path in the wall panels. See Figure 6.7-14 for a schematic flow diagram of the air cooling cycle.

The air handling units are designed to provide a discharge or duct entrance air temperature of 10°F, nominal. A temperature sensor is located in the cold air stream of each air handler outlet which provides local indication of the discharged air temperature.

The air handling units are designed for automatic self-defrost operation. The self-defrost cycle is initiated by a timer set to initiate defrosting at intervals to ensure less than 2/3 occlusion of the air passage across the cooling coils. The timer switch completes a circuit to the fans, through a relay, which in turn allows gravity closure of the damper in that air handler. A circuit is also completed to the power-operated valve of the air handling unit, stopping glycol flow through the coil. When the air damper closes, a limit switch is activated which initiates closure annunciation in the control room. The self defrost cycle is terminated by the timer upon completion of the defrost cycle. In addition to the coil defrost heaters mounted on the face of the coil, each unit has a drain pan heater and a condensate drain heater. These heaters prevent refreezing of the coil defrost water during the defrost cycle. Over-temperature protection is provided by a high temperature thermostat set to disable the defrost heaters if the fan box temperature reaches 180°F. This high temperature limit switch must be manually reset.

Provisions also exist for defrosting the wall panels by circulating heated air through the wall panels. The structural function and capabilities of the air cooling cycle components are discussed in the following sections:

1. Air handling unit - Section 6.7.7
2. Wall panels - Section 6.7.2
3. Air distribution ducts - Section 6.7.12

Table 6.7-18 has additional parameters for the air handling units.

6.7.6.3 Design Evaluation

The refrigeration system is sized to maintain the required ice inventory even under worst-case operating conditions. The chiller package total capacity is sufficient to maintain both ice condensers. Worst-case conditions are:

1. Lower containment air temperature 120°F

2. Upper containment air temperature 110°F

3. Equipment room air temperature 120°F

4. Exterior containment wall design air temperature 110°F

Items 1 and 2 are limits stated in the Technical Specifications. Item 4 is the design dry-bulb temperature in the region of Tennessee where the Watts Bar units are located for a 50 year hot summer, plus an additional margin of 9°F. The 1% factor is defined such that only 1% of the time the dry-bulb temperature during the summer months is above the specified temperature for a 50 year hot summer. Data was obtained from ASHRAE climatic guide for cooling and heating design conditions. For an average summer, the 1% design dry-bulb temperature is 96°F and, for a 50 year hot summer, is 101°F. The average (4 quadrant) sol-air temperature for vertical walls corresponding to a maximum dry-bulb temperature of 95°F is about 107°F.

The major thermal boundaries of the ice condenser, including the floor, cooled walls with ducts, lower inlet doors, and top deck support beams are analyzed using a Westinghouse-developed computerized technique, TAP-A, (or TAP-B), which is a program for computing transient or steady-state temperature distributions (WANL-TME-1872, Dec. 1969, Subcontract NP-1).

The TAP-A program is applicable to both "transient and steady-state heat transfer in multi-dimensional systems having arbitrary geometric configurations, boundary conditions, initial conditions, and physical properties. The program can be utilized to consider internal conduction and radiation, free and forced convection, radiation at external surfaces, specified time dependent surface temperatures, and specified time dependent surface heat fluxes."

The solution of the general heat conduction equation is determined with finite difference techniques. The program solves the equation as determined for the particular finite element or nodal model set up, either explicitly or implicitly. All cases studied for the ice condenser are solved implicitly.

The TAP-B program is a variation of TAP-A but includes fluid coupling to the finite element model. The TAP-B variation was used to analyze the cooled wall panels. Since the duct air temperature distribution is included in the model it is possible to evaluate the temperature distribution of the surface of the wall panel facing the ice condenser over the complete length of the duct.

The wall panel heat load comprises about 60% of the total heat load, through the thermal boundaries with the inner surface area of the wall panels covering just under 30,000 ft^2.

The wall panel model for the crane wall is 48 feet long, with 8 axial stations, each 6 feet in length. The width of the model covers the region from the centerline of the duct region to the centerline of the lap strip region.

There are approximately 1,000 interior and surface nodes for the 48-foot length of the model which consists of half of a duct section.

Roughly 70% of the thermal load through the wall panels flows through the mounting brackets (or about 50% of the total thermal load of the ice condenser). The cold boundary temperature of the model was assumed to be 12°F in the ice bed with a 10°F duct entrance temperature.

The basic floor model utilizes TAP-B. The basic floor design is analyzed with fluid coupling. The results of the basic model justify the design concept. Variations in the basic floor are checked by hand calculations for overall thermal load. The basic floor model is comprised of approximately 1200 nodes in 5 layers and covers one quarter of a typical floor bay, of which there are 24 bays. The air temperature over the floor is assumed to be 15°F. The temperature of the glycol boundary is calculated for each fluid node. Over 90% of the heat entering the floor region is found to be removed by the floor cooling system. Use is made of the transient capabilities of the program to determine the defrost or warmup time required when the glycol is heated. The heat transfer through the top surface of the floor is in two directions, both into and out of the wear slab. The net flow from the top surface to the ice condenser chamber is about 1000 Btu/hr. About 75,000 Btu/hr total is absorbed by the floor glycol coolant using the basic model.

The lower inlet door region while not contributing significantly to the overall thermal load on the refrigeration system is extremely important when considering sublimation. Various models of portions of the door are postulated to determine effective means of limiting the heat flux through the lower inlet doors.

The total heat load through doors with appropriate insulation is maintained at less than 10,000 Btu/hr to the ice bed. The door assembly is analyzed in two segments. There are 24 complete 2 door assemblies in the ice condenser. The first door model covers the region from the centerline of one door panel to the central seal region. Hand calculations are used to determine the nature of the convection between the two door panels in the central seal region, and in the outer hinge region. The information on the type of convection present is necessary to be gained from positioning flaps or boots around the door perimeter. Flaps are not considered necessary in the door center because the convection is determined to be laminar with air conduction dominating.
The central door model contains about 150 internal nodes including insulation. The second region covered by a model is the hinge region. The hinge model is 15-inches deep (about 1/6 of the door length) and includes effects of the reinforcement channels along the full width of the door. The extremities further away from the hinge region are only grossly modeled. There are a total of 168 internal nodes in the "hinge" model including a protective boot around the hinge.

The hinge model also includes effects of the pillar in the crane wall upon which the door is mounted. The hinge region is of major importance in contributing to the internal thermal load with most of the heat input coming from the massive concrete pillar. It is necessary to protect the hinges with boots to limit the convective heat transfer which is quite effective in reducing the heat flow.

The top deck support beams are similarly modeled using TAP-A. The beams are a major source of thermal load in the plenum are thermal boundaries but only a small fraction of the total thermal load on the air handlers (not including air handler motor heat).

The modeling required for analysis of the components is extensive and detailed. The admittance of each node and connection; involving the determination of the length, volume, and area of each element was conservatively estimated where simplification of the model was required. The models are realistic since sufficient detail was considered and all significant modes of heat transfer were considered. Hand calculations backup all major assumptions used to arrive at a model.

The summation of the thermal analysis gives a total nominal thermal load of 36 tons or 432,000 Btu/hr.

The breakdown is listed below. The values given are considered to be nominal expected loads. Design change required as a result of change in air distribution duct configuration or other design reevaluations would, of course, change the final summation. The final thermal load is still maintained at the same level consistent with stated refrigeration requirements.

	Btu/hr (10^4)
Wall panels	29.2
Plenum and Top Deck	10.11
Leakage 50 cfm	1.1
Lower inlet doors	2.0*
Floor	10.0
End walls	1.17
Total thermal load	53.58

The calculated heat loads show that a heat gain of 432,000 Btu/hr per containment may be expected from thermal boundaries of the ice condenser. Additionally each air handling unit fan motor generates less than 6,000 Btu/hr (subtotal 30 AHU x 6,000 Btu/hr = 180,000 Btu/hr) based on 30 operating air handlers with a design allowance of 1.5 in. of H_2O over the air delivery system. The floor cooling system, including pump heat, has a heat gain of 90,000 Btu/hr nominal.

* Calculated Load < 1 x 10^4 Btu/hr
Design Allowance = 2 x 10^4 Btu/hr (includes miscellaneous items in addition to door load.)

The circulating pumps (2 operating) add a total of 100,000 Btu/ hr. The piping is estimated to pick up 7,000 Btu/hr. Therefore a chiller package capacity of about 800,000 Btu/hr per containment (base load) is required. The glycol chillers are common equipment serving both Units, and the total chiller capacity was chosen to be three (3) times the total two Unit base load which is 2,400,000 Btu/hr. Since each chiller is rated nominally at 300,000 BTU/hr, depending on cooling water temperatures, the total installed capacity is therefore, 3,000,000 BTU/hr (nominal rating) which exceeds the required total capacity and provides operational margin.

The six circulating pumps (2 operating, 4 standby) are conservatively sized to deliver the required coolant. Four standby pumps are included in the design to assure adequate cooling solution flow even in the event of a pump failure. Similarly the air handling units are conservatively sized to handle the worst case cooling load. Thirty dual air handling packages are installed based on a 10/7 ratio of installed capacity to base load.

The ice bed is sufficiently subcooled and insulated so that even a complete breakdown of the refrigeration system, or of all air handlers, does not permit the average temperature of the ice bed to rise above the melting point of the borated ice for a period of approximately one week. Anomalous conditions in the ice condenser are indicated by alarm annunciation from expansion tank level switches, the temperature monitoring system, or the door position monitoring system. Refer to Section 6.7.15 for a discussion of the ice condenser instrumentation system.

If one bay in the floor is not cooled because the glycol flow has to be isolated from that bay, the heat load from that bay is about 4,500 Btu/hr. The additional sublimation rate would be under 0.35% per year per bay. It would be expected that one bay would not be permitted to go uncooled for extensive length of time. Once an operational sublimation rate is established, it would not be unreasonable to assume that possibly three isolated, uncooled floor bays could be permitted to be uncooled for about 1 year. If the floor cooling system is shut off completely, it should be put back in operation as soon as convenient. An annual sublimation rate of about 5% per year will result with no cooling in the floor, which would require ice bed replenishing in 3 years.

6.7.7 Air Handling Units

6.7.7.1 Design Basis

Air Handling Units (AHU)

During normal operation the air handling units serve to cool the air and to circulate the cooled air through the panels in the ice condenser walls to keep the ice subcooled in the ice beds. Normal structural loads expected are dead weight, seismic, and thermal loads. During an accident the AHU structure is designed to resist the normal structural loads plus SSE + DBA induced loads. Welding, welder qualification and weld procedures are in accordance with USASI B31.5 Refrigeration Piping and the ASME Boiler and Pressure Vessel Code, Section IX "Welding Qualification".

AHU Support Structure

1. Function

   The AHU support structure supports the air handling unit package under various design conditions which are detailed below.

2. Design Criteria and Codes

Refer to Section 6.7.16

3. Design Conditions

A. Normal Operation

Deadweight loads due to AHU, structure	2500 lbs
Design temperature, min.	15°F

B. Accident Conditions

Post-accident temperature (no uplift)	250°F

6.7.7.2 System Design

Air Handling Units

Each AHU is supported from its support structure, transmitting its major loads to top deck cross beams.

The air is drawn by each AHU from the upper plenum, is cooled in the AHU and is discharged into the air distribution header. The gross cooling capacity of each AHU package is 30,000 Btu/hr with the plenum air entering, at 19°F estimated and cooled by the AHU to 10°F nominal. Each package has a 2,200 cfm nominal air delivery capacity. The entering glycol mixture is at -5°F nominal temperature and the discharge glycol temperature is 1.0°F nominal. Electrical power is provided for fan motor and defrost heaters as well as for control circuits.

In order to limit seismically induced loads the AHU and supports are designed to have a natural frequency in excess of 20 Hz. All materials used in the AHUs are compatible with both normal and post-LOCA environments.

AHU Support Structure

The support structure supports the air handling unit vertically and tangentially from the cross beam of the top deck structure and is radially hinged from channels attached to the crane or containment wall. All parts are coated with a paint suitable for use inside containment. Figure 6.7-15 shows the design of the structure.

6.7.7.3 Design Evaluation

The pressure drop through the ducts and manifolds was estimated by using loss coefficients determined by using a standard reference[8] as a guide. The pressure drop through the air handlers was determined by test. The overall system flow rate was established by superimposing the system flow versus ΔP curve over the fan flow versus ΔP curve.

With the flow rate established the capacity of the air handlers was determined. First the air handler capacity was theoretically determined for a set of design conditions approximating operating conditions. Next the air handler units were tested by the manufacturer to the set of specified design conditions. It was determined that the theoretical relationships adequately predicted air handler performance and these techniques were then used to adjust the test values to those of actual operation. The gross operating capacity of one air handler is just under 30,000 Btu/hr by test and calculation.

The nominal heat load of 432,000 Btu/hr is adjusted by a factor of 10/7 to ensure adequate capacity under operating conditions for fouling, defrosting or isolated instances of one or several unit failures. Maintenance and inspection ensures reliable mechanical operation and cooling performance.

An estimate of the number of air handlers required is made to initiate the calculation, the flow pressure and rates drops are then calculated and the fan motor heat and heat transfer rates of the air handler unit predicted. The predicted performance is compared with the required capability and the calculation is reiterated varying the number of AHUs until the predicted performance just exceeds the required capability. The final number of required air handlers was determined to be 30 dual units.

A modal frequency analysis was performed for the air handling unit housings and support structure. The results indicate that the design frequency is approximately 20 Hz, so that the fundamental mode is well out of the frequency range of peak amplification on the response spectra. In the process of designing the structure on the basis of stiffness, strength of members subjected to various combinations exceeds specified limits by generous margins.

6.7.8 Lower Inlet Doors

6.7.8.1 Design Basis

Function

The ice condenser inlet doors form the barrier to air flow through the inlet ports of the ice condenser for normal unit operation. They also provide the continuation of thermal insulation around the lower section of the crane wall to minimize heat input that would promote sublimation and mass transfer of ice in the ice condenser compartment. In the event a loss-of-coolant accident causes a pressure increase in the lower compartment, the doors open, venting air and steam relatively evenly into all sections of the ice condenser.

The door panels are provided with tension spring mechanisms that produce a small closing torque on the door panels as they open. The magnitude of the closing torque is equivalent to providing approximately a one pound per square foot pressure drop through the inlet ports with the door panels open to a position equivalent to the full port flow area. The zero load position of the spring mechanisms is set such that, with zero differential pressure across the door panels, the gasket holds the door slightly open. This setting provides assurance that all doors will be open slightly, upon removal of cold air head, therefore eliminating significant inlet maldistribution for very small incidents.

For larger incidents, the doors open fully and flow distribution is controlled by the flow area and pressure drops of inlet ports. The doors are provided with shock absorber assemblies to dissipate the larger door kinetic energies generated during large break incidents.

Design Criteria

1. Radiation Exposure

   Maximum radiation at inlet door is 5 rad/hr gamma during normal operations. No secondary radiation due to neutron exposure.

2. Structural Requirements

   Refer to Section 6.7.16

3. Loading Modes

   A. The door hinges and crane wall embedments, etc., must support the dead weight of the door assembly during all conditions of operation. Door hinges are designed and fabricated to preclude galling and self-welding.

   B. Seismic loads tend to open the door.

   C. During normal operations the outer surface of the door operates at a temperature approaching that of the lower compartment while the inner surface approaches that of the ice bed. During LOCAs, the outer surface is subjected to higher temperatures on a transient basis. Resultant thermal stresses are considered in the door design.

   D. During large break accidents, the doors are accelerated by pressure gradients then stopped by the shock absorber system. During small break accidents, doors open in proportion to the applied pressure with restoring force provided by springs. Upon removal of pressure, doors close as a result of spring action.

4. Design Criteria - Accident Conditions

   A. All doors open to allow venting of energy to the ice condenser for any leak rate which results in a divider deck differential pressure in excess of the ice condenser cold head.

      The force required to open the doors of the ice condenser is sufficiently low such that the energy from any leakage of steam through the divider barrier can be readily absorbed by the containment spray system without exceeding containment design pressure.

   B. Doors and door ports limit maldistribution to 150% maximum, peak to average mass input for the accident transient, for any reactor coolant system release of sufficient magnitude to cause the doors to open.

C. The basic performance requirement for lower inlet doors for design basis accident conditions is to open rapidly and fully, ensuring proper venting of released energy into the ice condenser. The opening rate of the inlet doors is important to ensure minimizing the pressure buildup in the lower compartment due to the rapid release of energy to that compartment. The rate of pressure rise and the magnitude of the peak pressure in any lower compartment region is related to the confinement of that compartment. The time period to reach peak lower compartment pressure due to the design basis accident is approximately 0.05 seconds.

D. Doors are of simple mechanical design to minimize the possibility of malfunction.

E. The inertia of the doors is low, consistent with producing a minimal effect on initial pressure.

5. Design Criteria - Normal Operation

A. The doors restrict the leakage of air into and out of the ice condenser to the minimum practicable limit. The inlet door leakage has been confirmed by test to be within the 50 cfm total used for the ice condenser design.

B. The doors restrict local heat input in the ice condenser to the minimum practicable limit. Heat leakage through the doors to the ice bed is a total of 20,000 Btu/hr or less (for 24 pairs of doors).

C. The doors are instrumented to provide indication of their closed position. Under zero differential pressure conditions, all doors remain open by 3/8 inch.

D. Provisions are made for adequate means of inspecting the doors during reactor shutdown.

E. The doors are designed to withstand earthquake loadings without damage so as not to affect subsequent ice condenser operation for normal and accident conditions. These loads are derived from the seismic analysis of the containment.

F. The door system provides a flow proportioning capability for small break conditions in accordance with Figure 6.7-16.

6. Interface Requirements

A. Crane wall attachment of the door frame is via bolts into embedded anchor plates with a compressible seal. Attachment to the crane wall is critical for the safety function of the doors.

B. Sufficient clearance is required for doors to open into the ice condenser. Items to be considered in this interface are floor clearance, lower support, structure clearance and floor drain operation and sufficient clearance (approximately six inches) to accommodate ice fallout in the event of a seismic disturbance occurring coincident with a loss-of-coolant accident. Original ice basket qualification testing (Topical Report WCAP-8110, Supplement 9-A) has shown freshly loaded ice is considered fused after five weeks. In the event of an earthquake (OBE or greater) which occurs within five weeks following the completion of ice basket replenishment, plant procedures require a visual inspection of applicable areas of the ice condenser within 24 hours to confirm that opening of the ice condenser lower inlet doors is not impeded by any ice fallout resulting from the seismic disturbance. The 24 hour time frame for inspection is applicable during modes where the lower inlet doors are required to be operable; otherwise perform this inspection prior to startup. This alternative method of compliance with the requirements of GDC 2 is credible based upon the reasonable assurance that the ice condenser doors will open following a seismic event during the 5 week period and the low probability of a seismic event occurring coincident with or subsequently followed by a Design Basis Accident.

C. Door opening and stopping forces are transmitted to the crane wall and lower support structure, respectively.

Design Loads

Pressure loading during LOCA is provided by the Transient Mass Distribution (TMD) code from an analysis of a double-ended hot leg break in the corner formed by the refueling canal, with 100% entrainment of water in the flow. For conservatism, TMD results were increased by 40% in performing the design analysis for the lower inlet doors.

The lower inlet door design parameters and loads are presented in Table 6.7-19.

6.7.8.2 System Design

Twenty-four pairs of inlet doors are located on the ice condenser side of ports in the crane wall at an elevation immediately above the ice condenser floor. General location and details of these doors are shown in Figures 6.7-17 through 6.7-21. Each door panel is 92.5 in. high, 42 in. wide and 7.5 in. thick. Each pair is hinged vertically on a common frame.

Each door consists of a 0.5 in. thick fiber reinforced polyester (FRP) plate stiffened by six steel ribs, bolted to the plate. The FRP plate is designed to take vertical bending moments resulting from pressures generated from a LOCA and from subsequent stopping forces on the door. The ribs are designed to take horizontal bending moments and reactions, as well as tensile loads resulting from the door angular velocity, and transmit them to the crane wall via the hinges and door frame.

Seven inches of urethane foam (Unit 1/Unit 2) and/or polyisocyanurate (Unit 2) are bonded to the back of the FRP plate to provide thermal insulation. The front and back surfaces of the door are protected with 26 gauge stainless steel covers which provide a complete vapor barrier around the insulation. The urethane foam (Unit 1/Unit 2) and/or polyisocyanurate (Unit 2) and stainless steel covers do not carry overall door moments and shearing forces.

Three hinge assemblies are provided for each door panel; each assembly is connected to two of the door ribs. Loads from each of the two ribs are transmitted to a single 1.572-inch diameter hinge shaft through brass bushings. These bushings have a spherical outer surface which prevents binding which might otherwise be caused by door rib and hinge bar flexure during accident loading conditions.

The hinge shaft is supported by two self-aligning, spherical roller bearings in a cast steel housing. Vertical positioning of the door panel and shaft with respect to the bearing housing is provided by steel caps bolted to the ends of the shaft and brass spacer rings between the door ribs and bearings. Shims are provided between the shaft and caps to obtain final alignment. Each bearing housing is bolted to the door frame by four bolts, threaded into tapped holes in the housing. Again, shims are provided between the housings and door frame to maintain hinge alignment. Hinges are designed and fabricated to prevent galling and self welding.

The door frame is fabricated mainly from steel angle sections, 6 inches x 6 inches on the sides, and 6 inches x 4 inches on the top and bottom. A 4 inches central I beam divides the frame into sections for each door. At each hinge bracket, extensions and gusset plates, fabricated from steel plate, are welded to the frame to carry loads to the crane wall.

The door panel is sealed to the frame by a compliant rubber seal which attaches to channels welded to the door frame. During normal unit operations these seals are compressed by the cold air head of the ice bed acting on the door panels. As the seals operate at a much warmer temperature than the ice bed, frosting of the seal region is extremely unlikely.

Each door is provided with four flow proportioning springs. One end of each spring is attached to the door panel and the other to a spring housing mounted on the door frame. These springs provide a door return torque proportional to the door opening angle and thus satisfy the requirement for flow proportioning. In addition, they assure that the doors close in the event they are inadvertently opened during normal unit operations. The springs are adjusted during assembly such that, with no load on the doors, the doors are slightly open. For small door openings, the required 3/8-inch effective door opening is controlled by a 3/8 inch gap between panels and is, thus, independent of the door position as measured in degrees.

In order to dissipate the large kinetic energies resulting from pressures acting on the doors during a LOCA, each door is provided with a shock absorber assembly as shown in Figure 6.7-21. The shock absorber element is a sheet metal air box 93 inches high, 42 inches wide, and 29 inches thick at its thickest section. The air box is attached to a back plate assembly which is bolted to the ice condenser lower support structure.

Two edges of the sheet metal box are fastened to the ends of back plate by clamping bars and bolts, making them air tight joints. The sheet metal is bent such that it has an impact face and a prefolded side.

When the lower inlet doors open due to sudden pressure rise, they impact on the impact face of the air box. The impact face moves with the door. Because of a restraining rod within the box, the prefolded side of the air box collapses inwards. The volume of the air trapped in the air box decreases as the impact face moves towards the back plate, thereby increasing air pressure. Part of the kinetic energy of the door is used up in compressing air. To prevent excessive pressure rise, the air is allowed to escape through the clearance gap between the sheet metal and end plates. A portion of the energy of the doors is also dissipated in buckling of the stiffners.

Material

Door materials are consistent with the listing of acceptable materials as presented in Section 6.7.18. All exposed surfaces are made of stainless steel or coated with paint suitable for use inside the containment. All insulation material is compatible with containment chemistry requirements for normal and accident conditions.

6.7.8.3 Design Evaluation

The lower inlet doors are dynamically analyzed to determine the loads and structural integrity of the door for the design basis load conditions.

Using results from the computer program TMD (transient mass distribution) as input, the door dynamic analysis is performed using the "DOOR" Program. This computer program has been developed to predict door dynamic behavior under accident conditions. This program takes the door geometry and the pressures and calculates flow conditions in the door port. From the flow are derived the forces on the door due to static pressure, dynamic pressure and momentum. These forces, plus a door movement generated force, i.e., air friction, are used to find the moment on the door and from this are derived the hinge loads. Output from the program includes door opening angle, velocity and acceleration as functions of time, as well as both radial and tangential hinge reactions.

Analysis Due to LOCA

The net load distributions on the door for both opening and stopping are determined by considering the applied pressures acting on the door and then solving the rigid body equations of motion such that the net forces and moments at the hinge point are zero. In the process, this produces expressions for the inertial forces in the door and a hinge reaction as functions of the applied pressure.

The expressions for net load distribution are integrated to determine door shear and moment as functions of distance from the hinge point. The resultant load, shear and moment distribution curves and the total hinge loads, calculated by the "DOOR" Program, provides the inputs for subsequent stress analysis.

Using this input, the door assembly is analyzed as a stiffened plate structure with vertical bending being taken by the FRP outer plate and horizontal bending plus radial tensile loads being resisted by the steel ribs. Since inertial forces are directly accounted for in the analysis, no dynamic load factor was applied.

Hinge pin, hinge bracket, and frame stresses are analyzed under hinge reactions considering the effects of tension, shear bending, and torsion as appropriate. For these components, a dynamic load factor of 1.2 was calculated and applied.

Stresses in the flow proportioning springs are calculated considering dynamic effects as well as static ones. Welded and bolted connections are analyzed as part of the overall door, frame and hinge analysis.

All portions of the door and frame show factors of safety greater than one. The general acceptance criterion is that stresses be within the allowable limits of the AISC-69 Structural Code. This provides an additional margin of conservatism over the general ice condenser design criteria for D + DBA which permit stresses up to 1.33 times the AISC limits. For materials and components not covered by the Code, i.e., bearings, non-metallic materials, etc., conservative acceptance criteria are established on the basis of manufacturer's recommendations and/or engineering evaluations.

Flow proportioning characteristics of the door are evaluated by determining the door opening as a function of applied pressure. Assuming a triangular pressure distribution across the door, the flow area vs pressure at full door opening, is determined to be consistent with the curve shown on Figure 6.7-16. In addition the effects of door closure were evaluated assuming the pressure is suddenly released from a fully opened door and the door allowed to shut under the effect of the door proportioning springs. Stress levels in the door, gasket, and frame are found to be acceptable for this condition. In addition to the above analysis, full scale simulated blowdown tests have been performed on prototype door and shock absorber assemblies. These tests confirm the adequacy of these components at test levels up to 140% of maximum loading conditions predicted by the TMD Code.

Analysis of Seismic Load

Seismic analysis of the doors indicates that stresses are insignificant in comparison with those occurring during a LOCA. Under a SSE the doors could open several inches (actually, the crane wall will move away from the doors). At the termination of the earthquake, the doors immediately close and reseal under the effects of proportioning spring tension and the ice bed cold air head. Thus, any loss of cold air during a OBE or SSE is small and limited to a short period of time.

The dynamic testing of the air box shock absorber is discussed in Reference [13].

6.7.9 Lower Support Structure

6.7.9.1 Design Basis

Function

The lower support structure is designed to support and hold down the ice baskets in the required array, to provide an adequate flow area into the ice bed for the air and steam mixture in the event of a design basis accident, to direct and distribute the flow of air and steam through the ice bed, and to protect the containment structure opposite the ice condenser inlet doors from direct jet impingement forces.

The last two functions are accomplished by turning vanes that are designed to turn the flow of the air and steam mixture up through the ice bed in event of a design basis accident. For such an event, the vanes would serve to reduce the drag forces on the lower support structural members, reduce the impingement forces on the containment across from the lower inlet doors and to distribute the flow more uniformly over the ice bed. In addition to the turning vanes, the lower support structure has a continuous impingement plate around the outer circumference of the lower support structure, designed to reduce the jet impingement forces on the containment structure across from the lower inlet doors in the event of a design basis accident.

Design Criteria and Codes

The loading combinations, stress limits and material specifications used in the design of the lower support structure are given in Sections 6.7.16 and 6.7.18.

Design Conditions

The normal operating temperature range is 10°F to 25°F. The normal operational temperature change, including maintenance operations is 10°F to 70°F. The maximum temperature during a design basis accident is 250°F.

The loads used for the design of the lower support structure consist of dead weight (gravity), forces as a result of DBA, OBE and SSE seismic loads and loads as a result of thermal changes.

The dead loads include the weight of the crane wall insulated duct panels, the weight of the intermediate deck doors and frames, the weight of the lattice frames and columns, and the weight of the turning vanes. The weight of the ice baskets filled with ice, the slotted jet impingement plate assemblies and the door shock absorber, also act on the lower support structure.

Forces and loadings that occur during LOCA were provided by the Transient Mass Distribution (TMD) code from analysis of double-ended breaks in an end compartment, with 100% entrainment of water in the flow. For conservatism, all forces and loads that are a result of TMD were increased by 40% in performing the detail design and analysis for the lower support structure.

The lower support structure seismic design loads were developed using dynamic seismic analysis and the defined seismic response curves for the Watts Bar Nuclear Power Plant.

Thermal loading conditions, which result from two thermal excursions, were specified for the lower support structure. One thermal excursion from 10°F to 70°F is defined as a normal operating service load, and the other, defined as 70°F to 250°F, is the thermal excursion seen by the lower support structure following a LOCA.

The loading combinations considered in the design are given in Section 6.7.16.

6.7.9.2 System Design

The lower support structure is shown on Figures 6.7-22 and 6.7-23. The lower support structure is contained in a 300° circular arc of the containment. The three-pier lower support structure consists of 24 horizontal platform assemblies, 24 upper turning vane assemblies, 24 floor turning vane assemblies and 24 impingement plate assemblies. The aforementioned assemblies are supported by 25 radial-portal frame assemblies with columns at radii of 45 feet-6 inches, 49 feet-11¾ inches, and 55 feet-8½ inches. The 25 portal frame assemblies are spaced at approximately 12½° between adjacent portal frames. The total height of the structure is 9 feet-7 7/8 inches, measured from the top surface of the lower support structure to the pin. The design is such that the flow area at the ice basket interface for all 24 bays is at least 1088 square feet.

The horizontal platform consists of an inner and outer platform assembly for each bay. As assembled, the platform includes inner, middle and outer straight circumferential beams which span each portal frame. Nine radial beams formed by bar sections are welded to the inner, middle and outer circumferential beam. There is horizontal cross bracing between the inner and middle circumferential beams and the outer and middle circumferential beams.

The outer horizontal platform assembly consists of nine radial beams welded to the outer circumferential beam and welded to a channel which forms one half of the middle circumferential beam. The inner horizontal assembly is similar to the outer platform assembly. The channels of the inner and outer horizontal platform assemblies are field bolted to form a continuous middle circumferential beam.

For each bay, the platform inner and middle circumferential beams are connected to the portal frames with a shear connection, i.e., no moment is transmitted to the columns. The outer circumferential beam is connected to the portal column, but the connection is designed to transmit moment about a vertical axis. Every alternate horizontal platform (per bay) is connected to the columns at one side by bolted connections, which are slotted along the axis of the circumferential beams to accommodate circumferential thermal expansion. The adjacent bay is not slotted in the circumferential direction and supplies the tangential shear resistance for the slotted bay.

There are nine radial beams in each portal bay and each radial beam supports nine ice basket columns. Provision is made for attaching, by bolting, each ice basket column to the radial beams.

The inner and outer circumferential beams of the platforms assembly have the lattice frame column supports bolted to them. The insulated duct panels on the containment wall interface the floor and the insulated duct panels on the crane wall are supported by the inner circumferential beams of the lower support structure.

Each radial portal frame is comprised of three columns. The primary radial shear resistance is provided by a 2 inch thick plate with attached welded channels forming the inner and middle columns thus forming a steel shear wall. The outer column (radius 55 feet-9½ inches) is attached to the middle column assembly by a 2 inches thick plate. The 2 inches thick plate is pin-connected to the outer column by bars pinned at both ends and welded to the middle column. The column base plates are pin-connected to the ice condenser support floor. To accommodate thermal expansion, the middle pier column pin connections are designed to allow radial expansion, and every other outer column base plate pin connection is designed to allow circumferential expansion. The inner pier columns (near the crane wall) are designed to transmit all three force components. The base plate pin arrangement is shown on Figure 6.7-22. The lower inlet door shock absorbers are mounted to the 2 inches thick portal frame plate.

Tangential or circumferential rigidity of the lower support structure is provided by a cross bracing system between the outer columns. The cross bracing system is provided in alternate bays, which coincide with the bays in which the circumferential platform beams are not slotted in their axial direction at the column attachment points.

To turn, direct and distribute the flow through the lower inlet doors during a LOCA, each portal bay has five turning vanes that span between the adjacent radial portal frames. The vanes are as indicated on Figure 6.7-22. The vanes are slotted on one side in each bay to allow circumferential thermal growth.

In addition to the turning vanes, a beam grid work spans between adjacent outer columns (Figure 6.7-22) and acts as a jet impingement shield for the fluid flow not turned by the vanes. The slotted plate assembly is provided in each bay of the lower support structure and is attached to the outer columns with a bolted connection. Similar to the turning vanes, the slotted plate assembly is bolted on one side with slotted holes to allow for circumferential thermal growth.

The material for the lower support structure is ASTM-A588 steel. Bolting materials are ASTM-A320 Grade L7 and nut material is ASTM-194 Grade 7. These materials conform to the design criteria discussed in Section 6.7.18. All welding meets the requirements of the American Welding Society Structural Welding Code-1973-AWS Publication Dl.1-72.

The material used for the pins in the lower support structure is ASTM-A434 steel, E4340, Class BD. The material is normalized, then quenched and tempered. Chemical properties, physical test data and Charpy V-Notch test values at minus 20°F are required.

6.7.9.3 Design Evaluation

General

The lower support structure was analyzed using a finite element model. The ANSYS structural analysis program was used in the analysis. The seismic responses, in terms of equivalent acceleration and interface forces, in two horizontal directions (radial and tangential) and the vertical direction (z) were developed from a dynamic seismic response analysis performed for a combined lattice frame/ice basket/lower support structure model. The seismic loads, as well as loads due to dead weight, thermal and the forces due to DBA, were applied to the lower support structure as static forces.

Figures 6.7-24 and 6.7-25 show the finite element model used to represent the three pier lower support structure. The model is comprised of: three dimensional beam elements having six degrees of freedom per node; flat triangular shell elements, each having six degrees of freedom per node such that both membrane and bending action of the plates are considered; and general six degrees-of-freedom lumped masses having a 6 x 6 diagonal mass matrix with three values, Mx, My, Mz and three moments of inertia, Ix, Iy, and Iz. No horizontal ice mass is considered since this effect on the seismic response is accounted for in the results of the dynamic analysis of the combined lattice frames/ice baskets/lower support structure model. Rotary inertia terms are not used for the lumped masses.

Structural Representation

1. General

   Figure 6.7-24 shows an overall view of the one bay finite element model of the structural members. Each of the line members represents three dimensional beam elements. The loads generated from the model are used to design all the connecting joints to the AISC-69 Code, Section 2.8. A separate finite element model is used to determine the maximum stresses in the beams. The impingement plate which spans the chord between the two outer columns is modeled using equivalent beam elements.

   At beam connections where the beam centroidal axes do not intersect, either rigid links or specified offsets, which can be automatically accommodated for ANSYS beam elements, are used to preserve geometric compatibility between the elements. The connections of the horizontal platform to the portal frame are considered to be pin connections except at the outer column line where it is assumed that a moment around a vertical axis can be transmitted.

   The impingement plate is attached to the outer columns assuming no moment can be transmitted from the plate to the columns. Similarly, the upper and floor turning vanes are idealized as beam elements which are pin connected to the portal assemblies. The remaining structural connections are considered to be moment connections.

2. Mass Distribution

A. Structural Mass

The structural mass of the lower support structure is represented automatically in the ANSYS program through the use of consistent mass matrices associated with each of the structural finite elements. Thus, only the material density is input to account for the structural mass.

B. Ice Mass

The mass of the ice baskets is represented as lumped masses at node points along each radial beam. The mass is distributed based on the geometric placement of the ice baskets on the radial beams. Only mass in the vertical, Z direction, is assigned to the lumped masses representing the ice baskets, since the horizontal seismic effect of the ice basket mass is incorporated as loads on the radial beams. The horizontal seismic loads are determined from a dynamic analysis of a combined wall panel/lattice frame/ice basket/lower support structure model.

3. Displacement Boundary Conditions

Displacement boundary conditions are not specified for the tops of the column nor for other nodes contained in the column radial plane. However, forces are applied to the columns which account for the adjacent bay loading.

To accommodate the thermally induced loads in the structural members, the base plates of the two middle columns are free to expand in a radial direction. Likewise, to accommodate the circumferential thermal expansion, every other outer column base plate connection is free to expand circumferentially.

Referring to Figure 6.7-22, the above boundary conditions imply that the outer column bases at odd numbered column lines are restrained against motion in the vertical, radial and circumferential directions, while the outer column bases at even numbered column lines are free to displace circumferentially.

The middle columns are free to move in the radial direction at all column lines and the inside columns (near the crane wall) are restrained for all three translations at all column lines. These boundary conditions minimize the thermally induced stresses and floor loads.

Loading Conditions

1. Seismic Loads

A. General

Analysis indicates that the frequency of the lower support structure is sufficiently high relative to the peaks of the response spectra and is one mode dominant in the vertical direction, so that a seismic modal response analysis is not required. Instead, an equivalent static analysis was performed for vertical accelerations based on the assumption of one mode dominance. For horizontal seismic loads, the largest forces in the radial and tangential directions as determined from a dynamic analysis of combined ice basket/lattice frame/lower support structure moded are applied as static concentrated forces to the lower support structure. A schematic of the applied loads is shown in Figure 6.7-26.

B. Horizontal Radial Excitation

To account for the seismic loads transmitted from the ice baskets, lattice frames, and lattice columns, a dynamic analysis of the lattice frame and ice basket structures coupled to the lower support structure by means of flexibility coefficient which represents the lower support structure is performed. The loads transmitted to the lower support structure at the interface between the lower support structure and the ice baskets are applied as static concentrated forces. To account for the seismic loads transmitted from adjacent bays, radial forces are applied to the model at the required nodes.

C. Horizontal Tangential Excitation

The tangential loads transmitted from the lattice frames and ice baskets are determined in the same manner as the radial forces from the dynamic analysis performed.

The total tangential loads applied to the radial beams by the ice baskets are distributed in the same manner as the mass. Since the ice baskets are attached to the top surface of the radial beams, concentrated torques are applied at each of the nodes of the radial beams to account for the distance of approximately 6 inches from the top of the radial beam to the centroid of the cross section of the radial beam. The seismic loads from adjacent bays are considered by applying concentrated circumferential forces to the appropriate nodes.

2. Blowdown Loads

A. General

The blowdown forces applied to the lower support structure are divided into four classifications:

i. Vertical Forces
ii. Horizontal Radial Forces
iii. Lower Inlet Door Impact Forces
iv. Horizontal Tangential Forces

The following sections discuss the loads for each of the classifications and the application of the loads to the finite element model of the three pier lower support structure.

B. Vertical Blowdown Loads

The vertical uplift loads acting on the lower support structure arise from the following phenomena:

i. Uplift on the ice baskets
ii. Uplift on the radial beams
iii. Uplift on the horizontal platform bracing
iv. Uplift pressure across the intermediate deck
v. Uplift on lattice frames and lattice column

C. Horizontal Radial Blowdown Forces

The horizontal blowdown forces acting on the structure arise from the following phenomena:

i. Momentum forces on the middle circumferential beam turning vane.
ii. Momentum forces on the upper three turning vanes attached to the middle column.
iii. Momentum forces on the floor turning vane attached to the middle column.
iv. Momentum loading on the slotted impingement plate.
v. Forces on the outer circumferential beam.
vi. Radial forces on the ice baskets.

The forces are transient in nature. However, only the basic static values with dynamic load factors applied to account for the transient nature of the loading have been applied to the structural model, as concentrated forces on the appropriate nodes. To account for forces from adjacent bays, concentrated loads were applied to the portal frame connection points, as required.

D. Lower Inlet Door Impact Load

From results of studies and tests performed to determine the force-time history transmitted through the shock absorber which arrests the inlet door motion, a tangential load was applied to the lower support structure portal frame. The dynamic pulse characteristics of the force are accounted for by recommending a dynamic load factor of 2.0 for the pulse taken to represent the force versus time relationship for the shock absorber.

The door impact load is applied simultaneously in the same direction at both column lines 1 and 2 as a worst case. Thus, the loading considered is anti-symmetric tangential loading on the one bay model and creates an overturning moment about a radial axis through the lower support structure. In the design of the lower support structure, the bolt connections between the columns and the circumferential beams are designed to consider the possible loading from the door impact loads being applied in opposite tangential directions on the door arrestor plates.

E. Horizontal Ice Basket Forces

The tangential and radial forces acting on the ice baskets due to cross flow are assumed to act on the bottom, three feet of ice basket (one-half of the span between the top of the lower support structure and the attachment of the ice baskets to the first lattice frame). The loads are applied to the finite element model as uniformly distributed loads on each of the beam elements comprising a radial beam.

Dynamic Load Factors

1. General

To account for the dynamic nature of the blowdown forces, dynamic load factors are applied to the DBA forces applied statically to the finite element representation of the lower support structure. The dynamic load factors (DLFs) are as follows:

A.	Vertical Uplift Forces	DLF = 0 or 1.8
B.	Horizontal Radial Forces	DLF = 0 or 1.2
C.	Lower Inlet Door Impact Forces	DLF = 0 or 2.0
D.	Horizontal Tangential Forces	DLF = $\pm$ 1.2

2. Transient Analysis of Blowdown Loads

Following a LOCA, the inlet doors open admitting steam flow into the ice condenser chamber. The fluid flow through the lower support structure and upward through the ice bed cause time-dependent forces to be applied to the lower support structure. In general, there are four classifications of transient forces applied to the lower support structure: (a) vertical forces on the radial beams, ice baskets, lattice frames, lattice columns, and intermediate deck; (b) horizontal radial forces acting on the outer columns, the jet impingement plate, the outer circumferential beam, and turning vanes attached to the middle circumferential beam and middle column; (c) tangential forces, applied to the impact plates attached to the portal frames, resulting from arresting the motion of the inlet doors; and (d) tangential forces on the radial beams due to cross flow in the ice condenser compartment. The dynamic load factors are determined by performing a transient response spectrum analysis for each force-time history, as described below.

3. Single Degree of Freedom Representation

In general, the transient structural response of a multi-degree of freedom system is given by the expression:

$$y_i(t) = \sum_{j-1}^{N} r_j \, n_j(t) \, \psi_{ij}$$

where,

$y_i(t)$ is the structural response at any time (t).

ψ ij is the jth mode shape of the structure.

r_j is the participation factor of the jth mode shape for the transient load.

$n_j(t)$ is the generalized coordinate of the jth mode shape at any time (t).

The generalized coordinate nj of the jth mode is given in terms of the forcing function f(t) by Duhamel's integral, or the convolution integral as:

$$n_j(t) = \omega \int_0^t f(\tau) \sin \omega(t-\tau) \, d\tau$$

Thus, the expression for the generalized coordinate for each mode, j, is the same as the amplification factor, or dynamic load factor (DLF) definition for a single degree of freedom system:

$$\mathrm{DLF(t)} = \omega \int_0^t f(\tau)\ \sin\ \omega(t-\tau)\, d\,\tau$$

Assuming that 🖹$_j$ = 1 for some j = k and 🖹$_j$ ~ 0 for j ≠ k, amounts to the assumption that only one mode dominates, in the structural response to the transient. In this case, the structural response becomes:

$$y_i(t) = n_k(t)\ \Psi_{ik} \tag{4}$$

or,

$$y_i(t) = \mathrm{DLF}(t)\ \Psi_{ik} \tag{5}$$

In which case the maximum structural response is given by:

$$y_{i_{max}} \cong \mathrm{DLF}_{max}\, \Psi_{ik} \tag{6}$$

Assuming that the dominant mode ψ_{ik} can be approximated by the static deflection shape due to the loads applied to approximated by:

$$y_{i_{max}} \cong \mathrm{DLF}_{max}\, y_{i_{static}} \tag{7}$$

Thus, assuming that the response of the lower support structure to the transient blowdown forces may be represented by Equation 7, the dynamic effects of the transient may be investigated by evaluating the transient response spectra given by:

$$\mathrm{DLF}_{max}(\omega) = \max\,[\omega \int_0^t f(\tau)\ \sin\ \omega\ (t-\tau)\, d\,\tau] \tag{8}$$

evaluated for $\omega = \omega_\eta$ where ω_η is the natural frequency estimated for the lower support structure.

A typical force transient for a hot leg break is shown in Figure 6.7-27. The resulting dynamic load factor plot is shown in Figure 6.7-28.

4. Discussion

The recommended dynamic load factors are the maximum values from the transient response spectra for zero damping and for a frequency greater than 10 Hz (lowest estimated L.S.S. - Floor frequency).

As previously stated, transient response spectra used to determine the DLF are for zero damping, rather than, a damping of between 5 to 10%, which is more appropriated for the highly stressed, bolted lower support structure. Damping will reduce the dynamic response as indicated typically in Figure 6.7-28 which shows the response for horizontal forces for 0, 5, 10 and 20% damping. Thus the DLF recommended are conservative from this standpoint.

In addition to the conservatism used to derive the DLFs used for design, additional conservatism has been incorporated into the design by specifying that the forces scaled by the DLFs be applied to the structure in the worst manner to determine the maximum member forces. Since the maximum DLF for each transient will not occur at the same time, combining the member forces derived for each transient in this manner is conservative. In particular, an RMS combination similar to that used in earthquake analysis could be justified because of the time separation of peak occurrence.

The recommended DLFs have been conservatively derived and applied in the design of the lower support structure. Therefore, the resultant member forces determined for the

DBA, using the recommended DLF, result in a conservative prediction of the stresses induced in the structure.

Design Load Case

Because of the magnitude of the DBA forces, the proportions of all members and structural elements of the lower support structure are sized by the load combinations which include DBA forces. The DBA forces are 2 to 5 times larger than other forces that are applied to the lower support structure. The seismic, blowdown, and combined seismic and blowdown loads were considered in the design.

The combined load case is represented below:

DL + TN + EV + ER + ET + AV + AR + AT + LIDI

where:

DL	=	Gravity
TN	=	Thermal (70°F to 250°F)
EV	=	Safe shutdown earthquake forces in the vertical direction
ER	=	Safe shutdown earthquake forces in the radial direction
ET	=	Safe shutdown earthquake forces in the tangential direction
AV	=	Vertical forces due to DBA
AR	=	Radial horizontal forces due to DBA
AT	=	Tangential horizontal forces due to DBA
LIDI	=	Lower inlet door impact

Results of Stress Analysis

1. Members

   The stress in the various structural members for all of the design load cases was found to be below the design criteria as specified in Section 6.7.16.

2. Joints

   The member forces at connections from all load cases were used to proportion the connections. In the design of the connection for the load conditions, the recommendation of the AISC - 69 Code Section 2.8 were followed as specified in Section 6.7.16.

6.7.10 Top Deck and Doors

The top deck, intermediate decks containment shell, crane wall and end walls form the boundaries of the ice condenser upper plenum. The upper plenum houses the air handling units and the distribution ducts to the wall panels and provides a working space for loading, weighing and maintaining the ice baskets.

6.7.10.1 Design Basis

Function

An array of blanket panels forms a thermal and vapor barrier atop the upper plenum, allowing limited movement of air through vents during unit operation and free outflow of air during DBA.

A grating deck supports the blanket panels and accommodates traffic by inspectors. The top deck structure supports the grating as well as the bridge crane and rail assembly and the air handling units.

Loading Modes

The following loading conditions are considered in the design of the top deck: deadweight, seismic loads, blowdown loads, and live loads. The top deck structure withstand these loads and remain within the allowable limits established in Section 6.7.16.

Design Considerations

1. The blanket panels are hinged on top of the crane wall. The major loads are applied directly into the crane wall.

2. A blanket panel must be flexible, i.e., be capable of deforming out of its plane in response to relatively low forces without disintegrating. Deformation of panels during a DBA is permissible but formation of missiles must be averted.

3. The deck forms an integral part of ice condenser performance during a DBA. Structural loads are a function of air pressure and flow relationships, which in turn are affected by deck characteristics.

4. The top deck structures are subjected to loads from the air handling units and bridge crane in addition to the deck design loads.

Material Consideration

1. Refer to Section 6.7.18 for a discussion of design criteria for steel structures.

2. Blanket material is fire resistant by its own composition or by means of a suitable cover sheet.

3. Blanket material is not a significant source of halides in gaseous form, whether by gradual diffusion of inherent ingredients or by radiolysis of component material following a DBA.

4. Blanket material is not a significant source of leachable halides during exposure to containment spray following a DBA.

Thermal and Hydraulic Performance Requirements

1. Heat input to the plenum through the top deck assembly is limited to 13.5 Btu/hr-ft^2.

2. Resistance to air flow during a DBA is minimized, in terms of both inertia of panels and obstruction by grating. Panels may reclose or remain open following a DBA. Vents open on low differential pressure for small flow rates.

3. A vapor barrier is established on the upper surface of the blanket panels.

Interface Requirements

1. In the process of opening, adjacent blanket panels interfere with each other. This is acceptable in view of their flexibility.

2. Sealing strips are installed to connect panel vapor barrier to adjacent panels, to crane wall, to end walls and to containment shell, without transmitting appreciable loads to the containment shell.

3. The grating rests on, and is attached to, the cross beams between the top deck beams and transmits operating and drag loads to these structures. The structural members received loads from bridge crane and air handling units as well as the deck itself.

Design Loads

Loads used in the design of the top deck assembly are shown in Table 6.7-20.

6.7.10.2 System Design

The design of the top deck is shown in Figures 6.7-29 and 6.7-30.

The top deck doors consist of radially aligned flexible blanket panels resting on a grating deck and hinged on top of the crane wall.

A blanket pair covers one-half bay, extending from the radial centerline of a bay to the edge of the adjacent top deck beam. It consists of two blanket assemblies, one resting on the grating, the second one resting (mirror image) on the first one, with bands touching.

The parts of a blanket assembly and their respective functions are as follows:

1. Thermal insulation is provided by a 1-inch-thick flexible polyurethane foam blanket.

2. Approximately one-half of the centrifugal load is carried by 0.005-inch-thick fully hardened stainless steel bands.

3. A stainless steel cover sheet ("skin") or similar material serves as a vapor barrier (top surface), protects the blanket against wear and fire (top and bottom surfaces), and provides all of the lateral and about one-half of the centrifugal strength.

4. Parts 2 and 3 are bonded to the faces of the foam and extended along one edge to form a hinge.

The grating deck performs the structural functions of the top deck during non-accident conditions. It is supported from pairs of cross beams spanning the top deck beams, and its upper surface is flush with the top of the top deck beams. The bearing bars of the grating run parallel to the centerline of the particular bay. They are 2 inches high, 3/16 inch thick, and spaced on 2-3/8 inch centers. This design satisfied all requirements for open area and upward drag loads during DBA as well as for normal traffic loads. A clearance of no less than 4.0 inches is maintained between the grating and the containment.

The grating is fabricated from carbon steel, ASTM-A36, or A569 and provided with trim banding adjacent to top deck beams. Completed grating sections are galvanized for corrosion protection.

A hinge bar clamps one edge of each blanket assembly to the surface of the crane wall. Anchor bolts transmit the hinge loads into the crane wall.

Static insulation pads are attached to the top of the radial beams.

Flexible seal membranes are attached between vapor barrier (top) surfaces of the blanket panels and against vent base, and walls, and static insulation.

A pressure equalization "curtain" is suspended around the periphery of the top deck. The vent curtain minimizes diffusion of air under steady state conditions while permitting free movement of air in or out during momentary periods of pressure imbalance.

<u>Fabrication</u>

1. Grating sections are fabricated to specific shapes, complete with trim banding. The finished assemblies are cleaned and hot dip galvanized.

2. Structurals are cut and welded to suit.

3. Blanket assemblies are fabricated by an insulation contractor using specified bonding methods.

4. Hinge bars are machined from rectangular steel bars and painted or galvanized.

<u>Installation</u>

1. Radial and cross beams are installed.

2. The grating sections are placed and bolted down.

3. Static insulation pads and blankets are placed in position all around top deck.

4. Vent assemblies are installed.

5. Seals are installed.

6. Hinge bars are installed. Blankets are clamped. Static insulation is attached.

Top Deck Blanket Doors

The top deck doors were dynamically analyzed to determine the loads and structural integrity of the door for the design basis load conditions.

Using TMD results as input, the door dynamic analysis was performed using a separate computer code named the "DOOR" Program. This computer program has been developed to predict door dynamic behavior under accident conditions. This program takes the door geometry and the pressures and calculates flow conditions in the door port. From the flow are derived the forces on the door due to static pressure, dynamic pressure and momentum. These forces, plus a door movement generated force, i.e., air friction, are used to find the moment on the door and from this are derived the hinge loads. Output from the program includes door opening angle, velocity and acceleration as functions of time as well as both radial and tangential hinge reactions.

Analysis Due to LOCA

The net load distributions on the door opening are determined by considering the applied pressures acting on the door and then solving the rigid body equations of motion such that the net forces and moments at the hinge point are zero. In the process, this produces expressions for the inertial forces in the door and the hinge bar reaction as functions of the applied pressure. The resultant horizontal and vertical hinge loads, calculated by the DOOR Program, provide the inputs for subsequent stress analysis.

Using this input, the blanket assembly is analyzed with horizontal and vertical forces being taken by direct stress in the skin and bands. As inertial forces are directly accounted for in the analysis, no dynamic load factor is applied.

The hinge bar and anchor bolt stresses are analyzed under hinge reactions considering the effects of the horizontal and vertical components of the tension band. For these components, no dynamic load factor is applied since the bars are very rigid themselves and are rigidly attached to the crane wall. Stresses in the blanket floor grating due to aerodynamic drag are also calculated. Loads used for stress calculations include 40% margin above computed TMD values. Certain aspects of the dynamic performance of a flexible door (e.g., tangential distortion, whipping, bowing) cannot be modeled with sufficient confidence.

A summary of the analysis performed and results are presented in Table 6.7-21. All portions of the door show factors of safety equal to or greater than one. The general acceptance criterion was that stresses be within the allowable limits of the AISC-69 Structural Code. For materials and components not covered by the Code, i.e., spring temper stainless steel nonmetallic materials, floor grating, etc., conservative acceptance criteria are established on the basis of manufacturer's recommendations or ASTM minimum tensile specifications.

Dynamic Test

A full scale test of a blanket pair (one-half bay) is performed for verification of analysis. Observed dynamic characteristics are found to correlate well with computed TMD values, and integrity of blankets is maintained within acceptable limits.

Top Deck Structure

The top deck structure is analyzed using the ANSYS finite element computer program, with three-dimensional beams representing the structural members, three-dimensional lumped masses representing the mass elements, and a stiffness matrix to represent the flexible connections in the system.. Geometric compatibility is maintained using three-dimensional rigid elements.

Two bays considered representative of the system were isolated and modeled. Conservatively, four air handling units are assumed to be located in the two-bay region, two next to the crane wall and two next to the containment wall.

Stresses are calculated for the various combinations of dead load, thermal, seismic and accident conditions. A modal analysis is performed to determine seismic amplification. Blowdown stresses are calculated using a computed dynamic load factor. Maximum stresses produced in major members are within the limits given in Section 6.7.16. The circumferential struts, air handling unit beams and crane rails have been analyzed and are structurally acceptable.

6.7.11 Intermediate Deck and Doors

6.7.11.1 Design Basis

Function

The intermediate deck forms the ceiling of the ice bed region and the floor of the upper plenum. It serves as a thermal and vapor barrier, which allows limited air movement, through vents, between regions during normal plant operation and free out-flow of air and steam following a DBA.

Criteria

Refer to Section 6.7.16 for structural design criteria.

Loading Modes

The following loading conditions are considered in the design of the intermediate deck: deadweight, seismic loads, blowdown loads, and loads due to personnel traffic on deck. The intermediate deck structure withstands these loads and remain within the allowable limits established in Section 6.7.16.

1. Design Criteria - Accident Conditions

   a. Resistance to air flow during a DBA is minimized, in terms of both inertia of door panels and obstruction by the frames. Panels may reclose or remain open. Panels open on low pressure differential for small flow rates.

   b. At the end of their movement, pairs of doors collide. Distortion at the time is acceptable, provided doors do not become missiles.

   c. The doors are of simple mechanical design to minimize the possibility of malfunction.

2. Design Criteria - Normal Conditions

   a. Heat conduction through the intermediate deck is limited to 0.6 Btu/F-hr-ft^2.

   b. The design of the deck permits its use as a walking surface for maintenance of the air handling units and inspection of the ice bed.

   c. The design of the deck provides a vapor barrier between the ice bed and upper plenum area.

   d . The design of the deck provides convenient access to selected ice baskets for weighing and visual inspection.

3. Interface Requirements

   a. Sealing strips are installed to seal deck frames to wall panels as a continuation of the vapor barrier.

   b. Hinge loads, drag loads, and live loads are transmitted from the deck through support beams to the lattice frame support columns.

   c. Instrumentation cables from the temperature monitoring system penetrate the seal area of the deck.

Design Loads

Pressure loading during LOCA is provided by the Transient Mass Distribution (TMD) code from an analysis of a double-ended hot leg break in the corner formed by the refueling canal, with 100% entrainment of water in the flow.

The intermediate deck design parameters and loads are presented in Table 6.7-22.

6.7.11.2 System Design

The intermediate deck is shown in Figure 6.7-31. For ease of manufacture and installation, the deck is separated into 48 subsections. Each subsection covers an area extending over a length of three lattice frames and a width of approximately half the ice condenser annulus. Two types of subsections are used; the inner subsection has overall dimensions of 11 feet long by 5 feet, 7 inches wide; and the outer subsection has dimensions of 12 feet by 4 feet, 7 inches. Except for dimensional differences, the designs of inner and outer subsections are identical.

Each subsection consists of four door panels mounted on a steel frame. The door panels are sandwich structures, consisting of 26 gauge galvanized steel sheets bonded to a 2.5-inch thick urethane (Unit 1/Unit 2) and/or polyisocyanurate (Unit 2) foam core. Loads developed in the sandwich structures are transmitted to two panel hinge points by a 2.5-inch x 5-inch rectangular steel tube which forms a backbone for the panel. The panel is reinforced and sealed by a peripheral channel and two internal ribs, formed from 18 gauge steel sheet.

Plates, which are welded to the ends of the tubular backbone, are drilled to accommodate 1-in. diameter stainless steel hinge pins. These pins in turn are supported by welded steel support brackets which are bolted, through the door frame, to intermediate deck support beams. Thus, hinge loads are taken directly into the support beams and not into the frame itself.

The door frame is fabricated from steel angle and T-sections. A formed channel on the frame holds a compliant bulb-type rubber seal which is compressed by the door in its closed position. In addition to being clamped in place by the hinge support brackets as described above, additional bolts in the frame angles fasten the corners of the frame to the support beams and connect adjacent members of the inner and outer assemblies to each other.

The intermediate deck support beams are 8-inch wide flange steel members, which radially span the ice condenser annulus. They are bolted to the lattice frame support columns via welded plate bracket assemblies and compliant pads. The latter feature assures that beam end moments are not transmitted to the relatively flexible support columns.

Flexible membranes are installed between the intermediate deck frame and adjacent wall panels to provide a continuous vapor barrier.

Pressure equalization vents are installed at the containment wall side of the intermediate deck. Vertical flaps minimize diffusion of air under steady state conditions while permitting free movement of air in or out during momentary periods of pressure imbalance.

6.7.11.3 Design Evaluation

The intermediate deck doors are dynamically analyzed to determine the loads and structural integrity of the door for the design-basis load conditions.

Using TMD results as input, the door dynamic analysis was performed using a separate computer code named the DOOR Program. This computer program was developed to predict door dynamic behavior under accident conditions. This program takes the door geometry and the pressures and calculates flow conditions in the door port. From the flow are derived the forces on the door due to static pressure, dynamic pressure, and momentum. These forces, plus a door movement generated force, i.e., air friction, are used to find the moment on the door and from this are derived the hinge loads. Output from the program includes door opening angle, velocity, and acceleration as functions of time, as well as both radial and tangential hinge reactions.

Analysis Due to LOCA

The net load distributions on the door during opening are determined by considering the applied pressures acting on the door and then utilizing an analysis similar to that derived for the lower inlet doors (Section 6.7.8), to obtain shear, moment, and hinge reactions.

Using this input the door panel is analyzed as a sandwich panel; i.e., the outer steel skins are assumed to carry tensile and compressive membrane loads, while the urethane (Unit 1/Unit 2) and/or polyisocyanurate (Unit 2) core carries transverse shear loads between the outer skins. The tubular backbone is analyzed as a beam with biaxial bending and torsion under the combined effects of panel shear loading, panel centrifugal loading and hinge reactions. Hinge pins and support brackets, including bolting, are analyzed by considering the effects of tension, shear, and bending as appropriate. No dynamic load factor is applied, as inertial forces are directly accounted for in the analysis.

The door frame and attachment bolting are analyzed under loadings created by the differential pressure acting on the frame members. The intermediate deck beams and attachments are analyzed under the effects of loads transmitted to them by the door hinges and frames. For these latter analyses, appropriate dynamic load factors are calculated and applied.

All results indicated positive margins of safety in comparison with the criteria contained in Section 6.7.16.

During a LOCA, stopping of the doors is accomplished by impacting adjacent door panels against each other. In the process, a significant portion of the door kinetic energy is absorbed through plastic deformation of the door panels. This is an acceptable mode of behavior as long as the doors do not break up and lose their insulation or otherwise generate missiles. During simulated blowdown tests on full-scale prototype doors at levels of maximum pressures predicted by TMD, the ability of the doors to withstand opening and stopping loads is confirmed. Only local deformation of the panels results and no missiles or insulation are released.

Seismic Analysis

A response spectra nodal analysis is performed on the intermediate deck structure to determine maximum seismic loadings during ½SSE and SSE. Resultant loadings on the structure are found to be negligible in comparison with LOCA loadings. Further, calculations indicates the doors will not open during either earthquake.

6.7.12 Air Distribution Ducts

6.7.12.1 Design Basis

Function

The air distribution ducts distribute the cold air from all air handling units uniformly to the wall panels (see Figures 6.7-32 and 6.7-33).

The loss of the air distribution function does not affect the safety of the unit as the ice bed is a passive component and can tolerate refrigeration system failures.

Design Criteria

The air distribution ducts are permitted to deform during accident conditions but must not affect any safety related components located nearby.

Design Conditions

1. Normal Operation

Design temperature normal	10°F - 15°F
ΔP normal	2 inch WG

2. Accident Conditions

Accident temperature maximum (without ΔP)	190°F

6.7.12.2 System Design

The air distribution ducts are located in the upper plenum. The ducts are made of galvanized sheet steel. The design includes flexible connections separating each duct and each air handling unit. The flexible connections also serve as vibration breaks.

6.7.12.3 Design Evaluation

The air distribution ducts are a part of the refrigeration system and serve to distribute cold air to the wall panels thereby maintaining the readiness of the ice in the ice bed. The air distribution ducts are not required to function during an accident. The air distribution ducts are, therefore, non-safety related components. Refer to Section 6.7.6 for detailed discussions of the refrigeration system performance during normal operating conditions and of its ability to tolerate refrigeration component failures.

During a LOCA the air distribution ducts are permitted to deform. Any deformation is outward toward the crane and liner wall insulation and therefore presents no problem to nearby safety related components.

6.7.13 Equipment Access Door

6.7.13.1 Design Basis

Function

The equipment access door permits movement of crane, equipment and personnel into and out of the ice condenser plenum for ice loading and maintenance. Personnel access doors are provided in the equipment access doors to provide entry during power operation.

In closed position, the door constitutes a thermal and vapor barrier (normal unit operation) and a pressure barrier (accident condition) between ice condenser air and upper containment atmosphere.

The basic functions of the equipment access and personnel door are non-safety related. It is important, however, to prevent failure of the door in any manner that may effect safety-related components located nearby.

Design Criteria and Codes

The door is designed to comply with structural requirements of Section 6.7.16.

Design Conditions

1. Normal Operation

Design temperature inside	15°F nominal
Design temperature outside	100°F

2. Accident Conditions

Maximum surface temperature (without ΔP)	190°F

6.7.13.2 System Design

An equipment access door is provided in each end wall thereby providing ample access to the upper plenum. The equipment access door includes: the insulated door panel, Personnel Access Door, frame and hoist assembly, gasketing, and fasteners. The Equipment Access door frame slides from closed to open position within a fixed frame embedded in the concrete end wall. Personnel access doors are provided to open into the condensers.

It is noted in Unit 2 only, the hoist assembly is abandoned in place and a chain hoist is used by personnel when the equipment door is required to be opened.

All exposed surfaces are protected against corrosion by appropriate coating. In Unit 1, limit switches are provided to monitor movement of each door and to indicate position as a part of the door position monitoring system. IN Unit 2, limit switches are provided to monitor personnel access door latch position and the equipment access door seal inflation with the personnel access door position monitoring system.

6.7.13.3 Design Evaluation

The equipment access door is a non-safety related component. The door stresses during SSE + DBA loadings are below the allowable levels.

6.7.14 Ice Technology, Ice Performance, and Ice Chemistry

6.7.14.1 Design Basis

The operational principle of the ice condenser is the condensation of steam by means of melting ice. Approximately one and a half pounds of ice per pound of reactor coolant are required to absorb the coolant energy to prevent excessive containment pressure and temperature buildup. The liquid resulting from the thawing process drains to the containment sump where it is utilized during the recirculation phase of cooldown by the emergency core cooling system. It is, therefore, necessary that the boron concentration of the recirculated primary coolant not be diminished through the action of the ice condenser. Hence, the ice condenser utilizes borated ice, which upon bulk melting delivers an aqueous solution containing 1900 $\pm$ 100 ppm boron to the containment sump. The solution used in this case to produce the ice for the condenser is one containing approximately 1900 $\pm$ 100 ppm boron as sodium tetraborate ($Na_2B_4O_7 \bullet 10H_2O$).

The complete equilibrium freezing of this solution forms a eutectic composition with a melting point of -0.42°C (31.2°F).

On a microscopic scale, the complete equilibrium freezing of a 1900 $\pm$ 100 ppm aqueous solution of boron as sodium tetraborate, results in a solid consisting of crystals of pure ice (approximately 91% of the original water), surrounded by frozen eutectic. Microscopically this eutectic solid consists of individual crystals of pure ice and pure ($Na_2B_4O_7 \bullet 10H_2O$)[9].

6.7.14.2 System Design

The ice for the ice condenser is produced in machines that yield ice in the form of a continuous ribbon, approximately 1/8 inch thick which is deposited in storage bins via gravity chutes.

The ice is kept at subcooled temperatures by chilled air flowing through the hollow walls and floor of the bin and over the exposed surface of the ice.

Ice is pushed out of the bin by a mechanized rake and carried to an ice chopper via two screw conveyor. The chopper reduces the size of the ice flakes to approximately 2 inches x 2 inches x 1/8 inch. The ice chopper discharges through a metering hopper into a pneumatic conveying valve.

The pneumatic conveying valve feeds ice at a measured rate into a stream of chilled compressed air, which carries the ice through temporarily erected piping to either one of the ice condenser units. The air/ice mixture is fed into a cyclone receiver atop of the ice baskets where the ice drops into the basket while the air is released into the containment vessel. The air that is fed into the containment vessel during this operation is removed by a vacuum receiver in order to maintain a stable containment vessel pressure.

The ice baskets are weighed after loading is completed and the intermediate deck and top deck beams are put in place. Several tools, which utilize the same weighing device, are necessary to weigh all the baskets at this time due to varying degrees of accessibility. During later periodic inspections, additional weighings of selected baskets are performed using the same tools.

6.7.14.3 Design Evaluation

As the ice condenser is to be available to perform its engineered safety feature function for the life of the unit, ice storage characteristics are an important consideration. Two mechanisms influence the long-term storage of the ice: (1) the diffusion of sodium borate crystals through the ice crystals and (2) the sublimation of the ice.

Diffusion

For a discussion of the first mechanism, refer to the phase diagram presented in Figure 6.7-34. When the temperature of an aqueous sodium tetraborate solution is continuously lowered, freezing begins with the formation of crystals of pure water surrounded by the salt solution. The temperature at which the first ice crystals form (assuming no supercooling) depends on the initial concentration of the solution. For example, a solution of ($Na_2B_4O_7 \cdot 10H_2O$) containing 2000 ppm boron begins to freeze at -0.41°C (+31.27°F), under one atmosphere pressure (Point A in Figure 6.7-34). If the freezing process is allowed to continue reversibly, i.e., under conditions of the thermodynamic equilibrium, more ice crystallizes and the surrounding solution increases in concentration according to line AB in Figure 6.7-34. Finally, when the system temperature is -0.42°C (+31.24°F), the remaining liquid freezes to a solid with a boron concentration of 2220 ppm. The composition of this solid is known as the eutectic composition.

If the borated ice is made by the very slow freezing process just described, the pure water crystals first formed become the centers for further crystallization and therefore grow until the liquid reaches the eutectic composition. The total number of these relatively large pure ice crystals is determined by the number of nucleation sites available in the solution during the initial phase of the process. If the freezing rate is made extremely large, i.e., the process is carried out in an irreversible manner, the initial crystals do not have time to grow appreciably before all the water sodium borate has crystallized. Such a path is represented by the line CD in Figure 6.7-34. The solid obtained by this process is a uniform mixture of very small crystals of two kinds, ice and sodium tetraborate.

When a collection of various-sized crystals of a substance are maintained at constant temperature and pressure in contact with a solution saturated with respect to the substance, two processes tend to occur. The larger crystals tend to grow at the expense of the smaller ones, and the crystals of irregular form tend to become of regular form. Both of these phenomena are manifestations of systems tending toward thermodynamic equilibrium where the total free energy of the system (in this case the surface free energy) is at a minimum. The solution referred to above can also be a vapor and in the simplest case can be the pure saturated vapor of the crystalline substance. Note that kinetically the two processes are competitive and that both are subject to diffusional control. Therefore, diffusion of molecules, from one site to an adjacent one of the same crystal, would be favored over migration to another larger crystal in the case where rapid cooling of very dilute solutions causes many crystals to form that are small compared to the separation between them. Such is the case in practice with the ice condenser.

The driving force for diffusion between crystals of sodium borate through the pure ice matrix is a concentration gradient. If a large crystal is tending to grow, it causes depletion of sodium and borate ions in the immediately surrounding ice. If a small crystal tends to give up sodium and borate ions to feed the growth of the larger crystal then there is an increase in the concentrations of sodium borate surrounding the shrinking crystal. Since ice and sodium borate do not form an appreciable solid solution (note eutectic mixture of ice and sodium borate crystals), then the concentration of sodium borate around the shrinking crystal can not be large. For the sake of constructing an upper bound on diffusional effects in the borated ice, assume the maximum concentration to be approximately 10% of the eutectic solution concentration (i.e., 220 ppm).

Diffusion of sodium borate across a slab of pure ice can be estimated as follows:

Data for the diffusion of sodium borate in ice are not available, but the self-diffusion coefficients for deuterium, tritium and oxygen in ice have been reported by Franks[10]. At -11°C (+12°F) the value for all species is approximately 10^{-11} cm^2/sec. Assuming that the coefficient for sodium and borate ions is of the same order of magnitude, the rate of diffusion of sodium borate through a 1/32-inch slab of pure ice is estimated to be approximately 2×10^{-13} g/cm^2-sec. for an initial concentration of 220 ppm boron. If the concentration of boron in the ice phase on one face of the slab remained constant at 220 ppm while diffusion through the pure slab took place, it would take over 100 years for an amount of boron in a single piece of condenser ice to diffuse 1/32 in., or halfway through the ice flake.

Since the quick frozen-borated ice is of stable uniform composition, upon bulk melting, there should be formed a solution of borax of uniform concentration. If the entire borated ice-mass were to be uniformly warmed above -0.42°C (+31.24°F), melting would begin at the points of contact between water crystals and $Na_2B_4O_7{\bullet}10H_2O$ crystals, and the ice-mass would lose structure. This is a phenomenon known as "rotting" and has been observed at times in sea-ice which has been subjected to slow (order of hours or days) temperature excursions to just above the melting point. If the melting process is rapid then the fact that the borated ice-mass is a mixture of crystals and not a homogeneous solid solution does not affect the performance of the ice condenser. Melting in the ice condenser occurs over a time span of the order of seconds, beginning at the contact between the steam and the ice-mass and progressing inwardly.

The above arguments are greatly simplified, but lead to conservative results. It can therefore be concluded from the above arguments that while some local changes undoubtedly occur in the quick-frozen borated ice, a mal-distribution of the solute boron in the ice condenser, of such magnitude as to affect the operation of the condenser as described in the first paragraph, is extremely remote. Furthermore, the microscopically heterogeneous composition of the borated ice-mass does not reflect itself in the ice condenser performance.

<u>Sublimation - *Historical Information*</u>

The following information was developed during the design and initial operation of the ice condenser system. Actual sublimation rates have been established during the operation of Watts Bar and are discussed in the Section entitled Sublimation - Operational.

The other mechanism that affects the long-term storage of the ice is sublimation. Sublimation has several effects inside the ice condenser. The geometry of the ice mass changes where sublimation occurs, and the resulting vapor is deposited on a colder surface at another location inside the ice condenser.

In normal cold storage room application, the cooling coil is exposed to the air in the room, and moisture in the air freezes on the coil. If ice is stored in the room, all of the ice eventually migrates to the coil (which is defrosted periodically, draining the water outside the room) through a sublimation-mass transfer mechanism.

To avoid the mechanism, and maintain a constant mass of ice, the ice condenser is provided with double wall insulation. The annular gap between the insulated walls is provided with a heat sink in the form of a flow of cool, dry air that enters arid and leaves through the insulated panels.

However, a small amount of heat enters the system through the inlet doors, which are not double insulated, and also through the double layer insulation system. The effect of this heat gain on the ice condenser has been examined analytically.

An analytical model of the sublimation process has been developed to provide an estimate of the expected sublimation rate as well as identify the significant parameters affecting the sublimation rate. The model developed a relationship identifying the fraction of total heat input which sublimes ice (the rest of the heat raises the temperature of the air, which transports the vapor to the cold surface where it freezes). The sublimation fraction depends on the difference in vapor pressure between warmest and coldest air temperatures within the ice condenser. The sublimation fraction decreases as the ΔT decreases and also as the average ice condenser

temperature decreases. For an average temperature of 15°F in the ice condenser compartment, the analytical model predicts a sublimation rate of about 1% of the ice mass sublimed per year per ton (12,000 Btu/hr) of heat gain to the ice storage compartment. The final heat gain calculations identified a heat gain into the ice storage compartment of 1 to 1.5 tons, most of which enters the compartment through the doors.

For the purposes of this report, it is assumed that the reference heat gain for the unit is 1 ton, and therefore, the calculated reference sublimation rate would be 1% of the ice weight per year. Selected baskets are weighed as indicated in Technical Specifications to verify that the actual sublimation rate has not excessively depleted the ice inventory.

Sublimation - Operational

The method used for determining the sublimation rate is from cycle to cycle ice basket weighing. The as-left basket weight is compared to the as-found basket weight in the next cycle for the same basket. Overall trends for the ice bed at large and on a row group basis are also used to validate the sublimation rate. Historical data for Watts Bar Unit 1 and for the Sequoyah Nuclear Plant have shown cycle to cycle sublimation rates of around three percent. The selection of six percent is based on engineering judgment to provide a large safety margin.

The surveillance acceptance criteria contain a sublimation allowance of six percent. The value does not include a margin for measurement uncertainty. Instrument uncertainty must be added to the surveillance requirement ice weight value to determine an acceptable basket weight. The allowance for instrument uncertainty is approximately +15 lbs.

The Ice Bed Temperature is maintained between 15°F and 20°F during plant operation. The empirical sublimation rates described above are the results of operating in this temperature range. Procedures for ice bed temperature monitoring recommend the implementation of actions if the mean ice bed temperature exceeds 19°F. Procedures require actions to restore normal operating temperatures when ice bed temperatures reach 23°F.

Chemical Additives

Sodium tetraborate is used as a chemical additive to the ice in the plant. The boron is needed for recirculation through the core and the tetraborate is used for iodine removal and containment sump pH control. Boron or sodium tetraborate was also added to the ice used in the long-term-storage tests. Chemical analyses were performed before and after certain storage tests to identify any change in boron concentration in the ice. These chemical tests showed that the boron concentration did not significantly change during long-term ice storage. Also, the tests proved that the boron is not transferred with the ice during the sublimation process. It remains as a residue at the original point of sublimation.

Samples of flake ice with sodium tetraborate additive were placed in the cold storage room at Waltz Mill on August 29, 1969, and chemical analyses were made of the ice used in the test samples. The samples were suitably isolated so that sublimation would be minimized or prevented. The tests were terminated on June 19, 1970, approximately 9½ months after initiation, and chemical analyses were again made of several samples taken from different locations in the test section. These analyses indicated that there was essentially no change in the boron concentration from beginning to end of testing, confirming the diffusion theory discussed above.

Testing

1. General

The ice condenser design consists of 48-foot columns of ice contained within perforated metal baskets.

In the long-term storage of ice, the compression, shear, and creep characteristics are important considerations. Several years of testing at the Waltz Mill facility in these areas of interest has indicated that the ice bed maintains its geometry for its design life. While the construction of the ice baskets has changed since these tests were performed, the data is still applicable as the basic geometric configuration of the baskets has remained the same, and the same type of ice to be used in the unit was incorporated in the final series of tests. These Waltz Mill tests provide background on testing and additional information for evaluating the mechanical performance of ice.

A number of mechanical loading test series have been performed at Waltz Mill to determine compaction, shear, or creep rates in the ice bed. The first series of test initiated in 1966 used the tube ice (hollow cylinders, 1.50-inch o.d. by 0.5-inch i.d. by 2-inch length) produced in a commercial ice machine. The ice used in the above tests was made with no chemical additive, or with boron as a chemical additive to the ice. In some of these tests lead weights were placed on top of the ice samples to simulate the weight of various ice column heights.[14,15]

The final series of tests initiated in 1969 used flake ice in the same type of baskets to determine the compaction and shear rates of the ice.

As the flake ice represents the basis for the configuration used in the ice condenser, only those test results applicable for this ice form are discussed.

2. Compaction Tests

Table 6.7-23 lists and describes the flake ice compaction tests performed, the duration of the tests, and the resulting compaction after one year of testing for these tests. The results of all of the tests showed that the greatest amount of compaction occurred during the first several months of testing. The amount of compaction varied with the equivalent height of the ice column, and depended on the type of ice employed. Figure 6.7-35 presents the percent compaction versus time for flake ice test D'. Compaction of flake ice occurs much more rapidly than the other forms of ice due to the smaller and random size of the individual pieces of ice. After the initial year of compaction, the rate of compaction reduces significantly. The rate of compaction reduced almost to zero as the ice density approaches some value close to the density of solid ice. Inspection of the compaction tests indicated no evidence of ice being extruded out through the sides of the baskets.

For these tests the compaction measured is for the bottom Section of the ice bed only; the ice above this level (simulated by lead blocks) would be compacted to a lesser extent since it is loaded with less weight. Therefore, the test results were corrected for the effect of continuously reducing load from bottom to top of the ice column. When this correction was made, the results of the flake ice tests (D', E') suggest that the amount of compaction of an increment in the ice bed varies linearly with the height of the ice bed

above the increment, as shown by Figure 6.7-36. For flake ice the compaction rate must eventually change, as indicated by the dotted line, as the density of solid ice is approached. Application of this relationship would result in the estimated compaction relationship, shown in Figure 6.7-37, for total compaction (in the first year) versus unsupported height of the ice bed. Since the baskets provide supports for the ice every 6 feet, the compaction of any 6-foot section of the ice bed would be limited to less than 4 inches. While the ice bed drain temperature is a measure of ice condenser efficiency and the reduced surface area of a fused ice mass results of testing indicate that the overriding factors for determining ice condenser efficiency are initial ice mass and the geometrical arrangement of the ice columns and flow passages.

3. Shear Tests

In these tests, ice was loaded into the basket on top of a temporary bottom support which was removed within one or two weeks after loading. The initial series of tests employed tube ice in expanded metal baskets with lead weights added to simulate additional weight of ice. All of the tests experienced an initial settlement within the first two months (after the temporary support was removed). Afterwards, the results show very low creep rates, which appear to be proportional to the weight added. Subsequently, it was concluded that each increment of ice in the basket would support its own weight by shear on the adjacent basket walls.

To evaluate this theory with flake ice, additional shear tests (G',H',I') were initiated. In these tests, unsupported ice bed heights of 1 foot, 3 feet and 5 feet were tested, with no lead weights added. In theory, the shear rate should be the same since each foot of ice column had the same shear support.

The results presented in Table 6.7-23, confirmed that the shear rates for the three ice bed heights were of similar magnitude for a period of about 6 months. The rate measured was about 1 inch per year and was about 10 times the rate measured in the previous tests with tube ice in expanded metal baskets. From this information it is concluded that the shear capability of flake ice on the sides of the wire baskets is small. However, in the unit design the ice is supported by the horizontal supports at the bottom and center of each ½-foot section of ice column, so the stability of the ice bed does not depend on the shear forces existing between the ice and the baskets.

6.7.15 Ice Condenser Instrumentation

6.7.15.1 Design Basis

The ice condenser is a passive device requiring only the maintenance of the ice inventory in the ice bed. As such there are no actuation circuits or equipment which are required for the ice condenser to operate in the event of a LOCA. The instrumentation provided for the ice condenser serves only to monitor the ice bed status. Since the ice bed has a very large thermal capacity, postulated off-normal conditions can be successfully tolerated for a week to two weeks. Therefore, the ice condenser instrumentation provides an early warning of any incipient ice condenser anomalies. In this way the operator can evaluate the anomaly and take the proper remedial action. Depending upon the anomaly, the operator typically may perform a local or system defrost, switch to a backup glycol circulation pump, start a backup chiller package, provide glycol makeup, isolate a glycol leak, or perform a safe and orderly shutdown. Since the ice condenser instrumentation can in no way actuate, nor prevent, a reactor trip or

engineered safeguards action, there are no codes which apply to the design of the instrumentation systems.

Any instrumentation failures or anomalies, however, are apparent in the control room, where ice condenser temperature monitors, door position monitors, coolant liquid level and valve position indications are displayed and alarmed. Ample time is available to investigate and alleviate or eliminate any off-normal condition without seriously degrading the ice inventory. The instrumentation is nevertheless designed for reliable operation which includes sufficient redundancy to ensure that the operator can accurately monitor the ice condenser status. There are no special provisions for periodic testing of the instrumentation since normal testing and maintenance can be performed and is sufficient.

6.7.15.2 Design Description

Each equipment package (e.g., air handler, ice machine, chiller package) is provided with controls needed to regulate its normal operation. The ice condenser instrumentation serves to monitor the operation of the equipment packages and the ice bed status by providing to the operator the following control room information:

Ice Bed Temperature Monitoring

Resistance temperature detectors (RTD) are located in various parts of the ice condenser. They serve to verify attainment of a uniform equilibrium temperature in the ice bed and to detect general gradual temperature rise in the cooling system if breakdown occurs.

UNIT 1 Only

Forty-two resistance temperature detectors are mounted on ice bed probes which are located throughout the ice bed (Five other RTDs are provided to monitor floor temperature and monitor air temperature above the ice baskets). These forty-two resistance temperature detectors tie into a temperature scanner unit, located in the incore instrument room. The scanner multiplexes the ice condenser RTD's signals to a temperature recorder in the main control room. There are also six temperature switches located at various points in the ice bed to serve as backup indication should the scanner unit or recorder fail to operate. These inputs provide an alarm on the control room annunciator panel should the ice bed temperature exceed preset value.

UNIT 2 Only

Forty-two RTDs are mounted on ice bed probes which are located throughout the ice bed. (Five other RTDs are provided to monitor floor temperature and monitor air temperature above the ice baskets). These forty-seven RTDs, and an additional thirty–eight RTDs which serve to monitor various ice condenser temperatures, tie into 2 temperature recorders located in the In-Core Instrument Room. The recorders multiplex the RTD signals via an Ethernet connection to the Integrated Computer System for Main Control Room indication. There are also six temperature switches located at various points in the ice bed to serve as backup indication should the recorders fail to operate. These inputs provide an alarm on the control room panel should the ice bed temperature exceed preset value. Refer to Table 6.7-24 and Figure 6.7-38 for location of these detectors. Refer to Figure 6.7-39 for a monitor system block diagram.

Lower Inlet Door Position Indication

Ninety-six limit switches are mounted on the lower inlet door frames with two limit switches on each of forty-eight door panels per containment unit. The position and movement of the switches are such that the doors must be effectively sealed before the switches are actuated. A single annunciator window in the control room gives a common alarm signal when any door is open.

For door monitoring purposes, the ice condenser is divided into six zones (refer to Figure 6.7-40). Each zone contains four inlet door assemblies, or a total of eight door panels. Each lower inlet door is provided with two single pole double throw, or equivalent limit switches, herein designated Switch X and Switch Y.

Within each zone, the normally open contacts of all the "X" switches are connected in series to a monitor light ("Door Closed") on the lower inlet door position display panel located in the main control room on Panel M-10 (Refer to Figure 6.7-41).

Within each zone, the normally closed contacts of all the "X" switches are connected in parallel to a monitor light ("Door Open") on the door position display panel. (Refer to Figure 6.7-41).

The normally open contacts of all "Y" switches are not used. The normally closed contacts of all "Y' switches in the ice condenser are connected in parallel to the alarm on the annunciator panel ("Ice Condenser Door Open") in the main control room (refer to Figure 6.7-41).

Equipment Access Doors

Eight limit switches are provided to monitor the position of the equipment access door and the personnel access door with two switches per door. These switches are fitted in a single series circuit providing control room indication of the position of all the doors.

Each equipment access door is provided with two single pole double throw or equivalent switches to indicate door latched and door seal inflated, respectively. The normally closed set of contacts of switches on the equipment access doors, latched and inflated, are all connected in series to a monitor light ("Access Door Closed"). The normally open set of contacts are connected in parallel to a monitor light ("Access Door Open"). Refer to Figure 6.7-41.

Expansion Tank Level

Annunciation and display are provided to warn the operator of coolant level excursions in the glycol expansion tank. Four indications are displayed corresponding to HI-HI, HI, LO, and LO-LO liquid levels. A loss of level would indicate a leak somewhere in the system or an erroneous valve operation. High level would result from mal-operation or failure of the refrigeration system. Two independent sensors are provided for each pair of level indications.

Isolation Valves

Two position lights (Open and Closed) located in the control room are provided for each of the glycol containment isolation valves. Individual annunciator windows in the control room alarm on isolation valve closure.

6.7.15.3 Design Evaluation

The ice condenser design provides adequate time for the proper evaluation of any adverse

situations such that corrective action can be performed or an orderly unit shutdown can be scheduled and accomplished within the Technical Specification limits. The ice condenser monitoring instrumentation is tested and/or inspected on a periodic basis. Sufficient redundancy is provided in the ice condenser instrumentation to assure accurate monitoring of the ice condenser status.

6.7.16 Ice Condenser Structural Design

6.7.16.1 Applicable Codes, Standards, and Specifications

The ice condenser structural design analysis are based on the AISC specification[11] where applicable. Material codes are discussed in Section 6.7.18.

6.7.16.2 Loads and Loading Combinations

1. Dead Load + Operating Basis Earthquake loads (D + OBE).*

2. Dead Load + Accident induced loads (D + DBA).

3. Dead Load + Safe Shutdown Earthquake (D + SSE).

4. Dead Load + Safe Shutdown Earthquake + Accident induced loads (D + SSE + DBA).

The loads are defined as follows:

Dead Load (D)

Weight of structural steel and full ice bed at the maximum ice load specified.

Live Load (L)

Live load includes any erection and maintenance loads, and loads during the filling and weighing operation.

Thermal Induced Load

Includes those loads resulting from differential thermal expansion during operation plus any loads induced by the cooling of ice containment structure from an assumed ambient temperature at the time of installation.

Accident Fluid Dynamic and Pressure Loads (DBA)

Accident pressure load includes those loads induced by any pressure differential drag loads across the ice beds, and loads due to change in momentum.

Operating Basis Earthquake (OBE)

The operating basis earthquake loads are those induced loads determined from the response of the ice bed and supporting structure to the OBE defined for the site.

*Also considered is D + L.

Safe Shutdown Earthquake (SSE)

The safe shutdown earthquake loads are those induced loads determined from the response of the ice bed and supporting structure to the SSE defined for the site.

6.7.16.3 Design and Analytical Procedures

Analysis meeting the criteria presented in Section 6.7.16.4 is based on elastic system and component analyses. Limited load analysis is an alternative to this elastic analysis. Limit loads are defined using limit analysis by calculating the lower bound of the collapse load of the structure. Load factors are applied to the defined design-basis loads and compared to the limit loads. The load factors determined for design-basis loads provide a margin of safety for the structure against collapse. A load factor of 1.43 is used when considering the mechanical loads due to dead weight and OBE. A load factor of 1.3 is used for either D + SSE or D + DBA. A load factor of 1.18 is used for D + SSE + DBA. The material is assumed to behave in an elastic-perfectly-plastic manner. The minimum specified yield strength is used. Mechanical plus thermal induced load combination and fatigue is analyzed on an elastic basis and satisfy the limits of Section 6.7.16.4.

Experimental or Test Verification of Design

In lieu of analysis, experimental verification of design using actual or simulated load conditions was used in some cases.

In testing, account is taken of size effect and dimensional tolerances (similitude relationships) which may exist between the actual component and the test models, to assure that the loads obtained from the test are a conservative representation of the load carrying capability of the actual component under postulated loading. The load factors associated with such verification are: 1.87 for D + OBE, 1.43 for D + DBA or D + SSE, and 1.3 for D + SSE + DBA. If the load factor of 1.87 for D + SSE cannot be met, a load factor of 1.7 is used.

A single test sample is permitted but in such cases test results are derated by 10%. Otherwise, at least three samples are tested and the design is based on the minimum load carrying capability.

6.7.16.4 Structural Acceptance Criteria

Table 6.7-25 provides a summary of the allowable limits to be used in the design of the ice condenser components.

For all cases the stress analysis is performed by considering the load combinations producing the largest possible stress values.

When limit analysis is performed on the ice condenser structure, or parts thereof, using the alternate analytical criteria method, Section 6.7.16.3, justification is provided to show that the results of the elastic systems analysis are valid.

Stress Criteria

The stress limits for elastic analysis are:

1. D + OBE

   Stress is limited to normal AISC, Part 1 Specification allowables (S). The members and their connections are designed to satisfy the requirements of Part 1, Sections 1.5, 1.6, 1.7, 1.8, 1.9, 1.10, 1.15, 1.16, 1.17, 1.20, 1.21, and 1.22 of the AISC Specification (stress increase in Sections 1.5 and 1.6 is disallowed for these loads). Where the requirements of Section 1.20 are not met, differential thermal expansion stresses are evaluated and the maximum range of the sum of mechanical and thermal-induced stresses are limited to three times the appropriate allowable stresses provided in Sections 1.5 and 1.6 of AISC Specification.

2. D + SSE, D + DBA

   Stresses are limited to normal AISC Specifications allowables given in Sections 1.5 and 1.6, increased by 33% (1.33 S). No evaluation of thermal-induced stresses or fatigue is required.

3. D + SSE + DBA

   Stresses are limited to normal AISC Specification allowables given in Sections 1.5 and 1.6, increased by 65% (1.65 S). No evaluation of thermal-induced stresses for fatigue is required.

   For all cases, direct (membrane) mechanical stresses are not to exceed 0.7 S_u, where S_u is the ultimate tensile strength of the material.

   The summary of the ice condenser allowable limits is given in Table 6.7-25.

6.7.17 Seismic Analysis

6.7.17.1 Seismic Analysis Methods

The lattice frames, ice baskets, wall panels on the crane wall side, and lower support of the ice condenser structure form a complex structural system. In order to perform a realistic seismic analysis of this structure, it is necessary to consider the gaps between the ice baskets and the lattice frame. It is not feasible to perform a response spectrum model analysis when considering gaps because the structure is non-linear, thus requiring a dynamic time history analysis. Six different non-linear, models are used to develop the design loads. Results are documented in Section 6.7.17.2.

Linear Seismic Analysis

Each level of lattice frames encompasses an approximate 300° horizontal arc and consists of 72 lattice frames. One level of eight levels of lattice frames is modeled so that the structural coupling between individual lattice frames could be evaluated.

The dynamic model used to determine the horizontal response characteristics of one level of lattice frames is shown in Figure 6.7-44. It is a lumped-mass beam representation. Cantilever beam elements are used to represent the bending and shear stiffness of six interconnected lattice frames as shown in Figure 6.7-45. For the model shown in Figure 6.7-44, the mass associated with a set of six lattice frames is lumped at the end of the cantilever beam. The length used for the cantilever beam is representative of the distance to the center of gravity of the ice baskets associated with one lattice frame. The lumped masses are connected by tie members representing the combined coupling stiffness of six lattice frames.

The dynamic response characteristics of one level of lattice frames is obtained by computer program. It was determined that the structural coupling between individual lattice frames is negligible and that the fundamental response of the ice bed lattice frame is essentially that of the individual lattice frames acting independently. Therefore, a lattice frame can be uncoupled from those in the same level for modeling purposes.

Non-Linear Seismic Analysis

1. Ice Condenser Seismic Load Study of the Effect of Gaps

   A clearance or gap is required at the ice basket supports for installation and maintenance reasons. A schematic view of the ice basket gap is shown in Figure 6.7-46. The design value for the cap is ¼-inch radially or ½-inch on the diameter.

   The effect of the gap during a seismic excitation is two-fold. First, impact loads are applied to the ice basket as it bounces within the clearance, which produce higher loads in the ice basket than would exist if there were no gap. Second, the repetitive impacting at the ice basket supports dissipate substantial amounts of energy. Stated differently, there is a higher damping within the structure than would exist if there were no gaps. This effect is illustrated with actual test results in Figure 6.7-47.

2. Description of Non-Linear Models

   Four non-linear models of lattice frames uncoupled from those in the same level were used to determine the effect of ice basket impact on the ice condenser loads. Two additional models with adjacent lattice frame bays coupled by a phasing link were used to investigate lattice frame phasing. The six models are shown in Figures 6.7-48 through 6.7-53 and are described as follows:

Shown in Figure 6.7-48 is the two-mass model which is composed of two non-linear elements which represent the local impact stiffness existing between the lattice frame and ice basket, and a lattice frame spring between the lattice frame mass and the crane wall. The impacting mass represents twenty-seven, six-foot long ice baskets.

Five other models were developed to assess the validity of the two mass model.

Figure 6.7-49 shows the three mass tangential model whose purpose was to assess the effect of phasing between ice baskets in the tangential direction. There are three rows of ice baskets in the tangential direction across each lattice frame. Each lumped mass represents one ice basket. The lattice frame is modeled as truss members spanning each ice basket.

Figure 6.7-50 shows the nine-mass radial model whose purpose is to assess phasing in the radial direction. Nine rows of ice baskets in the radial direction going out from the crane wall are represented in the model. Each basket has its associated impact elements on each side and the effective properties of the lattice frame spanning each ice basket.

Figure 6.7-51 shows the 48-ft beam model which is a non-linear model containing twenty-seven ice baskets modeled as a continuous beam. The local effect of each lattice frame is represented by a pair of impact elements, one on each side of the ice basket. The lattice frame-wall panel stiffness is represented by a stiffness element. The lower support structure is modeled by a stiffness element at the bottom of the ice basket. The purpose of this model is to investigate the influence of the full 48-feet ice basket column.

Figure 6.7-52 shows the phasing mass model whose purpose is to evaluate the phasing link loads and crane wall reactions when adjacent bays of lattice frames respond out of phase with each other. The phasing mass model consists of a pair of two-mass models representing adjacent bays of the ice condenser. The lattice frames of the adjacent bays are coupled together with a phasing link. The design value for the phasing link gap is 1/16- inch between adjacent lattice frames.

Figure 6.7-53 shows the non-linear 300° phasing model. The non-linear 300° phasing model is similar to the linear model shown in Figure 6.7-44 except that it incorporates a phasing connector between lattice frames with a phasing gap of 1/16 inch between adjacent lattice frames. The purpose of this model was to demonstrate that the phasing link creates "phasing" within a specified tolerance and to demonstrate that it is still valid to model the basic ice condenser structure using only one lattice frame per level even though a phasing connector is used.

Analytical Procedure and Typical Results

Using typical results obtained from the two-mass dynamic model, the procedure used in the non-linear analysis will now be discussed. First, the input acceleration-time histories are converted to displacement-time histories by double integration. The displacement time histories as shown in Figure 6.7-54 were then input to the non-linear dynamic model.

Results are shown in Figures 6.7-55 through 6.7-57 for the case corresponding to a one-half inch gap between the ice basket and lattice frame, for tangential excitation.

Figure 6.7-55 shows the output displacement-time history of the ice basket mass superimposed on the input displacement. It shows that the response generally follows the input displacements except for some amplification in the neighborhood of the peaks.

Figure 6.7-56 shows the impact loads on the ice baskets for this particular case. Note the short duration time of the impact loads.

Figure 6.7-57 shows the forces induced in the wall panels on the crane wall side as obtained from the two-mass dynamic model.

6.7.17.2 Seismic Load Development

Time History Dynamic Input

Crane wall seismic time histories for the OBE and SSE, in the EW and NS directions, were developed using four synthesized earthquakes. These earthquakes are the same as used to develop the Watts Bar response spectra. These time histories were the actual earthquake records as modified by the building, i.e., as filtered through the building to the points of interest on the crane wall.

The structural response is computed for each earthquake and then averaged by computing the arithmetic mean of the four sets of response values. The seismic design loads are based on the seismic loads obtained by averaging.

Design Load Verification Analyses

Non-linear seismic results obtained using the two-mass dynamic model are shown in Tables 6.7-27 and 6.7-28 for the tangential and radial cases, respectively. The wall panel loads and impact loads are shown for the OBE and-SSE north-south and east-west earthquakes with the respective design loads. The lattice frame-wall panel stiffness used to obtain the analysis results shown were 24,000 lb/in for the tangential case and 50,000 lb/in for the radial case. These values are consistent with stiffness obtained from tests.

The analyses made using the two-mass dynamic model used the time histories associated with 807.82 feet elevation on the crane wall which has high seismic response characteristics.

Table 6.7-29 gives a summary of SSE load results obtained from the five non-linear dynamic models.

Seismic tangential and radial load distributions along the crane wall were found using the 48 ft beam model and are presented in Figures 6.7-58 and 6.7-59. They represent the portion of the seismic design load used at the various lattice frame locations. All loads obtained from analysis are within the seismic load distribution design "envelope".

Many seismic studies have been performed to understand the dynamic behavior of the ice condenser system. The effect of sublimation on the ice condenser system response has been studied. Phasing studies have been performed. The findings from these studies have been reported in other submittals, and therefore are not reported here. For a discussion of these

studies, see References [12] and [19].

Seismic Design Loads

Seismic design loads have been developed for the lattice frames, ice baskets, and the wall panels. They are shown in Tables 6.7-27 and 6.7-28. The seismic design load distributions developed using the 48 feet beam model are shown in Figures 6.7-58 and 6.7-59.

The non-linear analyses performed to develop seismic design loads uses 5% structural damping and 10% impact damping, and nominal gap size of ½-inch on the diameter between the baskets and the lattice frames. Note that a nominal gap size of 1/16-inch exists in the link between adjacent lattice frames.

Table 6.7-30 gives a summary of parameters used in the seismic analyses. These parameters are based on analyses and tests of the ice condenser system.

6.7.17.3 Vertical Seismic Response

The combined floor and lower support structure are modeled in the vertical direction. The full weight of the baskets and ice were considered. It was found that the fundamental frequency, the dominant mode, of the combined structure in the vertical direction is above 14.7 Hz. There is no amplification of the crane wall in the vertical direction at the elevation of the lower support structure. Therefore, the vertical response spectra have the shape of the ground response spectra and are normalized to two-thirds of the seismic ground acceleration.

6.7.18 Materials

6.7.18.1 Design Criteria

Structural steels for ice condenser components are selected from the various steels listed in the AISC Manual of Steel Construction or ASTM Specifications[11]. When materials such as steel sheets, stainless steel or non-ferrous metals are required and are not obtainable in the AISC Code, these materials are chosen from ASTM specifications. Proprietary materials such as insulating materials, gaskets and adhesives are listed with the manufacturer's name on the component drawings.

Material certifications for chemical analysis and mechanical properties are required with testing procedure and acceptance standards meeting the AISC or ASTM requirements.

Because the concept of non-ductile fracture of ferritic steel is not a part of the AISC Code and Westinghouse recognizes its importance in certain ice condenser components where heavy plates and structurals are used such as the lower support structure, Charpy V-notch (CVN) energy absorption requirements are stipulated as shown in Table 6.7-26. Bolting material for bolts one inch in diameter and larger meets the impact energy absorption requirement, for full size CVN specimen, of 20 ft-lbs at -20°F.

These criteria apply to the design of the following ice condenser components:

1. Lattice frame and columns including attachments and bolts greater than one inch in diameter.

2. Structural steel supporting structures comprising the lower support structure, door frames and bolts greater than one inch in diameter.

3. The supports of auxiliary components which are located within the ice condenser cavity but which have no safety function.

Wall panels and cooling duct support studs attached to the crane wall and end walls are tested as follows:

1. A hammer bend test on the gun-welded and fillet-welded studs is performed in accordance with AWS D 1.1. This test is performed at temperatures of +70°F. +20°F, and -20°F .

2. A bend test to measure the flexural strength of the studs at the above temperatures is performed. The studs are welded to a plate of similar physical and chemical properties by the method and position (flat, vertical, overhead, sloping) used in installation. Acceptance is based on the stud's ability to meet the minimum ultimate strength prior to failure.

The various candidate materials, i.e., steel sheets, structural shapes, plates and bolting used in the ice condenser system, are selected on the following criteria:

1. Provide satisfactory service performance under design loading and environment and pressure or construction performance.

2. Assure adequate fracture toughness characteristics at ice condenser design conditions.

3. Be readily fabricated, welded, and erected.

4. Be readily coated for corrosion resistance, when required.

The candidate materials are of high quality and are made by steel making practices to be specified by Westinghouse. Principal candidate materials meeting the above bases are discussed below. Other materials for specific applications are selected on a case-by-case basis.

6.7.18.2 Environmental Effects

The atmosphere in the ice bed environment is at 10°F - 20°F and the absolute humidity is very low. Therefore, corrosion of uncoated carbon steel is negligible.

To ensure that corrosion is minimized while the components of the ice condenser are in storage at the site or in operation in the containment, components are galvanized, painted, or placed in a protective container. Galvanizing is done in accordance with ASTM A123.

Materials such as stainless steels with low corrosion rates are used without protective coatings.

Corrosion has been considered in the detailed design of the ice condenser components, and it has been determined that the performance characteristics of the ice condenser materials of construction are not impaired by long term exposure to the ice condenser environment.

Since metal corrosion rates are directly proportional to temperature and humidity, corrosion of ice condenser components at operating temperatures has been considered to be almost non-existent. Data available in the open literature do not reflect the exact temperature range and chemistry conditions that are expected to exist in the ice condenser, but do indicate that corrosion rates decreased with decreasing temperatures for the materials and conditions being considered. Although the data in the literature indicated that corrosion of components is not expected, several preventive measures were used in the construction of the ice condenser system. To inhibit corrosion, the ice baskets are galvanized. Other structural members are either galvanized or protected by corrosion resistant paints that meet the requirements of ANSI 101.2-1972 ("Protective Coatings (Paints) for Light Water Nuclear Reactor Containment Facilities"), as a minimum, or are constructed of stainless steel. Heavy plate and structural fabrications made from A588 steel may be installed in the blasted and/or bare condition. A tightly adherent scale forms on the surface of this steel when it is exposed to the atmosphere.

Quenched and tempered steel components are not hot dip galvanized, but are painted or left in the base condition.

With due consideration of the non-corrosive environment, and judicious selection of component materials based upon sound engineering judgment, the structural integrity of the ice condenser components is not jeopardized, and the design criteria for the plant are met.

6.7.18.3 <u>Compliance with 10 CFR 50, Appendix B</u>

The following sections of this report address themselves to demonstrating compliance with 10 CFR 50, Appendix B. The design process control policy defines the criteria that must be considered when establishing design process control procedures. The design process procedures represent how Westinghouse controls its design processes relative to 10 CFR 50, Appendix B requirements. The subject procedures are supplementary to the flow diagram and cross-reference is obtained through the use of activity numbers.

The products and scope of responsibility at Westinghouse are defined by the shop order description. From this base, the shop order flow diagrams were developed for the purpose of subdividing the job into its component activities and thereby creating generic categories of activities that require similar control systems. These categories are:

1. <u>Interface Control</u>

   A. Interfaces are controlled by specifically identifying the relationship on the flow diagram and also by quantifying the information transmitted across the interface.

   B. Document Control - A procedure employing a file log book is carefully maintained and provides control of document issue.

2. <u>Analysis (includes review and comment, approval responsibilities)</u>

   These activities involved in providing or commenting on design information are controlled according to methods outlined in design process procedures. The nature of the product and its relative technical importance determine the level of controls applied.

3. <u>Verification</u>

These activities fulfill the requirement that design information must be validated by the originator prior to communication to a user. Techniques used vary due to the diverse nature of the products involved.

There is a design process control procedure for each shop order which consists of at least, a flow diagram, a shop order description, and design process procedures.

The design process procedures are related to the flow chart through the use of the activity numbers and the specific control methods.

Regarding quality assurance of material, when required, parts or welded fabrications will be inspected by visual, magnetic particle (MT), liquid penetrant (PT), or ultrasonic (UT) methods according to ASTM procedures, AWS D1.1, or Westinghouse process specifications. The method and extent of inspection are designated on the component drawings.

6.7.18.4 Materials Specifications

Sheets

Carbon steel sheets are commercial quality (CQ), drawing quality (DQ), or drawing quality-special kilned (DQ-SK). The selection of the quality depends upon the part being formed. When higher strength, structural quality sheets are required, ASTM Specification A607 (Unit 1/Unit 2) and A1011 (Unit 2) are used.

The ice baskets are made from perforated sheet material. The wall duct panels are made from sheet material.

Structural Sections, Plates, and Bar Flats

Structural sections, plates, and bar flats are generally high strength, low alloy steels selected for suitable strength, toughness, formability and weldability.

The high strength low alloy steels are A441, A588, A572, or A633. These steels are readily oxygen cut and possess good weldability.

Bolting

High strength alloy steel Type A320 L7 bolting for low temperature service is used for the lower support structure. Stocked bolting made from A325, A449, and ASTM A354 Grade BD (SAE J429 Grade 8) materials is used for other parts. The above bolts meet CVN 20 ft-lb at 20°F for sizes greater than 1-inch in diameter.

Non-Metallic Materials

Non-metallic materials such as gaskets, insulation, adhesives and spacers are selected for specific uses. Freedom from detrimental radiation effects is required.

Welding

Welding was in accordance with the American Welding Society (AWS), "Structural Welding

Code," AWS D1.1 with revisions 1-73 and 1-74, except later editions may be used for prequalified joint details, base materials, and qualification of welding procedures and welders. Nuclear Construction Issues Group documents NCIG-01 and NCIG-02 may be used after June 26, 1985, for weldments that were designed and fabricated to the requirements of AISC/AWS. Visual inspection of structural welds will meet the minimum requirements of NCIG-01 and NCIG-02 as specified on the design drawings or other design output. Inspectors performing visual examination to the criteria of NCIG-01 are trained in the subject criteria.

Magnetic particle examination is performed on at least 5% of the welds in each critical member of the lower support structure. Magnetic particle or liquid penetrant examinations where applicable, are performed on 5% of the welds in each critical member of the balance of the ice condenser structure. The welds selected for non-destructive test examinations are designated on the component drawings or in the design specifications.

6.7.19 Tests and Inspections

The tests and inspections are given in the Technical Specifications.

REFERENCES

1. Deleted.

2. Test Plans and Results for the Ice Condenser System, WCAP-8110, April 16, 1973.

3. Test Plans and Results for the Ice Condenser System, WCAP-8110, Supplement 1, April 30, 1973.

4. Test Plans and Results for the Ice Condenser System, WCAP-8110, Supplement 2, June 19, 1973.

5. Test Plans and Results for the Ice Condenser System, WCAP-8110, Supplement 3, July 19, 1973.

6. Test Plans and Results for the Ice Condenser System, WCAP-8110, Supplement 4, November 15, 1973.

7. Plantema, F. F., Sandwich Construction, John Wiley, New York, 1966.

8. Idel'chik, Handbook of Hydraulic Resistance, AECTR 6630, NTIS, Springfield, Va.

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11. Specification for the Design, Fabrication and Erection of Structural Steel for Buildings, American Institute of Steel Construction, 1969 Edition.

12. Donald C. Cook Nuclear Plant, FSAR, American Electric Power Service Corporation, Docket Numbers 50-315 and 50-316.

13. Ice Condenser System Lower Inlet Door Shock Absorber Test Plans and Results, WCAP-8336, May 1974 (Westinghouse NES Proprietary), and WCAP-8110, Supplement 5, May 1974.

14. Final Report - Ice Condenser Full-Scale Tests at the Waltz Mill Facility, WCAP-8282, February 1974 (Westinghouse NES Proprietary), and WCAP-8110, Supplement 6, May 1974.

15. Final Report - Ice Condenser Full-Scale Tests at the Waltz Mill Facility, WCAP-8282, Addendum 1, May 1974 (Westinghouse NES Proprietary), and WCAP-8110, Supplement 7, May 1974.

16. Stress and Structural Analysis and Testing of Ice Baskets, WCAP-8304 (Westinghouse NES Proprietary) and WCAP-8110, Supplement 8, May 1974.

17. Ice Fallout From Seismic Testing of Fused Ice Basket, WCAP-8110, Supplement 9-A, May 1974.

18. Static Testing of Production Ice Baskets, WCAP-8110, Supplement 10, September 1974.

19. Sequoyah Nuclear Plant Final Safety Analysis Report, Tennessee Valley Authority, Docket Numbers 50-327 and 50-328.

20. TVA letter Number W-7678, “Stepped Boron Concentration/Refueling Water Storage Tank Boron,” dated May 16, 2003, transmitting maximum stored ice weight in Ice Condenser.

TABLE 6.7-1 (Sheet 1 of 2)

WALL PANEL DESIGN LOADS[1]

A.	Service Loads	
	Weight of Panels on Containment and End Wall (58 ft-length)	100 lbs/linear ft
	Weight of Panels Crane Wall (48-ft length)	60 lbs/linear ft
	Pressure (Wall panel internal)	0 to 0.5 psig
B.	OBE Lattice Frame Column Loads [2] (Maximum at 45-ft elevation)	
	Radial at 90° (acting alone)	± 7920 lbs
	Tangential at 0° (acting alone)	± 9600 lbs
	Combined Load at 45°	
	Radial	± 6190 lbs
	Tangential	± 6190 lbs
C.	SSE Lattice Frame Column Loads [2] (Maximum at 45-ft elevation)	
	Radial at 90° (acting alone)	+ 8800 lbs
	Tangential at 0° (acting alone)	± 11200 lbs
	Combined Load at 45°	
	Radial	± 7070 lbs/ea
	Tangential	± 7070 lbs/ea
D.	DBA[2] (Maximum at 15 ft elevation)	
	Lattice Frame Column Load	
	Radial	± 6210 lbs
	Tangential	± 8259 lbs
	Pressure (D.L.F.) = 1.5; M = 1.4)*	18.9 psig

*DLF = Dynamic Load Factor
M = Margin

TABLE 6.7-1 (Sheet 2 of 2)

WALL PANEL DESIGN LOADS[1]

E. SSE plus DBA[2]

15-ft Elevation

Lattice Frame Column Load @ 0°	
Radial	± 6211 lbs
Tangential	±13260 lbs
Lattice Frame Column Load @ 45°	
Radial	±10701 lbs
Tangential	±12750 lbs
Pressure (D.L.F. = 1.5; Margin = 1.4)	18.9 psig

33-ft Elevation

Lattice Frame Column Load @ 0°	
Radial	0
Tangential	±14920 lbs
Lattice Frame Column Load @ 45°	
Radial	± 6916 lbs
Tangential	±13336 lbs
Lattice Frame Column Load @ 90°	
Radial	±11060 lbs
Tangential	± 6420 lbs
Pressure (D.L.F. = 1.5; Margin = 1.4)	18.9 psig

(1) Design Pressure loads, as stated, are applied uniformly to the wall panel transverse beams. Radial and tangential loads are applied at lattice frame column to wall panel attachment. These are maximum load combinations.

(2) Vertical seismic loads (0.35 and 0.55 times dead load for 1/2 SSE and SSE, respectively) and vertical design basis accident loads are neglected in the analyses because they are small in comparison to the radial and tangential loads.

TABLE 6.7-2

ICE BASKET LOAD SUMMARY

Minimum Test Loads

Elevation* (ft)	Case I D + OBE		Case II D + DBA		Case III D + SSE		Case IV D + SSE + DBA	
	H	V	H	V	H	V	H	V
0	463	4933	429	-2283	496	433O	841	-3473
6	1131	4316	423	-1998	1211	3789	1486	-3039
12	1296	3698	414	-1713	1387	3248	1638	-2605
18	1543	3083	357	-1427	1652	2707	1826	-2171
24	1748	2466	333	-1142	1872	2164	2005	-1736
30	1790	1849	303	-856	1916	1623	2017	-1301
36	1810	1232	252	-531	1938	1082	1991	-831
42	1687	617	213	-285	1806	541	835	-434
48	823	0	192	0	881	0	976	0

Basic Design Loads

Elevation* (ft)	D		OBE		SSE		DBA	
	H	V	H	V	H	V	H	V
0	0	1776	225	622	315	977	143	-2536
6	0	1554	550	544	770	885	141	-2219
12	0	1332	630	466	882	733	138	-1902
18	0	1110	750	389	1050	611	119	-1585
24	0	888	850	311	1190	488	111	-1268
30	0	666	870	233	1218	366	101	-951
36	0	444	880	155	1232	244	84	-614
42	0	222	820	78	1148	122	71	-317
48	0	0	400	0	560	0	64	0

* Above lower support structure

TABLE 6.7-3

SUMMARY OF STRESSES IN BASKET DUE TO DESIGN LOADS

Elevation from Lower Support Structure, ft	Design Load, lb[1] H	V	Maximum Stress, psi	Allowable Stresses, psi
0	[3] 304	3029	11,508	25,536[2]
12	[3] 650	2271	17,100	25,536[2]
24	[3] 761	1514	17,967	25,536[2]
36	[3] 835	378	17,435	25,536[2]
48	[4] 1017	-2003	23,988	31,104[5]

Notes:

(1) With 10% margin

(2) Allowable stress = 0.6 x s_y x 1.33 per Section 6.2.2.16

(3) Design load, D + SSE

(4) Design load, D + SSE + DBA, 10% margin on weight, 40% margin on pressure and 1.5 dynamic load factor

(5) Allowable stress = 0.6 x s_y x 1.65

TABLE 6.7-4

ICE BASKET MATERIAL MINIMUM YIELD STRESS

Item	Material	Minimum Yield Stress (ksi)
Clevis Pin	ASTM A434 Class BC Grade 4140	110
U-Bolts	SAE-J 429 Grade 8	130
Basket End Coupling and Stiffener	ASTM A-622	32
Nut	SAE J995 Grade 8	96
Mounting Bracket Assembly	ASTM A-148 + AMS 5334C Grade 80-50	50
Plate	ASTM A-36	36
Grid Bars	ASTM A-570 Grade B and/or ASTM A1011 Gr 33	40 33
Wire Mesh	AISI 1010-1015	40
Perforated Basket	ASTM A-569	32
Couple Screw	AISI 1022 Rc32	112

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TABLE 6.7-5

ALLOWABLE STRESS LIMITS (D + OBE)
FOR ICE BASKET MATERIALS

Material	Specified Minimum Yield (ksi)	Tension $F_t = 0.6Fy$ (ksi)	Allowable Shear $F_y = 0.4Fy$ (ksi)	Limits Bearing $F_p = 0.9Fy$ (ksi)	Bending $F_b = 0.66Fy$ (ksi)
Carbon Steel					
130 KSI					
Minimum Yield	130	78	52	117	85.8
ASTM					
A588	50	30	20	45	33
ASTM					
A570	30	18	12	27	19.8
ASTM					
A622	32	19.2	12.8	28.8	21.1
ASTM					
A36	36	21.6	14.4	32.4	23.8
ASTM					
A641	40	24	16	36	26.4
ASTM					
A569	32	19.2	12.8	28.8	21.1

TABLE 6.7-6

ALLOWABLE STRESS LIMITS (D + SSE), (D + DBA)
FOR ICE BASKET MATERIALS

Material	Specified Minimum Yield (ksi)	Tension S_t=1.33F_t (ksi)	Allowable Shear S_v = 1.33F_v (ksi)	Limits Bearing S_p = 1.33F_p (ksi)	Bending S_b=1.33F_b (ksi)
Carbon Steel					
130 KSI					
Minimum	130	103.7	69.2	155.6	114.1
ASTM-A588	50	39.9	26.6	59.8	43.9
ASTM					
A570	30	23.9	16.0	35.9	26.3
Grade B					
ASTM					
A622	32	25.5	17.0	38.3	28.1
ASTM					
A36	36	28.7	19.1	43.0	31.6
ASTM					
A641	40	31.9	21.3	47.9	35.1
ASTM					
A569	32	25.5	17.0	38.3	28.1

TABLE 6.7-7

ALLOWABLE STRESS LIMITS (D + SSE + DBA)
FOR ICE BASKET MATERIALS

Material	Specified Minimum Yield (ksi)	Tension $S_t = 1.65F_t$ (ksi)	Allowable Shear $S_v = 1.65F_v$ (ksi)	Limits Bearing $S_p = 1.65F_p$ (ksi)	Bending $S_b = 1.65f_b$ (ksi)
Carbon Steel					
130 KSI					
Minimum	130	128.7	85.8	193.1	141.6
ASTM-A588	50	49.5	33.0	74.2	54.4
ASTM					
A570	30	29.7	19.8	44.6	32.7
Grade B					
ASTM					
A622	32	31.7	21.1	47.5	34.8
ASTM					
A36	36	35.6	23.8	53.5	39.2
ASTM					
A641	40	39.6	26.4	59.4	43.6
ASTM					
A569	32	31.7	21.1	47.5	34.8

TABLE 6.7-8

ICE BASKET CLEVIS PIN STRESS SUMMARY

Load Case No.	Horiz. Load H (lbf)	Vert. Load V (lbf)	Pin Bending Stress f_b (10^3 psi)	Pin Shear Stress f_y (10^3 psi)	Pin-Lug Bearing Stress f_p (10^3 psi)
I	251	2638	67.3 (97.5)[1]	13.5 (52)	10.6 (45.0)
II	300	-1596	41.2 (129.7)	8.3 (69.2)	6.5 (59.8)
III	251	3028	77.1 (129.7)	15.5 (69.2)	12.1 (59.8)
IV	551	-2671	69.3 (160.9)	13.9 (85.8)	10.9 (74.2)

Notes:

[1]Parenthetical values are stress allowables.

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TABLE 6.7-9

ICE BASKET MOUNTING BRACKET ASSEMBLY STRESS SUMMARY

Load Case No.	Horiz. Load H (lbf)	Vert. Load V (lbf)	Load Case Factor N	Point Interaction Formula Value[1] X	Washer Bearing Stress f_p (psi x 10^3)	Shear Tear Out Stress f_v (psi x 10^3)	Weld Shear Stress f_v (psi x 10^3)
I	251	2638	1.0	0.90	34.6 (45.0)[2]	- (20.0)	7.8 (20.0)
II	300	-1596	1.33	0.57	36.6 (59.8)	5.3 (26.6)	5.4 (26.6)
III	251	3028	1.33	1.02	34.6 (59.8)	- (26.6)	8.7 (26.6)
IV	551	-2671	1.65	0.96	53.0 (74.2)	8.9 (33.0)	9.2 (33.0)

Notes:

[1] $X \leq N$ indicates safe condition.

[2] Parenthetical values are stress allowables.

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TABLE 6.7-10

ICE BASKET PLATE STRESS SUMMARY

Load Case No.	Horiz. Load H (lbf)	Vert. Load V (lbf)	Load Case Factor N	Point 1 Interaction Formula Value[1] X	Point 2 Interaction Formula Value[1] X
I	251	2638	1.0	0.25	0.27
II	300	-1596	1.33	0.23	0.29
III	251	3028	1.33	0.28	0.27
IV	551	-2671	1.65	0.42	0.53

Notes:

(1) $X \leq N$ indicates safe condition.

TABLE 6.7-11

ICE BASKET V-BOLT STRESS SUMMARY

Load Case No.	Horiz. Load H (lbf)	Vert. Load V (lbf)	Tensile Stress f_b (10^3 psi)
I	521	2638	42.8 (78.0)[1]
II	300	-1596	55.1 (103.7)
III	251	3028	42.8 (103.7)
IV	551	-2671	65.6 (128.7)

Notes:

(1) Parenthetical values are stress allowables.

TABLE 6.7-12

ICE BASKET - BASKET END STRESS SUMMARY

Load Case No.	Horiz. Load H (lbf)	Vert. Load V (lbf)	Load Case Factor N	Point 1 Interaction Formula Value X[1]	Point 2 Interaction Formula Value X[1]
I	251	2638	1.0	0.74	0.97
II	300	1596	1.33	0.85	0.63
III	251	3028	1.33	0.76	1.10
IV	551	2671	1.65	1.56	1.08

Notes:

[1] $X \leq N$ indicates safe condition.

TABLE 6.7-13

ICE BUCKET COUPLING SCREW STRESS SUMMARY
3 INCH ELEVATION[1]

Load Case No.	Horiz. Load H (lbs)	Vert. Load V (lbs)	Screw Bending Stress f_b (ksi)	Screw Shear Stress f_v (ksi)	Basket Bearing Stress f_p (ksi)	Shear Tear-Out Stress f_{vt} (ksi)
I	251	2638	65.8 (85.8)[2]	12.0 (52.0)	16.8 (28.8)	4.3 (12.8)
II	300	-1596	43.1 (114.1)	7.8 (69.2)	11.0 (38.3)	2.8 (17.0)
III	251	3028	74.7 (114.1)	13.6 (69.2)	19.1 (38.3)	4.8 (17.0)
IV	551	-2671	73.1 (141.6)	13.3 (85.8)	18.7 (47.5)	4.7 (21.1)

Notes:

[1] Above top of lower support structure.

[2] Parenthetical values are stress allowables.

TABLE 6.7-14

ICE BUCKET COUPLING SCREW STRESS SUMMARY
12 FOOT ELEVATION[1]

Load Case No.	Horiz. Load H (lbs)	Vert. Load V (lbs)	Screw Bending Stress f_b (ksi)	Screw Shear Stress f_v (ksi)	Basket Bearing Stress f_p (ksi)	Basket Tear-Out Stress f_{vt} (ksi)
I	818	1977	81.8 (85.8)[2]	14.9 (52.0)	20.9 (28.8)	5.3 (12.8)
II	289	-1198	40.2 (114.1)	7.3 (64.2)	10.3 (38.3)	2.6 (17.0)
III	818	2271	88.5 (114.1)	16.1 (64.2)	22.6 (38.3)	5.7 (17.0)
IV	1108	-2004	95.3 (141.6)	17.4 (85.8)	24.4 (47.5)	6.2 (21.1)

Notes:

(1) Above top of lower support structure.

(2) Parenthetical values are stress allowables.

TABLE 6.7-15

ICE BASKET COUPLING SCREW STRESS SUMMARY
24 FOOT ELEVATION[1]

Load Case No.	Horiz. Load H (lbs)	Vert. Load V (lbs)	Screw Bending Stress f_b (ksi)	Screw Shear Stress f_v (ksi)	Basket Bearing Stress f_p (ksi)	Basket Tear-Out Stress f_{vt} (ksi)
I	1122	1319	82.1 (85.8)[2]	15.0 (52.0)	21.0 (28.8)	5.3 (12.8)
II	233	-799	29.0 (114.1)	5.3 (64.2)	7.4 (38.3)	1.9 (17.0)
III	1122	1513	86.5 (114.1)	15.8 (69.2)	22.1 (38.3)	5.6 (17.0)
IV	1355	-1335	93.2 (141.6)	17.0 (85.8)	23.9 (47.5)	6.0 (21.1)

Notes:

(1) Above top of lower support structure.

(2) Parenthetical values are stress allowables.

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TABLE 6.7-16

ICE BUCKET COUPLING SCREW STRESS SUMMARY
36 FOOT ELEVATION[1]

Load Case No.	Horiz. Load H (lbs)	Vert. Load V (lbs)	Screw Bending Stress f_b (ksi)	Screw Shear Stress f_v (ksi)	Basket Bearing Stress f_p (ksi)	Basket Tear-Out Stress f_{vt} (ksi)
I	1161	658	66.9[2] (85.8)	12.2 (52.0)	17.1 (28.8)	4.32 (12.8)
II	176	-371	16.4 (114.1)	3.0 (64.2)	4.2 (38.3)	1.1 (17.0)
III	1161	757	69.1 (114.1)	12.6 (69.2)	17.7 (38.3)	4.5 (17.0)
IV	1338	-639	74.4 (141.6)	13.6 (85.8)	19.0 (47.5)	4.8 (21.1)

Notes:

(1) Above top of lower support structure.

(2) Parenthetical values are stress allowables.

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TABLE 6.7-17

CRANE AND RAIL ASSEMBLY DESIGN LOADS

Normal Operation		
Crane Weight (excluding rails)	7200 lbs	
Maximum Capacity During Plant Erection	6000 lbs	(each of two cranes)
Maximum Capacity	6000 lbs	(one crane)
Maximum Load Expected	2400 lbs	

TABLE 6.7-18 (Sheet 1 of 2)

REFRIGERATION SYSTEM PARAMETERS

		UNIT 1	UNIT 2
1.0	General - per twin containment station		
	Cooling water temperature, maximum design	85°F	85°F
	Number of ice condenser units	2	2
2.0	Refrigeration - per twin containment station		
2.1	Glycol Chilling Machines - 5 dual packages installed		
	Refrigeration capacity per chiller (half-pkg), nominal	25 tons*	25 tons*
	Total plant capacity, nominal, 5 x 2 x 25	250 tons	250 tons*
	Glycol flow per evaporator, normal	~127 gpm	~127 gpm
	Glycol flow per evaporator at max. - P	200 gpm	200 gpm
	Glycol pressure, maximum design	180 psig	180 psig
	Pressure drop through evaporator, normal	16 feet	16 feet
	Maximum allowable ªP through evaporator	17 feet	40 feet
	Glycol entering temperature, estimated	2°F	2°F
	Glycol exit temperature	-5.2°F	-5°F
	Cooling water flow per condenser, normal	110 gpm	110 gpm*
	Total cooling water flow, 5 x 2 x 110	1100 gpm	1110 gpm*
	Cooling water pressure, maximum design	150 psig	150 psig
	Pressure drop through condenser	3.6 feet	3.6 feet
	Approximate refrigerant charge per chiller	235 lbs	150 lbs
	Refrigerant	R-507	R-502
2.2	Glycol Circulation Pumps - 6 installed; 2 required		
	Design flow per pump	190 gpm	190 gpm
	Total design capacity, 6 x 190	1140 gpm	1140 gpm
	TDH at design flow	220 feet**	220 feet**
	Shut-off head	250 feet	250 feet
	NPSH required at design point	~12 feet	~12 feet
2.3	Pressure Relief Valves		
2.3.1	External Headers 2 - installed		
	Set pressure (for thermal expansion of glycol)	150 psig	150 psig
	Capacity at set pressure (each)	2.9 gpm	2.9 gpm
2.3.2	Floor Cooling System Heater (1 per containment)		
	Set pressure	180 psig	180 psig

* Nominal refrigeration rating based on 88°F RCW Condenser inlet temperature (Unit 1).
Nominal refrigeration rating based on 85°F cooling water (Unit 2).

** During preoperational testing with glycol, the Glycol Circulation Pumps did not meet vendor pump performance curve. However, review of the test data indicates the pumps will deliver sufficient flow and head to satisfy operations requirements and test results are acceptable.

TABLE 6.7-18 (Sheet 2 of 2)

REFRIGERATION SYSTEM PARAMETERS

2.4	Refrigeration Medium (glycol) - UCAR Thermofluid 17 or equal			
	Concentration, ethylene glycol in water - 50 weight % or 47.8 volume %			
	At temperature:	-5°F	0°F	100°F
	Specific gravity	1.08	1.082	1.056
	Absolute viscosity (centipoises)	25.0	20.5	2.3
	Kinematic viscosity (centistokes)	23.1	18.9	2.18
3.0	Ice Condenser (per one containment unit)			
3.1	Ice Bed			
	Amount of ice stored per unit, maximum	3.0×10^6 lbs*		
	Minimum amount of ice	2.26×10^6 lbs		
	Ice displacement per year, design objective	2%		
	Design predicted ice displacement per year to wall panels for normal operation	<0.3%		
	Ice melt during maximum LOCA, calculated, approx.	See Section 6.2.1		
	Temperature of ice & static air	15-20°F nominal		
	Pressure at lower doors due to cold head, Nominal	1 psf		
	Inlet opening pressure	1 psf		
3.2	Air Handling Units - 30 dual packages installed per Containment			
	Refrigeration requirements per containment, calculated, nominal	51.5 tons		
	Gross capacity per dual package rated	2.5 tons		
	Glycol entering temperature, approx.	-5°F		
	Glycol exit temperature, approx.	1°F		
	Glycol flow per air handler (1/2 package)	6 gpm nominal		
	Total glycol flow, 30 x 2 x 6	360 gpm nominal		
	Glycol pressure drop, estimated	50 feet		
	Air blower head	2' H_2O		
	Air entering temperature, estimated	15°F		
	Air exit temperature	10°F nominal		

* Maximum ice weight not to exceed 3.0×10^6 lbs.[20]

Table 6.7-19

LOWER INLET DOOR DESIGN PARAMETERS AND LOADS

A. Normal Operation

Temperature, Lower Compartment, °F	120, Maximum
Temperature, Ice Bed, °F	10, Minimum
Pressure across Doors, psf	1.0, Nominal

B. Seismic

Response of Crane Wall at Door Elevation	
Horizontal, 1/2 SSE, g	0.20
Vertical, 1/2 SSE, g	0.05
Horizontal, SSE, g	0.40
Vertical, SSE, g	0.10

C. Accident Conditions

Temperature, Lower Compartment, °F	250, Maximum
Pressure across Doors, psf (refer to Figure 6.7-16)	1.0, Nominal

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TABLE 6.7-20

DESIGN LOADS AND PARAMETERS
TOP DECK

Plant Parameters	
Ambient temperature before cooldown, maximum °F	100
Ambient temperature, upper surface and hinge bar, range, °F	75-100
Ambient temperature, lower surface, minimum, °F	15
Post-LOCA temperature, lower surface, minimum, °F	15
Post-LOCA temperature (no - P applied), maximum, °F	190
Dead Weight	
Air handling unit and support structure, lbs/bay	2500
Grating, lbs per ft^2	7.7
Blanket panel, lbs per ft^2	1.33
Hinge bar, lbs per ft	53
Static design equivalent of live load (personnel) traffic), psf	100
LOCA Loading	
Maximum drag load on horizontal beam surfaces, lbs/ft^2	177
Maximum drag load on grating, lbs/ft^2	25.7
Maximum back pressure following LOCA, psi	0.28
Maximum drag load on AHU, lbs	1,250

Note:

(1) Margin and dynamic load factor are to be applied to tabulated values as appropriate.

TABLE 6.7-21

SUMMARY OF RESULTS
UPPER BLANKET DOOR STRUCTURAL ANALYSIS - LOCA

Item	Area	Code Allowable Stress / Max. Calculated Stress	Design[1] Basis
1	Skin and bands, direct tension	4.17	B
2	Hinge bar - bending	6.30	A
3	Anchor bolts - tension	6.50	C
4	Floor grating - bending	4.55	D
5	Insulation tip stress - tear - tensile	2.01 16.70	D

Notes:

(1) Key to Design Basis

A. Allowable value per AISC-69 limits
B. ASTM-177 minimum tensile with AISC allowable
C. ASTM-A325 minimum tensile with AISC allowable
D. Strength values per Manufacturer's literature

TABLE 6.7-22

DESIGN LOADS AND PARAMETERS
INTERMEDIATE DECK

Normal Operations

Ambient temperature before cooldown, maximum, °F	100
Ambient temperature, minimum, °F	15
Temperature differential across deck, estimated, °F	5

B. Dead Weight

Panel, lbs/ft^2, maximum	5.5
Static design equivalent of live load (personnel traffic), psf	100

C. Accident Conditions

Post-LOCA temperature (no ΔP applied), maximum, °F	190

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TABLE 6.7-23

SUMMARY OF WALTZ MILL TESTS

Compaction Tests

One foot diameter wire mesh baskets, loaded with flake ice to various heights, lead weights added to simulate additional height of ice.

Test	Started	Terminated	Length of Test (months)	Equivalent Height of Bed (feet)	Compaction (% Volume in First Year)
D'	2/21/69	8/28/70	18.0	22	24.5
E'	2/21/69	8/28/70	18.0	7.5	5.5

Shear Tests

One foot diameter wire mesh baskets, loaded with flake ice to various heights, temporarily supported between two wooden discs by pegs which are removed after one month.

Test	Started	Terminated	Length of Test (months)	Actual Height of Bed (feet)	Shear Rate[(1)] (inches/year)
G'	9/16/69	8/28/70	11.4	5	0.9
H'	9/16/69	8/28/70	11.4	3	0.9
I'	9/16/69	8/28/70	11.4	1	0.4

Notes:

(1) Shear rate approximated based on 6 months of data; not applicable for greater than 6 months.

TABLE 6.7-24 (Sheet 1 of 3)

ICE CONDENSER RTDs

Ice Bed RTDs:

TE No.	Bay No.	Radial Loc.	Elev. Above Wear Slab	Detail	TE No.	Bay No.	Radial Loc.	Elev. Above Wear Slab	Detail
180	24	3	55 ft 0"	(2)	153	10	2	55 ft 0"	(1)
181	24	3	30 ft 9"	(2)	154	10	2	30 ft 9"	(1)
182	24	3	0 ft 0"	(7)	155	10	2	10 ft 6"	(1)
177	21	2	55 ft 0"	(1)	150	7	1	55 ft 0"	(1)
178	21	2	30 ft 9"	(1)	151	7	1	30 ft 9"	(1)
179	21	2	10 ft 6"	(1)	152	7	1	10 ft 6"	(1)
174	18	1	55 ft 0"	(1)	147	7	2	55 ft 0"	(1)
175	18	1	30 ft 9"	(1)	148	7	2	30 ft 9"	(1)
176	18	1	10 ft 6"	(1)	149	7	2	10 ft 6"	(1)
171	18	2	55 ft 0"	(1)	144	7	3	55 ft 0"	(1)
172	18	2	30 ft 9"	(1)	145	7	3	30 ft 9"	(1)
173	18	2	10 ft 6"	(1)	146	7	3	10 ft 6"	(1)
168	18	3	55 ft 0"	(1)	141	4	2	55 ft 0"	(1)
169	18	3	30 ft 9"	(1)	142	4	2	30 ft 9"	(1)
170	18	3	10 ft 6"	(1)	143	4	2	10 ft 6"	(1)
165	15	2	55 ft 0"	(1)	138	1	3	55 ft 0"	(2)
166	15	2	30 ft 9"	(1)	139	1	3	30 ft 9"	(2)
167	15	2	10 ft 6"	(1)	140	1	3	0 ft 0"	(7)
159	13	1	55 ft 0"	(6)	162	Az107E	N/A	58 ft 6"	(3)
160	13	1	30 ft 9"	(6)	163	Az107E	N/A	58 ft 6"	(4)
161	13	1	10 ft 6"	(6)					
156	13	2	55 ft 0"	(1)					
157	13	2	30 ft 9"	(1)					
158	13	2	0 ft 0"	(7)					
183	13	3	55 ft 0"	(2)					
184	13	3	30 ft 9"	(2)					
185	13	3	10 ft 6"	(2)					

TABLE 6.7-24 (Sheet 2 of 3)

ICE CONDENSER RTDs

Floor Cooling RTDs:

TE No.	Bay No.	Radial Loc.	Elev. Above Wear Slab	Detail	TE No.	Bay No.	Radial Loc.	Elev. Above Wear Slab	Detail
129A	1	2	0 ft 6" (typ)	9 (typ)	129N	13	2	0 ft 6" (typ)	9 (typ)
129B	2	2	0 ft 6" (typ)	9 (typ)	129P	14	2	0 ft 6" (typ)	9 (typ)
129D	3	2	0 ft 6" (typ)	9 (typ)	129Q	15	2	0 ft 6" (typ)	9 (typ)
129E	4	2	0 ft 6" (typ)	9 (typ)	129R	16	2	0 ft 6" (typ)	9 (typ)
129F	5	2	0 ft 6" (typ)	9 (typ)	129S	17	2	0 ft 6" (typ)	9 (typ)
129G	6	2	0 ft 6" (typ)	9 (typ)	129T	18	2	0 ft 6" (typ)"	9 (typ)
129H	7	2	0 ft 6" (typ)	9 (typ)	129U	19	2	0 ft 6" (typ)	9 (typ)
129I	8	2	0 ft 6" (typ)	9 (typ)	129V	20	2	0 ft 6" (typ)	9 (typ)
129J	9	2	0 ft 6" (typ)	9 (typ)	129 W	21	2	0 ft 6" (typ)	9 (typ)
129K	10	2	0 ft 6" (typ)	9 (typ)	129X	22	2	0 ft 6" (typ)	9 (typ)
129L	11	2	0 ft 6" (typ)	9 (typ)	129Y	23	2	0 ft 6" (typ)	9 (typ)
129M	12	2	0 ft 6" (typ)	9 (typ)	129Z	24	2	0 ft 6" (typ)	9 (typ)

TABLE 6.7-24 (Sheet 3 of 3)

ICE CONDENSER RTDs

	TE No.	Bay No.	Radial Loc.	Elev. Above Wear Slab	Detail
Temperature Switches:	131A	1	2	57 ft 0"	(T)
	131B	4	2	57 ft 0"	(T)
UNIT 1	131D	8	2	57 ft 0"	(T)
	131E	18	2	57 ft 0"	(T)
UNIT 2	131D	7	2	57 ft 0”	(T)
	131F	21	2	57 ft 0"	(T)
	131G	24	2	57 ft 0"	(T)
Wall Panel RTDs:	132A	1		10 ft 6"	(8)
	132B	8		10 ft 6"	(8)
	132D	8		1 ft 0"	(8)
	132E	13		10 ft 6"	(8)
	132F	13		1 ft 0"	(8)
	132G	18		10 ft 6"	(8)
	132H	18		1 ft 0"	(8)
	132I	24		10 ft 6"	(8)
Wear Slab RTDs:	210A	1	1		(7)
	210E	17	1		(7)
	210F	17	1		(7)
	210B	8	1		(7)
	210D	8	1		(7)
	210G	24	1		(7)

Detail No.:

(1) (2) (6) - Lattice-frame mtd. ice bed temp. RTD
(3) (4) - Plenum-panel mtd. RTD
(7) - Wear slab (floor) mtd. RTD
(8) - Wall panel mtd. RTD
(9) - Glycol Return Piping from Bay Floor Mounted.
(T) - Temperature switch

TABLE 6.7-25

ICE CONDENSER ALLOWABLE LIMITS [1]

		Elastic Analysis			
Load Combination	Mechanical [2]	Mechanical and Thermal	Limit Analysis [3] Fatigue	Test (Load Factors)	(Load Factors)
D + OBE	S[4]	3S	AISC Part 1	1.43	1.87
D + DBA	1.33 S	N/A	N/A	1.3	1.43
D + SSE	1.33 S	N/A	N/A	1.3	1.43
D + SSE ± DBA	1.65 S	N/A	N/A	1.18	1.3

Notes:

(1) For particular components that do not meet these limits, specific justification is provided on a case-by-case basis.

(2) Membrane (direct) stresses are $\leq$ 0.7 S_u (70% of ultimate stress).

(3) For mechanical loads only. Mechanical plus thermal expansion, combination, and fatigue satisfy the elastic analysis limits.

(4) S = Allowable stresses as defined in Sections 1.5 and 1.6 of the AISC Part 1 Specification.

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Table 6.7-26

SELECTION OF STRUCTURAL STEELS IN RELATION TO PREVENTION OF NON-DUCTILE FRACTURE OF ICE CONDENSER COMPONENTS

Properties	Section Thickness: 5/8-inch thick and under	Section Thickness: Over 5/8-inch thick
Energy Absorption Level[2]	None required	i) 20 ft-lb CVN at -20°F for steel over 36,000 psi yield strength ii) 15 ft-lb CVN at -20°F for steel under 36,000 psi yield strength
Heat Treatment[1]	None required Steel can be used in the hot rolled condition	i) Normalizing ii) Quench and temper
Type of Steel	i) Rimmed[3] ii) Semi-Killed[4] iii) Killed[4],[5] iv) Killed - fine grain practice	i) Killed ii) Killed - fine grain practice

Notes:
(1) Hot rolled, normalized, or quenched and tempered steels are used where applicable.
(2) Charpy-V Notch (CVN) impact testing is performed in accordance with the requirements of ASTM-A370.
(3) Rimmed steel is used only for carbon steel sheet products.
(4) These types of steel are applied for components which remain within AISC Code stress limits for all load conditions.
(5) Killed steels for above AISC Code stress limits are upgraded by heat treatment, e.g., bolting.

Table 6.7-27

SUMMARY OF WATTS BAR LOADS - TANGENTIAL CASE
OBTAINED USING THE TWO-MASS DYNAMIC MODEL

Earthquake Condition and Direction	Wall Panel Load - kips	Impact Load - lbs	Design Values Wall Panel Load - kips	Design Values Impact Load - lbs
OBE, N-S	4.4	163	9.8	1165
OBE, E-W	2.1	81	9.8	1165
SSE, N-S	10.15	430	11.3	1400
SSE, E-W	7.5	298	11.3	1400

Table 6.7-28

SUMMARY OF WATTS BAR LOADS - RADIAL CASE
OBTAINED USING THE TWO-MASS DYNAMIC MODEL

			Design Values	
Earthquake Condition and Direction	Wall Panel Load - kips	Impact Load - lbs	Wall Panel Load - kips	Impact Load - lbs
OBE, N-S	4.2	145	13.5	1165
OBE, E-W	3.1	106	13.5	1165
SSE, N-S	11.9	474	15.5	1400
SSE, E-W	7.2	252	15.5	1400

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TABLE 6.7-29

SUMMARY OF LOAD RESULTS OF FIVE NON-LINEAR DYNAMIC MODELS

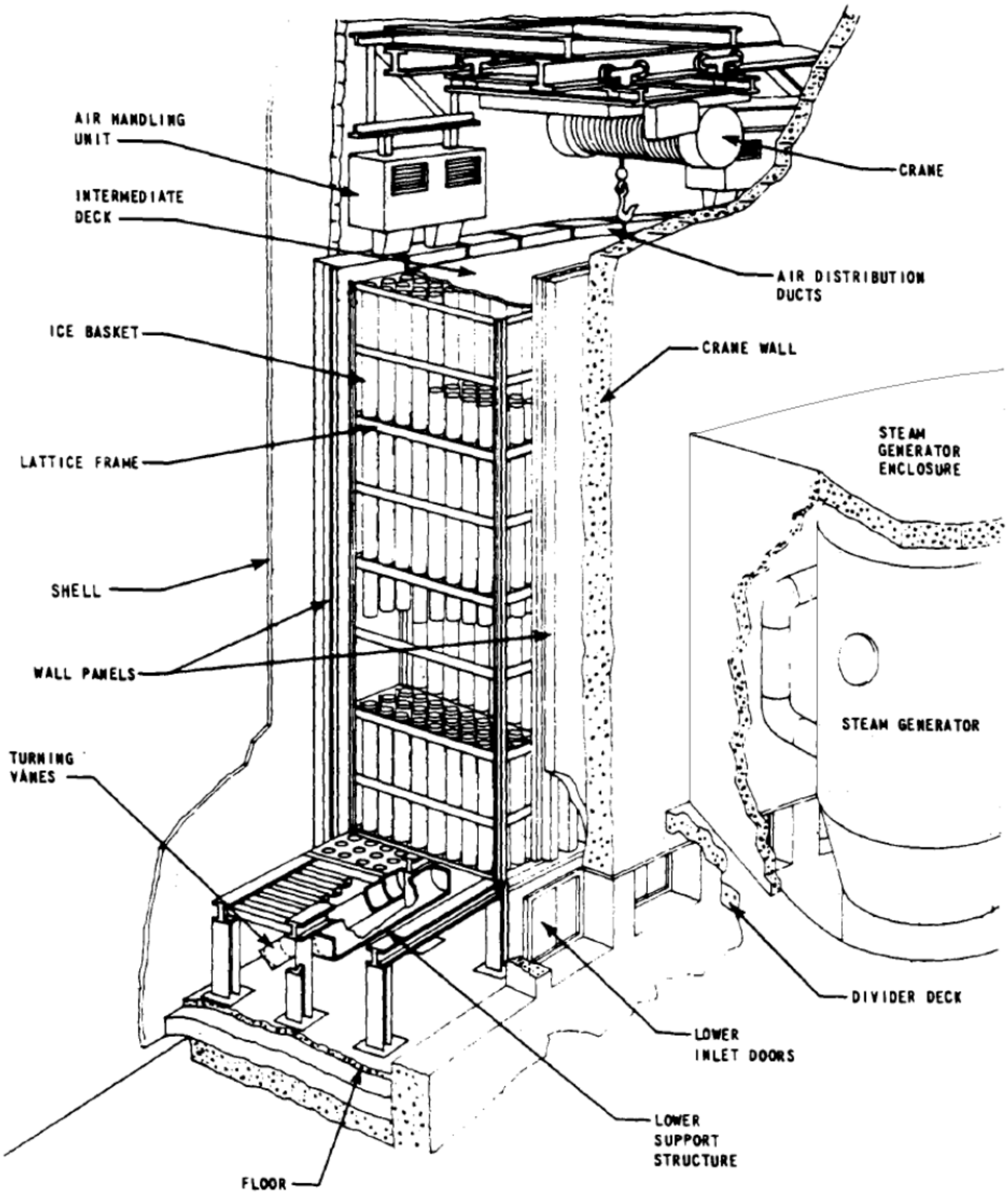
Maximum Load Average of 4 Earthquakes	2-Mass Model	3-Mass Model	9-Mass Model	48-Foot Beam Model	Phasing Mass Model	Design Load
Tangential Impact Load	430	787		609	521	1,400
Tangential Wall Panel Load	10,150			10,760	8,121	11,300
Radial Impact Load	474		1,106	605		1,400
Radial Wall Panel Load	11,889			14,588		15,500
Link Impact Load					8,216	12,000

Table 6.7-30

SUMMARY OF PARAMETERS
USED IN THE SEISMIC ANALYSIS

Item	Description	Watts Bar Parameters
1.	Lower Support Stiffness	
	a. Radial Direction	670,000 lbs/in
	b. Tangential Direction	319,000 lbs/in
2.	Lattice Frame Wall Panels Combined Stiffness	
	a. Radial Direction	20,900 to 40,410 lbs/in
	b. Tangential Direction	23,910 lbs/in
3.	Local Impact Stiffness	
	a. Radial Direction	127 kip/in
	b. Tangential Direction	130 kip/in
4.	Ice Basket Weight with Ice	43.5 lbs/ft*
5.	Gap Size	0.5 in
6.	Ice Basket Stiffness	
	a. Bending Rigidity (EI), where: E = modulus of elasticity I = moment of inertia	330×10^{6} lbs-in^{2}

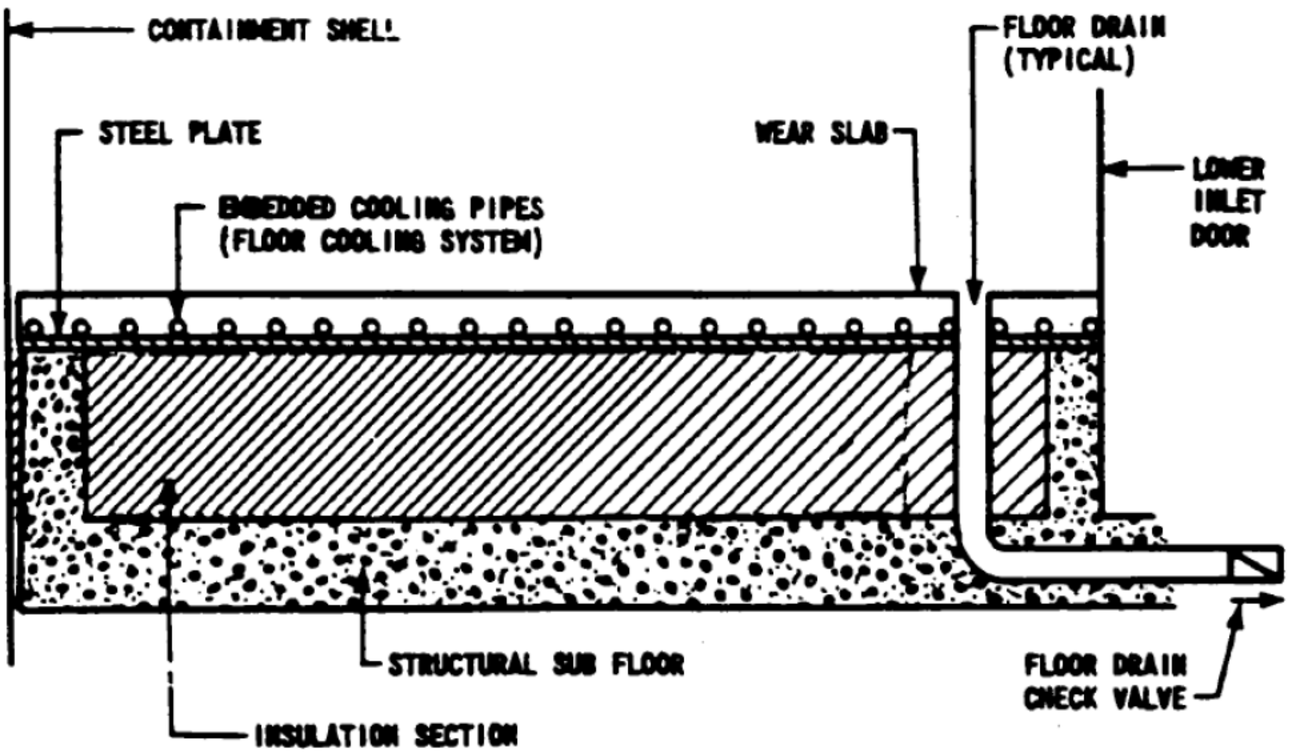
* Westinghouse design basis for the Watts Bar Nuclear Plant is 2090 lbs for the maximum individual ice basket weight (gross weight of ice and baskets) (includes 19 lbs for the concentrated mass at the lower support structure attachment) and 1809, 1909, 2009 lbs for the maximum average weight of the ice baskets, respectively for the inner third (closest to the crane wall), the middle third, and outer third (closest to the containment wall) (includes 19 lbs for the concentrated mass at the lower support structure attachment) 3x3 lattice frame array of baskets. (Reference: WAT-D-10850).



WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

Isometric of
Ice Condenser

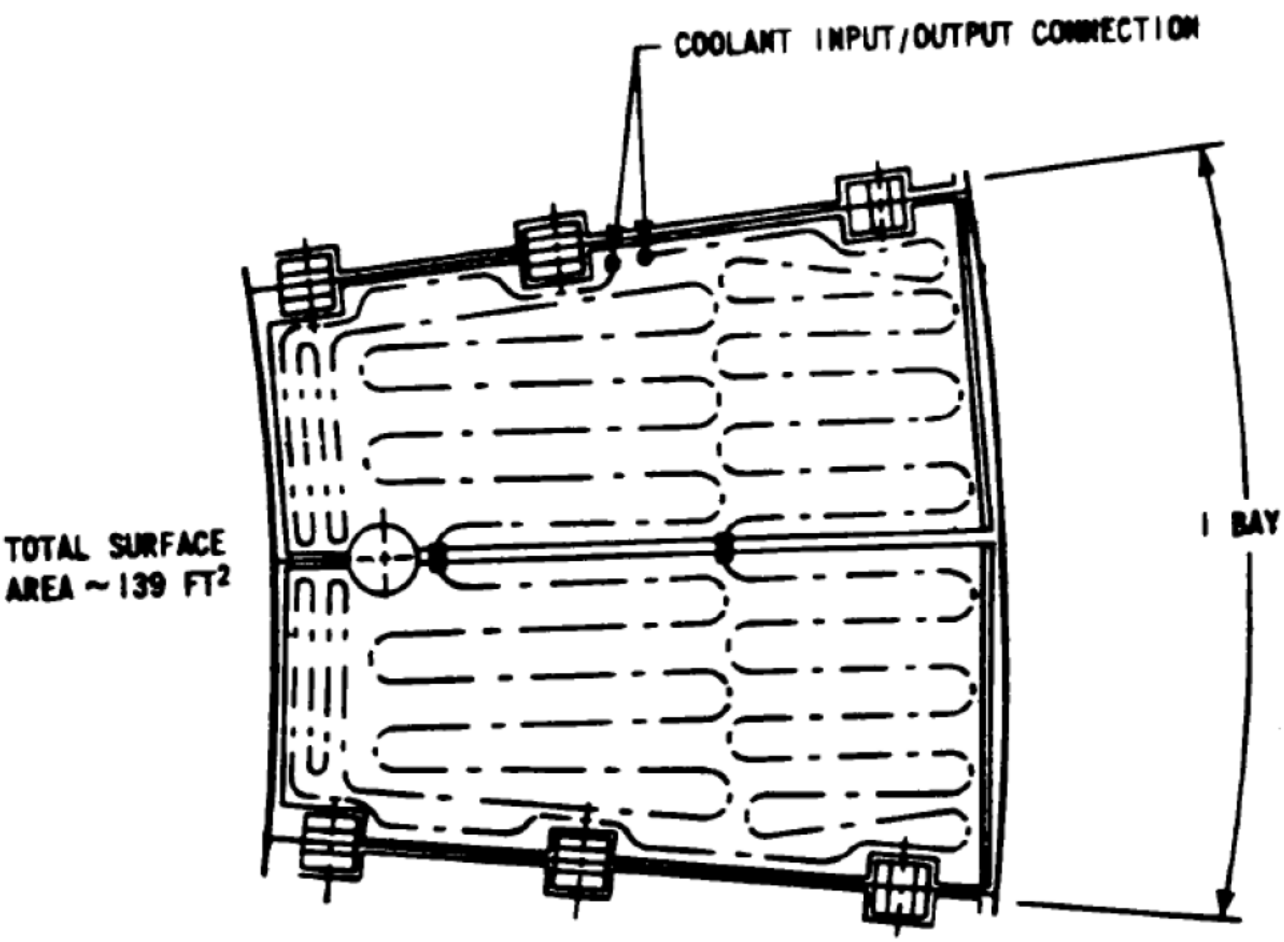
Figure 6.7-1



WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

Floor Structure

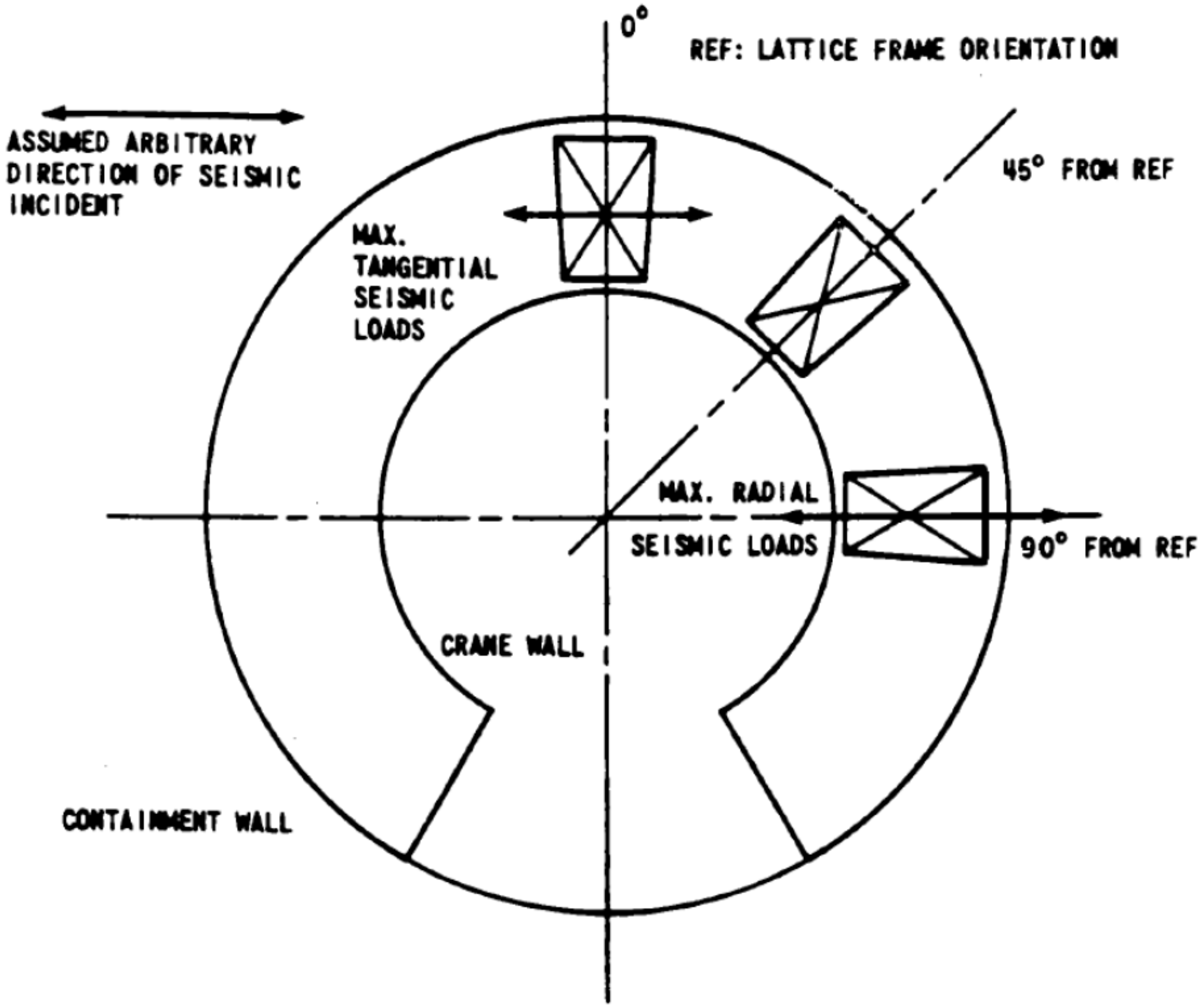
Figure 6.7-2



WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

Wear Slab Top Surface Area Showing Typical Coolant Piping Layout

Figure 6.7-3



NOTES:

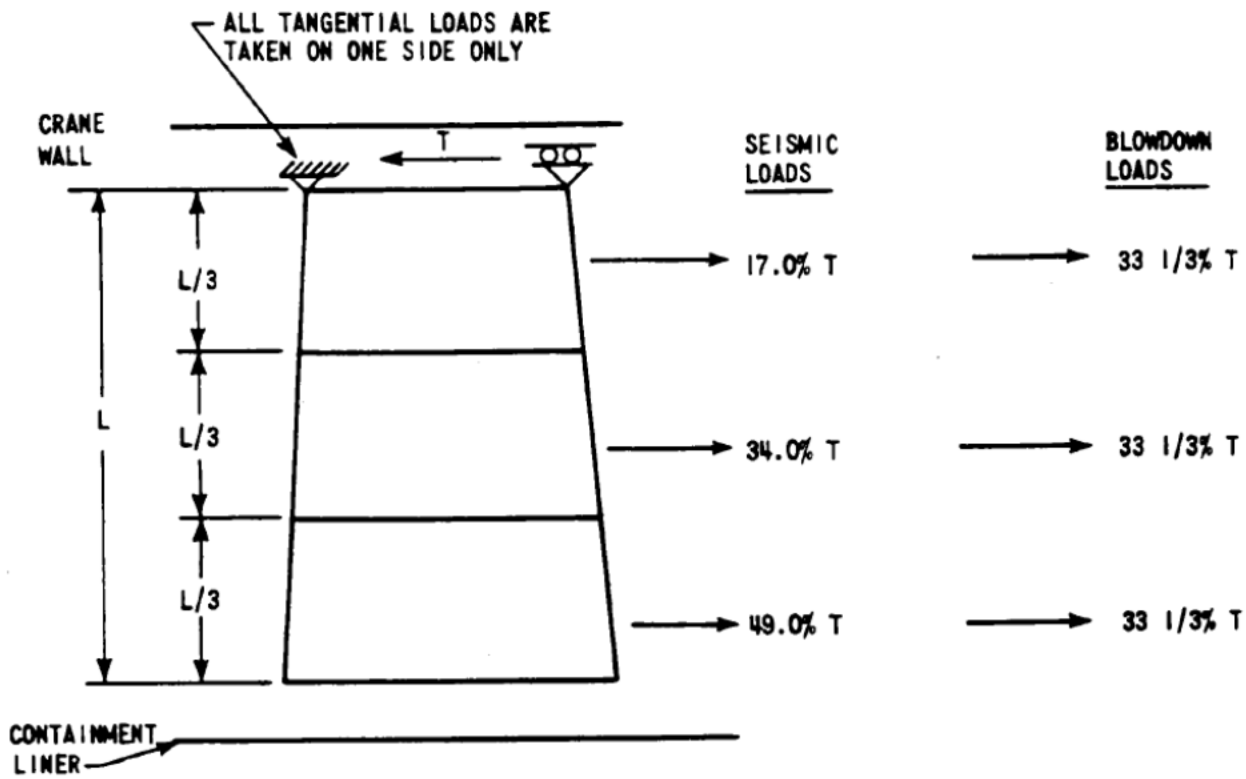
1. MAXIMUM TANGENTIAL AND RADIAL SEISMIC LOADS CANNOT OCCUR SIMULTANEOUSLY.
2. TANGENTIAL AND RADIAL SEISMIC LOADS 45 DEGREES FROM THE REFERENCE DIRECTION OF SEISMIC INPUT OCCUR SIMULTANEOUSLY AND THE MAGNITUDE IS THE AVERAGE OF MAXIMUM RADIAL AND MAXIMUM TANGENTIAL TIMES THE COSINE OF 45°, OR $\left(\frac{\text{RADIAL} + \text{TANGENTIAL}}{2}\right) .707$.
3. HORIZONTAL AND VERTICAL SEISMIC LOADS CAN OCCUR HORIZONTALLY.
4. BLOWDOWN LOADS, TANGENTIAL, RADIAL AND VERTICAL CAN OCCUR SIMULTANEOUSLY. RADIAL BLOWDOWN LOADS ALWAYS OCCUR IN THE DIRECTION OF THE CONTAINMENT WALL.

* In an individual lattice frame.

WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

Lattice Frame Orientation

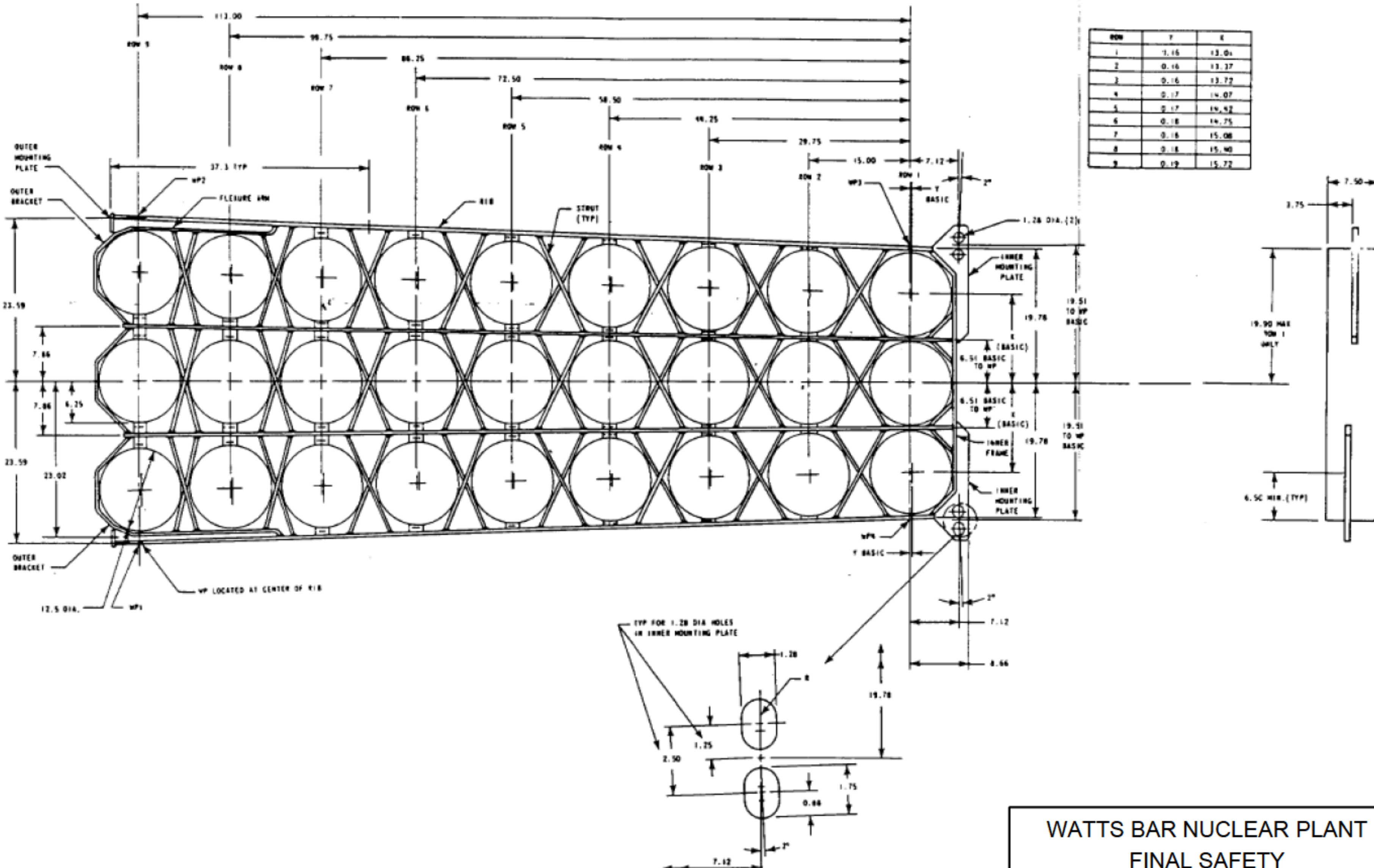
Figure 6.7-4



WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

Load Distribution for
Tangential Seismic and
Blowdown Loads
in Analytical Model

Figure 6.7-5

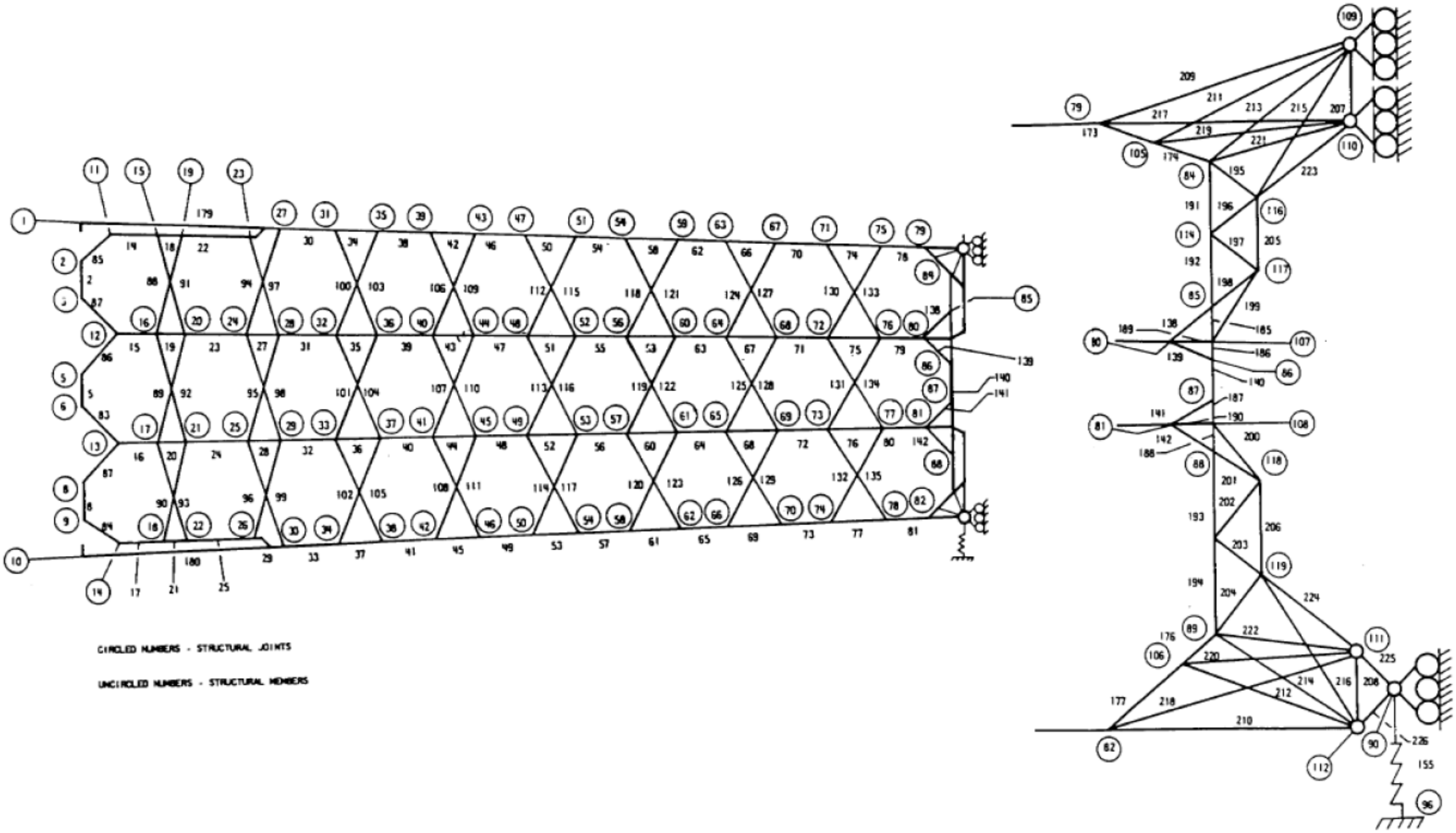


ROW	Y	X
1	1.16	13.0[illegible]
2	0.16	13.37
3	0.16	13.72
4	0.17	14.07
5	0.17	14.42
6	0.18	14.75
7	0.18	15.08
8	0.18	15.40
9	0.19	15.72

WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

Lattice Frame

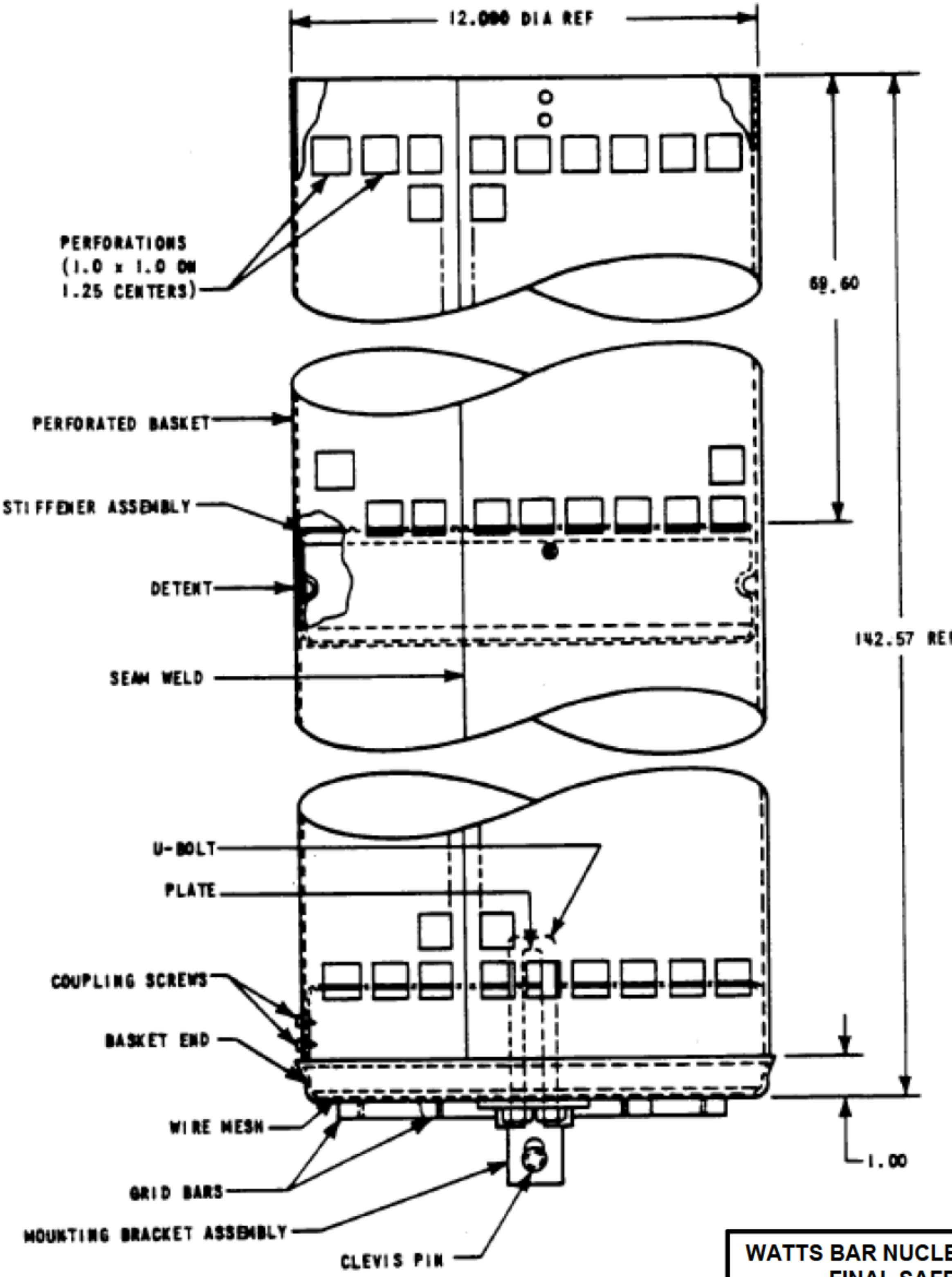
FIGURE 6.7-6



WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

Lattice Frame
Analysis Model

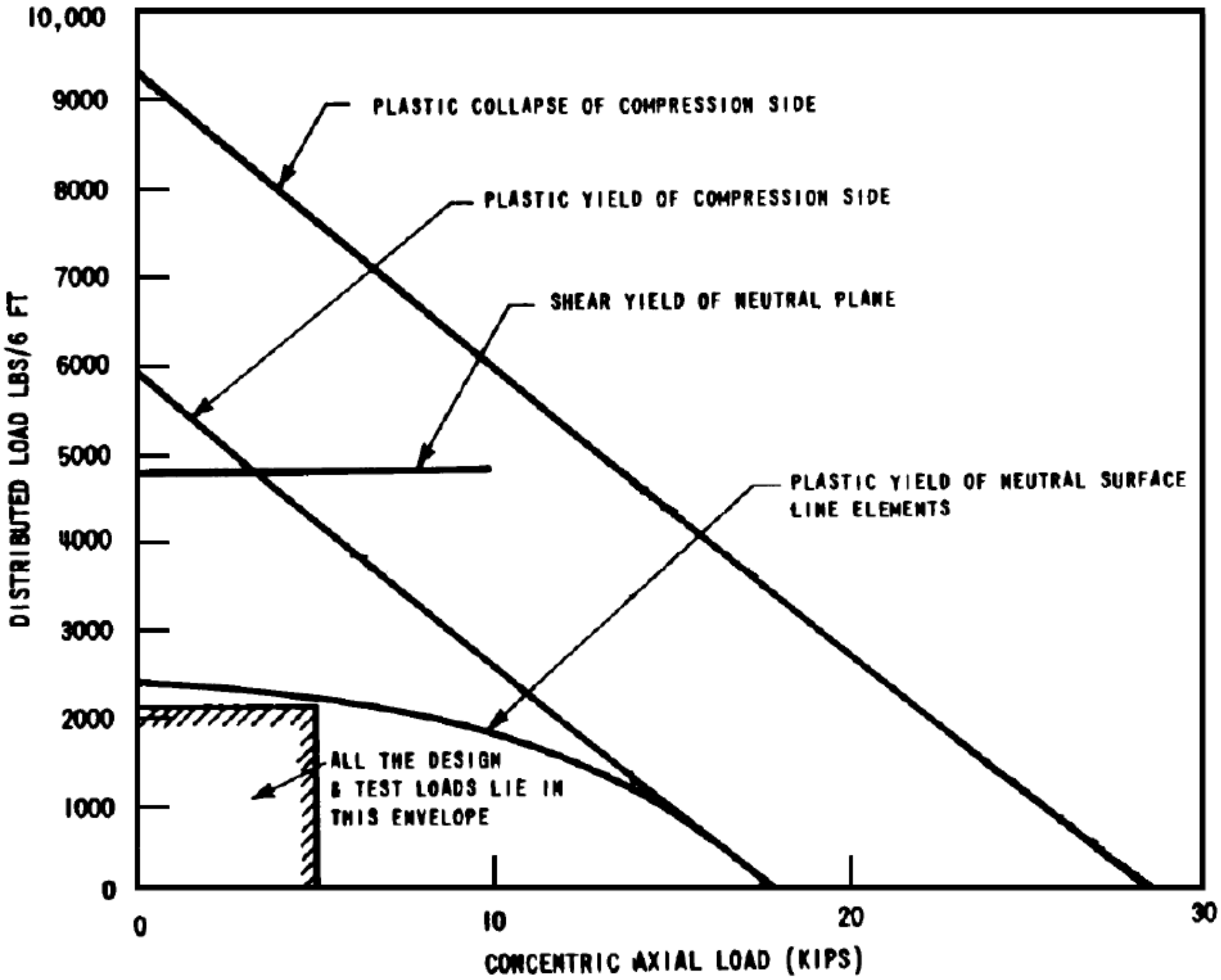
FIGURE 6.7-7



WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

Typical Bottom
Ice Basket Assembly

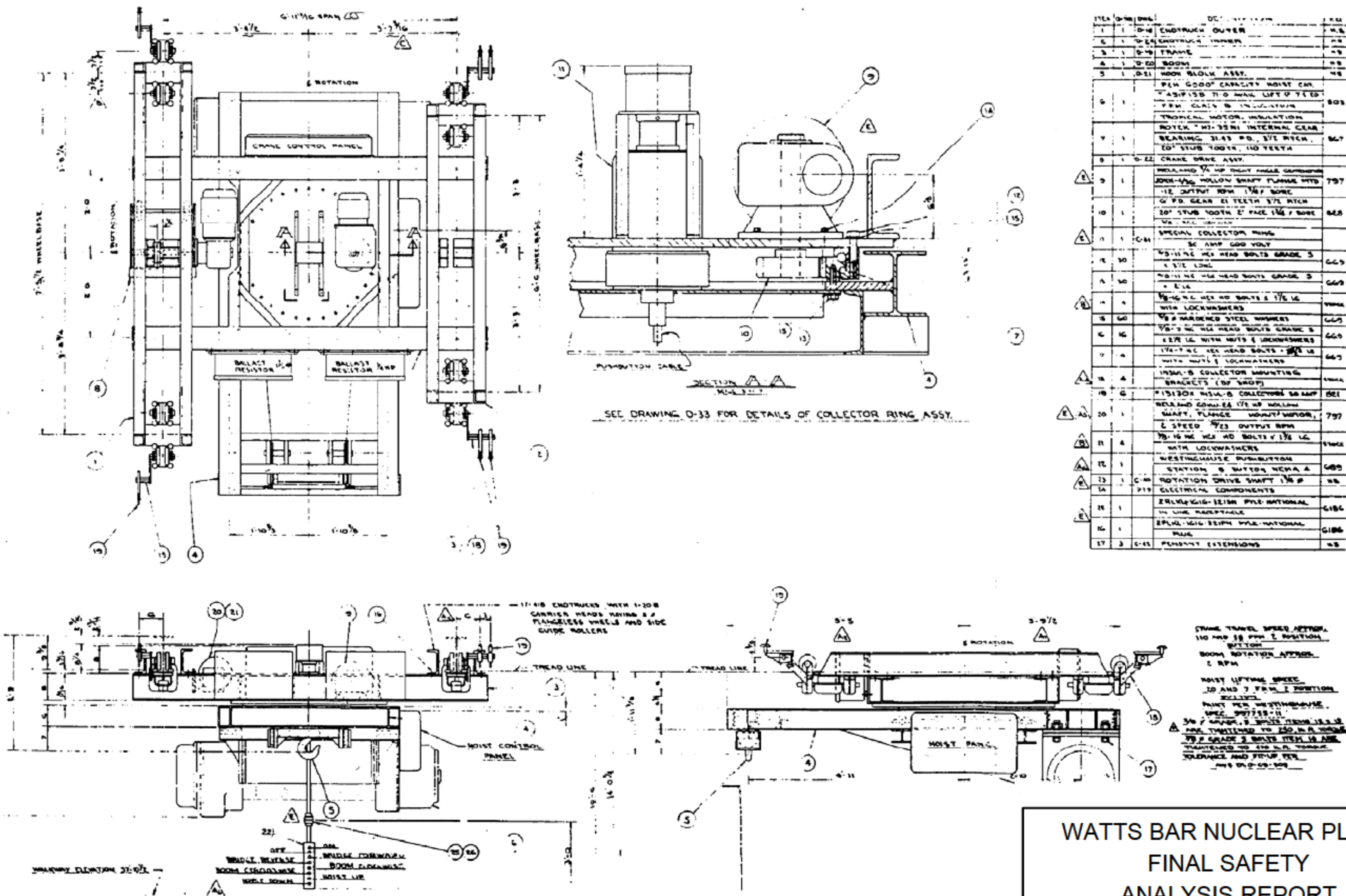
Figure 6.7-8



WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

Combinations of Concentric Axial Load and distribution Load That Will Cause Failure of a Perforated Metal Ice Condenser Basket Material

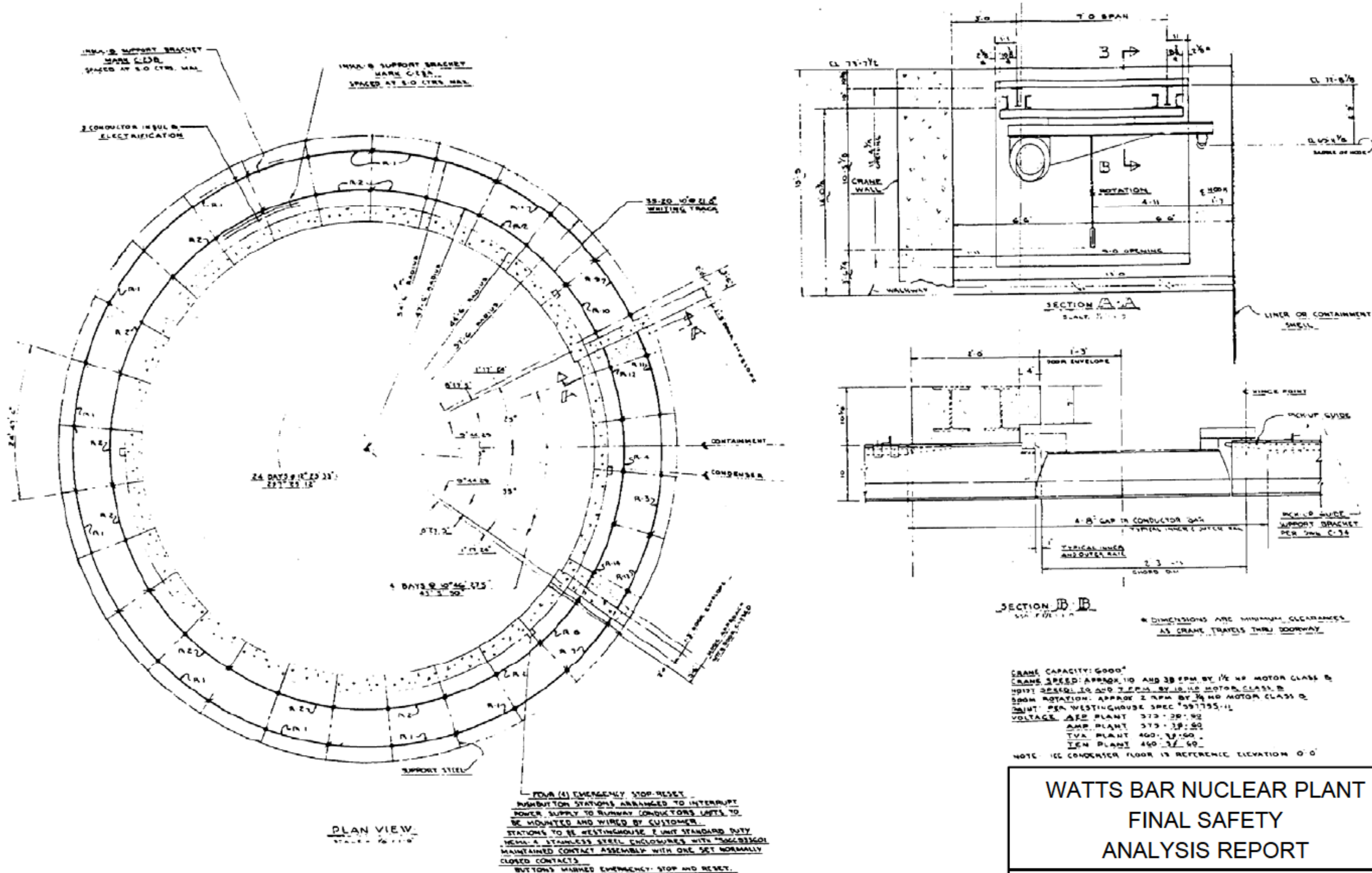
Figure 6.7-9



WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

Crane
Assembly

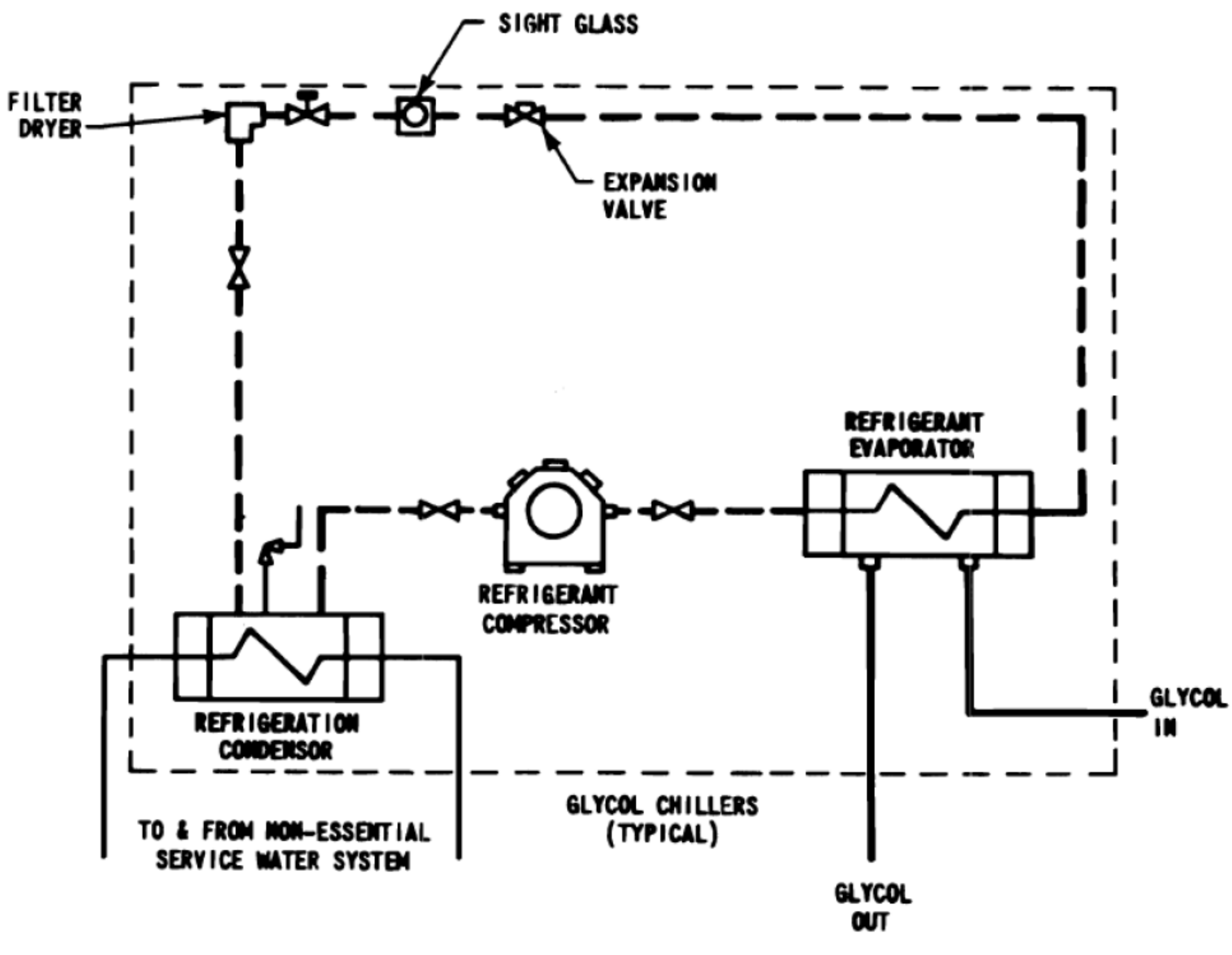
FIGURE 6.7-10



WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

Crane Rail
Assembly

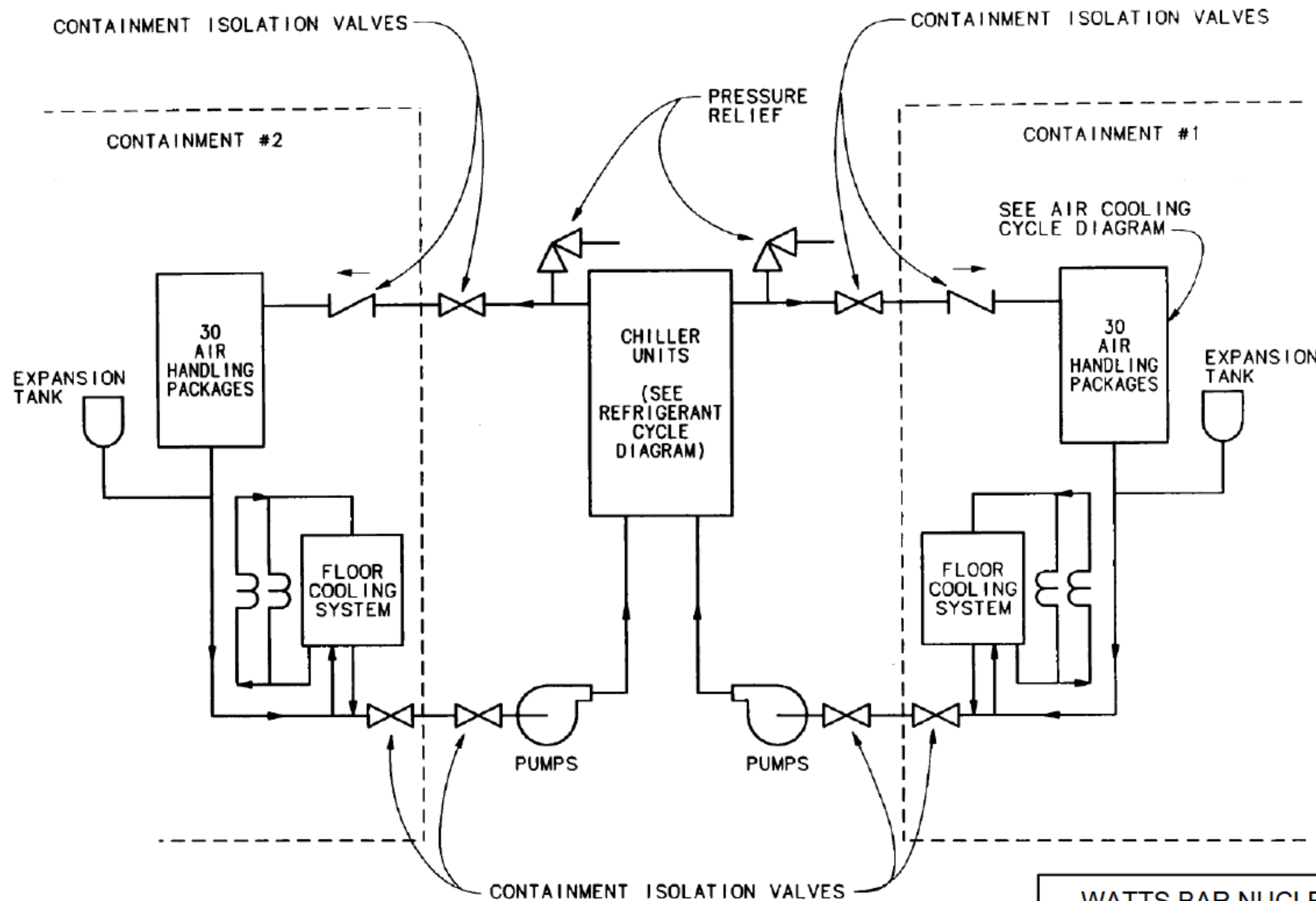
FIGURE 6.7-11



WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

Unit 2
Refrigerant Cycle
Diagram

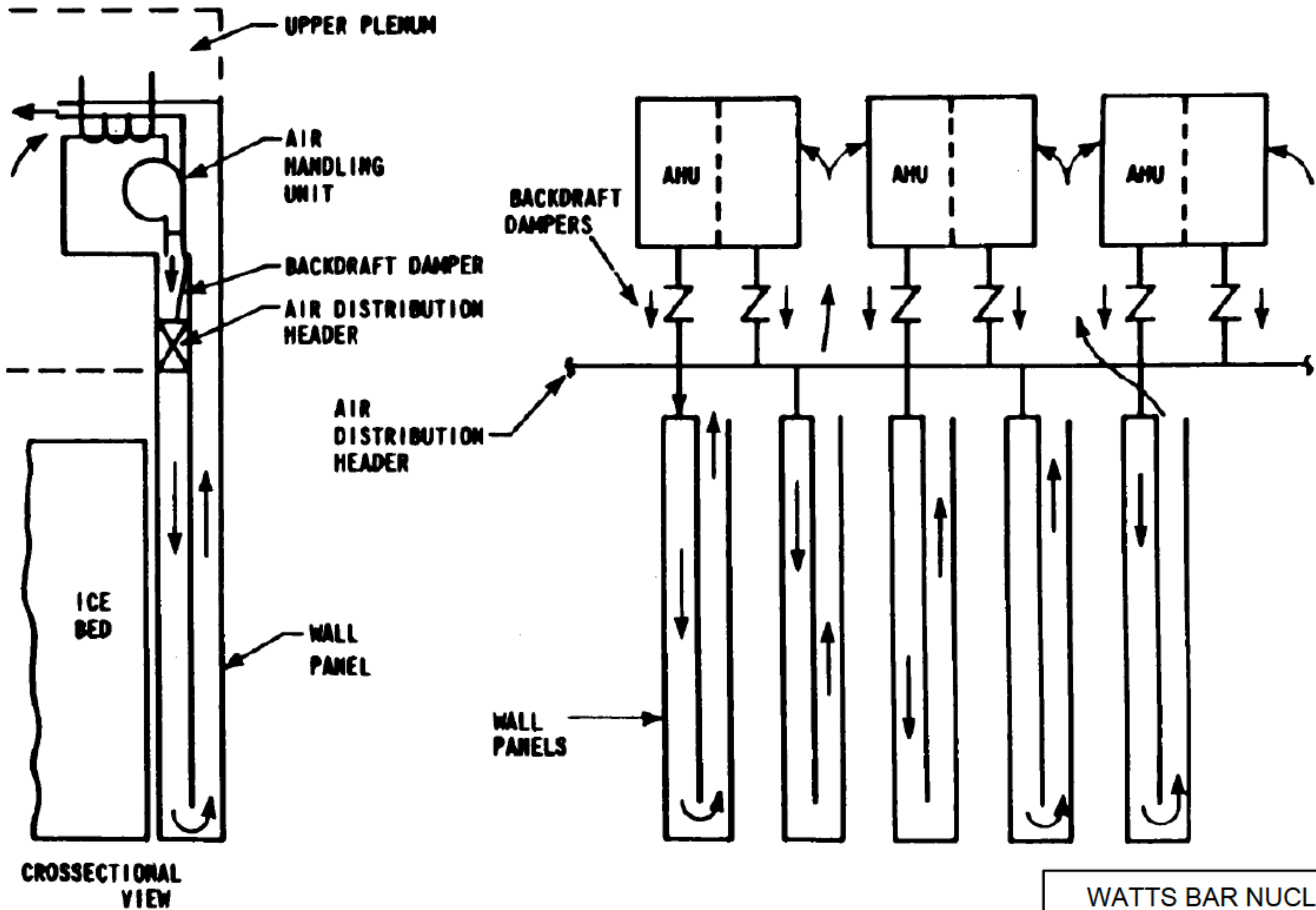
Figure 6.7-12



WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

Glycol Cycle to
Each Containment

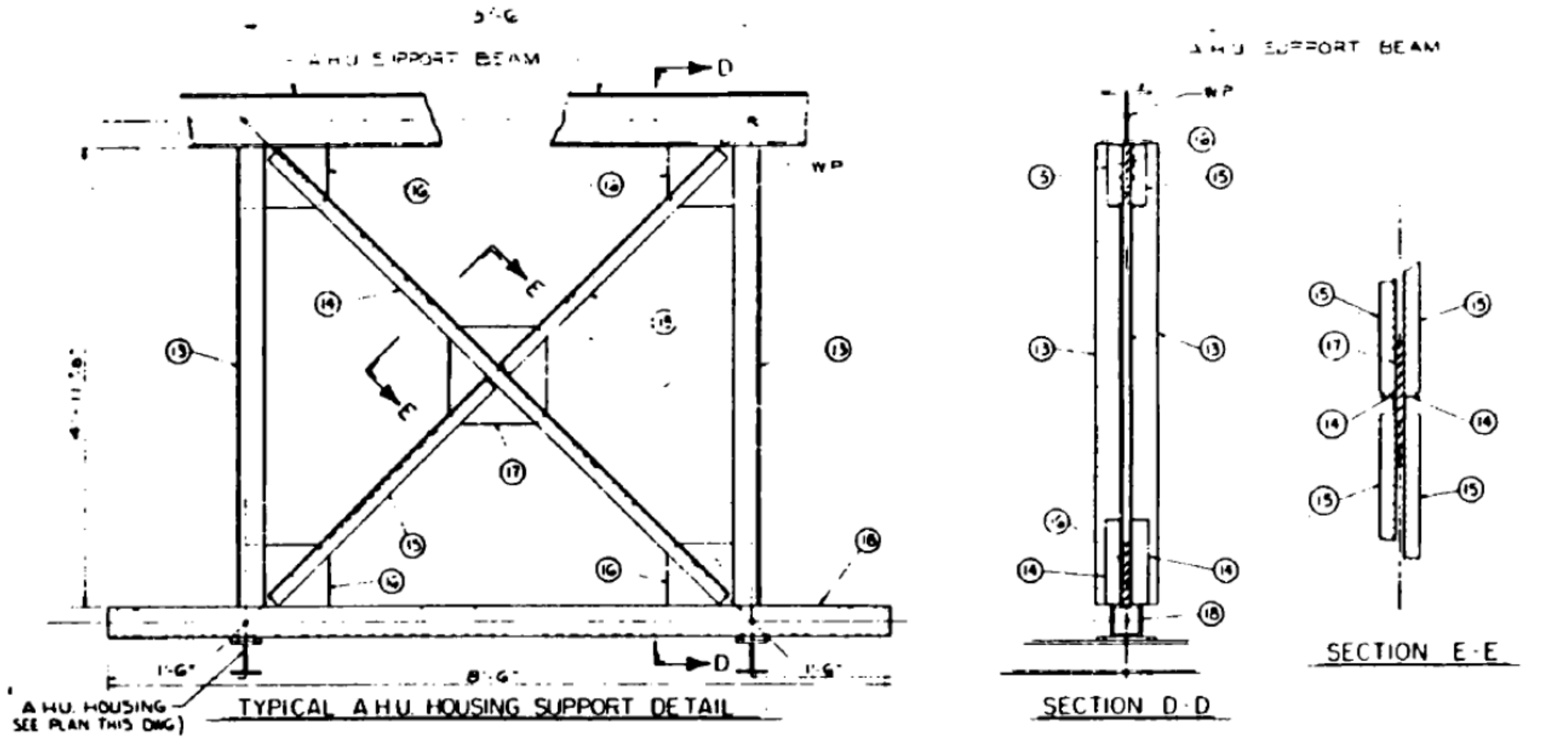
FIGURE 6.7-13



WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

Schematic Flow Diagrams of
Air Cooling Cycle

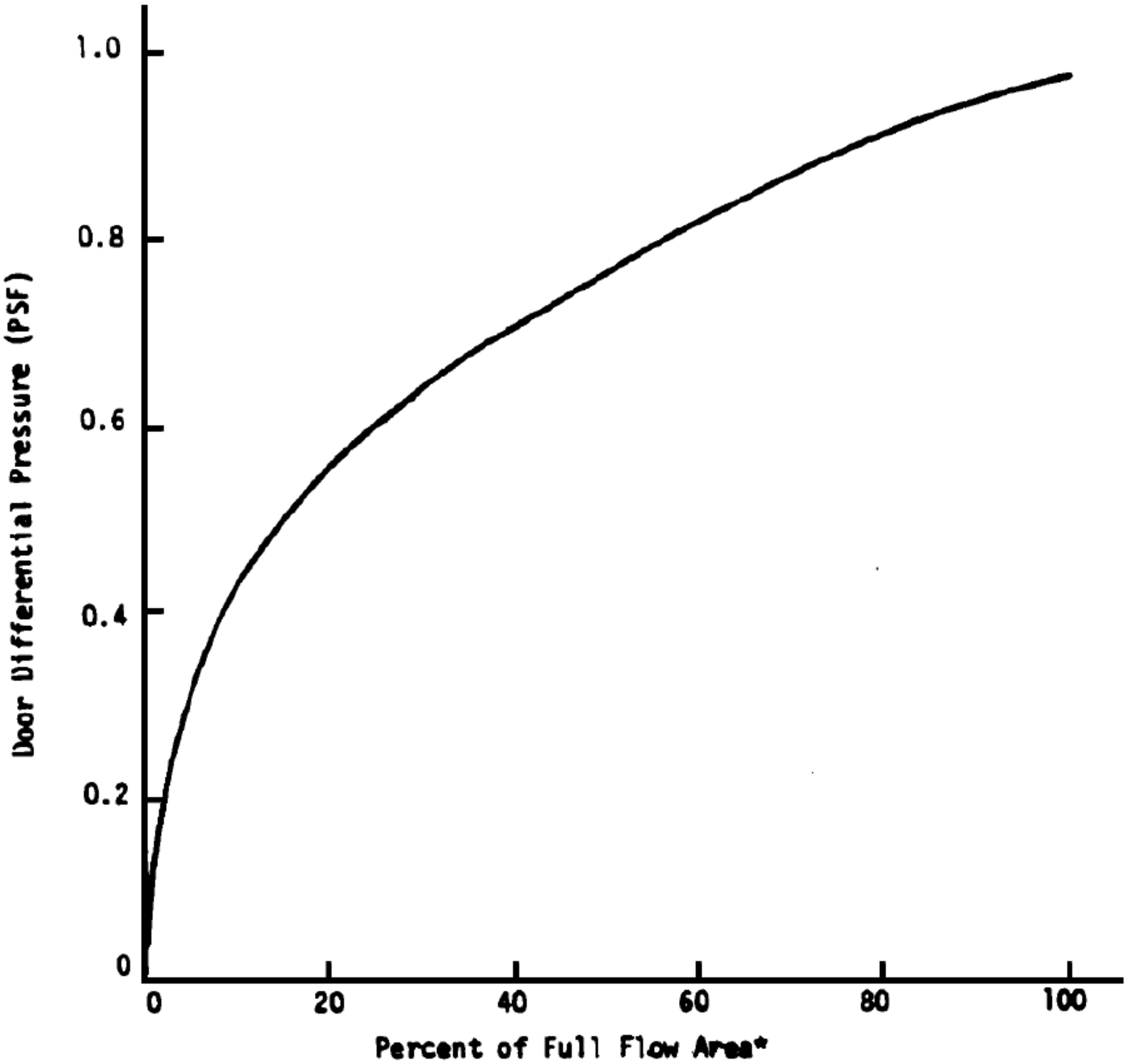
FIGURE 6.7-14



WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

Air Handling Unit
Support Structure

FIGURE 6.7-15

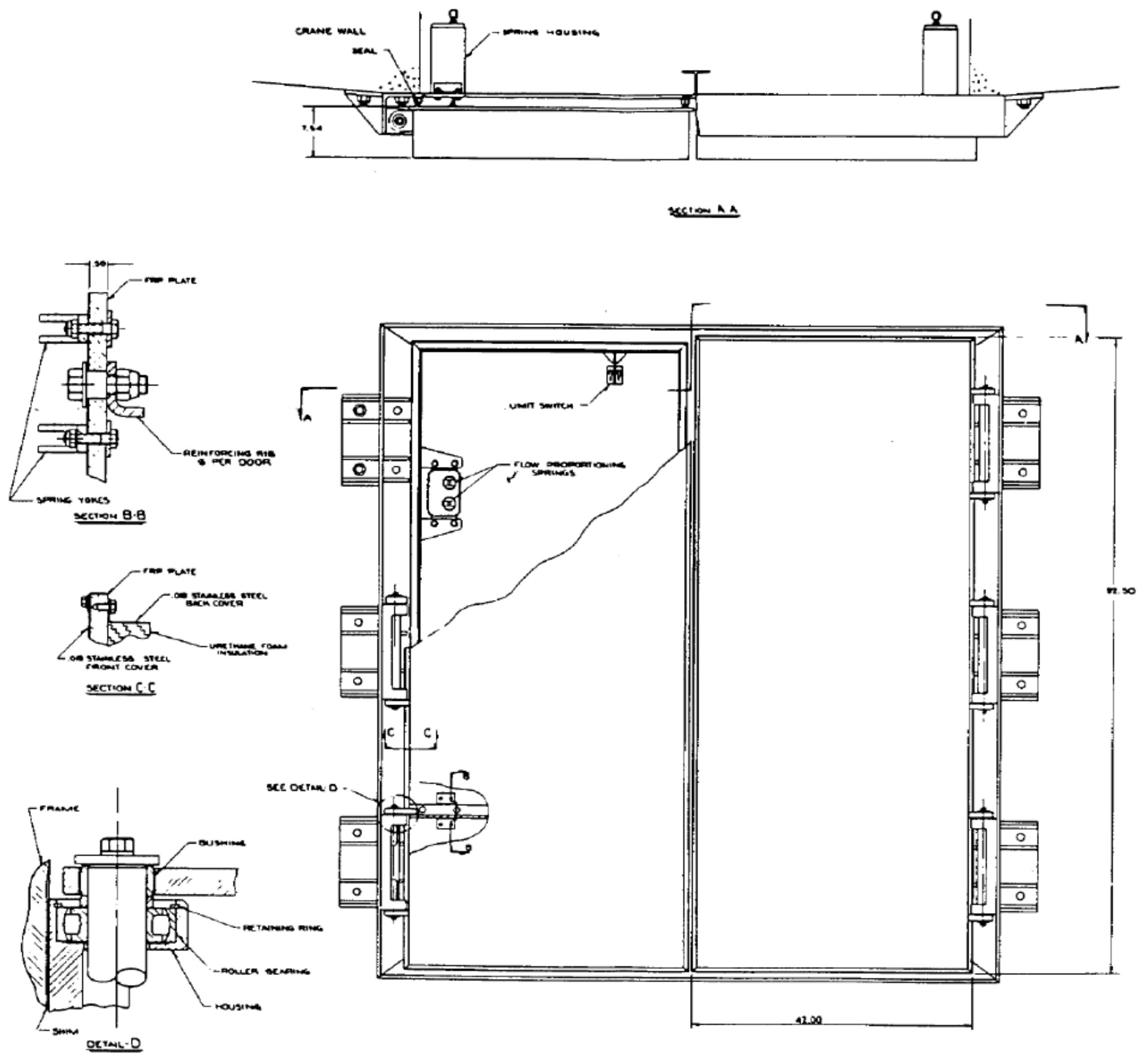


*Full Flow Area is defined as the minimum door port area with doors fully open

WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

Flow Area – Pressure Differential

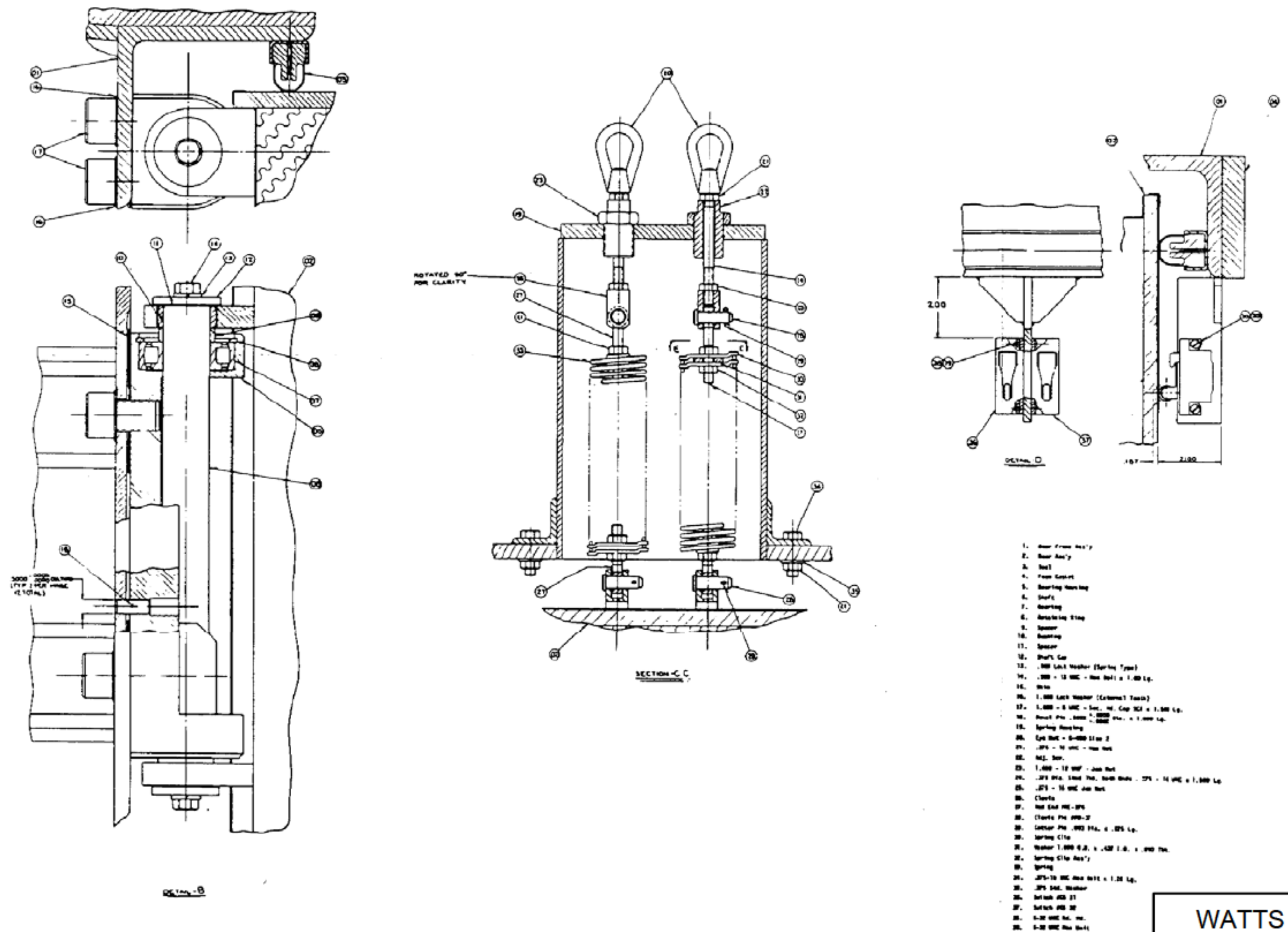
Figure 6.7-16



WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

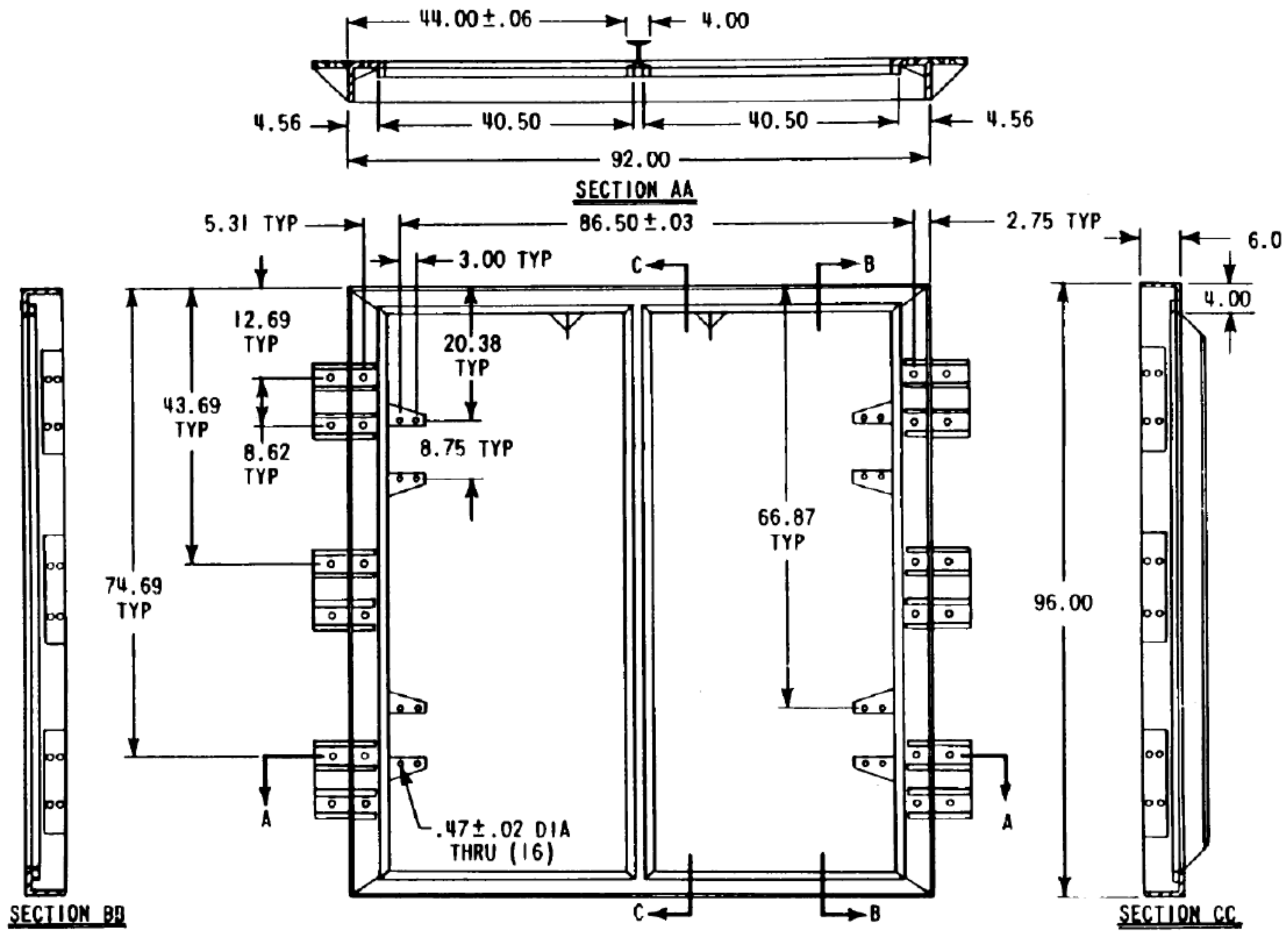
Lower Inlet
Door Assembly

FIGURE 6.7-17



WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

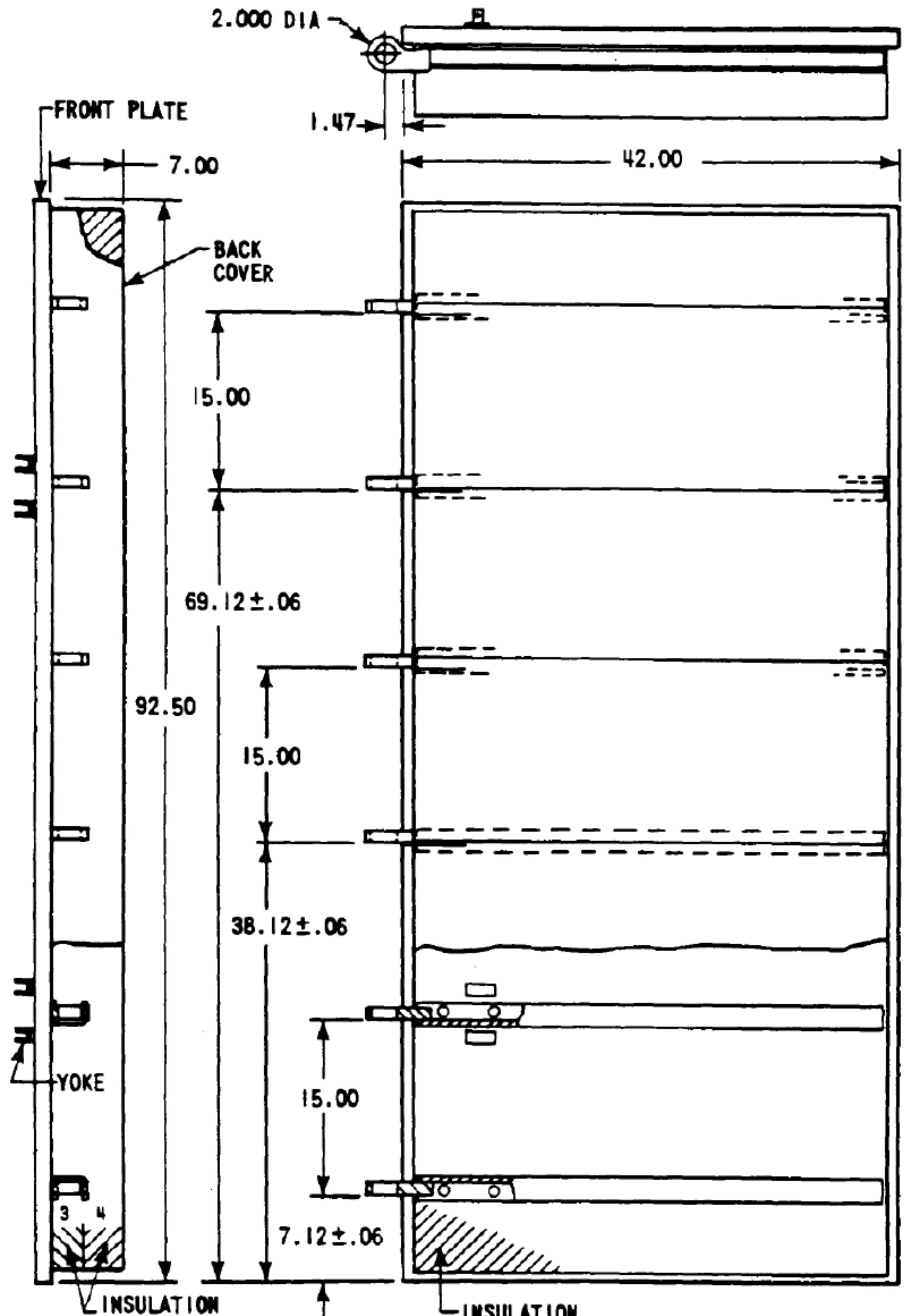
Details of Lower Inlet Door Showing Hinge, Proportioning Mechanism Limit Switches and Seals
FIGURE 6.7-18



WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

Inlet Door Frame Assembly

FIGURE 6.7-19



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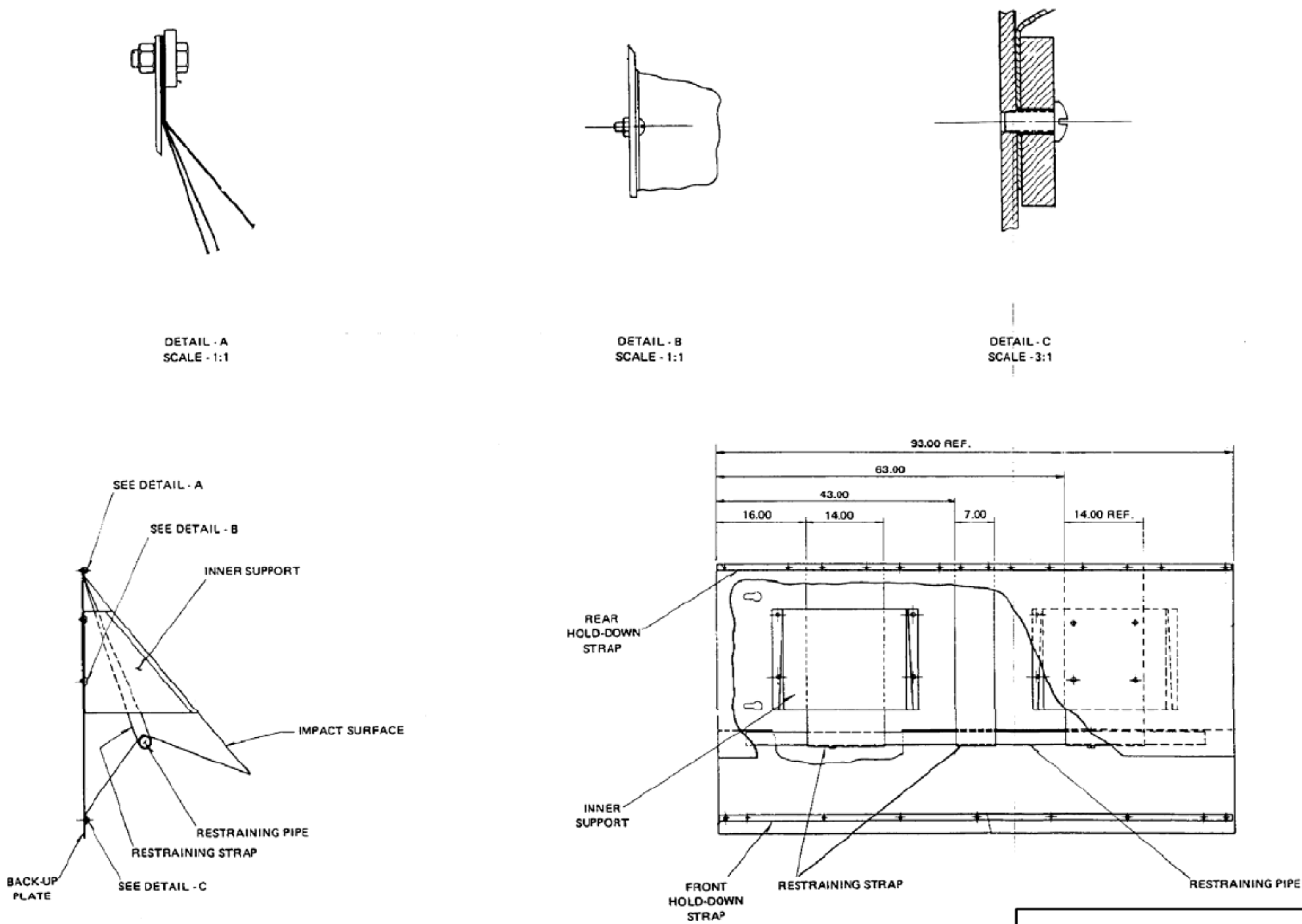
ANALYSIS REPORT

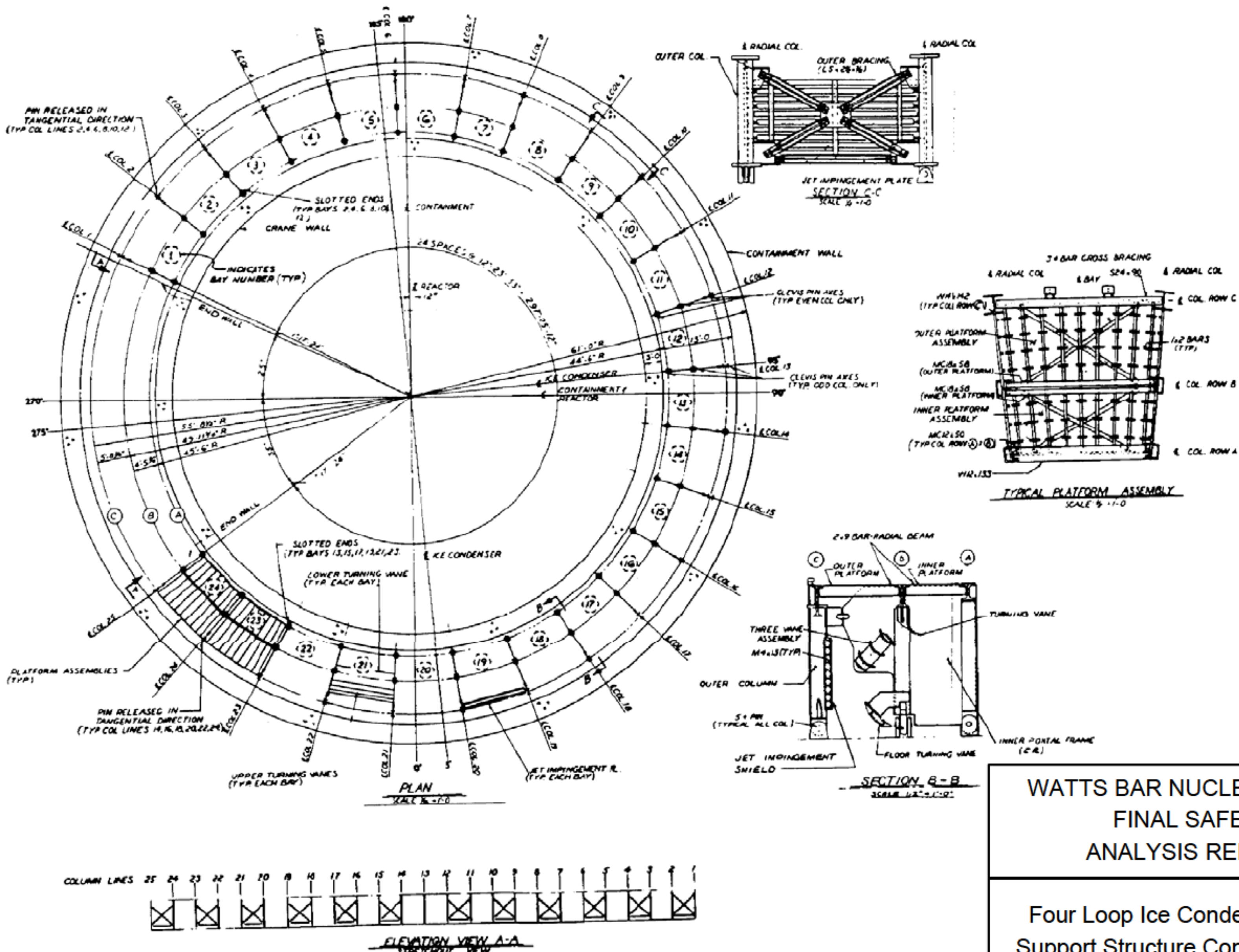
Inlet Door Panel Assembly

Figure 6.7-20



WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

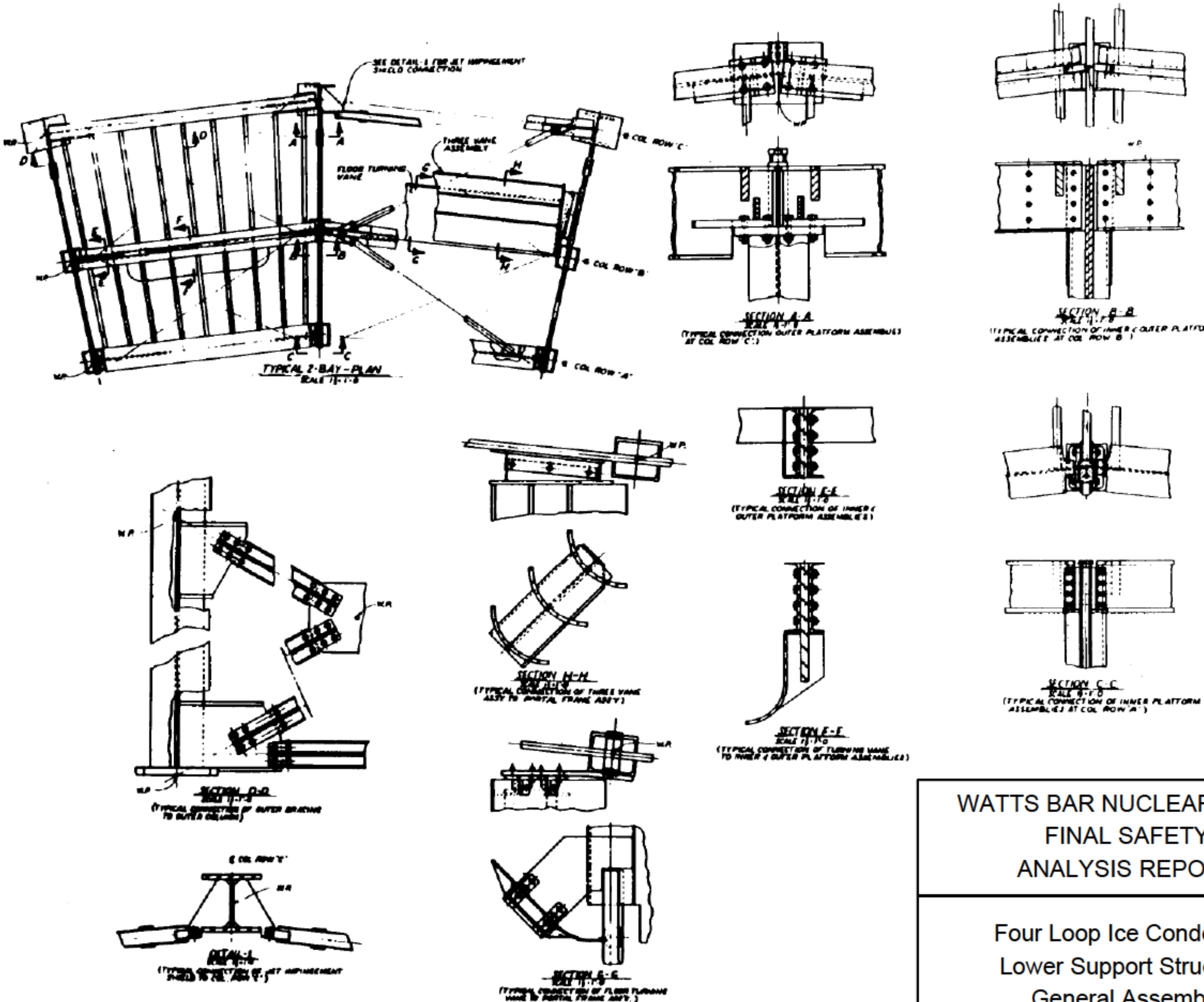
Lower Inlet Door
Shock Absorber Assembly
FIGURE 6.7-21



WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

Four Loop Ice Condenser Lower Support Structure Conceptual Plan and Sections

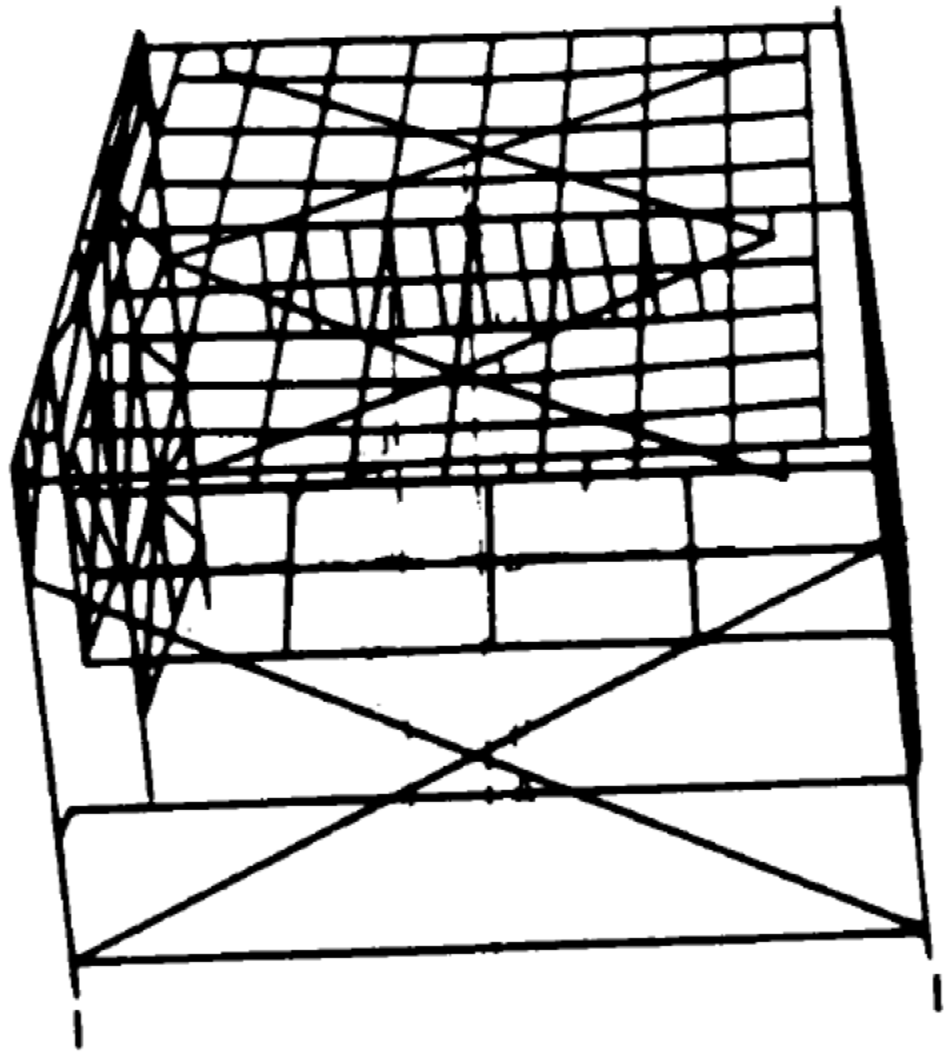
FIGURE 6.7-22



WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

Four Loop Ice Condenser
Lower Support Structure
General Assembly

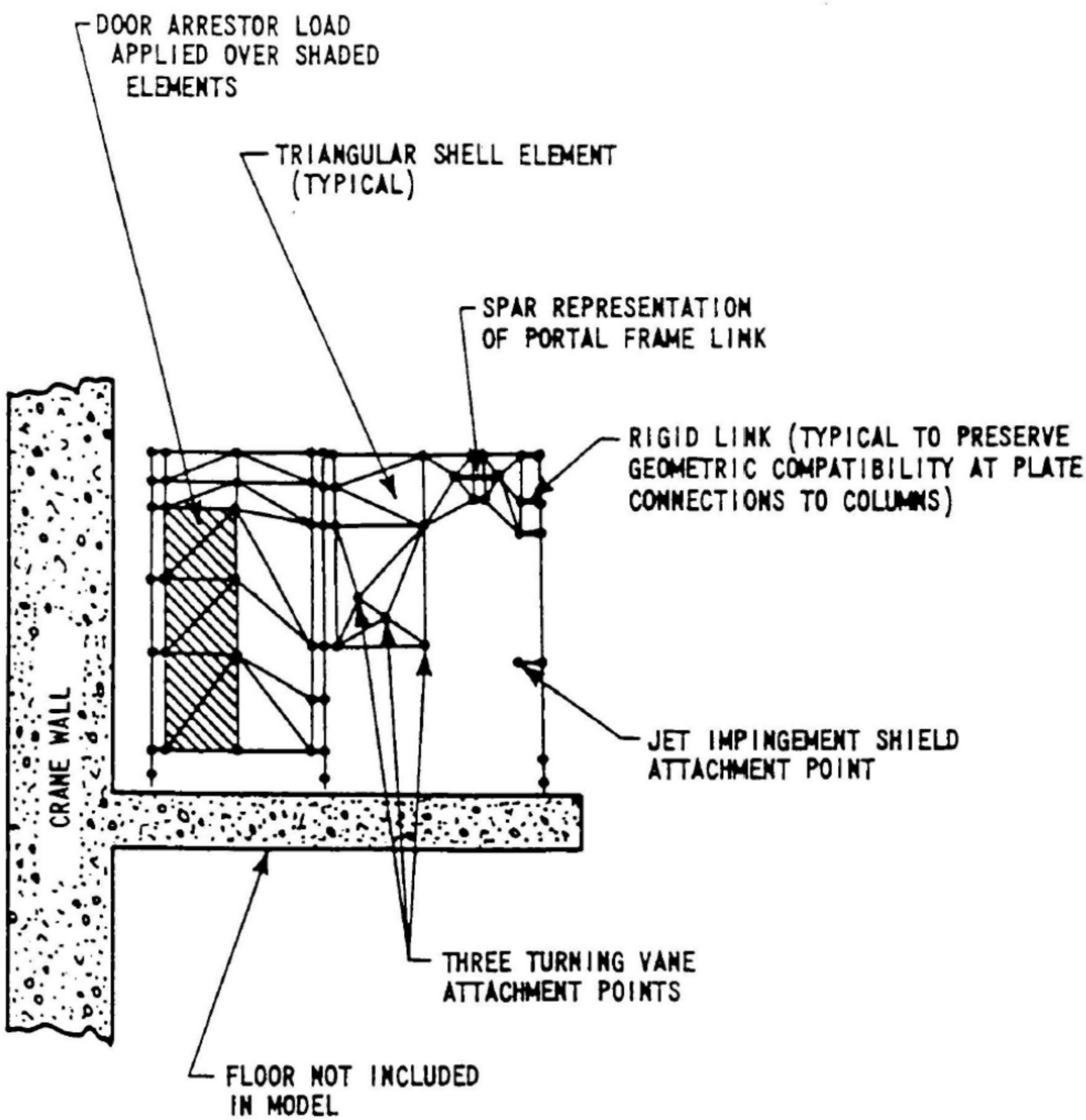
FIGURE 6.7-23



WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

ANTS Model Assembly

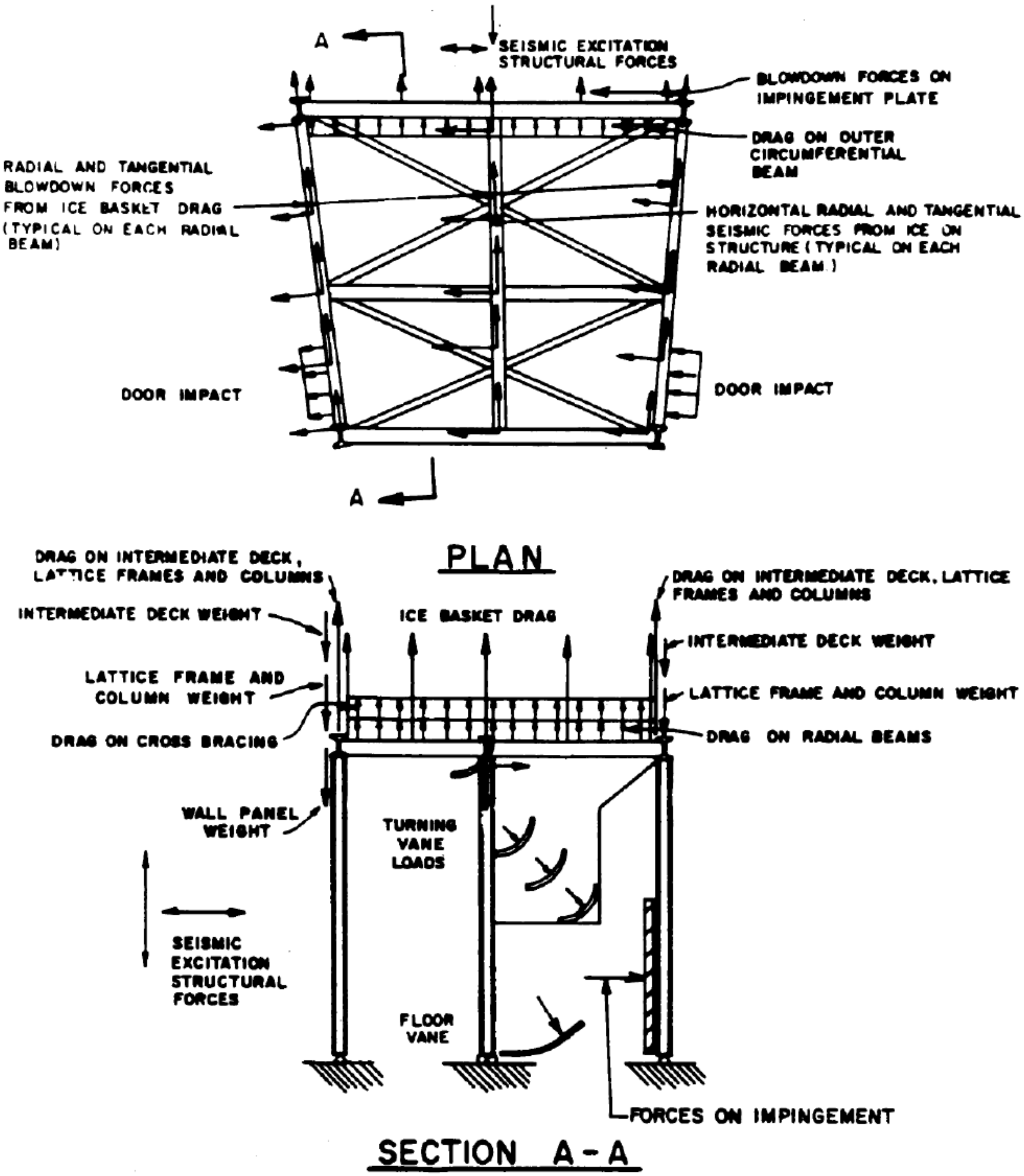
Figure 6.7-24



WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

Finite Element Model
of Portal Frame

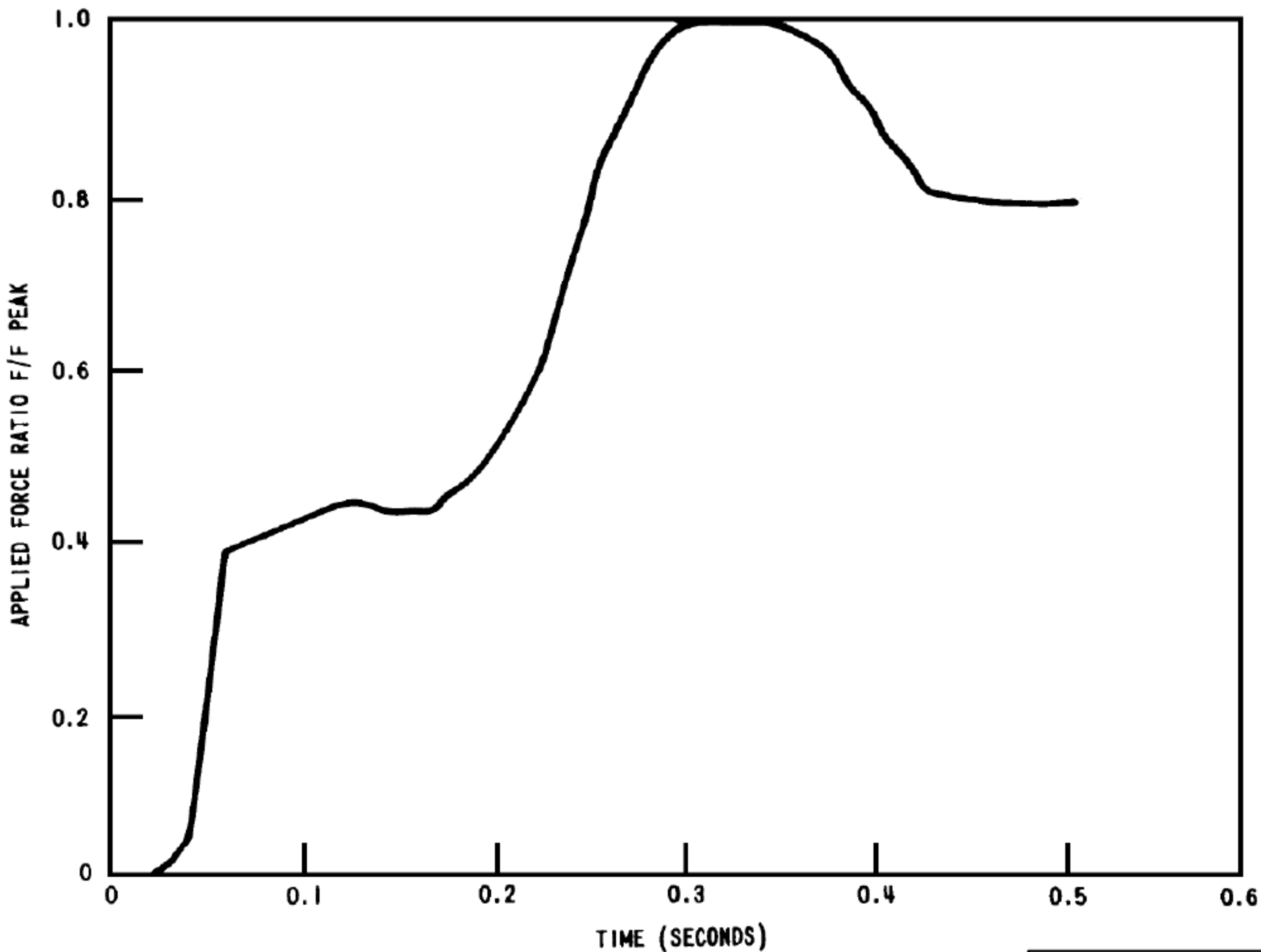
Figure 6.7-25



WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

Schematic Diagram of Forces Applied to Three Pier Lower Support Structure

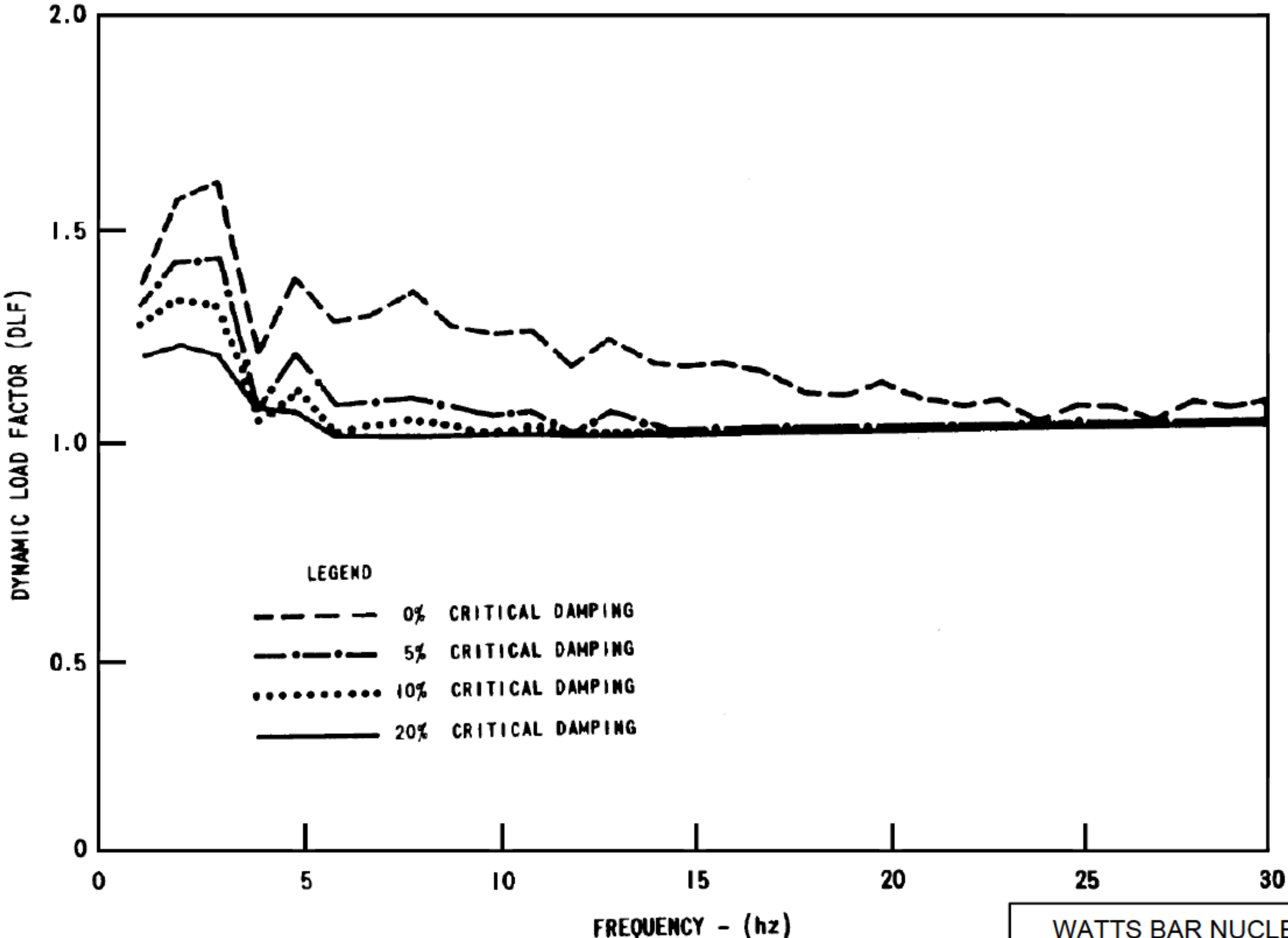
Figure 6.7-26



WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

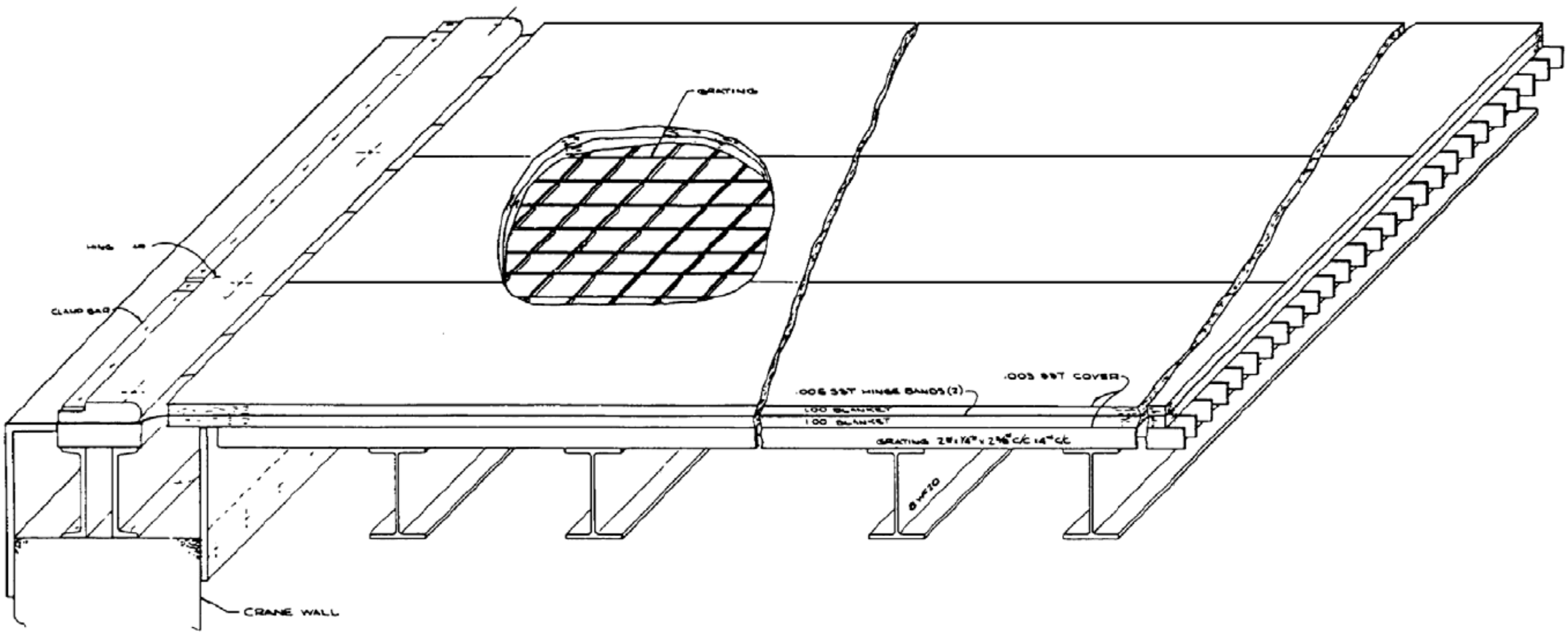
Force Transient
Hot Leg Break

FIGURE 6.7-27



WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

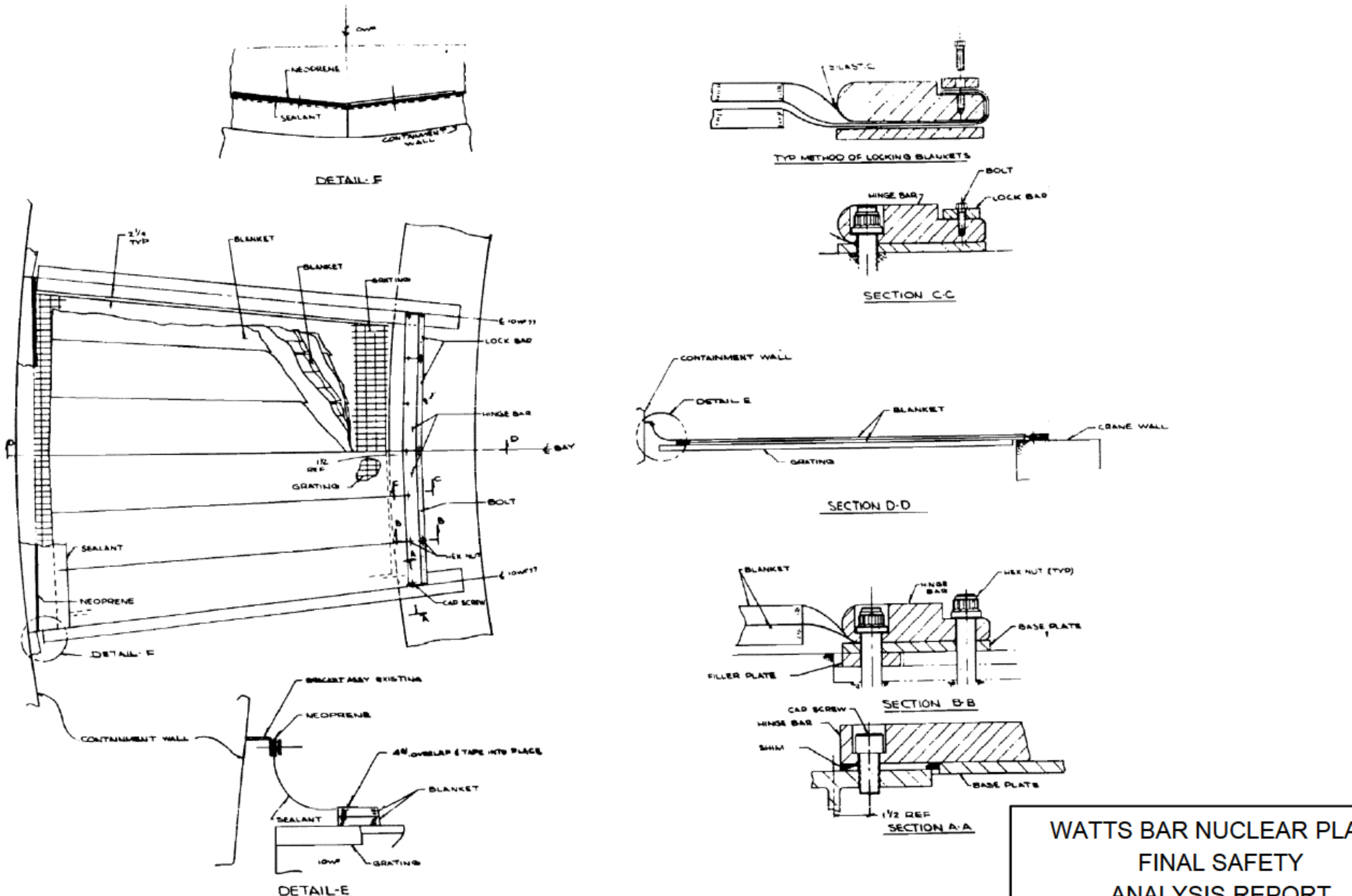
DLF Spectra
Hot Leg Break
Force Transient
FIGURE 6.7-28



WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

Top Deck
Test Assembly

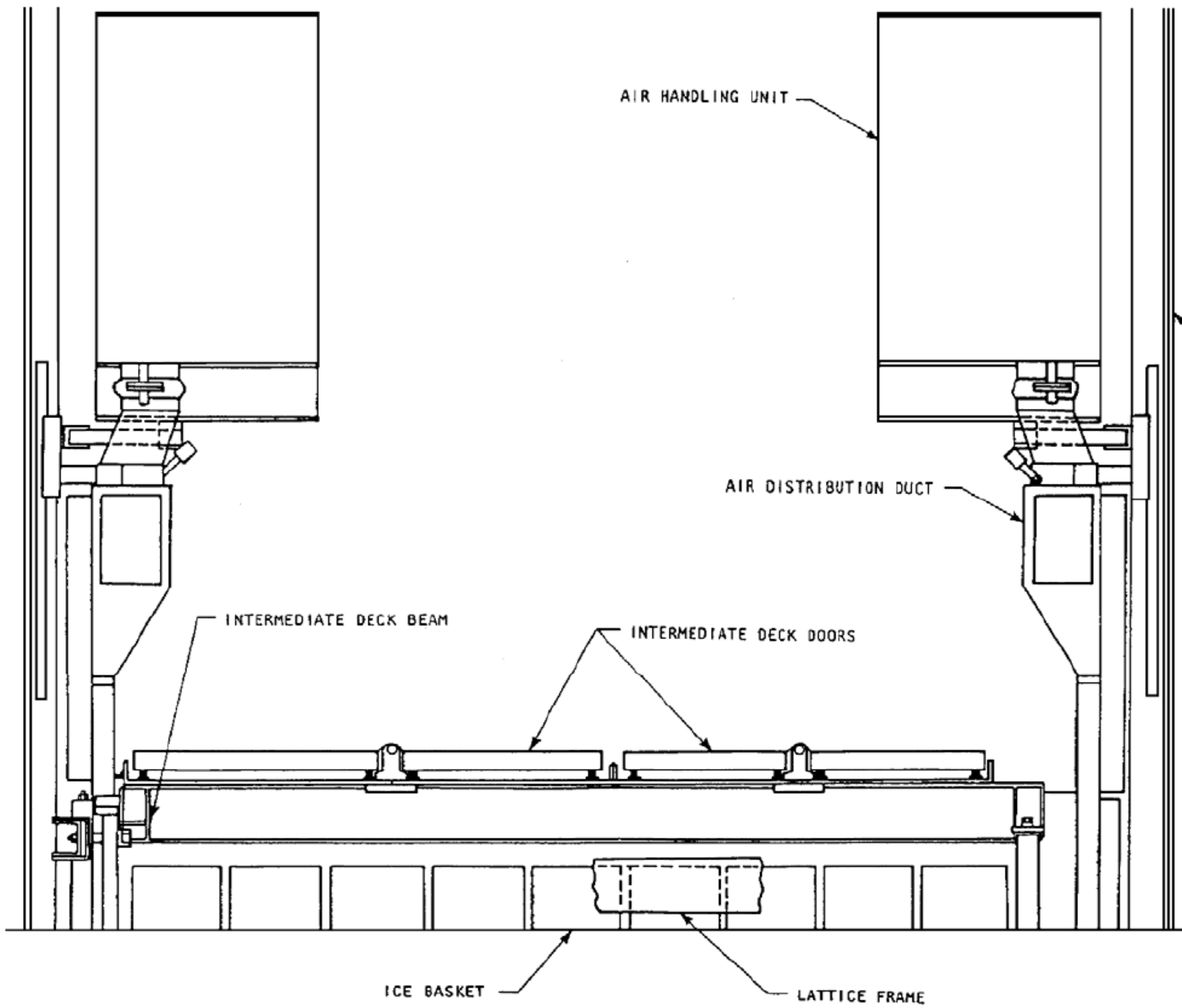
FIGURE 6.7-29



WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

Details of Top Deck
Door Assembly

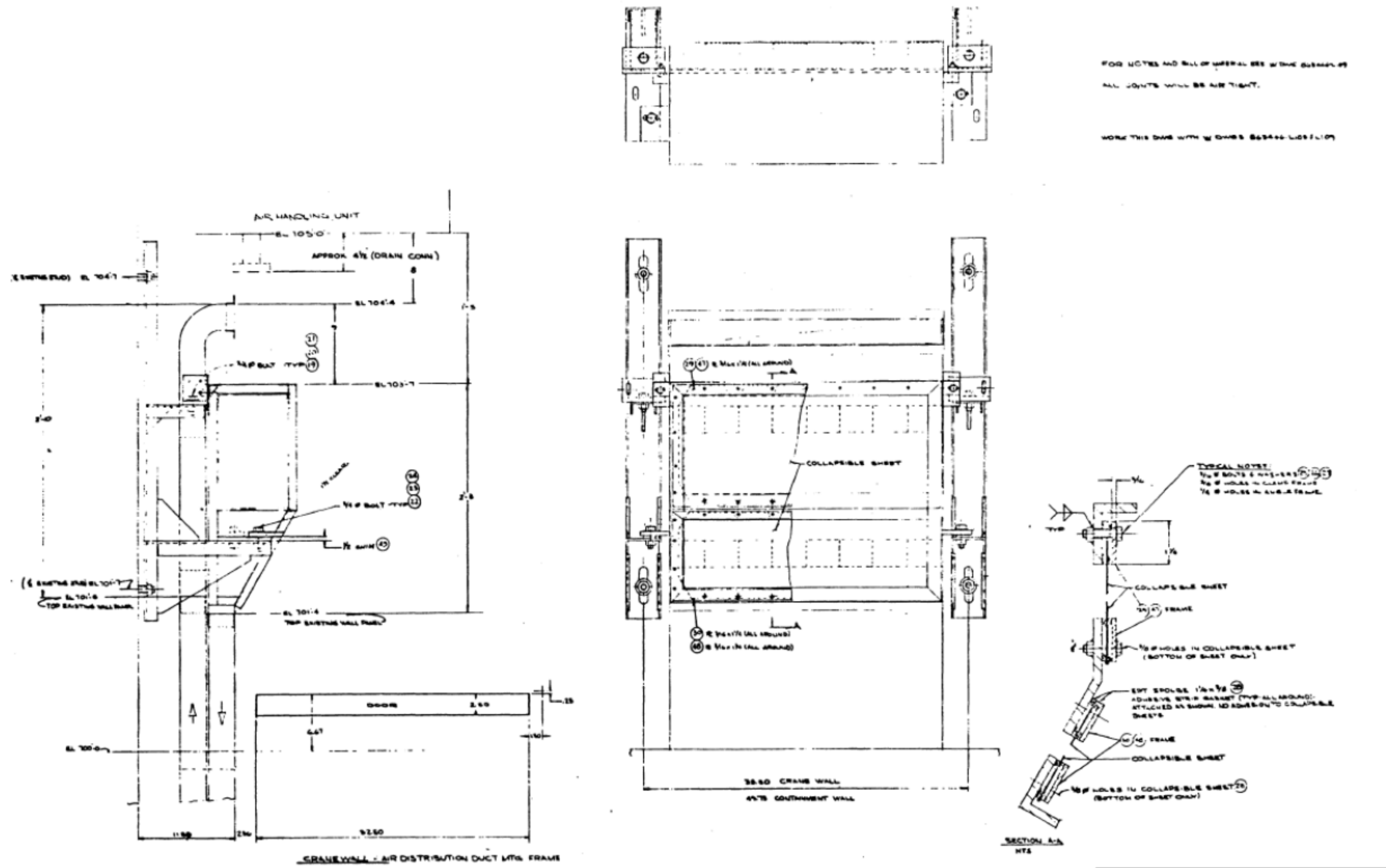
FIGURE 6.7-30



WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

Intermediate
Deck Doors

FIGURE 6.7-31



WATTS BAR NUCLEAR PLANT
FINAL SAFETY
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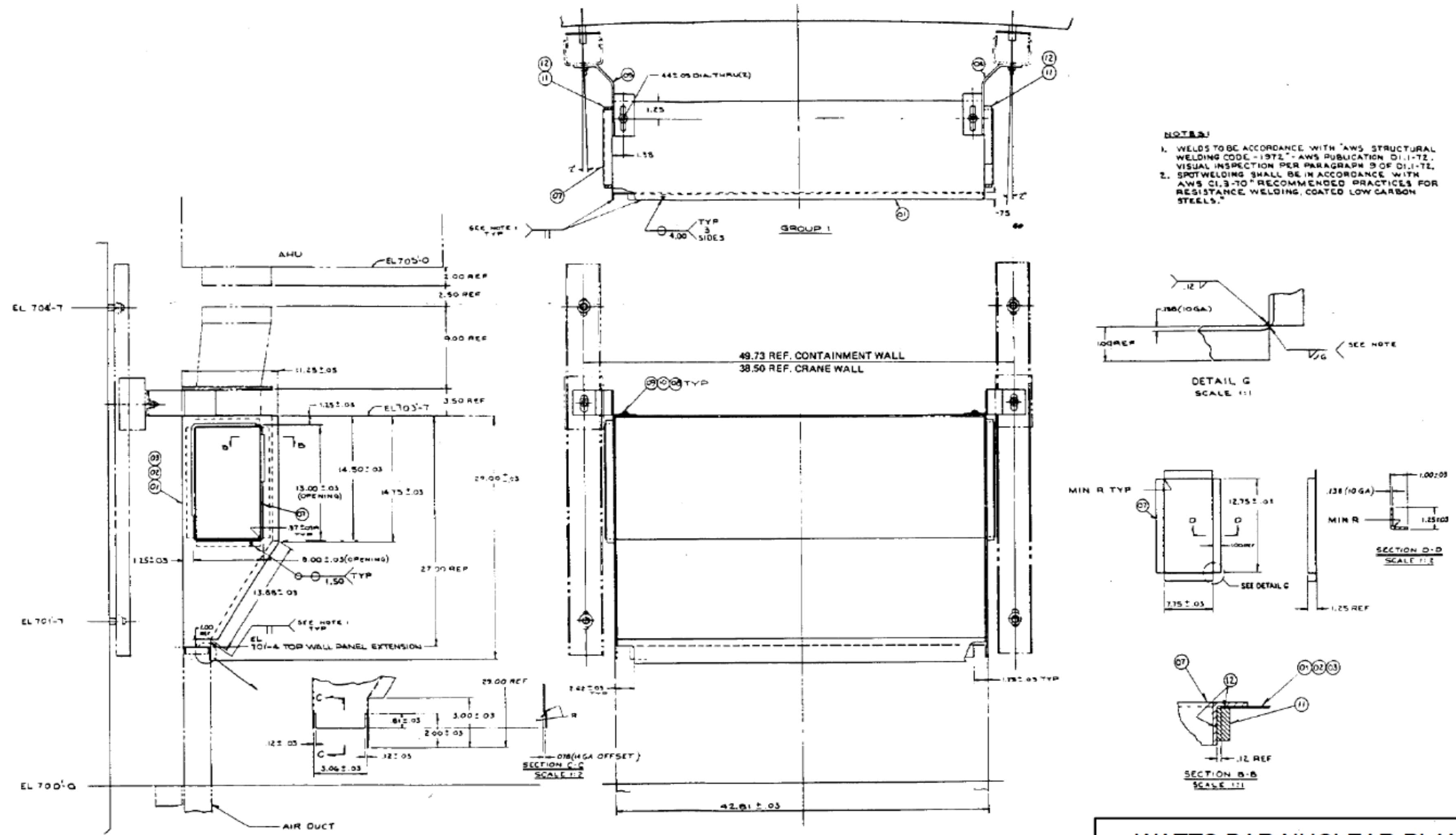
Air Distribution
Duct

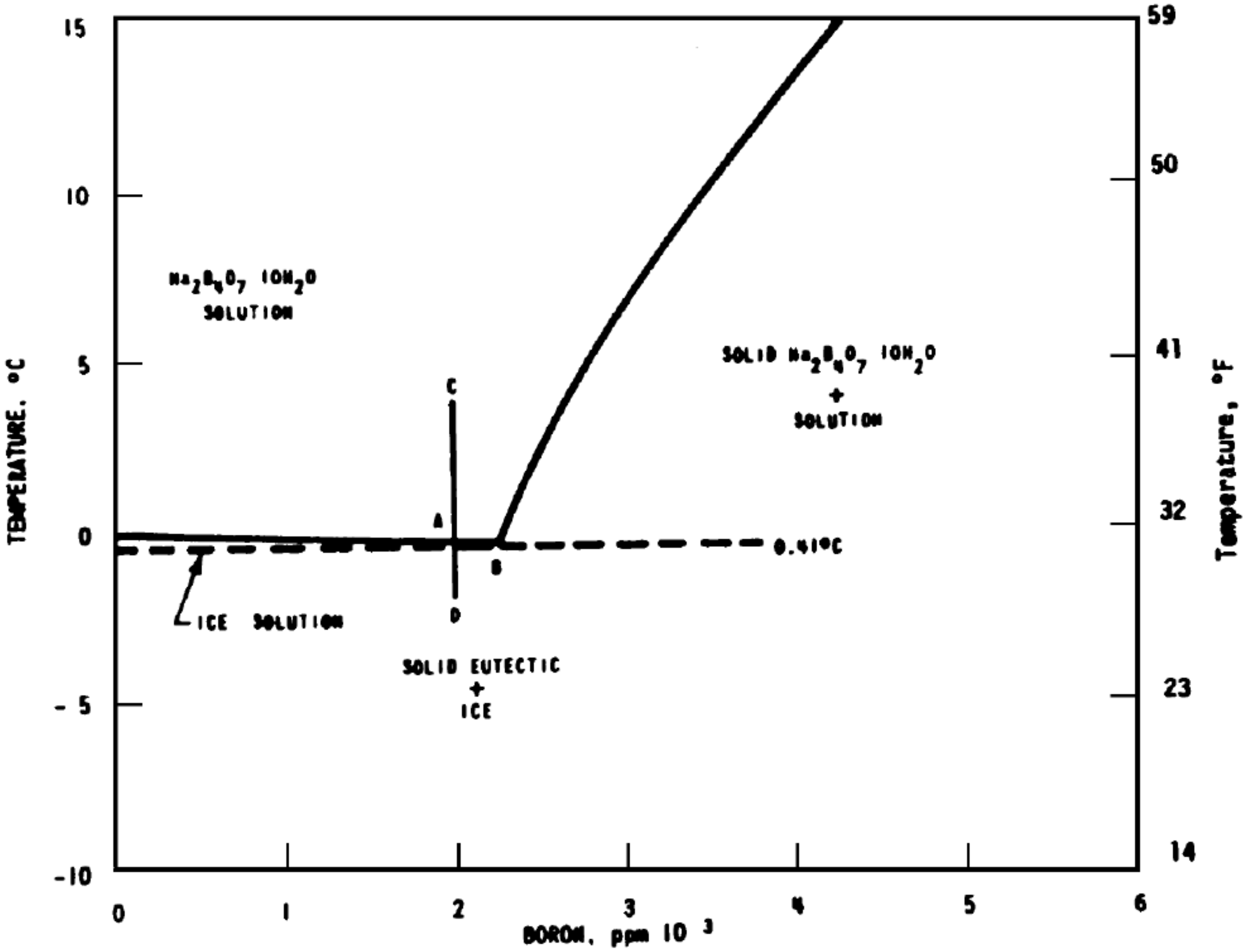
FIGURE 6.7-32



WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

Air Distribution
Duct

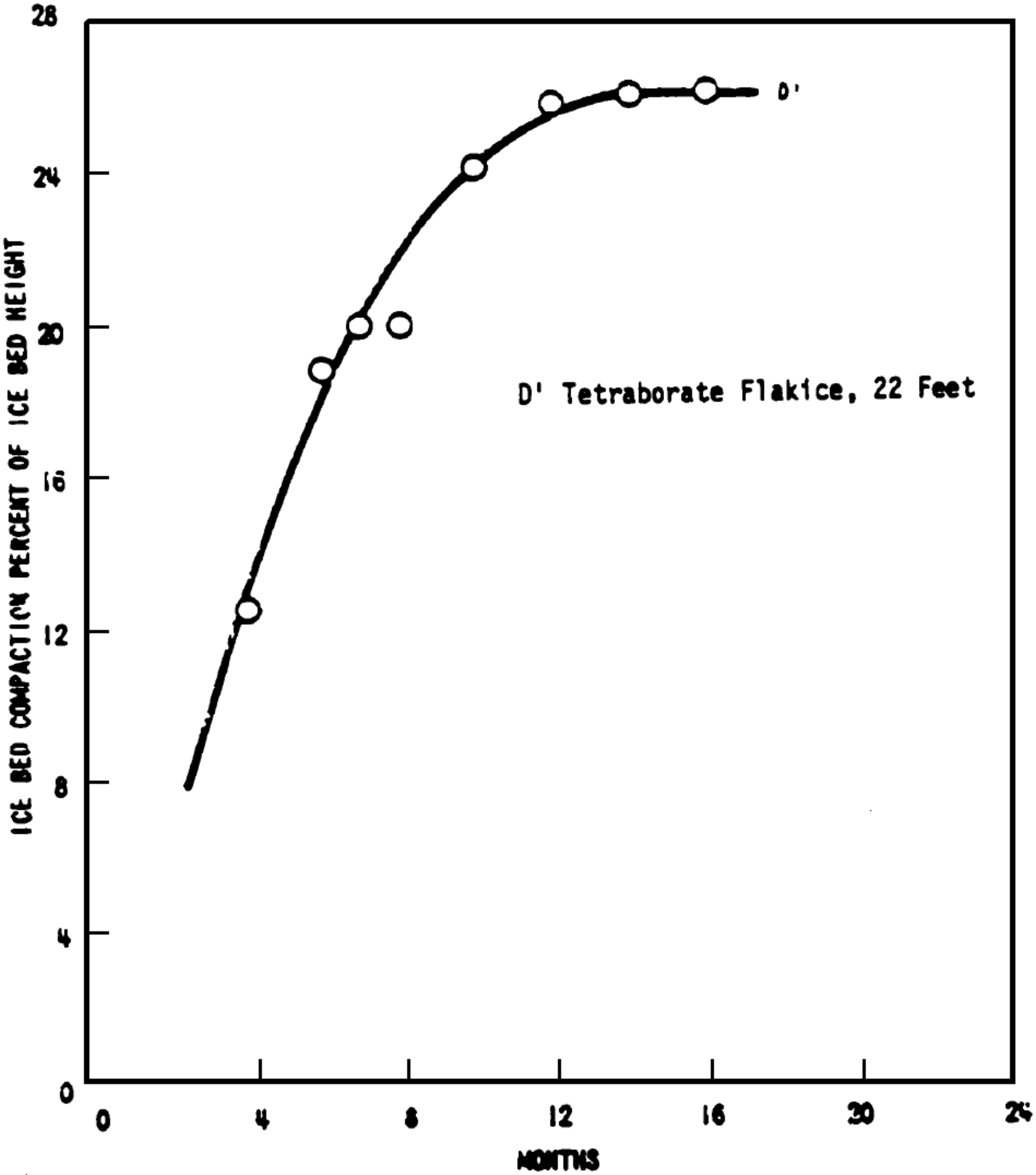
FIGURE 6.7-33



WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

Phase Diagram for $NA_2B_4O_7$ .10 H/Water System at One Atmosphere

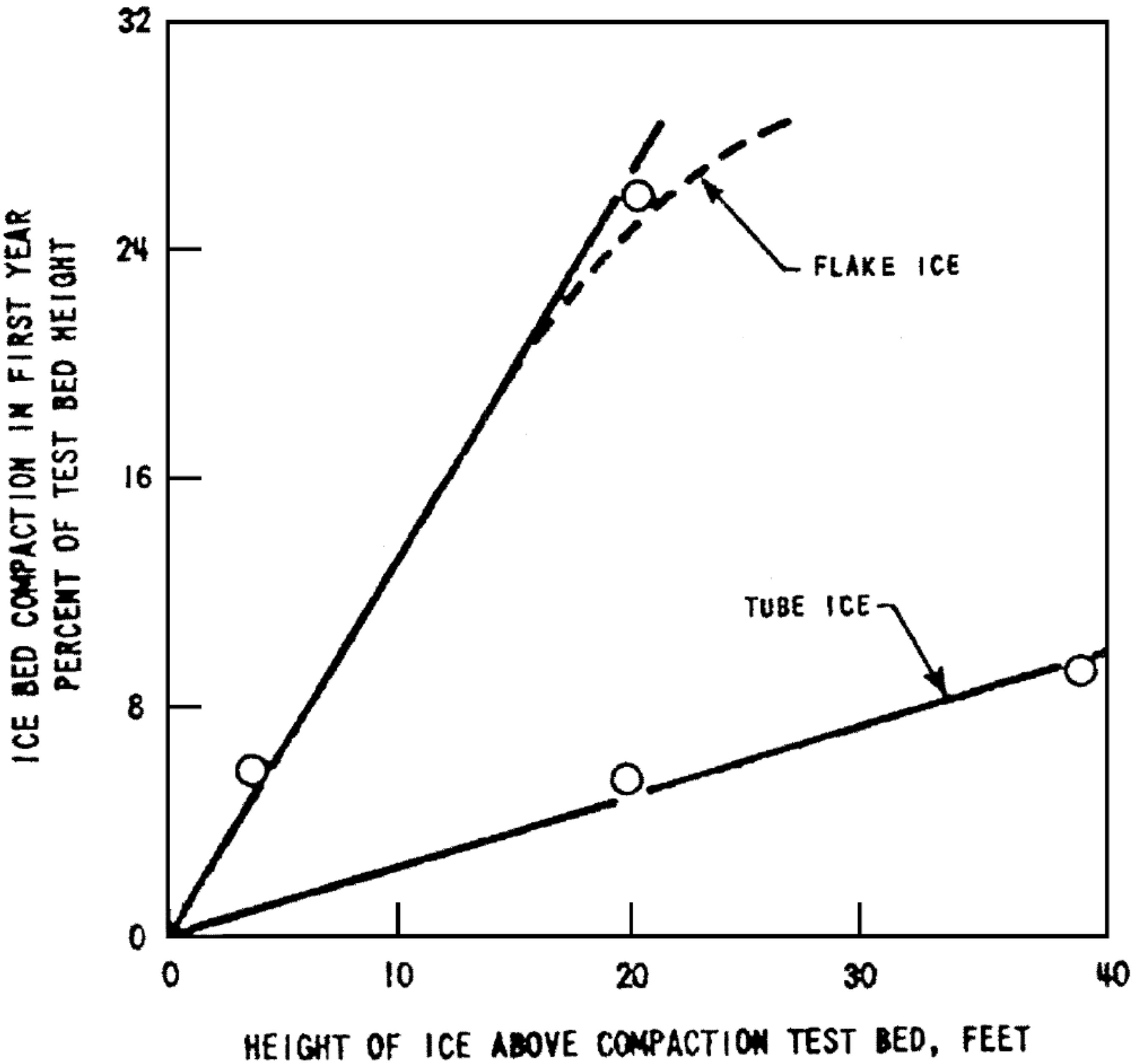
Figure 6.7-34



WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

Ice Bed Compaction
Versus
Time

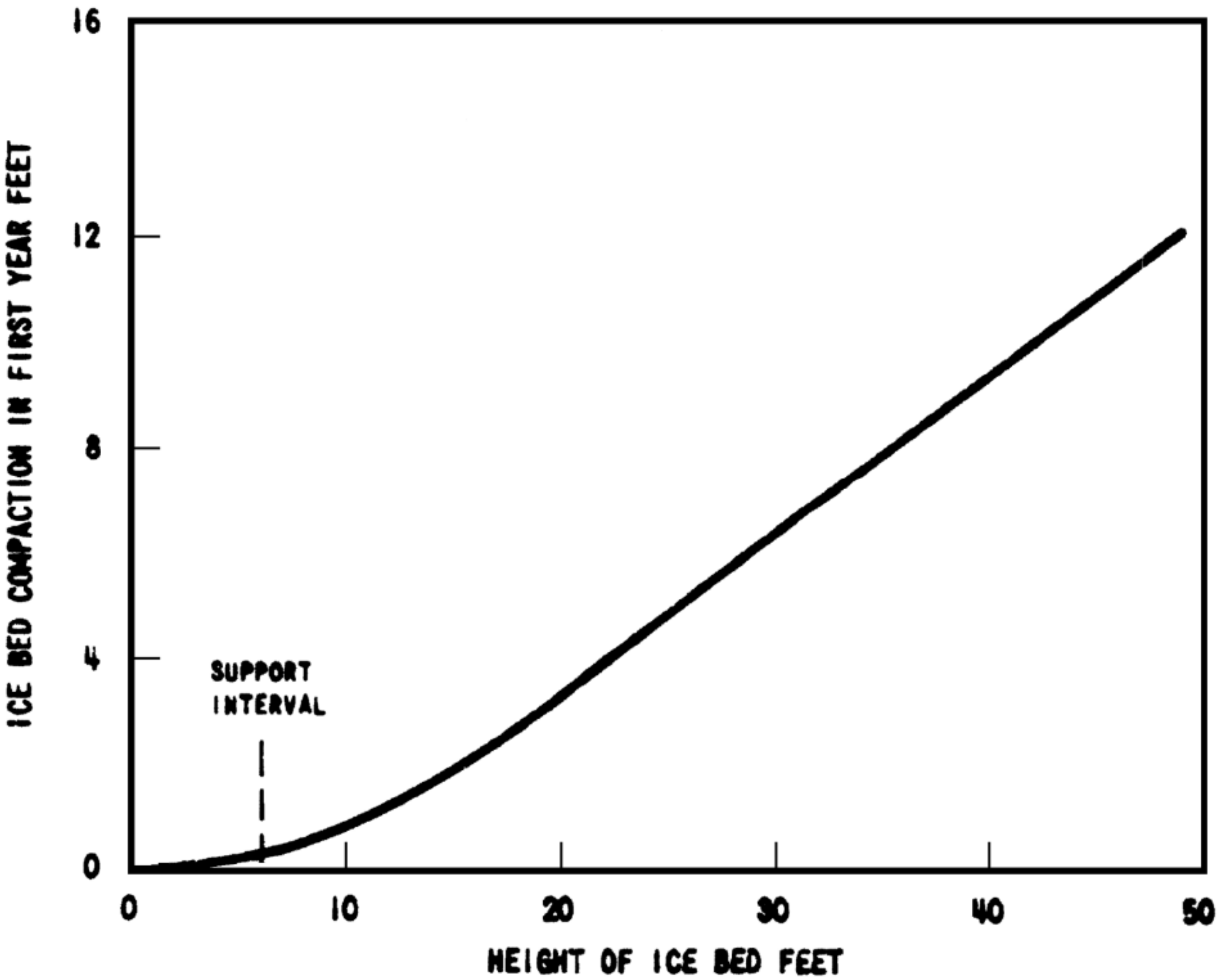
Figure 6.7-35



WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

Test Ice Bed
Compaction Versus
Ice Bed Height

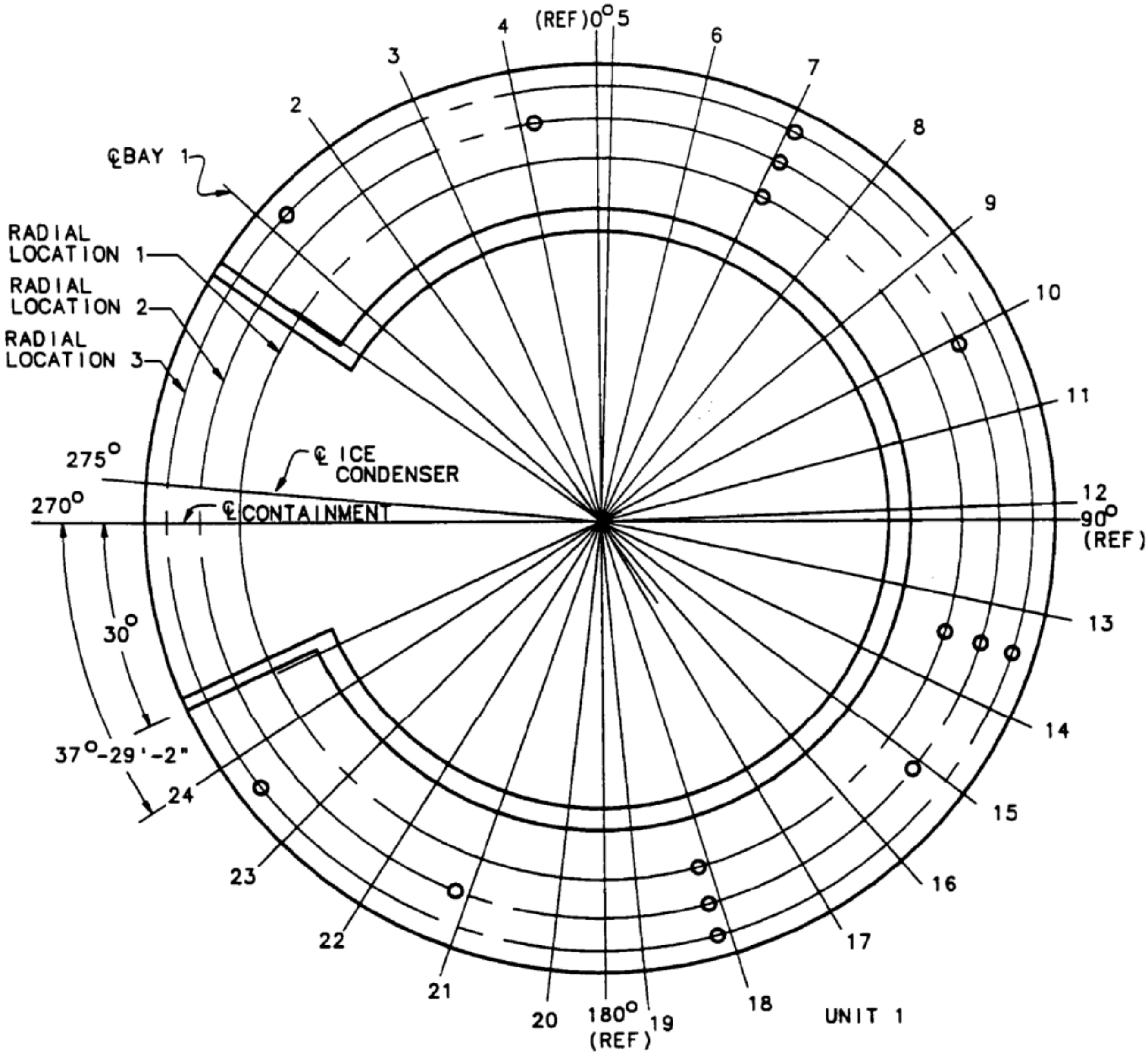
Figure 6.7-36



WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

Total Ice Compaction
Versus
Ice Bed Height

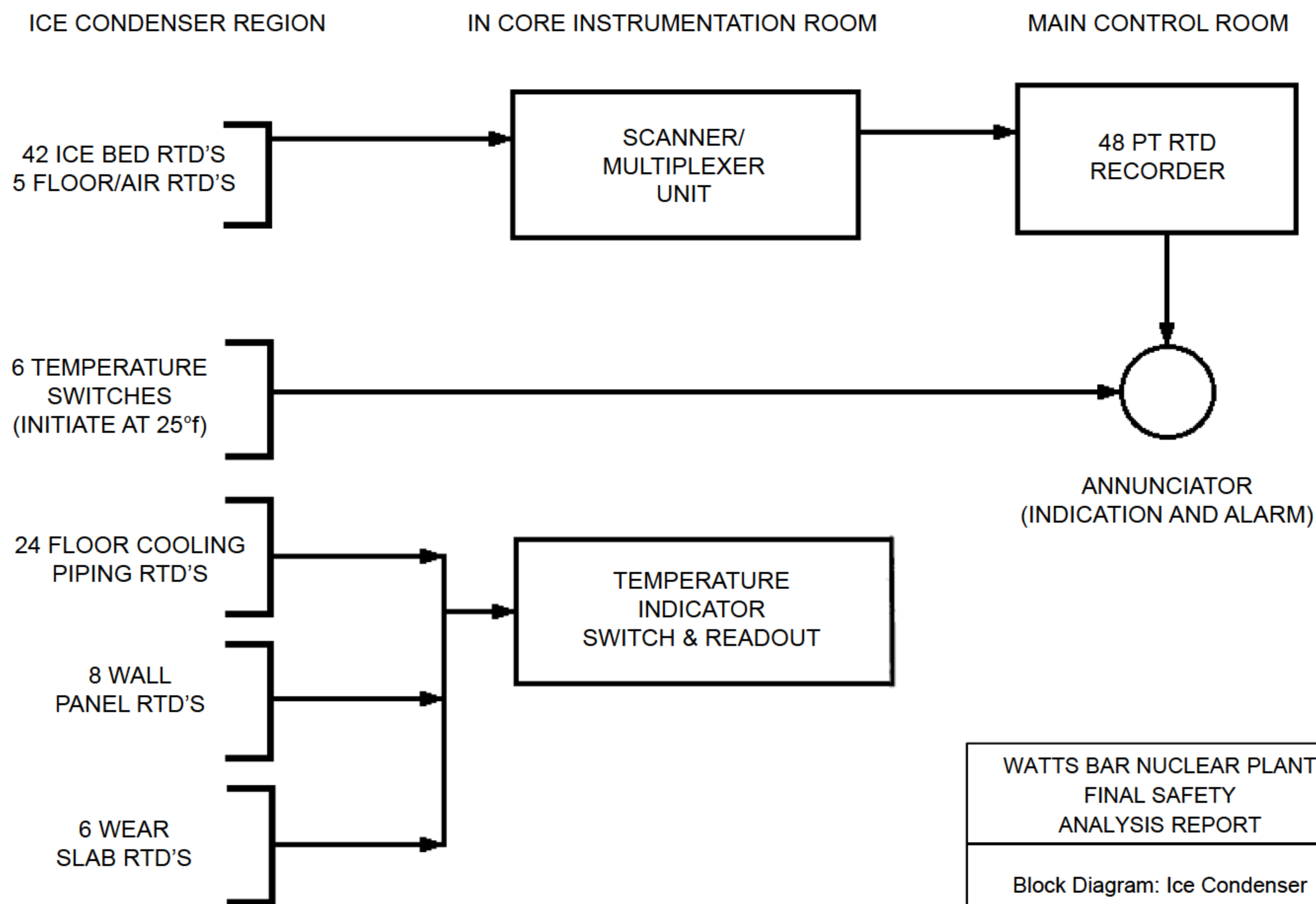
Figure 6.7-37



WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

Ice Condenser
RTD Location

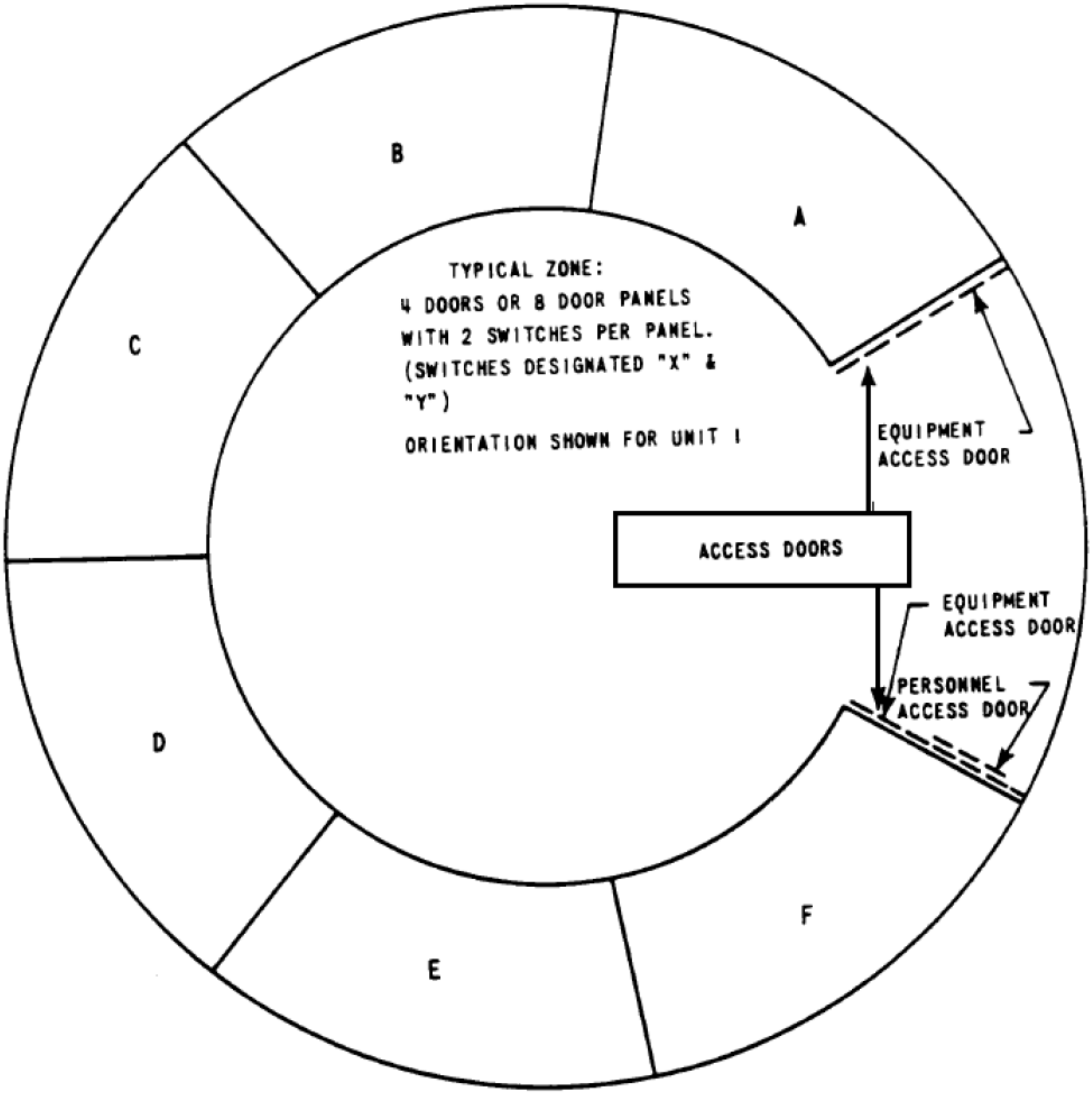
Figure 6.7-38



WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

Block Diagram: Ice Condenser Temperature Monitoring System

FIGURE 6.7-39

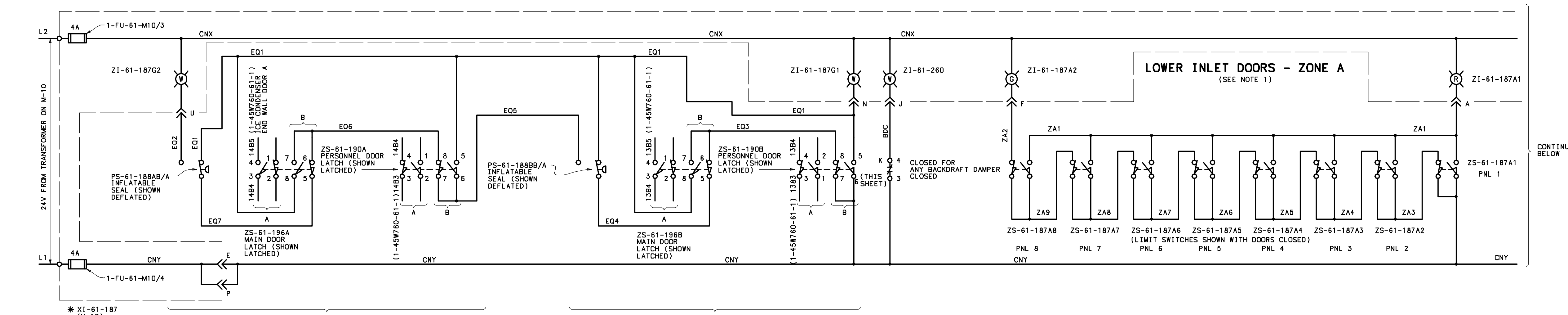


WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

Door Monitoring
Zones

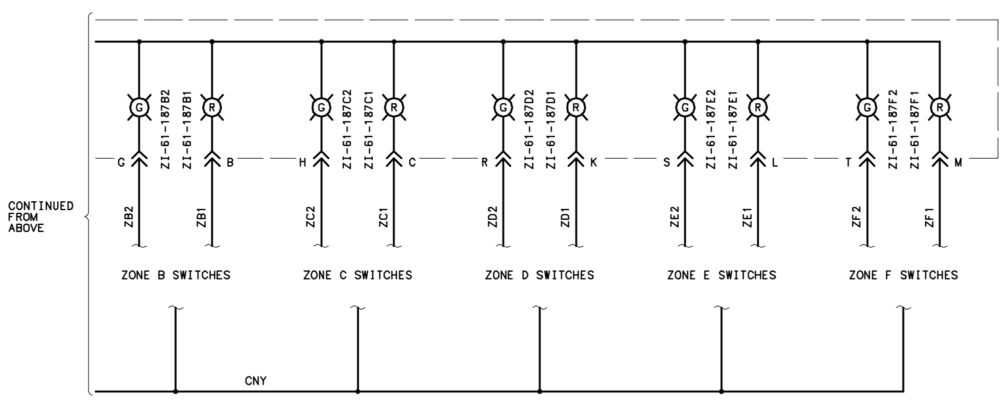
Figure 6.7-40

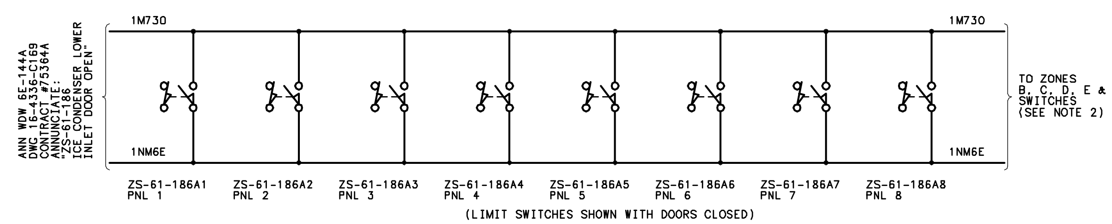
CAD MAINTAINED DRAWING



EQUIPMENT ACCESS DOOR A

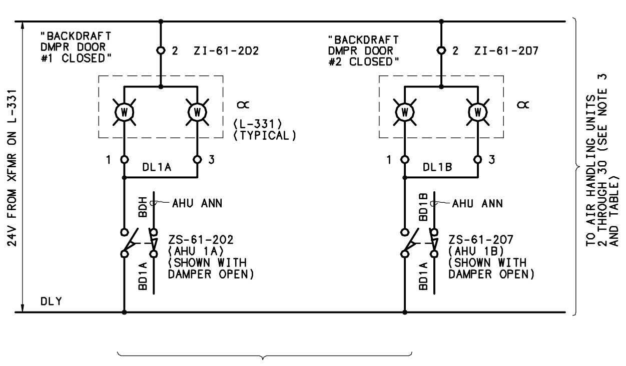
EQUIPMENT ACCESS DOOR B



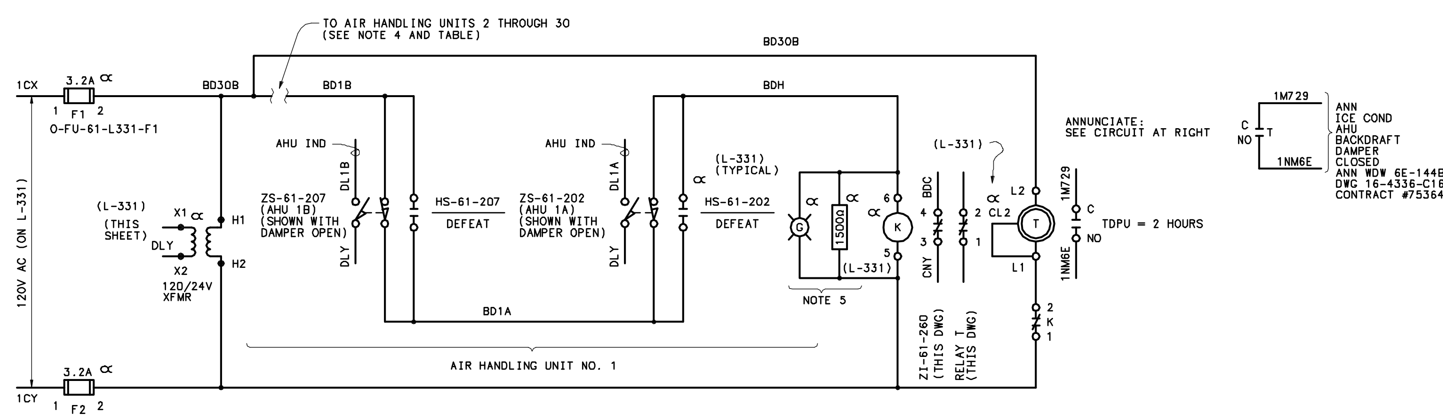


ZONE A

LOWER INLET DOOR ANNUNCIATION



AIR HANDLING UNIT BACKDRAFT DAMPER INDICATION



AIR HANDLING UNIT BACKDRAFT DAMPER ANNUNCIATION

NOTES:

1. THERE ARE SIX ZONES, A, B, C, D, E, AND F, IN THE ICE CONDENSER. ONLY THE SWITCHES FOR ZONE A INLET DOORS ARE SHOWN. THE WIRING FOR THE ZONES NOT SHOWN IS IDENTICAL TO THE ZONE A WIRING WITH THE EXCEPTION OF THE WIRE NUMBERS AND INSTRUMENT NUMBERS. WIRE NUMBERS FOR ZONE B ARE ZB1 THROUGH ZB9. INSTRUMENT NUMBERS FOR ZONE B ARE ZS-61-187B1 THROUGH ZS-61-187B8. THE OTHER ZONES ARE SIMILAR.
2. ALL LOWER INLET DOOR ZONE SWITCHES IN THE ANNUNCIATION CIRCUIT ARE IN PARALLEL. THE INSTRUMENT NUMBERS FOR ZONE B ARE ZS-61-186B1 THROUGH ZS-61-186B8. THE INSTRUMENTS FOR THE OTHER ZONES ARE SIMILAR.
3. THERE ARE 60 BACKDRAFT DAMPERS. THERE ARE 30 AIR HANDLING UNITS (AHU). EACH AIR HANDLING UNIT HAS TWO PARTS, A AND B. EACH OF THE TWO AHU PARTS HAS A BACKDRAFT DAMPER. THE LIGHT LENS ENGRAVINGS ARE NUMBERED FROM 1 TO 60 SUCH THAT THE ODD NUMBERS CORRESPOND TO THE A UNITS AND THE EVEN NUMBERS TO THE B UNITS. THE WIRE NUMBERS FOR THE OTHER AHU'S ARE FORMED BY PLACING THE AHU NO. (1A, 4B, 10A, ETC.) AFTER THE PREFIX "DL".
4. THE WIRE NUMBERS FOR THE BACKDRAFT DAMPER ANNUNCIATION CIRCUIT ARE FORMED BY PLACING THE AHU NO. (1A, 4B, 10A, ETC) AFTER THE PREFIX "BD". ALL THE SWITCHES FOR ALL THE UNITS ARE IN SERIES.
5. 1500Ω RESISTOR AND GREEN INDICATING LIGHT TO BE SUPPLIED AND INSTALLED BY CONSTRUCTION.

REFERENCE DRAWINGS:

47B601-61 SERIES ---- ELECTRICAL INSTRUMENT TABULATION
1-47W610-61 SERIES -- ELECTRICAL CONTROL DIAGRAM
47W611-61 ----------- ELECTRICAL LOGIC DIAGRAM
W1212E14 ------------ STATUS LIGHT PNL CONNECTION & WIRING DIAGRAM (71C62-54114-1)

SYMBOLS:

✻------------ DEVICES LOCATED IN UNIT CONTROL ROOM
∝ ----------- DEVICE LOCATED ON LOCAL PANEL

TABLE

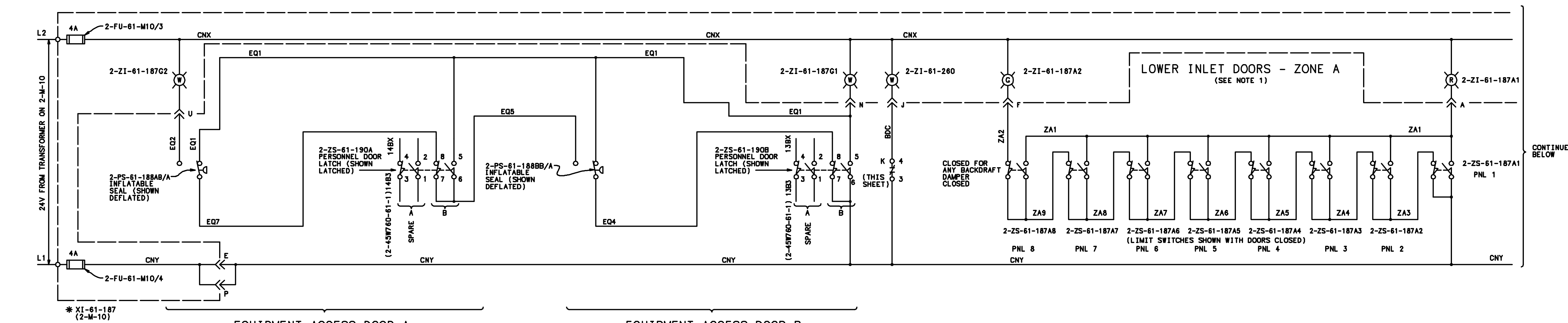
AHU	LOOP NO.	AHU	LOOP NO.
1A	202	16A	352
1B	207	16B	357
2A	212	17A	362
2B	217	17B	367
3A	222	18A	372
3B	227	18B	377
4A	232	19A	382
4B	237	19B	387
5A	242	20A	392
5B	247	20B	397
6A	252	21A	402
6B	257	21B	407
7A	262	22A	412
7B	267	22B	417
8A	272	23A	422
8B	277	23B	427
9A	282	24A	432
9B	287	24B	437
10A	292	25A	442
10B	297	25B	447
11A	302	26A	452
11B	307	26B	457
12A	312	27A	462
12B	317	27B	467
13A	322	28A	472
13B	327	28B	477
14A	332	29A	482
14B	337	29B	487
15A	342	30A	492
15B	347	30B	497

COMPANION DRAWINGS:
1-45W600-61-1

WATTS BAR
FINAL SAFETY
ANALYSIS REPORT

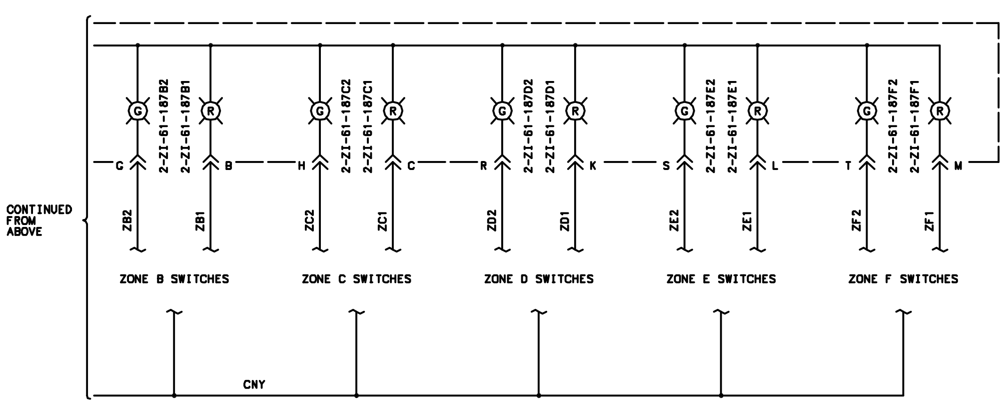
POWERHOUSE
UNIT 1
WIRING DIAGRAMS
ICE CONDENSER SYSTEM
SCHEMATIC DIAGRAMS
TVA DWG NO. 1-45W600-61-2 R12
FIGURE 6.7-41

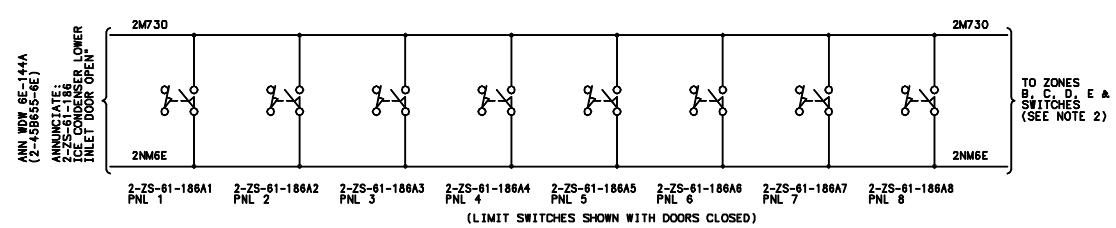
CAD MAINTAINED DRAWING



EQUIPMENT ACCESS DOOR A

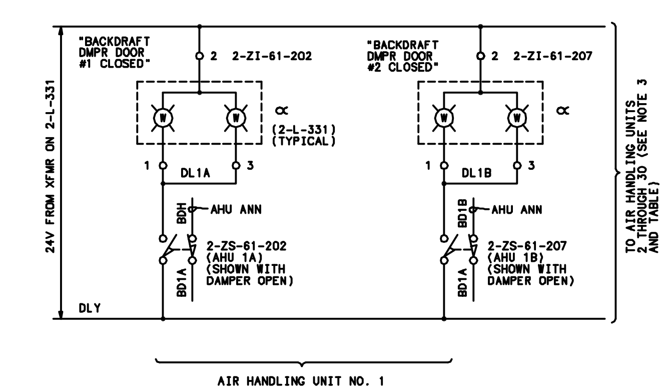
EQUIPMENT ACCESS DOOR B



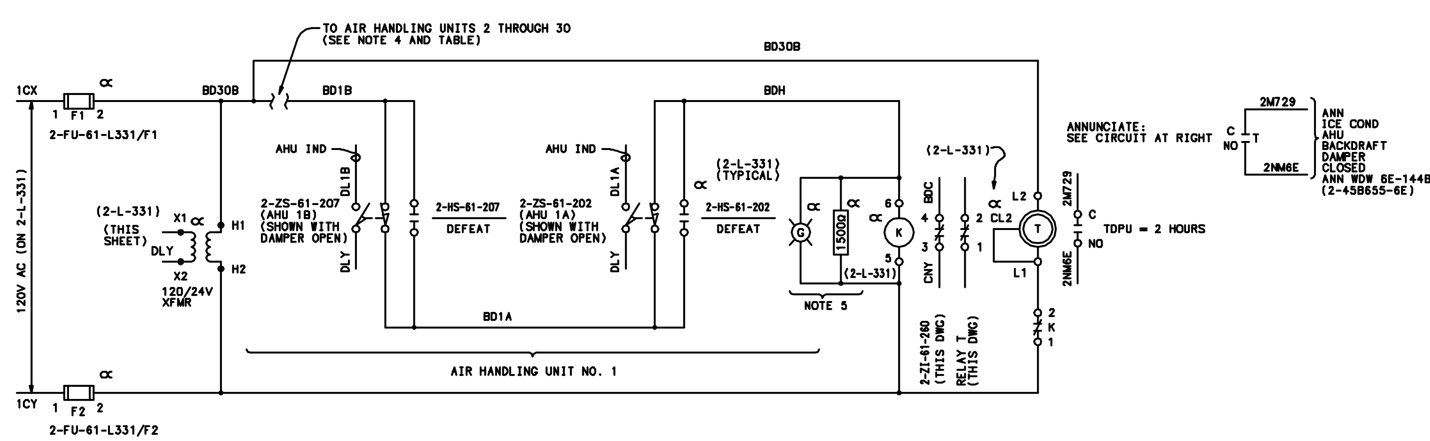


ZONE A

LOWER INLET DOOR ANNUNCIATION



AIR HANDLING UNIT BACKDRAFT DAMPER INDICATION



AIR HANDLING UNIT BACKDRAFT DAMPER ANNUNCIATION

NOTES:

1. THERE ARE SIX ZONES, A, B, C, D, E, AND F, IN THE ICE CONDENSER. ONLY THE SWITCHES FOR ZONE A INLET DOORS ARE SHOWN. THE WIRING FOR THE ZONES NOT SHOWN IS IDENTICAL TO THE ZONE A WIRING WITH THE EXCEPTION OF THE WIRE NUMBERS AND INSTRUMENT NUMBERS. WIRE NUMBERS FOR ZONE B ARE ZB1 THROUGH ZB9. INSTRUMENT NUMBERS FOR ZONE B ARE 2-ZS-61-187B1 THROUGH 2-ZS-61-187B8. THE OTHER ZONES ARE SIMILAR.
2. ALL LOWER INLET DOOR ZONE SWITCHES IN THE ANNUNCIATION CIRCUIT ARE IN PARALLEL. THE INSTRUMENT NUMBERS FOR ZONE B ARE 2-ZS-61-186B1 THROUGH 2-ZS-61-186B8. THE INSTRUMENTS FOR THE OTHER ZONES ARE SIMILAR.
3. THERE ARE 60 BACKDRAFT DAMPERS. THERE ARE 30 AIR HANDLING UNITS (AHU). EACH AIR HANDLING UNIT HAS TWO PARTS, A AND B. EACH OF THE TWO AHU PARTS HAS A BACKDRAFT DAMPER. THE LIGHT LENS ENGRAVINGS ARE NUMBERED FROM 1 TO 60 SUCH THAT THE ODD NUMBERS CORRESPOND TO THE A UNITS AND THE EVEN NUMBERS TO THE B UNITS. THE WIRE NUMBERS FOR THE OTHER AHU'S ARE FORMED BY PLACING THE AHU NO. (1A, 4B, 10A, ETC.) AFTER THE PREFIX "DL".
4. THE WIRE NUMBERS FOR THE BACKDRAFT DAMPER ANNUNCIATION CIRCUIT ARE FORMED BY PLACING THE AHU NO. (1A, 4B, 10A, ETC) AFTER THE PREFIX "BD". ALL THE SWITCHES FOR ALL THE UNITS ARE IN SERIES.
5. 1500Ω RESISTOR AND GREEN INDICATING LIGHT TO BE SUPPLIED AND INSTALLED BY CONSTRUCTION.

REFERENCE DRAWINGS:

47B601-61 SERIES ---- ELECTRICAL INSTRUMENT TABULATION
2-47W610-61 SERIES -- ELECTRICAL CONTROL DIAGRAM
47W611-61 ----------- ELECTRICAL LOGIC DIAGRAM
W1212E14 ------------ STATUS LIGHT PNL CONNECTION & WIRING DIAGRAM (71C62-54114-1)

SYMBOLS:

*------------ DEVICES LOCATED IN UNIT CONTROL ROOM
∝ ----------- DEVICE LOCATED ON LOCAL PANEL

TABLE

AHU	LOOP NO.	AHU	LOOP NO.
1A	202	16A	352
1B	207	16B	357
2A	212	17A	362
2B	217	17B	367
3A	222	18A	372
3B	227	18B	377
4A	232	19A	382
4B	237	19B	387
5A	242	20A	392
5B	247	20B	397
6A	252	21A	402
6B	257	21B	407
7A	262	22A	412
7B	267	22B	417
8A	272	23A	422
8B	277	23B	427
9A	282	24A	432
9B	287	24B	437
10A	292	25A	442
10B	297	25B	447
11A	302	26A	452
11B	307	26B	457
12A	312	27A	462
12B	317	27B	467
13A	322	28A	472
13B	327	28B	477
14A	332	29A	482
14B	337	29B	487
15A	342	30A	492
15B	347	30B	497

COMPANION DRAWINGS:
2-45W600-61-1

UFSAR AMENDMENT 1

WATTS BAR
FINAL SAFETY
ANALYSIS REPORT

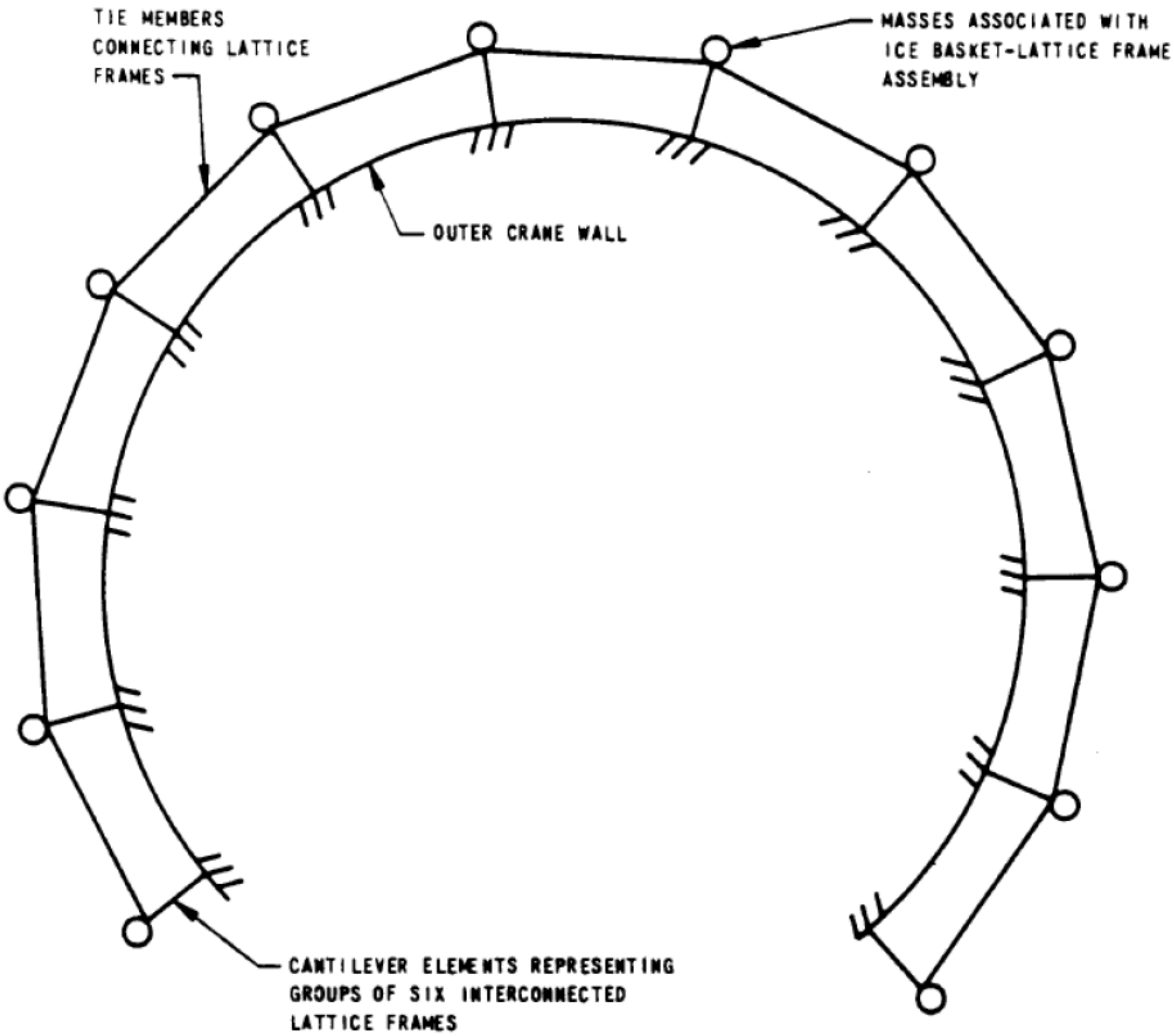
POWERHOUSE
UNIT 2
WIRING DIAGRAMS
ICE CONDENSER SYSTEM
SCHEMATIC DIAGRAMS
TVA DWG NO. 2-45W600-61-2 R7
FIGURE 6.7-41(U2)

FIGURE 6.7-42

DELETED

FIGURE 6.7-43

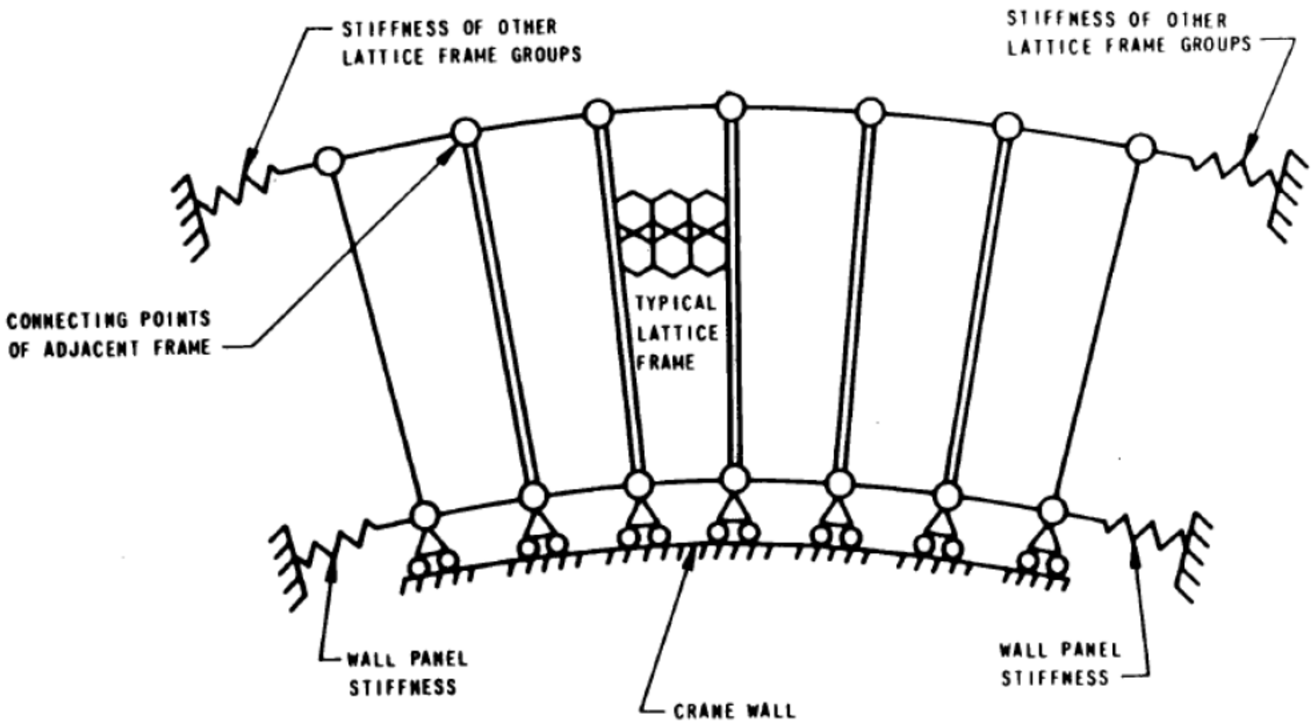
DELETED



WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

Model of Horizontal
Lattice Frame Structure

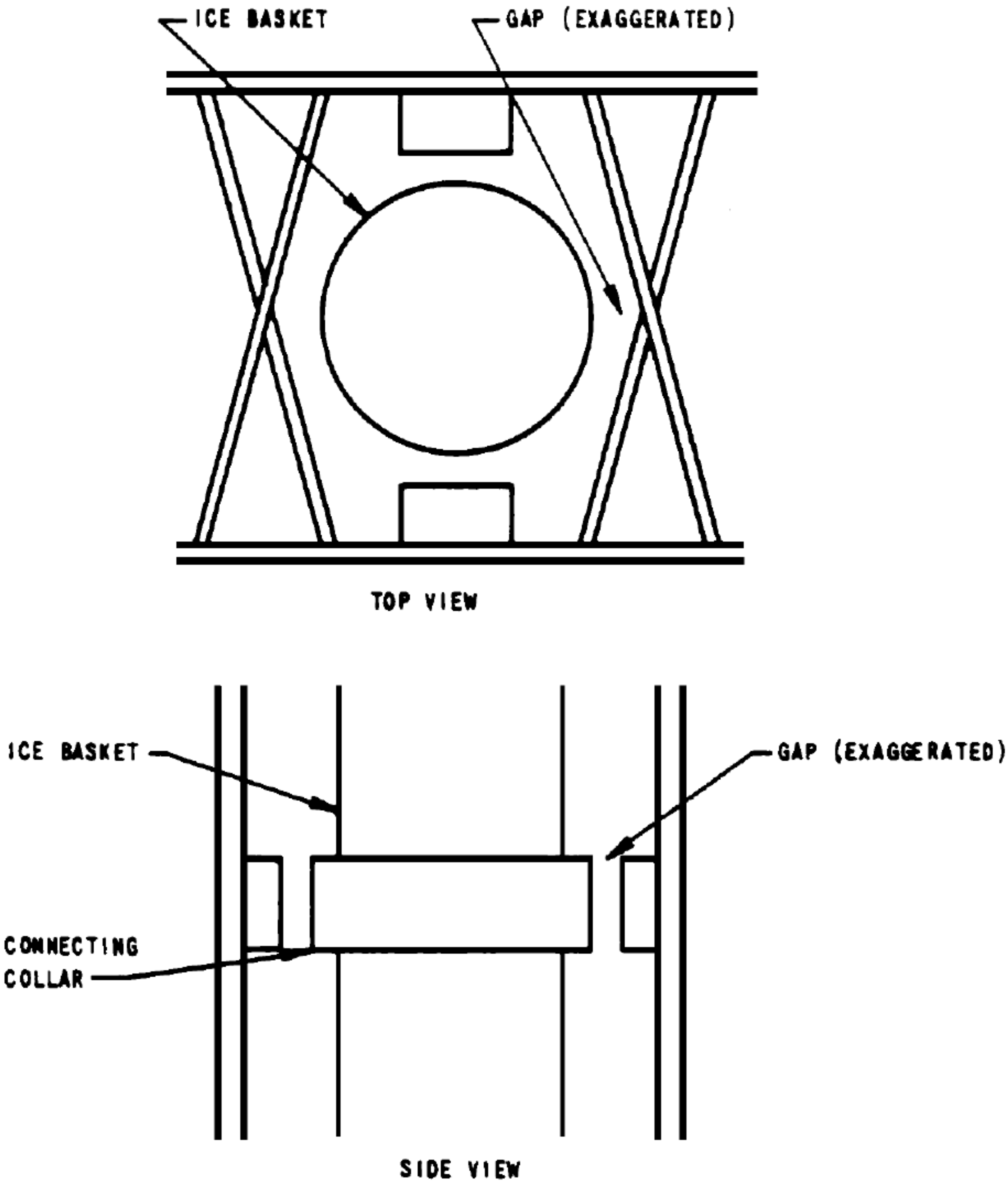
Figure 6.7-44



WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

Group of Six Interconnected Lattice Frames

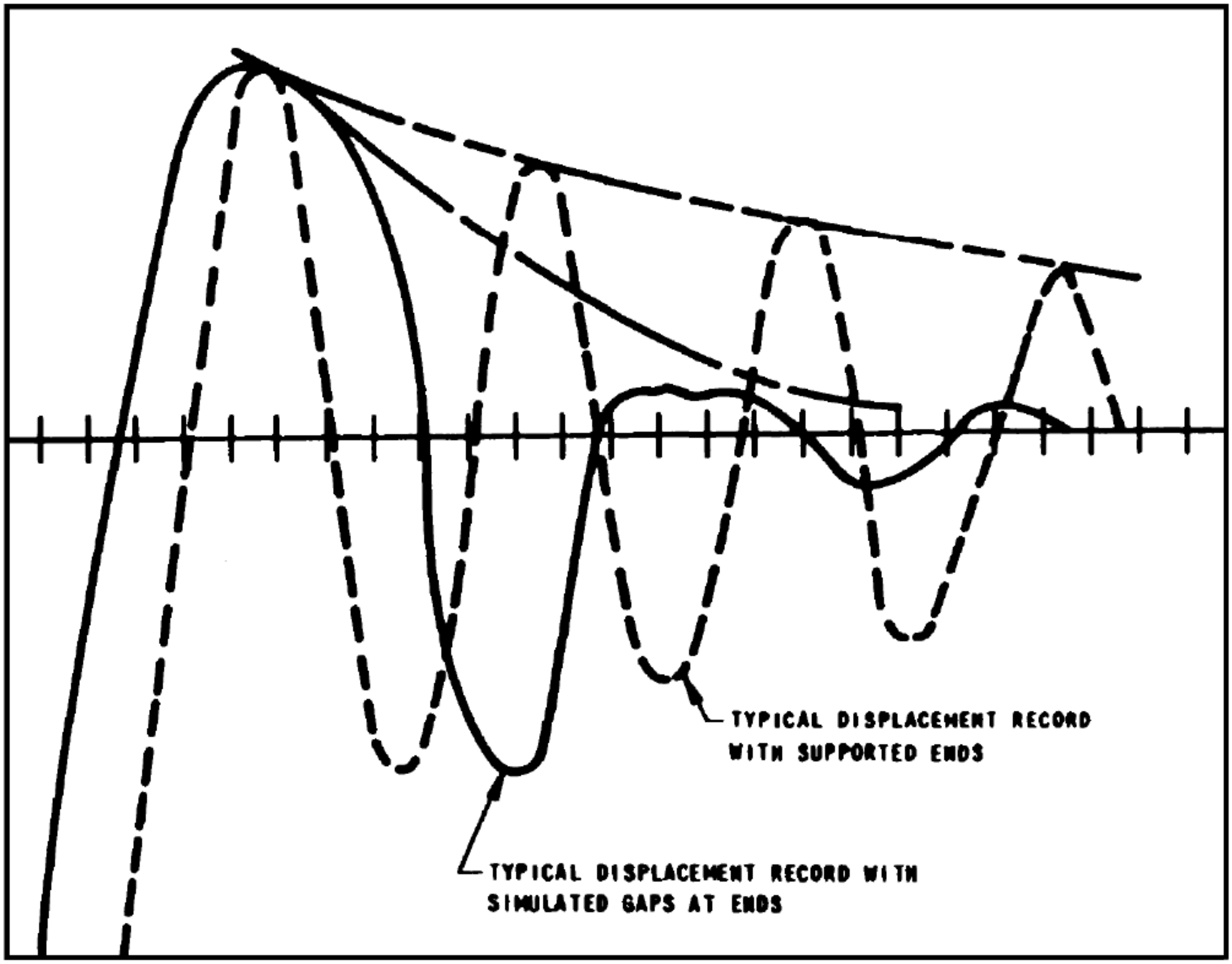
Figure 6.7-45



WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

Lattice Frame
Ice Basket Gap

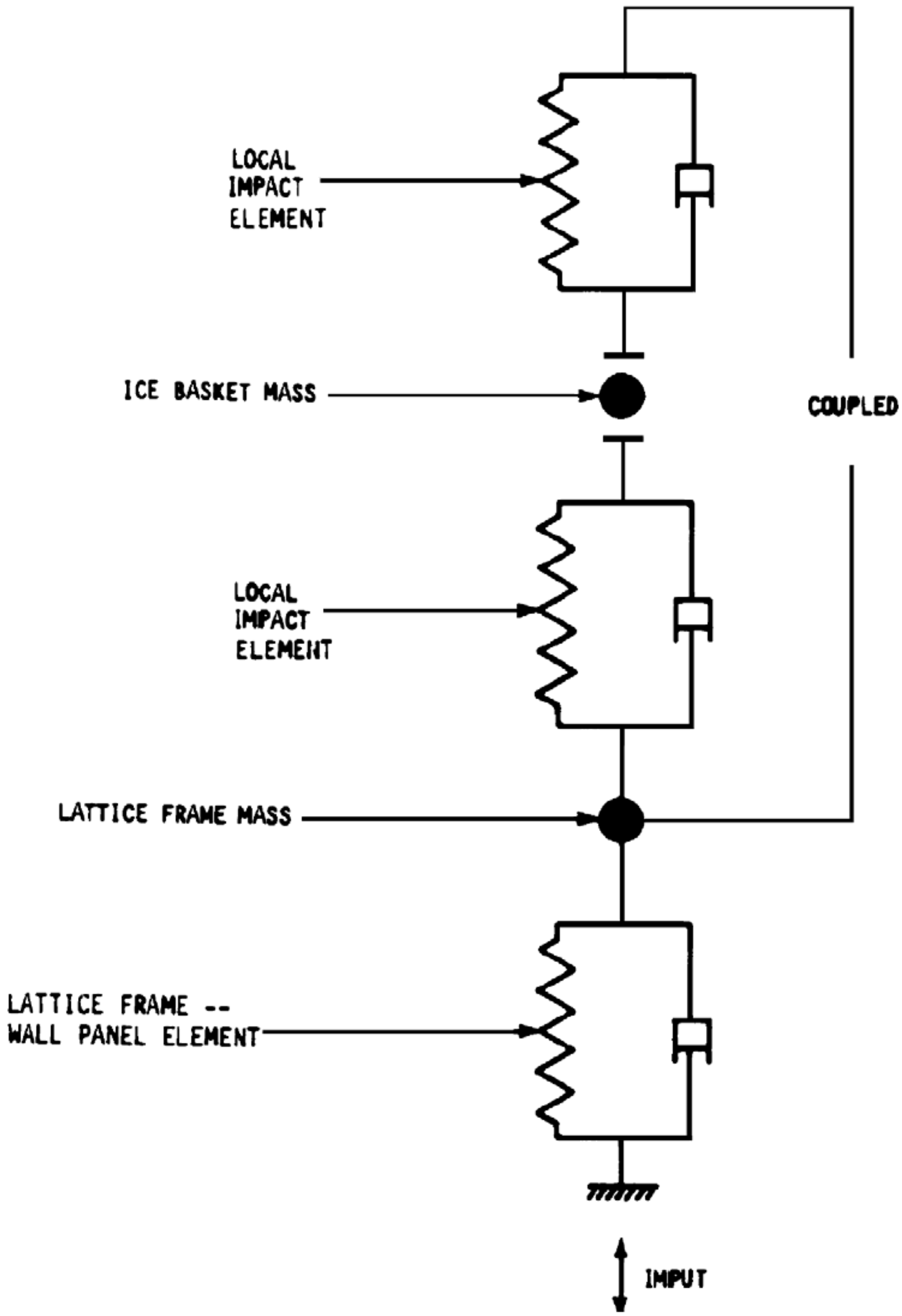
Figure 6.7-46



WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

Typical Displacement Time
Histories for 12 Foot Basket
With End Supports
Pluck Test

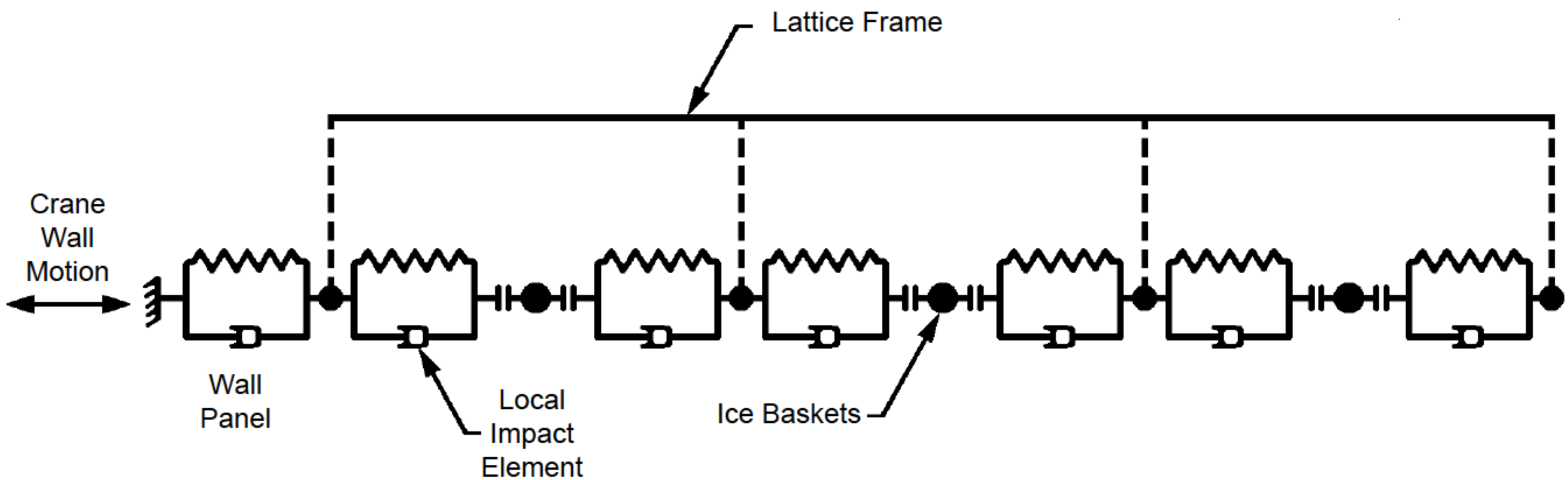
Figure 6.7-47



WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

Non Linear
Dynamic Model

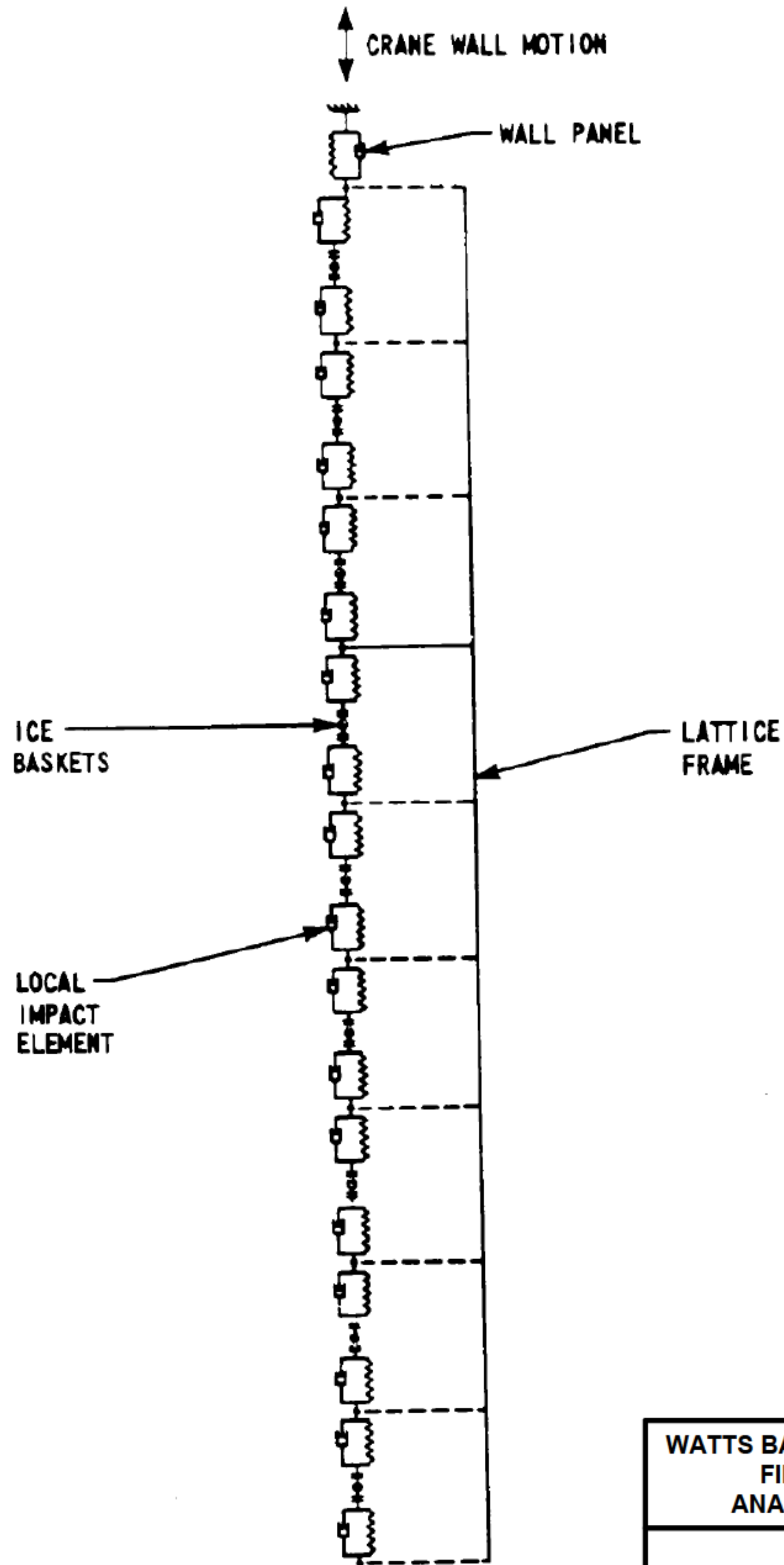
Figure 6.7-48



WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

3 Mass Tangential
Ice Basket Model

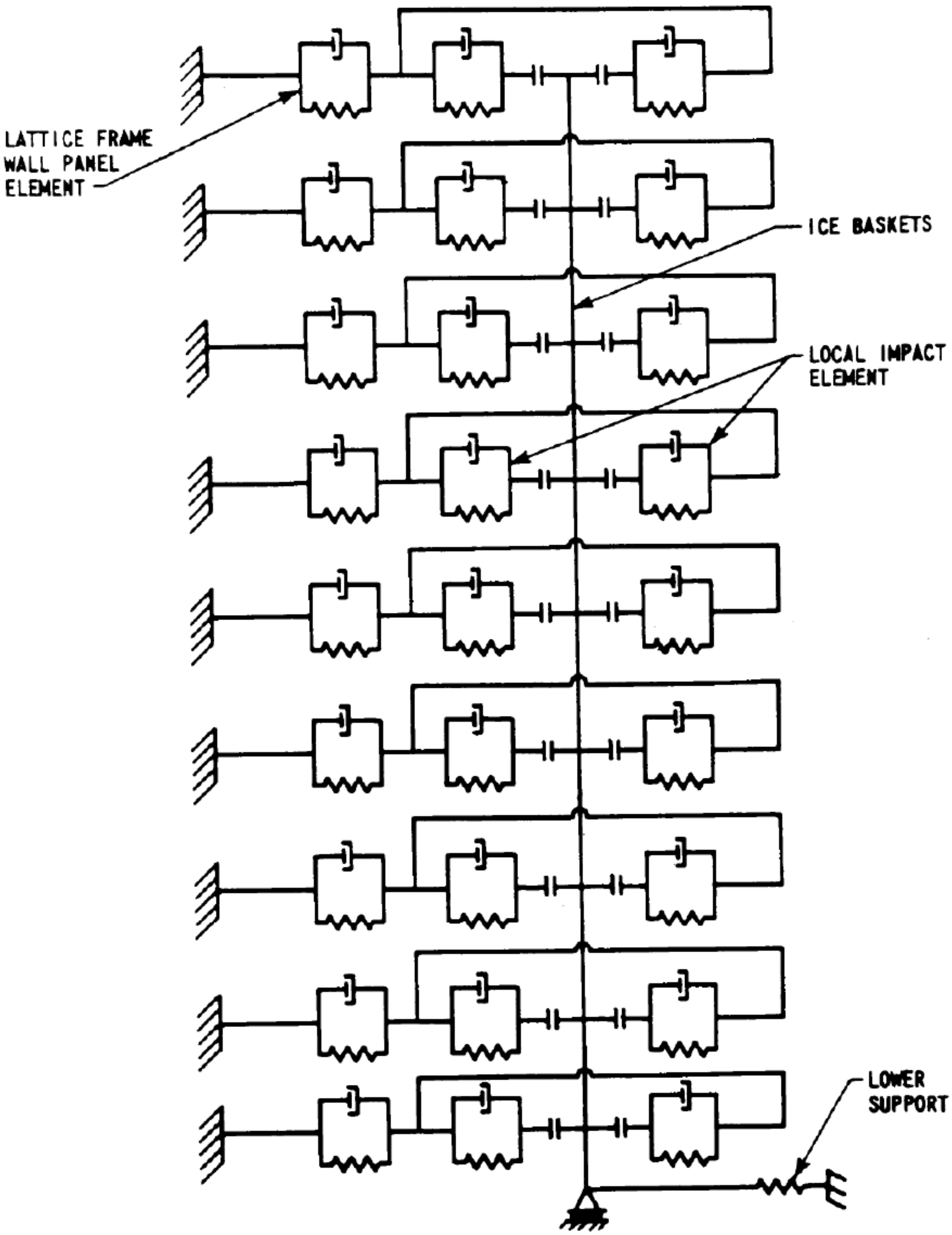
FIGURE 6.7-49



WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

9 Mass
Radial Ice Basket

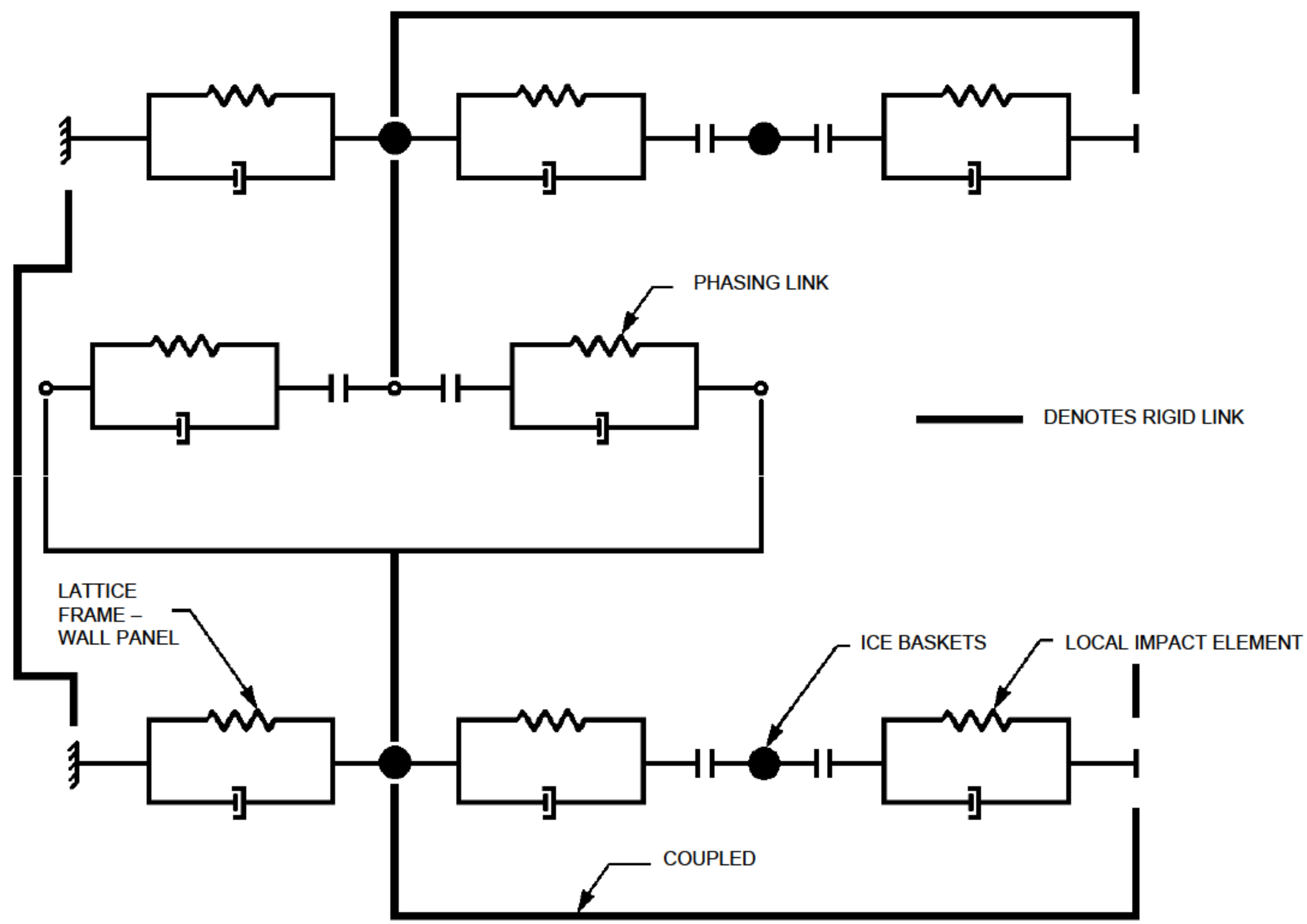
Figure 6.7-50



WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

48 Foot
Beam Model

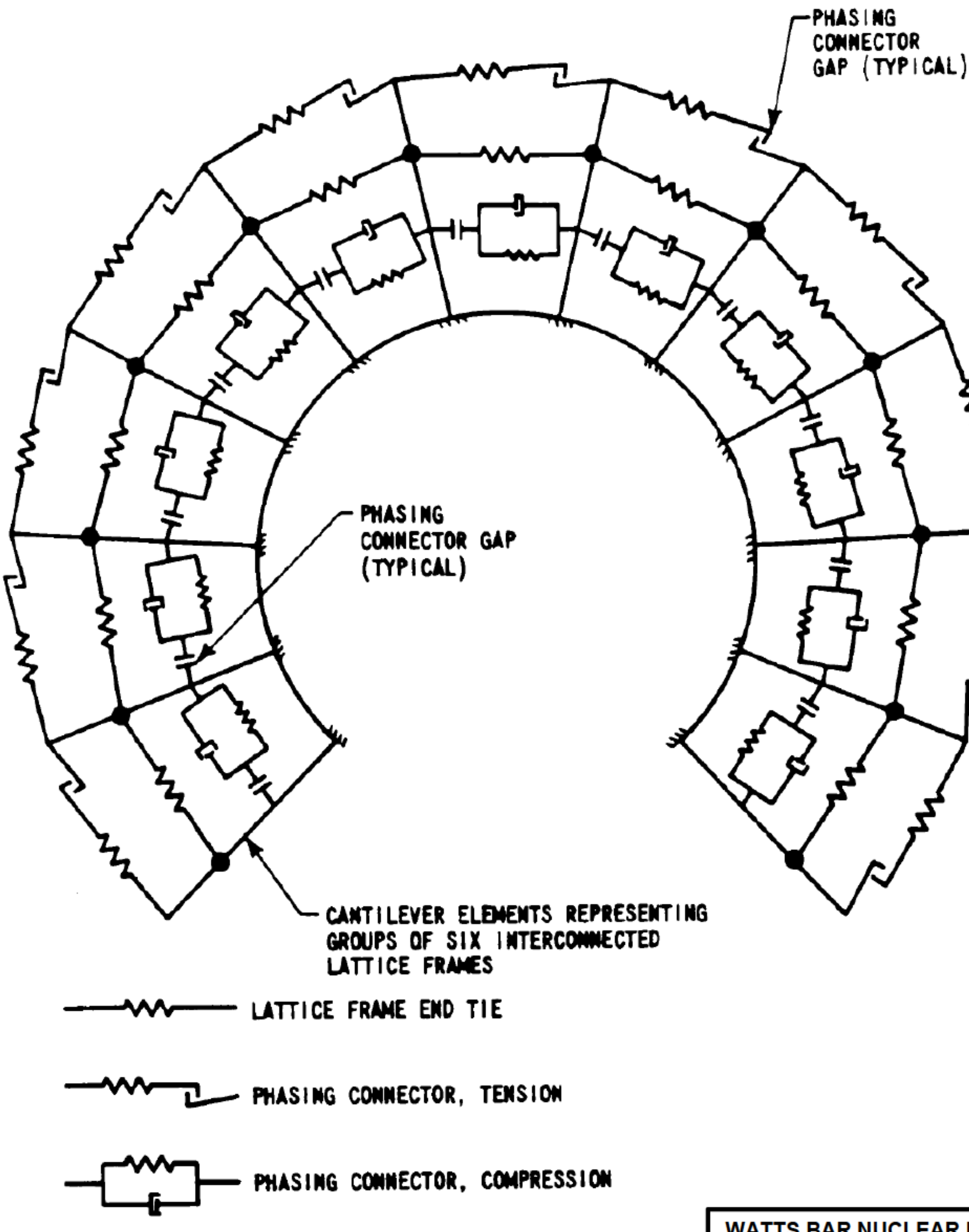
Figure 6.7-51



WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

Phasing Mass Model of
Adjacent Lattice Frame Bays

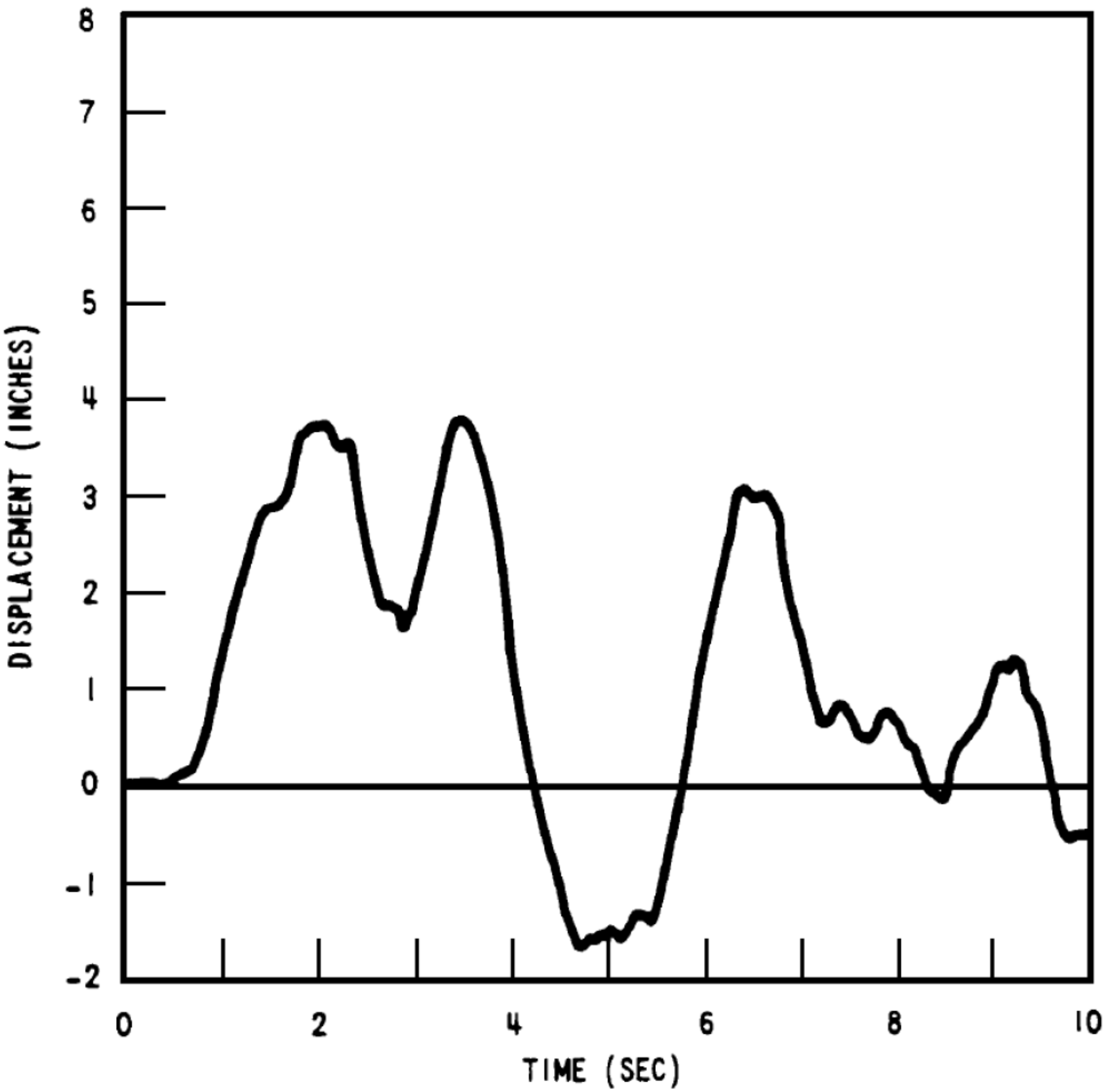
FIGURE 6.7-52



WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

Phasing Study Model, 1 Level
Lattice Frame 300 degrees
Non-Linear Model

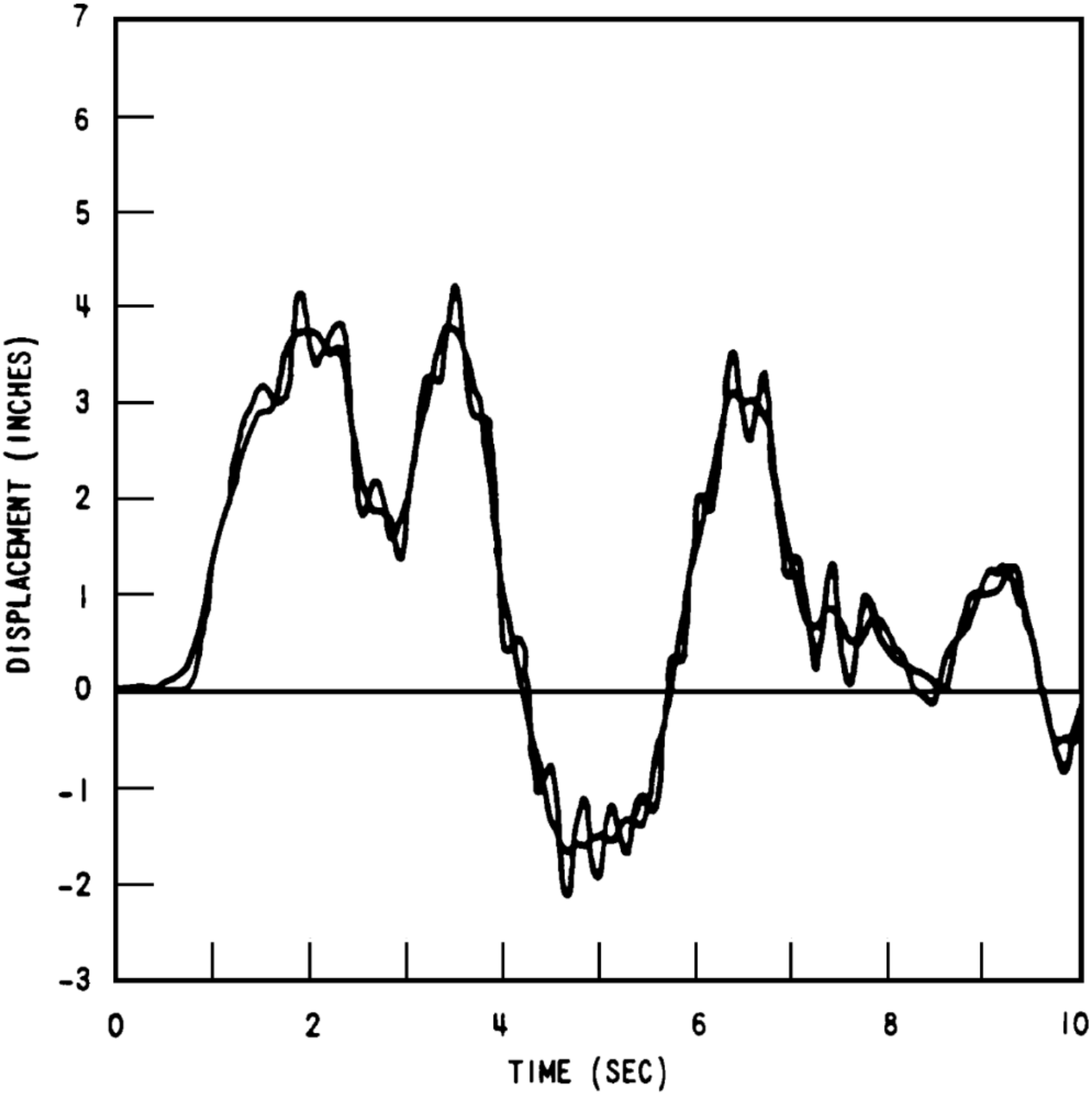
Figure 6.7-53



WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

Typical Crane Wall
Displacement

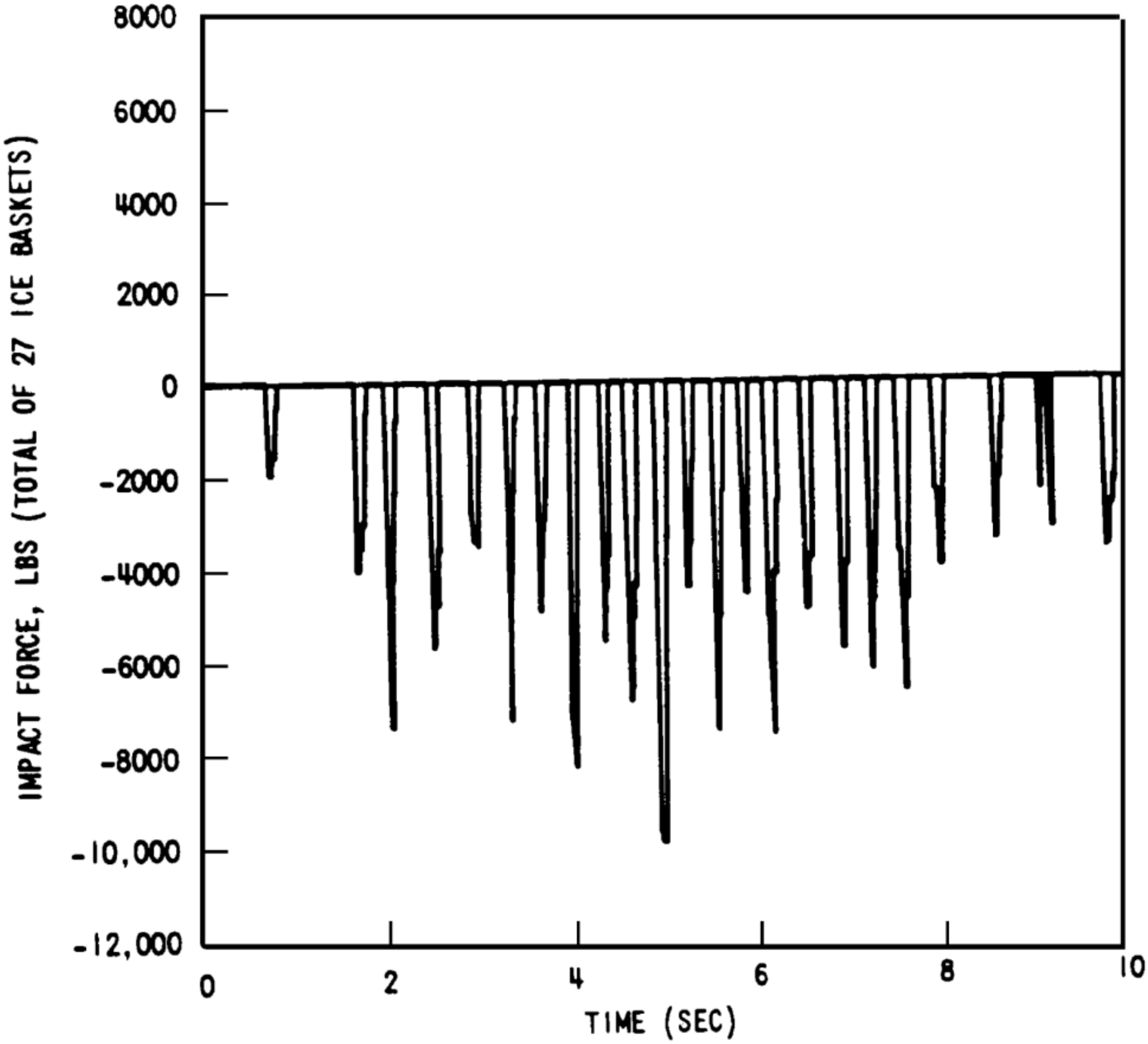
Figure 6.7-54



WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

Typical Ice Basket
Displacement Response

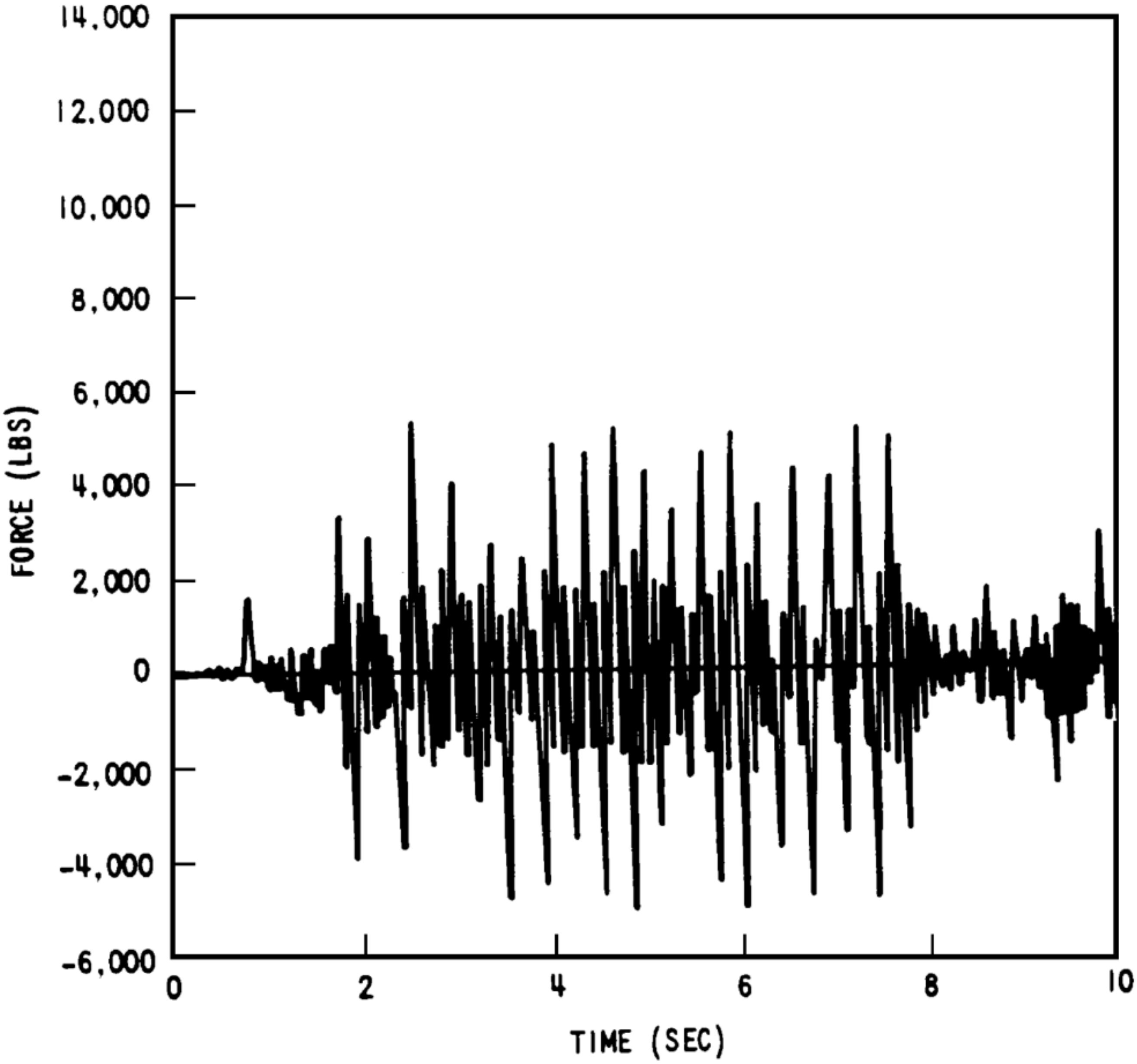
Figure 6.7-55



WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

Typical Ice Basket Impact
Force Response

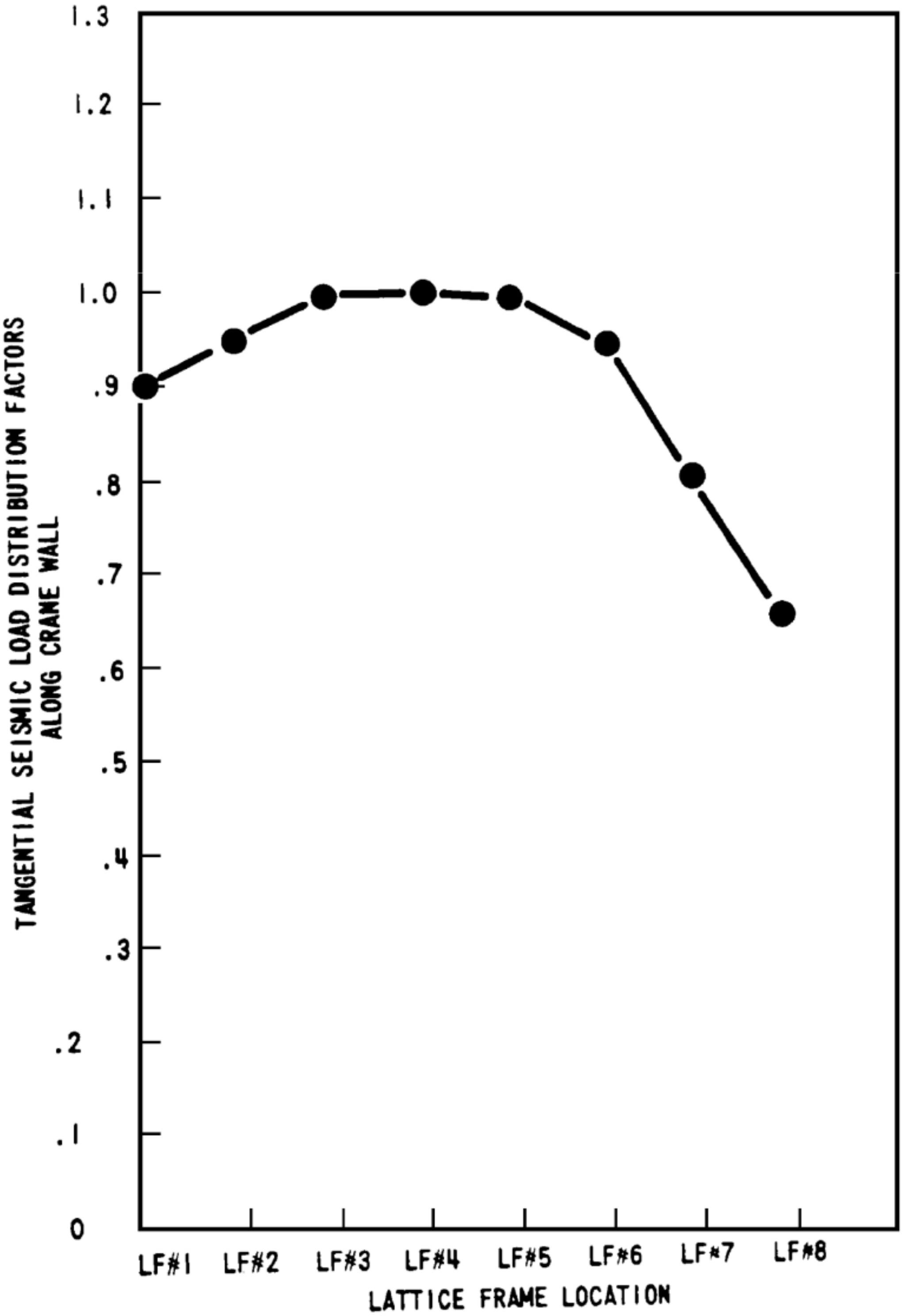
Figure 6.7-56



WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

Typical Crane Wall
Panel Load Response

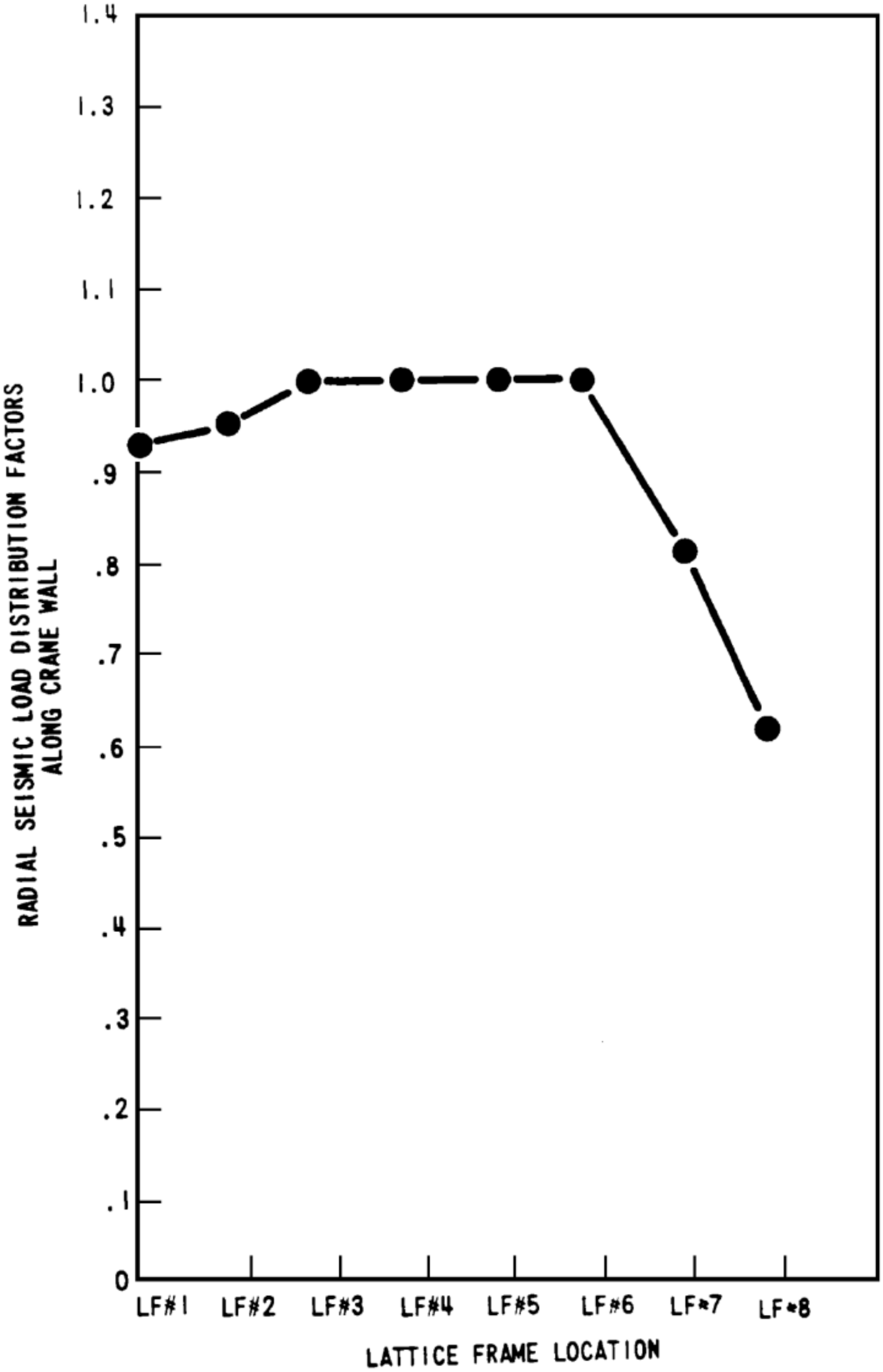
Figure 6.7-57



WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

Wall Panel Design Load Distribution Obtained Using the 48 Foot Beam Model Tangential Case

Figure 6.7-58



WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

Wall Panel Design Load Distribution Obtained Using the 48 Foot Beam Model Radial Case

Figure 6.7-59

6.8 AIR RETURN FANS

6.8.1 Design Bases

UNIT 1

The primary purpose of the air return fan system is to enhance the ice condenser and containment spray heat removal operation by circulating air from the upper compartment to the lower compartment, through the ice condenser, and then back to the upper compartment. This operation takes place at the appropriate time (Section 6.7) following a design-basis accident, which includes a LOCA. The secondary purpose of the system is to limit hydrogen concentration in potentially stagnant regions by ensuring a flow of air from these regions.

UNIT 2

The primary purpose of the air return fan system is to enhance the ice condenser and containment spray heat removal operation by circulating air from the upper compartment to the lower compartment, through the ice condenser, and then back to the upper compartment. This operation takes place at the appropriate time (Section 6.7) following a LOCA or other high energy line break within the containment. The secondary purpose of the system is to limit hydrogen concentration in potentially stagnant regions by ensuring a flow of air from these regions (Section 6.2.5).

6.8.2 System Description

Two 100% capacity air return fans, one redundant, are provided to remove air from the upper compartment through the divider deck to an accumulator room of the lower compartment. The discharged air flows from each accumulator room through the annular equipment areas into the lower compartment. Any steam produced by residual heat mixes with the air and flows through the lower inlet doors of the ice condenser. The steam portion of the mixture condenses as long as ice remains in the ice condenser and the air continues to flow into the upper compartment through doors at the top of the ice condenser. Air return fan suction side is equipped with a non-return damper which prevents flow from the lower compartment to the upper compartment during the initial stages of a LOCA.

Both fans are designed to start 9 $\pm$ 1 minutes after receipt of a Phase B isolation signal. In addition, either fan may be controlled manually from the main control room. Each fan can develop sufficient head to keep the non-return dampers and ice condenser inlet doors open after blowdown is complete.

The design life of the air return system is 40 years under normal (standby) conditions which are 120°F temperature, 100% relative humidity for brief periods of time, and an integrated radiation dose of 2×10^7 rads. The fan motors contain motor space heaters which operate normally to prevent condensation within the motor even when the ambient relative humidity is at 100%. Materials of the system are essentially steel, coated to prevent corrosion.

The system is designed to operate continuously during accident conditions. The air return fan system is an engineered safety feature and meets the qualification requirements for Seismic Category I. The design of the fans and controls of each 100% capacity system meets the intent of Regulatory Guides 1.29 and 1.53. Each air return fan is direct drive, vaneaxial, with a capacity of not less than 41,690 cfm. Each is driven by a 460-volt, 3-phase electric motor which

develops 100 horsepower at 1,770 rpm.

The non-return dampers are heavy duty and are designed to prevent airflow from the lower compartment to the upper compartment without first going through the ice condenser, under a differential pressure of 15 psig. The dampers are controlled to open when the differential pressure across the operating fan assures airflow from the upper to lower compartment. The gravity-loaded damper fails in the closed position upon loss of necessary flow head, and has a leakage area at 15 psig differential pressure of not more than 5.6 square inches. The position of the damper is monitored in the main control room.

Simultaneously with the return of air from the upper compartment to the lower compartment, post-severe accident hydrogen mixing capability is provided by the air return fan system in the following regions of the containment: containment dome, each of the four steam generator enclosures, pressurizer enclosure, upper reactor cavity, each of the four accumulator rooms, and the instrument room. These regions are served by separate hydrogen collection headers which terminate on the suction side of each of the two air return fans. A schematic of this system is shown in Figure 9.4-28. The minimum design airflow from each region is sufficient to limit the local concentration of hydrogen to not more than the allowable volume percent range as specified in Section 6.2.5.2. Minimum design flow rates are shown in Figure 9.4-28.

The header systems are airflow-balanced prior to initial plant operation to assure that the actual airflows are at least equal to the minimum design flow when either or both fans are in operation.

6.8.3 Safety Evaluation

UNIT 1

The design bases of the fans are to reduce containment pressure after blowdown from a LOCA or a non-LOCA pipe break, prevent excessive hydrogen concentrations in pocketed areas, and circulate air through the ice condenser. The containment air return fans turn on 9 $\pm$ 1 minute after Phase B containment isolation signal. Peak containment pressure, about 9.36 psig, is attained at approximately 8,959 seconds. The fans provide a continuous mixing of containment compartment atmosphere for the long-term post-blowdown environment. Mixing of the compartment atmospheres helps to bring fission products in contact with the ice bed and/or the upper compartment spray for removal from the containment atmosphere. The fans also aid in mixing the containment atmosphere to preclude hydrogen pocketing, which is assumed to be produced as a result of the accident.

Each fan located in the lower compartment, when operating alone, transfers 40,000 cfm from the upper compartment into the lower compartment and circulates 1,690 cfm from the enclosed areas in the lower compartment through the hydrogen collector duct headers to prevent excessive localized hydrogen buildup following a DBA. A back-draft damper, normally closed, is located upstream of each deck fan to prevent reverse flow during the initial loss-of-coolant blowdown.

The air return fans have sufficient head to overcome the compartment differentials that occur after the reactor coolant system blowdown. The fan head is sufficient to overcome the density effects of steam generation and resistance to airflow through the ice condenser and other system losses. After complete ice bed melt out, each fan has sufficient head to deliver 41,690 cfm with the containment pressurized to the design pressure rating.

The fans are designed to withstand the post-accident containment environment. Two 100% capacity air return systems are provided. Thus, if one fan fails, the other provides the necessary air flow from the upper to lower compartment. System redundancy also assures that the minimum design air flows required for hydrogen mixing capability are achieved even during operation of only one air return fan. As seen in Figure 9.4-28, the three main headers which serve the steam generator enclosures, pressurizer enclosure, accumulator rooms, and instrument room interconnect the suction side of each fan (downstream of the non-return damper). This arrangement permits flow from each compartment even if only one fan is in operation. The upper reactor cavity and containment dome areas have separate headers connected to each fan which accomplishes the same objective when only one fan is in operation.

UNIT 2

The design bases of the fans are to reduce containment pressure after blowdown from a LOCA or other high energy line break, prevent excessive hydrogen concentrations in pocketed areas, and circulate air through the ice condenser. The containment air return fans turn on 9 + 1 minute after Phase B containment isolation signal. Peak containment pressure, about 11.73 psig, is attained at approximately 3600 seconds. The fans provide a continuous mixing of containment compartment atmosphere for the long-term post-blowdown environment. Mixing of the compartment atmospheres helps to bring fission products in contact with the ice bed and/or the upper compartment spray for removal from the containment atmosphere. The fans also aid in mixing the containment atmosphere to preclude hydrogen pocketing, which is assumed to be produced as a result of the severe accident.

Each fan located in the lower compartment, when operating alone, transfers 40,000 cfm from the upper compartment into the lower compartment and circulates 1,690 cfm from the enclosed areas in the lower compartment through the hydrogen collector duct headers to prevent excessive localized hydrogen buildup following a DBA. A backdraft damper, normally closed, is located upstream of each deck fan to prevent reverse flow during the initial LOCA or other high energy line blowdown.

The air return fans have sufficient head to overcome the compartment differentials that occur after the reactor coolant system blowdown. The fan head is sufficient to overcome the density effects of steam generation and resistance to airflow through the ice condenser and other system losses. After complete ice bed melt out, each fan has sufficient head to deliver 41,690 cfm with the containment pressurized to the design pressure rating.

The fans are designed to withstand the post DBA environment and were shown to survive the beyond-design-basis accident containment environment (Section 6.2.5).

Two 100% capacity air return systems are provided. Thus, if one fan fails, the other provides the necessary air flow from the upper to lower compartment. System redundancy also assures that the minimum design air flows required for hydrogen mixing capability are achieved even during operation of only one air return fan. As seen in Figure 9.4-28, the three main headers

which serve the steam generator enclosures, pressurizer enclosure, accumulator rooms, and instrument room interconnect the suction side of each fan (downstream of the non-return damper). This arrangement permits flow from each compartment even if only one fan is in operation. The upper reactor cavity and containment dome areas have separate headers connected to each fan which accomplishes the same objective when only one fan is in operation.

6.8.4 Inspection and Testing

Preoperational performance tests are addressed in Chapter 14. Inservice tests and inspections are included in the Technical Specifications.

6.8.5 Instrumentation Requirements

UNIT 1

The essential instrumentation requirements are that at least one of the air return fans start at the appropriate time after LOCA and that the fan keeps running for one year. Instrumentation design details are shown on Figures 9.4-30 and 9.4-33. The logic, controls, and instrumentation of this engineered safety feature system are such that a single failure of any component does not result in the loss of functional capability for the system.

UNIT 2

The essential instrumentation requirements are that at least one of the air return fans start at the appropriate time after receipt of a Phase B isolation signal and that the fan is capable of running for one year. Instrumentation design details are shown on Figures 9.4-30 and 9.4-33. The logic, controls, and instrumentation of this engineered safety feature system are such that a single failure of any component does not result in the loss of functional capability for the system.

REFERENCES

None

6.9 MOTOR-OPERATED VALVE (MOV) PROGRAMS

The WBN MOV program elements were developed using the guidance provided in the following generic letters (GL):

GL 89-10, "Safety-Related Motor Operated Valve Testing and Surveillance"

A comprehensive MOV design bases, testing, and surveillance program has been established. This program provides for testing, inspection, and maintenance of safety-related MOVs and certain other MOVs in safety-related systems to provide the necessary assurance that the valves function when subjected to the design basis conditions that are to be considered during both normal operation and abnormal events. See Reference [1] for specific details.

GL 95-07, "Pressure locking and Thermal Binding of Safety Related Power-Operated Gate Valves"

The operational configurations of safety-related active gate valves were evaluated to identify those valves susceptible to either pressure locking or thermal binding. Corrective actions by modification or administrative controls were taken to ensure that these valves were capable of performing their intended safety functions. See Reference [2] for specific details.

GL 96-05, "Periodic Verification of Design Basis Capability of Safety-Related Motor-Operated Valves"

The Joint Owners Group (JOG) MOV Periodic Verification (PV) Program will be used to verify that the safety-related MOVs will continue to be capable of performing their safety functions within the current licensing bases of the facility. The JOG interim PV program is described in Topical Report (TR) MPR-1807 and has been completed. The final long term JOG PV program is described in the final TR MPR-2524, and has been endorsed in NRC's Safety Evaluation (SE) dated September 25, 2006. References [2 and 4]

REFERENCES

1. Generic Letter No. 89-10, "Safety Related Motor-Operated Valve Testing and Surveillance," June 28, 1989 and supplements.

2. Safety Evaluation for Watts Bar Nuclear Plant Unit 1 Response to Generic Letter 95-07, "Pressure Locking and Thermal Binding of Safety Related Power Operated Gate Valves," dated September 15, 1999.

3. Safety Evaluation for Watts Bar Nuclear Plant, Unit 1 Response to Generic Letter 96-05, "Periodic Verification of Design-Basis Capability of Safety-Related Motor-Operated Valves," dated July 21, 1999.

4. “Final Safety Evaluation on Joint Owners’ Group Program on Motor-Operated Valve Periodic Verification (TAC Nos. MC2346, MC2347, and MC2348),” Dated September 25, 2006.