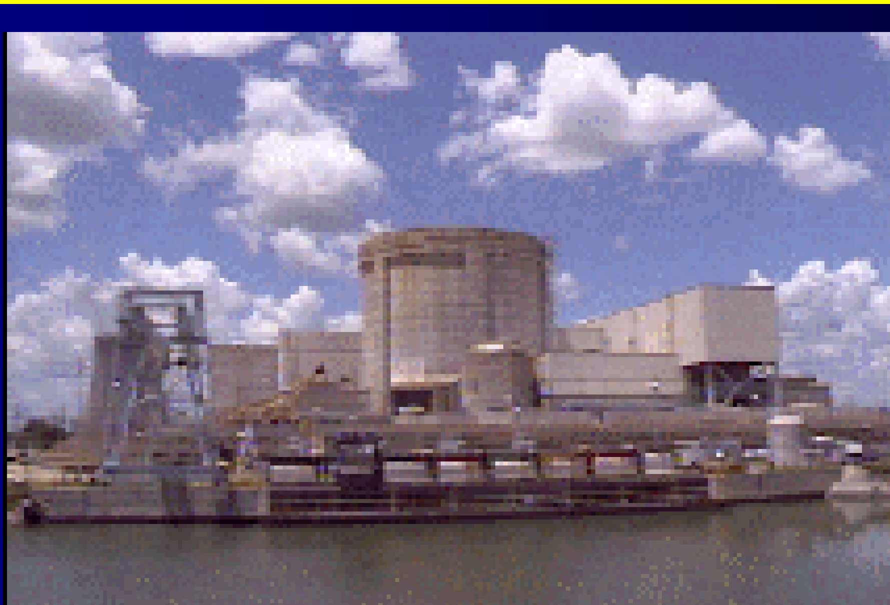
Defueled Safety Analysis Report

- **Duke Energy Florida**
- Crystal River Unit 3

This is Revision 2 of the Living DSAR.
All changes incorporated within are fully approved and this version should be utilized for Quality Activities.



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1. INTRODUCTION AND SUMMARY

1.1 INTRODUCTION

1.1.1 GENERAL INFORMATION

As documented in Nuclear Regulatory Commission (NRC) to Crystal River Nuclear Plant (CR3) letter dated March 13, 2013 (ADAMS Accession No. ML13058A380), the NRC has acknowledged CR3's certification of permanent cessation of power operation and permanent removal of fuel from the reactor vessel. The functional requirements of the majority of plant components no longer meet the definition of Safety Related in 10 CFR 50.2. As a result, the safety classification of the majority of plant components has been revised and updated in the Equipment Database (EDB). The only revision to affected plant documentation as a result of the cessation of power operation is the addition of this disclaimer for Defueled Safety Analysis Report (DSAR)-described Systems, Structures, and Components (SSCs) which indicates the EDB safety classification information has precedence over the safety classification of components described in the DSAR. This disclaimer is expected to be removed from the DSAR after the completion of the system abandonment process.

All spent nuclear fuel is in dry storage in the ISFSI and all new fuel has been shipped offsite to a new owner. As such, no SSCs at CR3 meet the definition of safety related in accordance with 10 CFR 50.2.

The DSAR was developed as the principal licensing source document describing pertinent equipment, structures, systems, operational constraints and practices, accident analyses, and decommissioning activities associated with the defueled status of the Crystal River Unit 3 Nuclear Generating Plant. As such, the DSAR is intended to serve in the same role as the Final Safety Analysis Report (FSAR) during the period of power operation.

The predecessor to the DSAR, the FSAR, was originally submitted in support of Florida Power Corporation's (FPC's) application for an operating permit and facility license for the one nuclear unit addition to the Crystal River Site located on the Gulf of Mexico in Citrus County, Florida, and designated Unit 3.

The organization of this report is in accordance with the United States Atomic Energy Commission (AEC) "A Guide for the Organization and Contents of Safety Analysis Reports," dated June 30, 1966. This report is responsive to that guide, the proposed AEC 70 Design Criteria, and to considered pertinent questions asked of other applicants up until the time of the FSAR submittal for the operating license. Revisions of 10 CFR 50 current at the time of original submission (FPC's letter dated January 25, 1971 to the Director, Division of Licensing, USAEC) were utilized in preparing this report.

Construction of Crystal River Unit 3 (CR3) was authorized by the AEC through issue of provisional construction permit CPPR-51 on September 25, 1968 in Docket 50-302. Construction of CR3 was completed and the operating license issued December 3, 1976. Fuel was loaded in 1976. CR3 last produced power in September 2009, while shutting down for Refuel 16. During activities to replace steam generators, a portion of the containment concrete wall delaminated. While completing repairs additional delamination occurred. CR3 was officially retired on February 5, 2013.

The CR3 Nuclear Steam Supply System (NSSS) is a pressurized water reactor type. It used chemical shim for reactivity control and generated steam with a small amount of superheat in Once-Through Steam Generators (OTSG). The NSSS and nuclear fuel were supplied by Babcock & Wilcox Company (B&W). The general arrangement of plant equipment and structures is shown in Figure 1-2 through Figure 1-21.

CR3 has an Independent Spent Fuel Storage Installation (ISFSI) located on the east berm of the plant. The ISFSI has the capacity for 40 Dry Shielded Canisters (DSCs), each holding up to 32 fuel assemblies. The ISFSI consists of the NUHOMS Reinforced Concrete Horizontal Storage Modules, each containing one 32PTH1-TYPE 2W DSC, manufactured for CR3 by Areva TransNuclear Corporation, under Certificate of Compliance 1004, Amendment Number 14. The 10 CFR 72.212 Report that documents how the site meets Part 72 has been issued as procedure ISFS-0212.

Duke Energy Florida (DEF) is fully responsible for the complete safety and adequacy of the plant. Aid in the design, construction, startup, and testing of CR3 was supplied principally by Gilbert Associates, Inc. (GAI) and B&W. Assistance was also rendered by other consultants and suppliers as required.

1.1.2 CRYSTAL RIVER UNIT 3 DSAR REVISIONS (SUBMITTED TO NRC)

Revision 0, Submitted August 31, 2017

Revision 1, Submitted May 24, 2018

1.2 SUMMARY PLANT DESCRIPTION

1.2.1 SITE CHARACTERISTICS

The 4,738-acre site is characterized by a 4,400-foot minimum exclusion radius centered on the Reactor Building; isolation from nearby population centers; sound foundation for structures; an abundant supply of cooling water; an ample supply of emergency power; and favorable conditions of hydrology, geology, seismology, and meteorology.

1.2.2 POWER LEVEL

The unit was operated at core power levels up to 2609 MWt corresponding to a maximum continuous gross electrical output of 924 MWe (nominal, peak cycle efficiency periods). Beginning with Fuel Cycle 16, a thermal power of 2609 MWt was used for fuel design. Licensed power level was 2609 MWt. Site parameters, and principal structures are evaluated for a core output of 2609 MWt, unless otherwise specified. All radiological analyses have been performed based on a licensed power limit of 2609 MWt.

1.2.3 PEAK SPECIFIC POWER LEVEL

The peak specific power level in the fuel for operation at 2609 MWt resulted in a maximum thermal output of 17.63 kW per ft of fuel rod.

1.2.4 REACTOR BUILDING

The leak-tight structure required to contain the Design Basis Loss-of-Coolant Accident (LOCA) was the Reactor Building.

The Reactor Building is a cylindrical reinforced concrete structure bearing on a sound foundation. The foundation slab is reinforced with conventional steel reinforcing. The cylindrical walls are prestressed with a post-tensioning tendon system in the vertical and horizontal directions. The dome roof is prestressed utilizing a three-way, post-tensioning tendon system. The inside surface of the Reactor Building is lined with a carbon steel liner to ensure a high degree of leak-tightness for containment.

The Reactor Building was left in a condition that will assure structural integrity throughout the SAFSTOR period.

The Reactor Building is similar in design to the containment buildings for the Three Mile Island Nuclear Station Unit 1 (Docket No. 50-289), the Turkey Point Plant (Docket Nos. 50-250 and 251), the Palisades Plant (Docket No. 50-255), the Point Beach Plant (Docket No. 50-226), and the Oconee Nuclear Station (Docket Nos. 50-269, 50-270, and 50-287).

1.2.5 ELECTRICAL SYSTEMS AND EMERGENCY POWER

Crystal River Unit 3 has the following sources of electric power:

a. Multiple 230 kV transmission lines.

b. Either Units 2 or 4 at the Crystal River Site.

The sources of power and associated electrical equipment will ensure safe functioning of the plant auxiliary loads (refer to Chapter 3).

1.3 DESIGN CHARACTERISTICS

1.3.1 DESIGN CHARACTERISTICS

The important design and operating characteristics of the nuclear steam supply system (NSSS) for Crystal River Unit 3 are no longer important for plant/ISFSI activities.

1.3.2 SIGNIFICANT DESIGN REVISIONS

The more significant design revisions made to the unit between the Preliminary Safety Analysis Report (PSAR) and the issuance of the Operating License are listed below.

1.3.2.1 Fuel Assembly

The initial Fuel Assembly design utilized Inconel spacer grids supported by the control rod guide tubes and center instrument tube rather than stainless steel grids supported by an external stainless steel perforated can. The "canless" assembly with Inconel grids was chosen to increase the neutron economy of the core, to improve local flow distribution, and to utilize the increased yield strength at operating temperature of Inconel versus stainless steel. All fuel rods are internally pressurized with helium. This change was made to improve the heat transfer characteristics and structural integrity of the fuel rod. Beginning with Batch 9 fuel assemblies first irradiated in Cycle 7, the Mark BZ fuel design was adopted. The Mark BZ fuel assembly design uses zirconium spacer grids. These grids allow greater fuel efficiency due to zirconium's lower neutron cross-section. Along with the Mark BZ fuel design, the Cycle 7 assemblies also include lower pre-pressure and annealed structural tubing. Lower fuel pin pre-pressure compensates for increased pressure from greater irradiation resulting from longer cycle lengths. Annealed structural tubing minimizes dimensional growth caused by irradiation.

The Mark-B10ZL Fuel Assembly was introduced into the CR3 core starting with Cycle 10, Batch 12. These fuel assemblies contain cruciform leaf-type holddown springs to provide an increase in the holddown spring load. The total holddown spring load, represented by a full core of Mark-B10ZL assemblies was evaluated for the effect on the reactor internals. The effect of the increased load was found acceptable.

Starting with Cycle 13, Batch 15, the Mark-B10 fuel assembly cruciform holddown spring was modified from an eight leaf design to a six leaf design, to reduce the stress on the fuel assembly guide tubes. Evaluation of the six leaf holddown spring indicates an adequate positive holddown margin remains. The total holddown spring load of a full core of the six leaf design Mark-B10 assemblies remains bounded by the eight leaf design analysis.

The Mark-B-HTP fuel assembly was introduced into CR3 in Cycle 14. This design incorporated, among other things, the use of M5 and substantial changes to spacer grid design.

<u>Fuel Assemblies</u>

Fuel Assemblies are designed for structural adequacy and reliable performance during core operation, handling, and shipping. Design criteria for core operation include steady state and transient conditions under combined effects of flow induced vibration, temperature gradients, and seismic disturbances. Topical Report BAW-10035, "Fuel Assembly Stress and Deflection Analysis for Loss-of-Coolant Accident and Seismic Excitation,"

summarizes an analysis of the Fuel Assembly for loads caused by the depressurization transient following an instantaneous RCS pipe rupture and/or seismic excitation.

Spacer grids located along the length of the Fuel Assembly position fuel rods in a square array and are designed to maintain fuel rod spacing during core operation, handling, shipping, and storage. Spacer grid-to-fuel rod contact loads are established to minimize fretting and allow axial relative motion resulting from fuel rod irradiation growth and differential thermal expansion.

For the Independent Spent Fuel Storage Installation (ISFSI) loading campaign, distorted fuel assemblies that will not fit within the cell of the selected Dry Shielded Canister (DSC) may be recaged. Recaging is performed on fuel assemblies by relocating the fuel rods from the distorted fuel assembly to a new Mark-B-HTP-1 or Mark-B-HTP cage. Engineering Change (EC) 408309 documents the equivalency of the Fit, Form and Function of the replacement cages. Spent fuel assemblies with damaged grid straps were addressed in procedure ISFS-0210, Fuel Selection Procedure, which assessed the damage and classified the fuel as intact or damaged.

1.3.2.1.1 Fuel Assembly Description

a. General

The fuel is sintered, low-enriched uranium dioxide cylindrical pellets. The pellets are clad in zirconium alloy tubing and sealed by zirconium alloy end caps, welded at each end. The clad, fuel pellets, end caps, and fuel support components form a "fuel rod." Two hundred and eight fuel rods, sixteen control rod guide tubes, one instrumentation tube assembly, eight spacer grids, and two end fittings comprise the basic "Fuel Assembly" (see Figure 1-22, Figure 1-23 and Figure 1-24). The guide tubes, spacer grids, and end fittings form a structural cage to arrange the rods and tubes in a 15 x 15 array. The center position in the assembly is reserved for instrumentation.

CR3 has 1,243 spent nuclear fuel assemblies stored at the site. The fuel assemblies are stored in dry storage on an Independent Spent Fuel Storage Installation (ISFSI) pad. The ISFSI is described in Section 1.1.1. In addition to the spent fuel assemblies, CR3 has one failed fuel basket stored within the ISFSI in a dry shielded canister with spent fuel.

Control components such as Control Rod Assemblies (CRA), Axial Power Shaping Rod Assemblies (APSRA), Burnable Poison Rod Assemblies (BPRA), and Orifice Rod Assemblies (ORA) are stored within selected fuel assemblies. Primary neutron source and regenerative neutron source assemblies are also stored within selected fuel assemblies.

b. Fuel Rod

The fuel is in the form of sintered and ground pellets of low enriched uranium dioxide. Pellet ends are dished to minimize differential thermal expansion between the fuel and cladding. Radial growth of the fuel during burnup is accommodated by pellet porosity, radial clearance between the pellets and the cladding, and by a small amount of permanent strain in the cladding. Fuel growth is calculated by the method given in Reference 2. All fuel rods are internally pressurized with helium. Starting with the Batch 9 fuel assemblies, internal fuel rod pre-pressure was lowered. The lower pre-pressure is used to compensate for increased pressure from greater irradiation caused by longer cycle lengths.

c. Fuel Assembly

1. General

   The Fuel Assemblies shown in Figure 1-22, Figure 1-23 and Figure 1-24 are of the canless type where the eight spacer grids, end fittings, and the guide tubes form the basic structure. Fuel rods are supported at each spacer grid by contact points integral with the walls of the cell boundary. The guide tubes are permanently attached to the upper and lower end fittings. Use of similar material in the guide tubes and fuel rods results in minimum differential thermal expansion.

2. Spacer Grids

   Spacer grids are constructed from strips which are slotted and fitted together in "egg crate" fashion. Each grid has 32 strips, 16 perpendicular to 16, which forms the 15 x 15 lattice. The square walls formed by the interlaced strips provide support for the fuel rods in two perpendicular directions. Contact points on the walls of each square opening are integrally punched in the strips. On each of the two end spacer grids, the peripheral strip is extended and rigidly attached to the respective end fitting. Batch 9 assemblies were the first to use zirconium spacer grids. The use of zirconium leads to greater fuel efficiency because of its lower thermal neutron cross-section.

3. Lower End Fitting

   The lower-end fitting positions the assembly in the lower core grid plate. The lower ends of the fuel rods rest on the grid of the lower end fitting. Penetrations in the lower end fitting are provided for attaching the control rod guide tubes and for access to the instrumentation tube.

4. Upper End Fitting

   The upper-end fitting positions the upper end of the Fuel Assembly in the upper core grid plate structure and provides means for coupling the handling equipment. An identifying number on each upper-end fitting provides positive identification.

   An internal hollow post in the center of the end fitting provides means for retention of either an ORA or a BPRA.

   Integral with each upper-end fitting is a hold-down spring and spider to provide a positive hold-down margin to oppose hydraulic forces.

   Penetrations in the upper-end fitting grid are provided for the guide tubes.

5. Guide Tubes

   The Zircaloy guide tubes provide continuous guidance to the control rods when inserted in the Fuel Assembly during operation and provide the structural continuity for the Fuel Assembly. Welded to each end of a guide tube are flanged and threaded sleeves, to secure the guide tubes to each end fitting by lock-welded nuts. Transverse location of the guide tubes is provided by the spacer grids. All guide tubes are made from annealed zirconium alloy, starting with Batch 9 fuel assemblies. Annealing the tubes results in less deformation and less rod bow than cold-worked tubes at the higher fuel burnups associated with extended operating cycles.

6. Instrumentation Tube

   This Zircaloy tube serves as a channel to guide, position, and contain the incore instrumentation within the Fuel Assembly. The instrumentation probe is guided up through the lower-end fitting to the desired core elevation. It is retained axially at the lower-end fitting by a retainer sleeve. Both the instrumentation tube and retainer sleeve are made from annealed zirconium alloy. The change from cold-worked zirconium alloy will result in less dimensional growth from irradiation. Annealed zirconium alloy instrumentation tubes and retainer sleeves were introduced in Batch 9 fuel assemblies.

7. Spacer Sleeves

   For Batch 13 fuel and earlier batches, the spacer tube segments fit around the instrument tube between spacer grids and prevent axial movement of the spacer grids during primary coolant flow through the Fuel Assembly. Spacer sleeves of annealed zirconium alloy were introduced in Batch 9 assemblies.

1.3.2.2 Reactor Building Liner Specifications

The exterior surface of the liner was not painted.

1.3.2.3 Piping

Power piping complies with B31.1.0 - 1967. "Nuclear" piping other than RCS piping is designed and field hydro-tested in accordance with B 31.1.0 - 1967; but is fabricated, shop-tested, erected, and inspected to B 31.7 - 1969 including Code Case B31-83 (October 1970) by N1, N2, N3 classification. "Nuclear" piping as described above is used in systems that normally contain a radioactive substance.

Alternately, welding fabrication of power piping and Nuclear Class N3 piping may comply with ANSI B31.1-86 or ASME B31.1-2001, 2002 Addenda, Chapter V. Nuclear Classes N1 and N2 piping may comply with ASME Section III–1986, articles NB–4000 and NC–4000, respectively.

All functions of the Intermediate Cooling Water System have been integrated into the Nuclear Services Closed Cycle Cooling System (SW), thereby eliminating the Intermediate Cooling Water System.

The source of supply of seawater for the Raw Water System (RW) has been shifted from the discharge canal to the intake canal.

1.4 PRINCIPAL ARCHITECTURAL AND DESIGN CRITERIA

Crystal River Unit 3 (CR3) has been designed and constructed taking into consideration the proposed 10 CFR 50.34 Appendix A, "General Design Criteria for Nuclear Power Plant Construction Permits" as published in the Federal Register (32FR10213) on July 11, 1967 which are applicable to this unit. As stated in the proposed rule at 32FR10215, "The criteria have been categorized as Category A or Category B. Experience has shown that more definitive information is needed at the construction permit stage for the items listed in Category A than those in Category B." In the discussion of each criterion, references are made to Sections of this DSAR where more detailed information is presented. The principal safety features that meet each criterion are summarized as follows:

On June 19, 1992, the Nuclear Regulatory Commission (NRC) published SECY-92-223, "Resolution of Deviations Identified During the Systematic Evaluation Program." SECY-92-223 proposed three Options to the NRC Commissioners regarding the applicability of the current 10 CFR 50, Appendix A General Design Criteria (GDC). On September 18, 1992, the NRC published a Staff Requirements Memorandum (SRM) that approved Option 1 in which the Staff chose not to apply the current GDC to plants with construction permits issued prior to May 21, 1971 and exemptions from the current GDC are not necessary. Since the CR3 construction permit is dated September 25, 1968, Option 1 of SECY-92-223 is applicable to CR3. The general design criteria presented in the following sections were found by the NRC to be acceptable for the design, construction, and operation of CR3.

Criterion that remain applicable to CR3 are not sequentially numbered and reflect the numbering scheme used in the initial plant licensing documents and as addressed in the CR3 Final Safety Analysis Report.

Following SSC Abandonment Activities the following criterion are those that remain applicable to CR3 during the Decommissioning Phase:

1.4.1 CRITERION 1 - QUALITY STANDARDS (CATEGORY A)

Those systems and components of reactor facilities which are essential to the prevention of accidents which could affect the public health and safety or to the mitigation of their consequences shall be identified and then designed, fabricated, and erected to quality standards that reflect the importance of the safety function to be performed. Where generally recognized codes or standards on design, materials, fabrication, and inspection are used, they shall be identified. Where adherence to such codes or standards does not suffice to assure a quality product in keeping with the safety function, they shall be supplemented or modified as necessary. Quality assurance programs, test procedures, and inspection acceptance levels to be used shall be identified. A showing of sufficiency and applicability of codes, standards, quality assurance programs, test procedures and inspection acceptance levels is required.

DISCUSSION

Essential Systems and Components

The integrity of systems, structures, and components (SSC) essential to accident prevention or mitigation of their consequences has been included in the reactor design evaluations and the quality program described in Section 1.7.

1.4.2 CRITERION 2 - PERFORMANCE STANDARDS (CATEGORY A)

Those systems and components of reactor building facilities which are essential to the prevention of accidents which could affect the public health and safety or to the mitigation of their consequences shall be designed, fabricated, and erected to performance standards that will enable the facility to withstand, without loss of the capability to protect the public, the additional forces that might be imposed by natural phenomena such as earthquakes, tornadoes, flooding conditions, winds, ice, and other local site effects. The design bases so established shall reflect: (a) appropriate consideration of the most severe of these natural phenomena that have been recorded for the site and the surrounding area and (b) an appropriate margin for withstanding forces greater than those recorded to reflect uncertainties about the historical data and their suitability as a basis for design.

DISCUSSION

The systems and components identified in Section 1.7 have been designed to performance standards that will enable the facility to withstand, without loss of capability to protect the public, the additional forces or effects which might be imposed by natural phenomena. The designs are based upon the most severe of the natural phenomena recorded for the site, with an appropriate margin to account for uncertainties in the historical data, or upon the most severe conditions which are susceptible to synthetic analyses.

The referenced Sections of the DSAR are as follows:

a.	Earthquakes	2.5.4, 2.5.4.1, 2.5.4.2, 3.2.1.1.6
b.	Tornado	2.3, 3.2.1.1.5
c.	Flood	2.4.1, 2.4.2, 3.2.1.1.7
d.	Wind	2.3, 3.2.1.1.4
e.	Hurricane	2.4.2, 2.4.2.1, 2.4.2.2
f.	Other Local Site Effects	2.3

1.4.3 CRITERION 3 - FIRE PROTECTION (CATEGORY A)

The reactor facility shall be designed (1) to minimize the probability of events such as fires and explosions, and (2) to minimize the potential effects of such events to safety. Noncombustible and fire resistant materials shall be used whenever practical throughout the facility, particularly in areas containing critical portions of the facility such as containment, control room, and components of engineered safety features.

DISCUSSION

The reactor facility is designed to minimize the probability of fire and explosion. Noncombustibles and fire resistant materials are used throughout the facility as indicated in the following referenced sections of the DSAR.

a.	Control Room	3.14.5
b.	Electrical Distribution Equipment	3.11.2.11

1.4.4 CRITERION 5 - RECORDS REQUIREMENTS (CATEGORY A)

Records of the design, fabrication, and construction of essential components of the plant shall be maintained by the reactor operator or under its control throughout the life of the reactor.

DISCUSSION

The records for the plant will be maintained as described in Section 1.7. On 11/05/2016, CR3 received an exemption from the record keeping requirements for SSC that are no longer part of the licensing basis.

1.4.5 CRITERION 11 - CONTROL ROOM (CATEGORY B)

The facility shall be provided with a control room from which actions to maintain safe operational status of the plant can be controlled. Adequate radiation protection shall be provided to permit access to equipment in the control room or other areas as necessary to maintain safe control of the facility without radiation exposures of personnel in excess of 10 CFR 50.67 and Regulatory Guide 1.183. It shall be possible to maintain the plant in a safe condition if access to the control room is lost due to fire or other cause.

DISCUSSION

Safe occupancy of the control room during abnormal conditions has been provided for in the design. Engineering Change 294154 has determined that the Control Complex no longer serves a safety function and has been reclassified as a Class III Structure. Adequate shielding is provided to maintain tolerable radiation levels in the control room even in the event of a dropped HIC event.

While limitation of personnel access to the control room is contemplated to be very unlikely; nevertheless, capability to maintain the plant in a safe condition is provided with the main control room inaccessible.

1.4.6 CRITERION 17 - MONITORING RADIOACTIVITY RELEASE (CATEGORY B)

Means shall be provided for monitoring the containment atmosphere, the facility effluent discharge paths, and the facility environs for radioactivity that could be released from normal operations, from anticipated transients, and from accident conditions.

DISCUSSION

Monitors are provided for the facility effluent discharge paths and the facility environment, as required, to monitor activity that is released as a result of normal operation, anticipated abnormal operating conditions, and postulated accidents. The monitors provided for the unit and their functions are described in Section 4.4.

1.4.7 CRITERION 18 - MONITORING FUEL AND WASTE STORAGE (CATEGORY B)

Monitoring and alarm instrumentation shall be provided for fuel and waste storage, and handling areas for conditions that might contribute to loss of continuity in decay heat removal and to radiation exposures.

DISCUSSION

Radiation monitors and alarms are provided in the fuel handling and auxiliary buildings as required to warn personnel of impending excessive levels of radiation or airborne activity. The radiation monitors provided for the unit and their functions are described in Section 4.4. These monitors meet the requirements for radiation monitors in 10 CFR 50.68(b), which Duke Energy has chosen to comply with instead of 10 CFR 70.24.

1.4.8 CRITERION 66 - PREVENTION OF FUEL STORAGE CRITICALITY (CATEGORY B) – WITH ALL FUEL IN DRY STORAGE, THIS CRITERION NO LONGER APPLIES.

1.4.9 CRITERION 67 - FUEL AND WASTE STORAGE DECAY HEAT (CATEGORY B) - WITH ALL FUEL IN DRY STORAGE, THIS CRITERION NO LONGER APPLIES

1.4.10 CRITERION 68 - FUEL & WASTE STORAGE SHIELDING (CATEGORY B)

Shielding for radiation protection shall be provided in the design of spent fuel and waste storage facilities as required to meet the requirements of 10 CFR 20.

DISCUSSION

Waste storage facilities, other equipment and piping requiring it, are shielded by reinforced concrete sufficient to meet or exceed the requirements of 10 CFR 20 for radiation protection. Section 4.3 specifies the design criteria for shielding throughout the unit and the design dose rates at various locations. The specified criteria comply with 10 CFR 20 limits.

1.4.11 CRITERION 69 - PROTECTION AGAINST RADIOACTIVITY RELEASE FROM SPENT FUEL AND WASTE STORAGE (CATEGORY B)

Containment of fuel and waste storage shall be provided if accidents could lead to release of undue amounts of radioactivity to the public environs.

DISCUSSION

Radioactive liquid waste processing and storage facilities for this unit are contained within or are housed in structures originally designed as Class I reinforced concrete structures. The analyses cover releases from a radwaste handling event.

1.4.12 CRITERION 70 - CONTROL OF RELEASES OF RADIOACTIVITY TO THE ENVIRONMENT (CATEGORY B)

The facility design shall include those means necessary to maintain control over the plant radioactive effluents, whether gaseous, liquid, or solid. Appropriate hold up capacity shall be provided for retention of gaseous, liquid, or solid effluent, particularly where unfavorable environmental conditions can be expected to require operational limitations upon the release of radioactivity effluents to the environment. In all cases, the design for radioactivity control shall be justified (a) on the basis of 10 CFR 20 requirements for normal operations and for any transient situation that might reasonably be anticipated to occur and (b) on the bases of 10 CFR 100 dosage level guidelines for potential reactor accidents of exceedingly low probability of occurrence except that reduction of the recommended dosage levels may be required where high population densities or very large cities can be affected by the radioactive effluents.

DISCUSSION

The ODCM dictates how CR3 controls effluent releases.

The design of the unit incorporates the means necessary to maintain control of releases of radioactive liquid, gas, and solid waste effluents such that these will be below 10 CFR 20 limits for normal decommissioning operation and reasonable transient situations and within 10 CFR 50.67 limits for accidents of low probability. Ample holdup capacity is provided in the unit for the retention of liquid and solid wastes to ensure that they can always be released in a controlled manner. The Liquid Waste Disposal System provides equipment for extensive decontamination of liquid wastes (if required) prior to their release. Ample storage for solid wastes prior to packaging for shipment offsite is provided. All normal releases of radioactive liquids are continuously monitored and controlled by the Radiation Monitoring System described in Chapter 4 or sampled prior to any release. The means provided to control releases of radioactive liquid and solid waste effluents, under normal and reasonable transient situations, are discussed in Chapters 3 and 4.

1.5 NOT USED

1.6 NOT USED

1.7 QUALITY PROGRAM

This Quality Program is revised and submitted to the NRC as required by 10 CFR 50.54(a). For later interim changes, contact the Nuclear Oversight (NOS) Section.

1.7.1 INTRODUCTION

With the Certification of Permanent Cessation of Power Operations DEF has permanently ceased power operations of Crystal River Unit 3 (CR3). To address this changing environment at CR3, a Decommissioning Quality Assurance Program (DQAP) has been developed to support the decommissioning activities of the station and ensure continued oversight of SAFSTOR, Decommissioning and the Independent Spent Fuel Storage Installation (ISFSI). Sections 1.7.1, 1.7.2 (and Table 1-3), 1.7.3 and 1.7.4 describe the Duke Energy Florida (DEF) Quality Program.

The DQAP includes a description of the organizational structure and functional responsibilities of the station management regarding the implementation of important to safety activities at CR3. The DQAP also outlines the oversight roles and responsibilities for Nuclear Oversight (NOS) staff and program expectations for the various station organizations. The DQAP satisfies the requirements of 10 CFR 50 Appendix B, Quality Assurance Criteria for Nuclear Power Plants and Fuel Processing Plants, 10 CFR 71, Subpart H, Quality Assurance for Packaging and Transportation of Radioactive Material, and 10 CFR 72, Subpart G, Quality Assurance for Independent Storage of Spent Nuclear Fuel and High-Level Radioactive Waste. Additional regulatory commitments are listed within Table 1-3.

1.7.1.1 Organization

CR3 has established the functional responsibilities and authorities of positions/organizations involved in the Quality Program. The functional departmental responsibilities described below, serves to document the relationships between organization positions. In certain instances, duties and authority to execute and audit the quality activities are delegated to other organizations. In all cases, DEF retains responsibility for the Quality Program for Crystal River Unit 3.

Verification of conformance to established quality requirements on safety-related structures, systems, and components is accomplished by those individuals or groups who do not have direct responsibility for performing or directly supervising the work being verified.

Persons and organizations performing quality assurance functions have sufficient authority and organizational freedom to identify quality problems; initiate necessary action to provide for resolution of nonconformances through designated channels; verify implementation of solutions; and, control further processing, delivery, or installation of a nonconforming item until the proper disposition of the deficiency or the unsatisfactory condition has been approved.

The following summarizes the functional responsibilities and authorities of positions/organizations involved in directing and managing the CR3 Quality Program (see Figure 1-26):

1. Duke Energy Corporation Corporate Organization

The Chairman, President and Chief Executive Officer has overall responsibility for Design, Construction, and Operation of generation and transmission facilities. The Executive Vice President and Chief Operating Officer of Duke Energy reports to the Chairman, President and Chief Executive Officer and is responsible for: the Duke nuclear operating fleet; enterprise project management and construction; new plant development and construction; and decommissioning activities. The Executive Vice President and Chief Operating Officer of Duke Energy has overall authority and responsibility for the QA Program. Reporting to the Executive Vice President and Chief Operating Officer of Duke Energy is the Chief Nuclear Officer (CNO) who directs several activities including the operation of the nuclear sites through the Senior Vice Presidents, Nuclear

Operations. Also reporting to the Chairman, President and Chief Executive Officer are Group Executives responsible for providing support to the Nuclear plants for the following: electrical transmission; electrical distribution; laboratory services; switchyard maintenance and technical support; support for the emergency response communications; Information Technology Services; document control and record management activities; and administration of the Access Authorization, Fitness for Duty, and Fatigue Rule programs. As such, the attainment of quality rests with those assigned the responsibility of performing the activity. The verification of quality is assigned to qualified personnel independent of the responsibility for performance or direct supervision of the activity. The degree of independence varies commensurate with the activity's importance to safety.

2. Duke Energy Nuclear

Duke Energy Nuclear is responsible for: the nuclear operating fleet; new nuclear plant development and construction; enterprise project management and construction; and decommissioning activities. These activities are directed by the Executive Vice President and Chief Operating Officer of Duke Energy.

3. Operations Support

The Senior Vice President Operations Support reports directly to the Executive Vice President and Chief Operating Officer of Duke Energy. Operations Support is responsible for defining and executing the decommissioning strategy for Crystal River Unit 3 to meet required regulations and commitments. The Vice President Project Management and Construction reports to the Senior Vice President Operations Support and is responsible for contracts, engineering oversight, and management related to existing plant upgrades, modifications, new plant construction and construction of the Independent Spent Fuel Storage Installation (ISFSI) at Crystal River Unit 3, as requested.

a. General Manager Decommissioning

The General Manager Decommissioning reports to the Senior Vice President Operations Support who reports to the Executive Vice President and Chief Operating Officer of Duke Energy. The General Manager Decommissioning shall have onsite responsibility for overall plant nuclear safety and shall ensure acceptable performance of the staff in operating, maintaining, and providing technical support to the plant to ensure nuclear safety. The General Manager Decommissioning also resolves disagreements with the Plant Nuclear Safety Committee.

The General Manager Decommissioning is given overall authority to staff, operate, and maintain CR3. In carrying out this assignment, the General Manager Decommissioning has the support of an organization consisting of the following:

- Operations & Maintenance Manager
- Decommissioning Technical Support Manager
- Radiation Protection & Chemistry Manager

The General Manager Decommissioning is responsible for:

- The safe, economic, and environmentally sound decommissioning of the nuclear plant, and shall delegate in writing the succession during his absence.
- Planning and monitoring of all activities necessary to achieve SAFSTOR conditions/dormancy with dry fuel storage.
- The CR3 ALARA Program.
- Development, maintenance, and interpretation of the monitoring program and the radiological effluent program as delineated in the Off-Site Dose Calculation Manual (ODCM).
- Investigation and reporting of abnormalities and corrective action taken including, but not limited to, the following:

  - Uncontrolled release of radioactivity
  - Personnel overexposure
  - Loss or theft of licensed radioactive material

- Corrective actions will be taken immediately in accordance with the PDTS (Permanently Defueled Technical Specifications).
- Approving prior to implementation, each proposed test, experiment, or modification to systems or equipment that affect stored nuclear fuel.

1) Operations & Maintenance Manager

The Operations & Maintenance Manager is responsible for:

- Operation of plant equipment required to support SAFSTOR and Spent Fuel cooling and monitoring
- Operations Support for decommissioning activities
- Implementation of the Emergency Planning Program as described in Section 6.4
- Implementation of the CR3 Fire Protection Plan.
- Managing, directing, and supervising the activities of the maintenance work controls and integrated daily work schedule.
- Plant Maintenance.
- Project Supervision.
- Construction.
- Facility Services.
- Directing the activities of the day-to-day scheduling.
- Initial and continuing training of Operations and Maintenance personnel.

2) Radiation Protection & Chemistry Manager

The Radiation Protection & Chemistry Manager is responsible for:

- Directing the overall plant chemistry activities of CR3 to ensure that the plant is functioning within prescribed procedures in the Chemistry area.
- Interfacing with regulatory agencies and organizations to ensure an effective working relationship between plant and agency personnel and to ensure cost effective compliance.
- Establishing program controls for offsite-dose-calculation methodologies, radioactive-effluent controls and radiological environmental-monitoring activities. These controls are contained in the Off-Site Dose Calculation Manual (ODCM) which shall contain the contents specified within Compliance Procedure CP-0500.
- Maintaining a regular interface with other departments within the company and is responsible for developing the processes needed to manage the Chemistry and Radiation Protection Departments including planning, monitoring, and follow-up to ensure consistent high performance.
- Directing the overall plant radiation protection activities of CR3 to ensure that the plant is functioning within prescribed procedures in the Radiation Protection area.
- Performing reviews as required by 10CFR20.1101(c).
- Generating, managing, and maintaining the Historical Site Assessment for use during and after the SAFSTOR period.
- Initial and continuing training of radiation protection, chemistry, and environmental personnel.

3) Decommissioning Technical Support Manager

The Decommissioning Technical Support Manager is responsible for:

- Supporting the Emergency Planning Program as described in Section 6.4.
- Coordinating changes to procedures as necessary for changing plant conditions as described in Section 1.7.1.5.
- Providing Document Control services as described in Section 1.7.1.6.
- Facilitating site personnel Training and qualification documentation.
- Maintaining Performance Improvement as described in Sections 1.7.1.15 and 1.7.1.16.

- Supporting licensing activities to prepare, control, and support regulatory activities associated with placing the facility in SAFSTOR and eventual decommissioning.
- Updating the DSAR and PDTS.
- Serving as the primary contact for the NRC.
- Initial and continuing training of Engineering personnel.
- Responsible for strategic planning and scheduling of decommissioning related activities.
- Coordinating site support and alignment associated with conduct of decommissioning.
- Providing engineering leadership and oversight to ensure the safe, reliable, and cost efficient decommissioning of the plant.
- Performing engineering activities that maintain configuration control of the plant design basis.
- Resolving engineering related issues to support safe and efficient operation of systems required to assure safe storage of spent fuel.
- Providing engineering leadership and support for managing engineering programs.
- Developing, controlling and monitoring the Fire Protection Program which includes ensuring that the program meets all regulatory requirements imposed by the regulatory agencies as they relate to acceptable implementation, and for ensuring the program is adequately implemented.
- Participating in the Plant Nuclear Safety Committee (PNSC), Project Management Oversight Board (PMOB), Plant Emergency Response Organization (ERO), SAFSTOR related organizations, Financial Management Steering Committee and Site programs as required.
- Identifying SSC designations throughout the period up to and including the SAFSTOR period.
- Updating required SSC designations and classifications as changes to SSCs occur.
- Revising engineering documents to reflect changing materiel condition.
- Ensuring the final plant configuration supports remediation activities following the SAFSTOR period.
- Coordinating engineering activities with the Historical Site Assessment.

4. Nuclear Oversight (NOS)

The senior manager for Nuclear Oversight (NOS) reports to the CNO. The senior manager for Nuclear Oversight (NOS) is responsible for and reports to the CNO on all matters related to the independent monitoring and auditing of activities performed by the line organizations for, or in support of Duke Nuclear plants and CR3 activities. NOS provides oversight of Nuclear Decommissioning through QA program audits. The senior manager for NOS has the authority and organizational freedom to: identify quality problems, initiate, recommend or provide solutions to quality problems through designated channels, verify the implementation of solutions to quality problems, and ensure cost and schedule do not influence decision making involving quality. This includes full access to Nuclear Decommissioning and all levels of management up to and including the Chief Executive Officer of Duke Energy Corporation.

The senior manager for NOS is delegated primary ownership of the CR3 QA program. If significant quality problems are identified, NOS personnel have the authority to stop work pending satisfactory resolution of the identified problem.

Also reporting to the senior manager for Nuclear Oversight is Employee Concerns, which investigates concerns identified through the Employee Concerns Programs to determine their validity and initiate corrective actions as appropriate. Employee Concerns also promotes the Safety Conscious Work Environment (SCWE) Program and is sensitive to SCWE concerns during investigations performed.

5. Department Interfaces

Departmental interfaces are identified in appropriate procedures. Quality related activities performed by departments other than Nuclear Generation or Operations Support are identified by and conducted in

accordance with approved departmental interface agreements. The following are generic descriptions of those other corporate departments and the services they provide. These generic organizations are referred to, as appropriate, within this document; however, approved departmental interface agreements establish and define the applicability of the DQAP to the services they provide:

a. Corporate Communications
Corporate Communications provides support for the nuclear site's emergency response organization.

b. Environmental Health and Safety
Environmental, Health and Safety will provide environmental and laboratory support services.

c. Nuclear Finance
Nuclear Finance provides support for the nuclear sites in the areas of decommissioning, workforce planning and development.

d. Customer Operations
Customer Operations provides electrical distribution and switchyard engineering, as well as providing electrical maintenance and testing support.

e. Information Technology
Information Technology provides a variety of services and technical support to the Duke Energy Nuclear fleet and CR3 for information technology applications and systems such as equipment databases, applications, infrastructure, and plant process information systems. They are also responsible for the development and maintenance of selected information technology services and support, including electronic document management, some of which support QA related activities.

f. Supply Chain
Supply Chain provides procurement services, storage, inventory control, and receipt inspection/testing.

6. Agents and Contractors

Duke may contract various activities such as engineering, procurement, and construction. These contracts will identify QAP requirements that are applicable to the contractors and their subcontractors, consistent with the requirements of Sections 1.7.1.4 and 1.7.1.7.

7. The Manager Internal Audits is responsible for performing internal audit activities and provides objective oversight of plant performance relating to nuclear safety and quality. The Manager Internal Audits reports to the senior manager for Nuclear Oversight.

8. The QC Corporate Functional Area Manager is responsible for plant inspections and non-destructive examinations. The QC Corporate Functional Area Manager reports to the senior manager for Nuclear Oversight.

9. Plant Nuclear Safety Committee

Refer to Section 6.8.1.

1.7.1.2 Quality Program

The Quality Program complies with the requirements of 10 CFR 50, Appendix B. This program requires that all persons performing quality activities associated with Crystal River Unit 3 comply with the program. CR3 conducts or delegates the responsibility to conduct audits of the program activities.

The Quality Program takes into account the need for special controls, processes, tests, equipment, tools and skills to obtain the required quality.

Managers/Supervisors selecting personnel shall assure that the qualifications stated in the job description meet or exceed the ANSI N18.1 education and experience requirements identified in the job description. Managers/supervisors will ensure that an individual being selected meets the minimum education and experience requirements of ANSI N18.1-1971, with the exception of the Radiation Protection Manager, who shall meet or exceed the qualifications of Regulatory Guide 1.8, September 1975 (as clarified in Table 1-3).

An indoctrination and training program is provided for personnel performing quality activities to assure that they are knowledgeable of the Quality Program's procedures and requirements. The indoctrination and training program includes appropriate procedures and personnel records.

Additionally, personnel responsible for performing safety-related activities are instructed as to the purpose, scope and implementation of the safety-related manuals, instructions, and procedures. Personnel performing safety-related activities are trained and qualified in the activity being performed.

Personnel who carry out Health Physics, or perform quality assurance functions, may report to the appropriate onsite manager; however, they shall have sufficient organizational freedom to ensure their ability to perform their assigned functions.

Quality activities such as inspection, examination and test are done with appropriate equipment and under suitable conditions.

CR3's organizational structure and the functional responsibility assignments assure that:

1. Attainment of program objectives is accomplished by those who have been assigned responsibility for performing work. Activities may include interim examinations, checks, and inspections of the work by the individual performing the work.
2. Verification of conformance to established program requirements is accomplished by a qualified person who does not have responsibility for performing or directly supervising the work. The method and extent of such verification shall be commensurate with the importance of the activity to plant safety and reliability as clarified by Table 1-3 (NRC Regulatory Guide 1.74, Item 4).
3. In structuring the organization and assigning responsibility, quality assurance should be recognized as an interdisciplinary function involving many organizational components and, therefore, should not be regarded as the sole domain of a single quality assurance group. Quality assurance encompasses many functions and activities and extends to various levels in all participating organizations, from the top executive to all workers whose activities may influence quality.

The CR3 Quality Program applies to structures, systems and components as defined in the "Safety Listing" and may be applied to other equipment and activities at management discretion (e.g., equipment important for SAFSTOR, Fire Service System, and Radioactive Waste System). Augmented Quality Assurance (non-safety-related with special quality/regulatory requirements) may be applied to programs such as Fire Protection, Radioactive Material Packages.

CR3 regularly reviews the status and adequacy of its Quality Program through periodic reviews conducted at least once every two years. In addition, responsible management reviews audit reports and corrective actions of that part of the Quality Program which they are implementing.

Changes to the Quality Program which result in more stringent requirements will be entered in appropriate implementing procedures within 90 days of the Quality Program change unless otherwise specified in the requirement to change the Quality Program or unless a longer period is evaluated and accepted by the senior manager for NOS. All other Quality Program changes will be reflected in appropriate implementing procedures at their next revision.

The CR3 Quality Program meets the requirements of the Regulatory Guides and ANSI Standards as defined in Section 1.7.2 (and clarified in Table 1-3).

For activities where quality considerations are subject to interpretation, the management responsible for the activity shall also be responsible for assuring that programmatic controls are applied. This in no way negates the need for clear management controls for all safety-related activities. Nuclear Oversight personnel evaluate and verify that controls are in place and effectively implemented through inspection and audit activities.

CR3's QAPD meets the Quality Assurance Requirements of 10CFR71 and 10CFR72. With respect to the package supplier's quality assurance, that CR3 relies on certifications of an NRC-approved Quality Assurance Program satisfying the requirements of 10CFR71.101.

1.7.1.3 Design Control

Measures shall be established to assure that applicable regulatory requirements and the design basis are correctly translated into specifications, drawings, procedures, and instructions. These measures shall include provisions to assure that appropriate quality standards are specified and included in design documents and that deviations from such standards are controlled. Measures shall also be established for the selection and review for suitability of application of materials, parts, equipment, and processes that are essential to the functions of the structures, systems and components.

Measures shall be established for the identification and control of design interfaces and for coordination among participating design organizations. These measures shall include the establishment of procedures among participating design organizations for the review, approval, release, distribution, and revision of documents involving design interfaces.

The design control measures shall provide for verifying or checking the adequacy of design, such as by the performance of design reviews, by the use of alternate or simplified calculational methods, or by the performance of a suitable testing program. The verifying or checking process shall be performed by individuals or groups other than those who performed the original design, but who may be from the same organization. Where a test program is used to verify the adequacy of a specific design feature in lieu of other verifying or checking processes, it shall include suitable qualifications testing of a prototype unit under the most adverse design conditions.

Design changes, including field changes, shall be subject to design control measures commensurate with those applied to the original design and be approved by the organization that performed the original design unless the applicant designates another responsible organization.

Safety-related design changes are verified by qualified personnel who are independent from the designer or the designer's immediate supervisor. This review assures that the design activities are in accordance with ANSI N45.2.11 as clarified in Table 1-3.

Maintenance or modifications which may affect safety-related structures, systems, or components are performed in a manner that ensures quality requirements, material specifications, and inspection requirements are met. Maintenance or modifications of safety-related equipment are planned and performed in accordance with written procedures, documented instructions, or drawings appropriate to the circumstances which conform to applicable codes, standards, specifications, and criteria as clarified in Table 1-3.

1.7.1.4 Procurement Document Control

Procurement documents are reviewed by qualified personnel, prior to purchase, to assure that quality and technical requirements have been specified. Individuals reviewing these procurement documents are not involved with the other phases of the procurement activity. These reviews are performed and documented in accordance with approved written procedures.

CR3's Quality Program contains provisions which require that:

1. Procedures are established which clearly delineate the sequence of actions to be accomplished in the preparation, review, approval and control of procurement documents and which identify those positions or groups responsible for performing those functions.
2. Review of and concurrence with the procurement documents are performed by qualified personnel to assure that the quality requirements are stated. This review is to determine that quality requirements can be inspected and controlled, that there is adequate acceptance or rejection criteria, and that the procurement document has been prepared in accordance with DEF Quality Program procedure requirements.
3. Documented evidence of the review and approval of procurement documents is provided and available for verification.

4. Procurement documents identify those 10CFR50 Appendix B requirements that must be complied with by the supplier's quality program. Details regarding evaluation and selection of suppliers are stated in written procedures.
5. Procurement documents contain or reference, as applicable, basic technical requirements such as regulatory requirements and design bases and identify the documentation to be prepared, maintained, submitted, and made available to DEF for review and/or approval. DEF's procurement documents include provisions for control of nonconformances.
6. Procurement documents contain the requirements for the retention, control, and maintenance of records as appropriate.
7. Procurement documents contain the right of access to vendor's facilities and records for source inspection and audit by DEF.
8. Changes and/or revisions to a procurement document are subject to review and approval requirements at least equivalent to those for the original document.
9. Procurement document control may be applied to Augmented Quality Assurance Programs (e.g., FP-Q, RW-Q, 10CFR71-Q, 10CFR72-Q, or Q Class B) as directed by DEF's procedural requirements.

1.7.1.5 Instructions, Procedures and Drawings

CR3's Quality Program contains requirements to assure that each of the 18 criteria within 10 CFR 50, Appendix B are delineated, accomplished, and controlled in accordance with approved written procedures.

CR3's Quality Program contains provisions which require that instructions, procedures, or drawings include appropriate quantitative (such as dimensions, tolerances, and operating limits) or qualitative (such as workmanship samples) acceptance criteria for determining that quality activities have been satisfactorily accomplished.

Written procedures are adhered to in matters relating to nuclear safety. The program identifies when written procedures shall be followed step-by-step. In order to properly document that procedural steps are verified as required, each step of the procedure requiring verification is initialed or otherwise acknowledged when completed.

1.7.1.6 Document Control

CR3 has a document control system for documents which prescribe activities affecting quality.

CR3's Quality Program contains provisions which require that:

1. Measures are established to review documents, such as instructions, procedures, and drawings (and changes thereto) prior to release to assure that the quality requirements are sufficiently, clearly, and accurately stated.
2. Changes to documents are reviewed and approved by the same organizations that performed the original review and approval unless delegated by the appropriate DEF organization to another qualified responsible organization.
3. The reviewing organization(s) has access to pertinent background information upon which to base its approval and has an adequate understanding of the requirements and intent of the original document.
4. Approved changes are promptly included with instructions, procedures, drawings, and other appropriate documents.
5. Obsolete or superseded documents are controlled to prevent their use.
6. Documents are available at the start of the work for which they are needed.
7. A method for identifying the current revision of instructions, procedures, and drawings is established and implemented. This information is updated and distributed as necessary to predetermined responsible personnel.

As a minimum under this criteria, the controlled documents include:

1. Design specifications.
2. Design, manufacturing, construction, and installation drawings.
3. Manufacturer inspection and testing instructions.
4. Procurement documents.
5. Maintenance, repair, and modification instructions.
6. Test surveillance instructions.
7. In-service inspection instructions.
8. Other procedures per Regulatory Guide 1.33, as clarified in Table 1-3.

1.7.1.7 Control of Purchased Material, Equipment and Services

Vendor evaluation surveys to qualify potential suppliers in accordance with approved written procedures and in compliance with ANSI N18.7, Section 5.2.13 and ANSI N45.2.13 as each is clarified in Table 1-3, are conducted by CR3 or through qualified contractors. Suppliers' quality assurance programs are reviewed and concurred with prior to implementation of activities except for commercial-grade calibration services from accredited calibration laboratories as described in Table 1-3 for Regulatory Guides 1.123 and 1.144.

Certain items classified as "Commercial Grade" may be purchased from a non-evaluated vendor and subsequently dedicated for use in a safety-related application. Suitability of an item for use in a safety-related application is determined by verification of engineering established critical characteristics.

DEF assures that quality requirements of the purchase document have been met, using source inspection, receipt inspection or document review, as appropriate.

Control of purchased material, equipment and services may be applied to Augmented Quality Assurance Programs (e.g., FP-Q, RW-Q, 10CFR71-Q, 10CFR72-Q, or Q Class B) as directed by DEF's procedural requirements.

1.7.1.8 Identification and Control of Materials, Parts and Components

CR3 has established measures for the identification and control of materials, parts, and components.

CR3's Quality Program contains provisions which require that:

1. Procedures are established which describe identification and control of material, parts, and components, including partially fabricated assemblies.
2. Identification requirements are determined during the initial planning stages (i.e., during generation of specification and design drawings).
3. Identification is specified to the extent that the item identified can be traced to the associated documentation, such as drawings, specifications, purchase orders, manufacturing and inspection documents, and physical or chemical mill test reports.
4. The degree of identification is specified on the design drawing or in referenced technical documents.
5. Measures are provided to assure that the location and method of identification do not affect the function or quality of the item being identified.
6. Measures are provided for the verification of correct identification of materials, parts, and components prior to release for manufacturing, shipping, construction and installation.

1.7.1.9 Control of Special Processes

Participating organizations provide written procedures for performance of special processes such as welding, heat treating, chemical cleaning, coating/painting, and non-destructive testing and include the requirements for qualification of personnel performing the work.

CR3's Quality Program contains provisions which require that:

1. Measures are established to assure adequate performance and control of special processes such as welding, heat treating, chemical cleaning, coating/painting and non-destructive testing.
2. Measures are established to assure that procedures, equipment and personnel connected with special processes are qualified in accordance with the requirements of applicable codes, standards, and specifications.
3. Measures are established to assure that special processes are performed by qualified personnel in accordance with approved written procedures. These procedures provide for recording evidence of verification and, if applicable, inspection and process results.
4. An active file is maintained on qualification records of all special process procedures and equipment, and personnel performing special processes.
5. Special process procedures and the credentials of qualified personnel are regularly reviewed to assure they are of the latest revision and that personnel qualifications have not expired.

1.7.1.10 Inspections

Written procedures are required for the performance of inspection. Inspection personnel are qualified in compliance with Regulatory Guides 1.8, 1.33, 1.58, and 1.146 as each Regulatory Guide is clarified in Table 1-3.

CR3's Quality Program contains provisions which require that:

1. Inspection personnel are independent from the individual or group physically performing and directly supervising the activity being inspected.
2. Inspection procedures, instructions and/or checklists are provided which document the date performed, by whom and/or by what equipment, the type of observation, the results, the data collected and its acceptability.
3. Inspection procedures or instructions are available for use prior to performing the inspection operation.
4. Measures are provided for qualifying the inspectors and maintaining the current status of each inspector's qualifications.
5. Measures are established to assure that inspection equipment is within calibration prior to performing an inspection operation.
6. Measures are provided for monitoring processing methods, equipment, and personnel if inspection of processed material is impossible or disadvantageous. Inspection and process monitoring are provided when control is inadequate without both.
7. Specific hold points are indicated in appropriate documents for mandatory witnessing or inspection beyond which work shall not proceed without the consent of DEF's designated representative.

1.7.1.11 Test Control

Required tests are performed in accordance with approved written procedures to assure compliance with design documents. Testing activities are conducted during the operational phase to verify the compliance of components to design requirements.

CR3's Quality Program contains provisions which require that:

1. A test program is established to assure that all testing required to demonstrate that the item will perform satisfactorily in service is identified, documented, and accomplished in accordance with approved written procedures.
2. The test program covers the required tests, including, where appropriate, prototype qualification tests, proof tests prior to installation, preoperational tests, and operational tests.
3. Written test procedures are prepared which incorporate or reference the requirements and acceptance limits contained in applicable design and procurement documents.
4. The written test procedures include, as appropriate, instructions for test method and identification of test prerequisites such as:
   a. calibrated instrumentation;
   b. adequate and appropriate equipment;
   c. trained, qualified, licensed and/or certified personnel;
   d. preparation, condition, and completeness of item to be tested; and
   e. suitable and controlled environmental conditions.
5. Test results are documented and evaluated to assure that test requirements have been satisfied.

1.7.1.12 Control of Measurement and Test Equipment

DEF has established and implemented appropriate test and calibration procedures for test devices used to verify the acceptability of items within the Quality Program. Calibration records and controls are provided for measurement and test equipment in accordance with the requirements of ANSI N18.7, Section 5.2.16, ANSI N45.2.4 and ANSI N45.2.8 as each ANSI Standard is clarified in Table 1-3.

CR3's Quality Program contains provisions which require that:

1. Procedures are established which describe the calibration technique, calibration frequency, maintenance and control of measuring and test instruments, tools, gauges, fixtures, reference standards, transfer standards, and non-destructive test equipment to be used in the measurement, inspection, and monitoring of safety-related components, systems, and structures.
2. Measurement and test equipment is uniquely identified and has traceability to the calibration test data.
3. Measurement and test instruments are calibrated and maintained at specified intervals, based on the required accuracy, purpose, the degree of usage, stability characteristics, and other conditions affecting the measurement.
4. Measurement and test equipment is calibrated on or before the designated due date or before use.
5. When measurement and test equipment is found to be out of calibration, an investigation is conducted and documented to determine the validity of previous inspections performed and the acceptability of those items previously inspected.
6. Calibrating instruments have known valid relationships to a nationally recognized standard. If no national standard exists, the basis for calibration is documented.
7. Facilities used for calibrating sensitive or close tolerance measurement and test equipment provide an environment that is sufficiently controlled to allow the measuring device to be evaluated and calibrated to its required accuracy.

1.7.1.13 Handling, Storage and Shipping

Approved written procedures and sound storage principles are used for material handling, storage, and shipping activities for plant spare parts and operating supplies. These procedures meet the requirements of ANSI N18.7, Section 5.2.13.4 and ANSI N45.2.2 as they are clarified in Table 1-3.

1.7.1.14 Inspection, Tests and Operating Status

Approved written procedures are used to assure the proper marking of equipment denoting its status.

CR3's Quality Program contains provisions which require that:

1. Measures are established and documented to identify the inspection, test, and operation status of structures, systems, and components, which provide means for assuring that required inspections and tests performed are known throughout manufacturing, installation, and operation.
2. Measures are established to control the use of inspection and status indicators, including the authority for application and removal of tags, markings, and labels.
3. Measures to preclude bypassing of required inspections, tests, and other critical operations are provided through approved written procedures.
4. The status of nonconforming, inoperative, or malfunctioning structures, systems, or components is clearly identified to prevent use.

1.7.1.15 Nonconforming Material, Parts, or Components

Written requirements are followed by persons performing quality activities, including contractors, to identify, document, segregate, disposition and report any nonconformance, deviation or other condition adversely affecting quality.

CR3's Quality Program contains provisions which require that:

1. Measures and procedures are established to control the identification, documentation, segregation, review, disposition, and notification of the affected organization of nonconformances.
2. Documentation is provided which clearly identifies the nonconforming item, describes the nonconformance and disposition of the nonconformance, inspection requirements, and includes signature approval of the disposition.
3. Measures are established and documented defining the responsibility and authority for determining the disposition of nonconforming items and approving the disposition.
4. Nonconforming items are segregated from acceptable items (where feasible) and uniquely identified as nonconforming until properly dispositioned for use.
5. Acceptability of "rework" or "repair" of materials, parts, components, systems, and structures are verified by reinspection and/or testing of the item in accordance with approved written procedures.
6. Nonconforming items which are dispositioned "use as is" or "repair" are formally controlled through approved procedures or design changes.
7. Nonconformance reports are made part of the quality assurance records.

1.7.1.16 Corrective Action

Approved written procedures are followed by persons performing quality activities, including contractors, to assure that corrective action is taken to preclude the recurrence of nonconformances, deviations or other discrepancies adversely affecting quality.

CR3's Quality Program contains provisions which require that:

1. Conditions adverse to quality, such as failures, deficiencies, deviations, defective material and equipment, and nonconformances are promptly identified and corrected.
2. Evaluation of nonconformance and determination of the need for corrective actions are in accordance with approved written procedures.

3. Measures are established to determine the cause of the nonconformance and institute corrective action to preclude the recurrence of those significant conditions adverse to quality.
4. Measures are established to follow up on corrective actions to assure proper implementation and close out of the corrective action documentation.
5. Measures are established to document and report to appropriate levels of management significant conditions adverse to quality, cause of the conditions, and corrective action taken.

1.7.1.17 Quality Assurance Records

A system has been established and implemented for the collection, storage, and maintenance of quality assurance records as required by the design documents, procurement documents, and Regulatory Guide 1.88 as clarified in Table 1-3. Quality assurance records transmitted to the quality files are done so in accordance with approved written procedures. Quality assurance records do not include vital records. Vital records are records maintained to meet regulatory or other commitments and are not required to meet the collection, storage and maintenance requirements of ANSI N45.2.9.

CR3's Quality Program contains provisions which require that:

1. Quality assurance records are of two categories, lifetime and nonpermanent. Nonpermanent records are required to show evidence that an activity was performed in accordance with applicable requirements but need not be retained for the life of the plant. Lifetime records are required to be maintained for the life of the plant while the particular item is installed in the plant or stored for future use.
2. Quality assurance records are those records that furnish documentary evidence of the quality of items and of activities affecting quality. A document is considered a quality assurance record when the document has been completed.

   These records include the results of reviews, inspections, examinations, tests, audits, assessments, monitoring of work performance and material analysis, the qualification of personnel, procedures, and equipment, training records, design drawings and subsequent modifications, specification reports, procurement documents, calibration procedures and reports, nonconformance and corrective action reports, and other records required by ANSI N18.7 as clarified in Table 1-3. The records are identifiable and retrievable per ANSI N45.2.9 as clarified in Table 1-3.
3. The inspection and test records contain the following:
   a. Description of the types of operation.
   b. Evidence of completion and/or verification of manufacturing inspection or test operation.
4. Records are stamped, dated, initialed, signed, or otherwise authenticated by authorized personnel. Measures are established to control the use of electronic signatures that prevent falsification or alteration of electronic records and provide a means to authenticate electronic signatures.
5. Storage facilities are constructed, located, and secured to prevent destruction and minimize deterioration or loss of records. Quality assurance records are maintained either in a vault for single copy records or as duplicate records in remote locations.

On November 30, 2016, the NRC granted an exemption request related to Record Keeping Requirements. This exemption permits the elimination of requirements to maintain records that are no longer necessary due to the permanently shut down and decommissioning status of CR3. Specifically, those QA Records are no longer required to be retained when: 1) the CR3 licensing basis requirements previously applicable to the nuclear power unit and associated systems, structures, and components (SSCs) are no longer effective (i.e., removed from the FSAR or Technical Specifications by the appropriate change mechanisms); or 2) for SSCs associated with safe storage of fuel in the spent fuel pools (SFPs) where spent fuel has been completely removed from the SFPs, and the associated licensing bases are no longer effective.

1.7.1.18 Audits and Independent Reviews

1.7.1.18.1 Independent Audit

The functions and activities affecting the nuclear programs at CR3 are independently audited. Nuclear Oversight is responsible for auditing activities that are performed by or for the Crystal River Plant.

1.7.1.18.1.1 Organization

Personnel performing independent audit activities have no direct responsibilities in the areas being audited.

Selection of audit personnel is based on experience and/or training which establishes that their qualifications are commensurate with the complexity or special nature of the area being audited. The process for qualification of personnel to perform and lead audits is established in procedures.

Personnel performing audits shall have access to records, procedures, and personnel to gather data.

1.7.1.18.1.2 Audit Process

The independent audit process includes gathering data, analyzing data, focusing on selected issues and identifying deficiencies. The results of independent audits are communicated to management in a manner that causes action to correct deficiencies and develop action to prevent recurrence. In addition, this process should evaluate corrective measures adopted to eliminate the deficiencies identified.

Planning activities identify the organizations to be evaluated, the characteristics to be focused on during the independent audit, and the applicable acceptance criteria. Independent audit activities are selected with flexibility based on various factors.

Preparation activities may include a review of performance data, relevant documentation, previous audit data, industry experience, team member experience, and management input. These activities enable the team to focus on issues which may impact safety and reliability when analyzing data.

Audits are scheduled on the basis of the status and safety importance of the activities or processes being performed. The schedule is flexible and dynamic to allow audits to be changed depending on plant conditions, events, or issues raised by senior management.

1.7.1.18.1.3 Nuclear Oversight (NOS) Audit Program

Audits of facility activities shall be performed by NOS. Audits will be scheduled based on plant performance and importance to safety, but with a frequency not to exceed twenty-four months. These audits shall encompass:

- The conformance of facility operation to provisions contained within the PDTS and applicable license conditions.
- The performance, training and qualifications of the staff supporting the Crystal River Nuclear Plant.
- The results of actions taken to correct deficiencies in facility equipment, structures, systems or method of operation that affect nuclear safety.
- The performance of activities required by the Quality Assurance Program to meet the criteria of 10 CFR 50 Appendix B, for activities performed by the staff supporting the Crystal River Nuclear Plant.
- The Radiological Environmental Monitoring Program, and the results thereof.
- The performance of activities required by the Quality Assurance Program for effluent and environment monitoring.
- The Offsite Dose Calculation Manual (ODCM) and implementing procedures.

- The Process Control Program and implementing procedures for processing and packaging of radioactive wastes.
- Audits of activities prescribed by the Code of Federal Regulations will be performed at the frequencies prescribed by the applicable regulation. These audits shall encompass:
  - Emergency Preparedness [per 10 CFR 50.54(t)]
  - Security [per 10 CFR 50.54(p)]

Results of audits will be provided to Nuclear Oversight management for review. A periodic briefing of Nuclear Oversight activities, along with potential issues and recommendations, shall be presented to the Senior Vice President, Operations Support.

Follow-up is accomplished to assure that corrective action is taken as a result of the audits and that deficient areas are reaudited, when necessary, to verify implementation of adequate corrective actions.

1.7.1.18.2 Independent Review Program

The PNSC provides the independent review of the following:

- Proposed changes to the facility as described in the DSAR. This review is to confirm that the change does not adversely affect safety and if a PDTS change or NRC review is required.
- Proposed changes to procedures as described in the DSAR and tests or experiments not described in the DSAR. This review is to confirm that the change does not adversely affect safety and if a PDTS change or NRC review is required.
- Proposed PDTS changes and license amendments, except in those cases where the change is identical to a previously reviewed proposed change.
- Licensee Event Reports that are required to be made to the NRC. This review includes results of any investigations made and recommendations resulting from such investigations to prevent or reduce the probability of recurrence of the event.
- Any other matter related to nuclear safety requested by the General Manager Decommissioning or referred for review by other organizations.

See Section 6.8.1.

1.7.2 PROGRAM COMMITMENT

During the SAFSTOR and Decommissioning phase, DEF will comply with the requirements of the ANSI Standards listed in Table 1-3 and implement the Regulatory Positions of the Regulatory Guides listed in Table 1-3, as clarified in that Table. DEF considers and will refer to such Regulatory Positions as requirements. When Regulatory Guides or ANSI Standards are superseded by an approved revision, that revision will not be implemented unless the DSAR is modified accordingly.

1.7.3 QUALITY ASSURANCE STAFF

Persons performing quality assurance functions, as defined in 10 CFR 50 Appendix B, conduct reviews and audits of departments, suppliers and contractors that perform safety-related functions in connection with Crystal River Unit 3. These reviews and audits are performed in accordance with approved written procedures and in compliance with approved minimum requirements for audit frequency. Persons performing quality assurance functions are authorized to identify quality problems and may recommend to management that work be stopped under totally unacceptable conditions. Such persons have the organizational freedom to effectively perform the quality assurance functions.

1.7.4 GLOSSARY OF TERMS

Terms used in the CR3 Quality Program are defined below or in those Regulatory Guides and ANSI Standards committed to by CR3, as clarified in Table 1-3.

1. Quality Activity

The term "quality activity" is a general term used to describe activities within the total Quality Program. The purpose of using the term "quality activity" is to reserve the words "control" and "assurance" for those specific functions of the Quality Program defined as "quality control" and "quality assurance."

1.8 INTERACTIONS BETWEEN CRYSTAL RIVER UNIT 3 AND THE FOUR FOSSIL FIRED PLANTS

AI-1300, "Engineering, Maintenance and Support Interfaces," is a CR3 document which contains descriptions of the numerous interactions between CR3 and other DEF organizations. It also defines the scope of the interfacing activities. The document is for use by organizations who perform activities which may affect the licensing/design basis of CR3 to identify those activities requiring the knowledge and participation of Nuclear Operations. A brief discussion of some of the interfaces follows:

1.8.1 WELL WATER SYSTEM

Well water to Units 1, 2, and 3 is furnished from a common system. Units 4 and 5 are on separate wells. The maintenance and operation of the Units 1, 2, and 3 system is under the supervision and direction of the Fossil Plant Superintendent.

1.8.2 DEMINERALIZED WATER SYSTEM MAKEUP

Demineralized makeup water is furnished to the nuclear plant from the site water treatment system. Normal makeup will be from a 450,000 gallon demineralized water storage tank with backup from two 147,000 gallon tanks. The operation of the water treatment facility is under administrative control.

1.8.3 INTAKE AND DISCHARGE CANALS

The intake and discharge canals are common between Units 1 and 2 and the nuclear unit. Maintenance of the canals is the responsibility of the Crystal River Fossil Operations.

1.8.4 FIRE SYSTEM MAKEUP

Makeup to the 300,000 (usable volume) gallon fire service water storage tank~~s~~ is supplied from Units 1 and 2 fire pumps.

1.8.5 TOXIC GAS STORAGE

Various chemicals are stored on the Crystal River Energy Complex. Events involving the limiting chemicals have been evaluated in establishing the Crystal River Unit 3 (CR3) control room habitability.

1.8.6 FLOATING DEBRIS FROM UNITS 1 AND 2 DURING A PROBABLE MAXIMUM HURRICANE

During a Probable Maximum Hurricane (PMH) event, floating debris from Units 1 and 2, or their related structures, will not impair the functionality of equipment at Unit 3 for the following reasons:

a. Floating debris from Units 1 and 2 will follow the path of the PMH, which will track on a northeasterly course away from Crystal River Unit 3, as shown on Figure 2-21.

b. The maximum draft depth for floating debris considered capable of striking the plant during a PMH is 28" (Reference 1).

c. Sections 3.2.1.1.5 and 3.3.3.2.2 state that the reactor building and other structures originally designed as Class I have been designed to withstand tornado generated missiles. Due to this design, Crystal River Unit 3 could withstand the impact from floating debris from Units 1 and 2.

1.9 CONCLUSIONS

On the basis of the information presented in this Defueled Safety Analysis Report and referenced material, DEF concludes that Crystal River Unit 3 was designed, constructed, and is being operated without undue risk to the health and safety of the public

1.10 REFERENCES

1. G/C, Inc., Structural Calculation FC 16.00.1/1A, Duke Calculation S02-0016, Revision 0, Attachment 1.

2. Daniel, R. C., et al., Effects of High Burnup on Zircaloy-Clad, Bulk UO_2, Plate Fuel Element Samples, WAPD-263, September, 1962.

TABLE 1-2 - CRYSTAL RIVER UNIT 3 IN-PLANT QUALITY PROGRAM FUNCTIONS

Quality Assurance	Quality Control	Work
Owner's surveillance and management evaluation of the operating staff activities through audit of the implementation of the commitments.	Plant staff personnel provide the Quality Control function. Included as a verification of compliance is the function identified as the Compliance Section which assures compliance by the Quality Control activities through audits and witness action.	The work starts with commercial operation and continues throughout the life of the plant.

TABLE 1-3 - CRYSTAL RIVER UNIT 3 QUALITY PROGRAM COMMITMENTS

This table presents the Regulatory Guides and ANSI Standards endorsed by Crystal River Unit 3 (CR3) as part of its Quality Program.

In each of the ANSI Standards, other documents (i.e., other Standards, codes, regulations, tables, or appendices) required to be included as part of the Standard are either referenced or described in a special section of the Standard. The specific applicability or acceptability of these referenced Standards, codes, regulations, tables, or appendices is either covered in other specific areas in the CR3 Quality Program description, including this Table, or such documents are not considered as Quality Program requirements, although they may be used as guidance. Whenever a standard endorsed in Table 1-3 invokes ANSI N45.2, CR3 shall interpret the statement to mean a QA Program which meets the requirements of 10 CFR 50, Appendix B.

When Sections of Standards are referenced within a clarification, it is understood that CR3 shall comply with the referenced Sections as clarified.

Table 1-3
Crystal River Unit 3 Quality Program Commitments
(continued)

Table 1-3
Crystal River Unit 3 Quality Program Commitments
(continued)

NRC REGULATORY GUIDE 1.8 - "PERSONNEL SELECTION AND TRAINING" (REVISION 1, 9/75) - ENDORSES ANSI N18.1-1971.

With the Certification of Permanent Cessation of Power Operations, DEF has permanently ceased power operations of Crystal River Unit 3 (CR3). CR3 follows this Regulatory Guide and associated Standard for the remaining important to safety activities associated with SAFSTOR, decommissioning, and the Independent Spent Fuel Storage Installation (ISFSI) with the following clarifications:

1) CR3 often uses additional non-CR3 employees and contract personnel to augment the facility staff. These personnel may or may not report to the General Manager, Decommissioning, Crystal River Nuclear Plant. These groups include, but are not limited to, supplemental HP and I&C technicians, NDT Examiners and QC inspectors, as well as CR3 System Maintenance Crew (SMC) personnel. When used to perform safety-related activities, these personnel shall meet the education, training and experience requirements of ANSI N18.1-1971 for equivalent positions or else they shall meet the requirements for certification as inspection, examination or testing personnel as set forth in CR3's commitment to ANSI N45.2.6-1978 given elsewhere in this Table.

2) Operators licenses, which are issued by the NRC and discussed in ANSI N18.1-1971, are no longer required based upon the NRC approval of TS Amendment 244.

3) The education, experience, and certification requirements of/for Shift Supervisor will meet that specified in Section 4.3.1.1.b and 4.3.1.1.d of ANSI/ANS 3.1-1981, as endorsed with clarification by Regulatory Guide 1.8, Rev. 2, 4/87, except for the experience requirements that refer to operating modes that are no longer applicable to the permanently shut down and defueled condition of CR3.

4) With regard to section 4 of ANSI N18.1-1971, titled <u>Qualifications</u>: Selection and qualification of personnel is based on the established requirements of that position through CR3 hiring and Nuclear Operations policies, thereby meeting the intent of ANSI N18.1 The hiring policies are governed by the CR3 Human Resources Department.

5) With regard to paragraph 4.3 of ANSI N18.1-1971, titled <u>Supervisors</u>: Supervisors fulfilling a multi-discipline position possess at least four (4) years cumulative work experience in multiple disciplines (electrical, I&C, etc.) or four (4) or more years work history of multi-discipline experience.

6) With regard to paragraph 5.5 of ANSI N18.1-1971, titled <u>Retraining and Replacement Training</u>: CR3's retraining and replacement training program for the facility staff shall be maintained under the direction of the General Manager, Decommissioning, Crystal River Nuclear Plant. Subjects addressed in Section 5.5 of ANSI N18.1-1971 will be included as appropriate per the System Approach to Training in accordance with 10CFR50.120.

7) With regard to paragraph 4.2.1 of ANSI N18.1-1971, titled <u>Plant Manager</u>, and paragraph 4.2.2 of ANSI N18.1-1971, titled <u>Operations Manager</u>: Operators licenses, which are issued by the NRC and discussed in ANSI N18.1-1971, are no longer required based upon the NRC approval of TS Amendment 244. The CR3 management structure does not require any positions to attend equivalent training.

Table 1-3
Crystal River Unit 3 Quality Program Commitments
(continued)

NRC REGULATORY GUIDE 1.30 - "QUALITY ASSURANCE REQUIREMENTS FOR THE INSTALLATION, INSPECTION AND TESTING OF INSTRUMENTATION AND ELECTRIC EQUIPMENT" (8/72) - ENDORSES ANSI N45.2.4 - 1972.

With the Certification of Permanent Cessation of Power Operations, DEF has permanently ceased power operations of Crystal River Unit 3 (CR3). CR3 follows this Regulatory Guide and associated Standard for the remaining important to safety activities associated with SAFSTOR, decommissioning, and the Independent Spent Fuel Storage Installation (ISFSI) with the following clarifications:

1) For operational phase maintenance and modification activities which are comparable in nature and extent to similar activities conducted during the construction phase, CR3 shall either control these activities under this Quality Program or under an NRC accepted Construction QA Program. When this Quality Program is used, CR3 shall comply with the requirements of the Regulatory Position documented in this Guide in that QA programmatic/administrative requirements included therein (subject to the clarifications below) shall apply to these maintenance and modification activities even though such requirements may not have been in effect originally. Technical requirements associated with the maintenance and modifications shall be the original requirements or better (e.g., code requirements, material properties, design margins, manufacturing processes, and inspection requirements).

   The Scope of this Standard (Section 1.1) and the method of identifying calibration status (third sentence in Section 2.5.2) are defined in items 2 and 3 below based on two classes of instruments.

2) Portable items of measuring and test equipment (M&TE) and reference standards shall be tagged or labeled indicating the date of calibration and/or the due date of recalibration as well as the identity of person performing calibration. These items are in a calibration program which requires recalibration on a specified frequency or, in certain cases, prior to use.

3) Instrumentation and electrical equipment which are (1) instruments installed as listed in the PDTS, (2) installed instrumentation used to verify PDTS parameters, and (3) installed safety-related instruments and electrical equipment that provide an active function during operation or shutdown (i.e., vice being designated safety-related solely because the instrument is an integral part of a pressure retaining boundary) shall be in a calibration program. This program provides, by the use of status cards, computer schedules, or tags, for the date that recalibration is due and indicates the status of calibration. In addition, each applicable instrument is identified with a unique number. This number is utilized in instrument maintenance records so that current calibration status, including data such as the date of calibration and identity of the person that performed the calibration, can be readily determined.

4) With regard to Section 1.4 of ANSI N45.2.4 - 1971 titled Definition: Definitions in this Standard which are not included in ANSI N45.2.10 shall be used; all definitions which are included in ANSI N45.2.10 shall be used as clarified in CR3's commitment to Regulatory Guide 1.74.

5) With regard to Section 2.1 of ANSI N45.2.4 - 1971 titled Planning: Planning requirements, when necessary, shall be incorporated into maintenance and modification procedures.

6) With regard to Section 2.3 of ANSI N45.2.4 - 1971 titled Procedures and Instructions: Procedures and instructions shall be implemented as set forth in Sections 1.7.1.5, 10 and 11 of the Quality Program Description and by compliance with the Crystal River Unit 3 PDTS and ANSI N18.7 as set forth in Table 1-3 to that Program in lieu of the requirements set forth here.

7) With regard to Section 2.4 of ANSI N45.2.4 - 1971 titled Results: These requirements are met by implementing Sections 1.7.1.10, 11 and 17 of the Quality Program Description and by compliance with ANSI N18.7 as set forth in Table 1-3 of that Program in lieu of the requirements set forth here.

Table 1-3
Crystal River Unit 3 Quality Program Commitments
(continued)

8) With regard to Section 3 of ANSI N45.2.4 - 1971 titled Preconstruction Verification: These requirements shall be implemented as follows: (1) They are required only for major modifications. (2) They shall be implemented with the clarification that "approved instruction manuals" shall be interpreted to mean the manuals provided by the supplier as required by the procurement order - these manuals are not necessarily reviewed and approved, per se, by CR3.* (3) No special checks are required to be made by the person withdrawing a replacement part from the warehouse -equivalent controls are assured by compliance with ANSI N45.2.2 as set forth in Table 1-3 of the Quality Program Description. (4) They shall be complied with as stated, by individual technicians as part of the maintenance/modification process.

   *See CR3's commitment to ANSI N18.7 in this Table for a description of how vendor manuals may be included in procedures.

9) With regard to Section 4 of ANSI N45.2.4 - 1971 titled Installation: These requirements shall be implemented by inclusion, as necessary in the appropriate maintenance or modification procedure, where such procedures are used. Standard CR3 maintenance practices require that care be exercised in the six areas listed whether a procedure is required or not.

10) With regard to Section 5.1 of ANSI N45.2.4 - 1971 titled Inspections: The requirements of Section 5.1, including Subsections 5.1.1, 5.1.2 and the first sentence of 5.1.3, shall be implemented as set forth in Section 1.7.1.10 of the Quality Program Description. The inspection program shall incorporate, as applicable, those items listed in these Subsections. The remaining sentence in 5.1.3 is covered in equivalent detail in CR3's commitment to ANSI N18.7, Section 5.2.6: The requirements as set forth in that commitment shall be implemented in lieu of the requirements stated here.

11) With regard to Section 5.2 of ANSI N45.2.4 - 1971 titled Tests: The requirements of Section 5.2, including Subsections 5.2.1 through 5.2.3, shall be implemented as set forth in Sections 1.7.1.3 and 11 of the Quality Program Description. The test program shall consider the elements outlined in this Section, where applicable, when developing test requirements for inclusion in maintenance and modification procedures. In some cases, testing requirements may be met by post-installation surveillance testing in lieu of a special post-installation test.

12) With regard to Section 6 of ANSI N45.2.4 - 1971 titled Post-Construction Verification: This activity is not generally considered applicable at operating facilities because of the scope of the work and the relatively short interval between installation and operation. Where considered applicable, the elements described in this Section shall be considered in the development and implementation of inspection and testing programs as described in Sections 1.7.1.3, 10 and 11 of the Quality Program Description.

13) With regard to Section 6.2.1 of ANSI N45.2.4 - 1972 titled Equipment Tests: The last paragraph of this Section deals with tagging and labeling items requiring calibration within post-construction verification equipment tests. CR3 shall meet the intent of this paragraph by complying with the provisions of Clarification 3 to this standard.

14) With regard to Section 7 of ANSI N45.2.4 - 1971 titled Data Analysis and Evaluation: These requirements shall be implemented as stated herein after adding the clarifying phrase "Where used" at the beginning of the paragraph.

15) With regard to Section 8 of the ANSI N45.2.4 - 1972 titled Records: CR3 shall maintain records in accordance with and to meet the requirements of Section 1.7.1.17 of the Quality Program Description and ANSI N45.2.9 as specified in Table 1-3.

Table 1-3
Crystal River Unit 3 Quality Program Commitments
(continued)

NRC REGULATORY GUIDE 1.33 - "QUALITY ASSURANCE REQUIREMENTS (OPERATION)" (REV. 2, 2/78) - ENDORSES ANSI N18.7 - 1976/ANS-3.2

With the Certification of Permanent Cessation of Power Operations, DEF has permanently ceased power operations of Crystal River Unit 3 (CR3). CR3 follows this Regulatory Guide and associated Standard for the remaining important to safety activities associated with SAFSTOR, decommissioning, and the Independent Spent Fuel Storage Installation (ISFSI) with the following clarifications:

1) Paragraph C.3 of Regulatory Guide 1.33 (and Section 4.3.4 of ANSI N18.7 which it references) regarding subjects requiring independent review shall be implemented as described in FSAR Section 1.7.1.18.2.

2) Paragraph C.4 and subsections a through c of Regulatory Guide 1.33 (and Section 4.5 of ANSI N18.7 which they reference) shall be implemented as required by FSAR Section 1.7.1.18.1 which defines the "Audit Program" to be conducted. The audit program is further defined and shall be implemented as required by the commitment to ANSI N45.2.12 as stated in Table 1-3.

3) Paragraph C.5.a of Regulatory Guide 1.33 (and the second paragraph of Section 4.4 of ANSI N18.7 which it references) regarding review activities of the onsite operating organization shall be implemented by meeting the requirements of FSAR Section 6.8.1.

4) Paragraph C.5.d of Regulatory Guide 1.33 (and Section 5.2.7.1 of ANSI N18.7 which it references) shall be implemented by adding the clarifying phrase "Where applicable" in front of the fourth sentence of the fifth paragraph. The Regulatory Guide's changing of the two uses of the word "should" in this sentence to "shall" unnecessarily restricts CR3's options on repair or replacement parts. It is not always practicable to test parts prior to use. For modifications where these requirements are not considered practicable, a review in accordance with the provisions of 10 CFR 50.59 shall be conducted and documented.

5) Paragraph C.5.e of Regulatory Guide 1.33 (and Sections 5.2.13.4 of ANSI N18.7 which it references) shall be implemented subject to the same clarifications made for ANSI N45.2.2 elsewhere in Table 1-3.

6) Paragraph C.5.f of Regulatory Guide 1.33 (and Section 5.2.19(2) of ANSI N18.7 which it references) shall be implemented with the substitution of the word "practical" for the word "possible" in the last sentence.

7) Paragraph C.5.g of Regulatory Guide 1.33 (and Section 5.2.19.1 of ANSI N18.7 which it references) shall be implemented with the addition of the modifier "normally" after each of the verbs (should) which the Regulatory Guide converts to "shall." It is CR3's intent to fully comply with the requirements of this paragraph, and any conditions which do not fully comply shall be documented and approved by management personnel. In these cases, the reason for the exception shall also be documented. The documentation shall be retained for the same period of time as the related pre-operational test.

8) With regard to Section 3.4.2 of ANSI N18.7 - 1976 titled Requirements for the Onsite Operating Organization: Training Standards referenced in this Section shall be implemented if such Standards are included in Table 1-3 of the Quality Program Description or are otherwise part of the license of Crystal River Unit 3. CR3's method of documenting and otherwise meeting the remainder of the requirements of this Section are set forth in Section 1.7.1.2 of the Quality Program Description and in other commitments for Crystal River Unit 3.

9) With regard to Section 4.1 of ANSI N18.7 - 1976 titled General: The CR3 audit program shall be implemented in accordance with and to meet the requirements of ANSI N45.2.12 as endorsed in Table 1-3; Sections 1.7.1.16 and 1.7.1.18.1 of the Quality Program Description; and FSAR Section 6.8. The review aspects called out in this section are accomplished by CR3 using the Nuclear Oversight Department, as described in DSAR Section 1.7.1.18.1.3.

10) With regard to Section 4.2 of ANSI N18.7 - 1976 titled Program Description: Two aspects are addressed in this Section: audits and independent reviews. The independent review program is implemented as described in

Table 1-3
Crystal River Unit 3 Quality Program Commitments
(continued)

FSAR Section 6.8. The CR3 audit program shall be described in accordance with and to meet the requirements of ANSI N45.2.12 as clarified in Table 1-3 and Sections 1.7.1.16 and 18 of the Quality Program Description.

11) With regard to Section 4.3 of ANSI N18.7 - 1976 titled Independent Review Program: The requirements of this Section are replaced by Section 6.8.1, Plant Nuclear Safety Committee (PNSC). This exception uses NRC Safety Evaluation dated January 13, 2005 to Nuclear Management Company (ADAMS ML050210276). The actions identified for the Independent Review Function in Section 4.5 are performed by NOS. Similarly, the review of corrective actions for significant conditions adverse to quality are reviewed by NOS in the Audit program rather than by the Independent Review Function, unless the condition relates to a Licensee Event Report.

12) With regard to Section 4.4 of ANSI N18.7 - 1976 titled Review Activities of the Onsite Operating Organization: The requirements of this section are met by compliance with FSAR Sections 6.8 and 6.8.1.

13) With regard to Section 4.5 of ANSI N18.7 - 1976 titled Audit Program: The CR3 audit program shall be implemented in accordance with and to meet the requirements of ANSI N45.2.12 as clarified in Table 1-3 and Sections 1.7.1.16, 17, and 18 of the Quality Program Description. The written audit reports are not formally reviewed as part of the Independent Review function. Audits are scheduled as identified in Section 1.7.1.18. Except when the frequency is specified by regulation, the following criteria for extending audit intervals apply:

    a) Schedules are based on the anniversary established for each audit.

    b) A maximum extension not to exceed 25 percent of the audit interval may be allowed (e.g., audits on a two year frequency may not be extended beyond 30 months, audits on an annual frequency may not be extended beyond 15 months).

    c) When an audit interval extension is used, the next audit for that particular audit area is scheduled from the original anniversary.

    d) Provision b) also applies to supplier audits and evaluations except that a total combined time interval for any three consecutive inspection or audit intervals should not exceed 3.25 times the specified inspection or audit interval.

14) With regard to Section 5.1 of ANSI N18.7 - 1976 titled Program Description: The fourth sentence in this Section required a "summary document"; CR3's Quality Program Description (Chapter 1.7 of the DSAR) is organized in accordance with the 18 criteria of 10 CFR 50, Appendix B. CR3 interprets this DSAR description to fulfill the requirements for a "summary document."

15) With regard to Section 5.2.2 of ANSI N18.7 - 1976 titled Procedure Adherence: Temporary changes to procedures recommended in Regulatory Guide 1.33, Appendix A and others specified in Clarification 29, below, may be made provided:

    a. The intent of the existing procedure is not altered; and

    b. The change is approved by two members of the plant management staff, at least one of whom is an ISFSI Shift Supervisor; and

    c. The change is documented and subsequently reviewed and approved in accordance with Administrative procedures or policies.

16) With regard to Section 5.2.6 of ANSI N18.7 - 1976 titled Equipment Control: CR3 shall comply with the "independent verification" requirements based on the definition of this phrase as given under our commitment to Regulatory Guide 1.74.

    The third sentence of the fourth paragraph requires independent verification "when appropriate." In the event of significant radiation levels is an area requiring independent verification, the CR3 Radiation Protection staff

Table 1-3
Crystal River Unit 3 Quality Program Commitments
(continued)

makes recommendations regarding access. However, the final decision regarding access for independent verification will be made by Operations.

Since CR3 sometimes uses descriptive names to designate equipment, the sixth paragraph, second sentence is replaced with: "Suitable means include identification number or other descriptions which are traceable to records of the status of inspections and tests."

The first sentence in the seventh paragraph shall be complied with after clarifying "operating personnel" to mean trained employees assigned to or under the control of CR3 management.

17) With regard to Section 5.2.7 of ANSI N18.7 - 1976 titled Maintenance and Modification: For other modifications, we are committed to the inspection requirements of Section 5.2.17 of ANSI N18.7 - 1976.

Further with regard to Section 5.2.7 of ANSI N18.7 - 1976 titled Maintenance and Modification: Since some emergency situations could arise which might preclude preplanning of all activities, CR3 shall comply with the following alternate to the first sentence in the second paragraph: "Except in emergency or abnormal operating situations where immediate actions are required to protect the health and safety of the public, to protect equipment or personnel, or to prevent the deterioration of plant conditions to a possibly unsafe or unstable level, maintenance or modification of equipment shall be preplanned and performed in accordance with approved written procedures. Where approved written procedures would be required and are not used, the activities that were accomplished shall be documented after-the-fact and receive the same degree of review as if they had been preplanned."

18) With regard to Section 5.2.7.1 of ANSI N18.7 - 1976 titled Maintenance Programs: CR3 shall comply with the requirements of the first sentence of the fifth paragraph, where practical. This clarification is needed since it is not always possible to promptly determine cause of the malfunction. In all cases, CR3 shall initiate proceedings to determine the cause, and shall make such determinations promptly, where practical.

19) With regard to Section 5.2.8 of ANSI 18.7 - 1976 titled Surveillance Testing and Inspection Schedule: In lieu of a "master surveillance schedule," the following requirement shall be complied with: "A surveillance testing schedule(s) shall be established reflecting the status of all in-plant surveillance tests and inspections."

20) With regard to Section 5.2.9 of ANSI N18.7 - titled Plant Security and Visitor Control: The requirements of the Crystal River Unit 3 Security Plan shall be implemented in lieu of these general requirements.

21) With regard to Section 5.2.10 of ANSI N18.7 - 1976 titled Housekeeping and Cleanliness Control: The requirements of this Section, beginning with the last sentence of the first paragraph and continuing through the end of the Section, shall be implemented as described in CR3's commitments to ANSI N45.2.3 and N45.2.1 as set forth in Table 1-3 of the Quality Program Description.

22) With regard to Section 5.2.13.1 of ANSI N18.7 - 1976 titled Procurement Document Control: The words "the same" in the last sentence are replaced with the words "an equivalent."

When purchasing commercial-grade calibration services from certain accredited calibration laboratories, the procurement documents are not required to impose a quality assurance program consistent with ANSI N45.2-1971. Alternate requirements described in Table 1-3 for Regulatory Guide 1.123 may be implemented in lieu of imposing a quality assurance program consistent with ANSI N45.2-1971.

23) With regard to Section 5.2.17 of ANSI N18.7 - 1976 titled Inspection: With respect to the fourth paragraph, not all inspections may require generation of a separate inspection procedure. Inspection requirements may be integrated into appropriate procedures or other documents with the procedure or document serving as the record. However, records of inspections shall be identifiable and retrievable.

Table 1-3
Crystal River Unit 3 Quality Program Commitments
(continued)

24) With regard to Section 5.2.19 of ANSI N18.7 - 1976 titled Test Control: Item (1) of this Section is not considered applicable since the CR3 Quality Program Description does not cover pre-operational testing.

25) With regard to Section 5.3.5(4) of ANSI N18.7 - 1976 titled Supporting Maintenance Documents: CR3 may choose to include material from vendor manuals in any of three ways: (1) The applicable section of the manual may be duplicated, referenced in and attached to the procedure. (2) The procedure may simply state that the manual or a specific section is to be followed for performing a particular function; the manual must then be used in conjunction with the procedure for performing the activity. (3) The pertinent material from the manual, either as originally written or as modified by the author/reviewers of the procedure, may be written into and become a part of the procedure.

    In options (1) and (3) above, the material meets the requirement to receive "the same level of review and approval as operating procedures" since the material is reviewed as part of the procedure review process. In option (2), the requirements shall be deemed to have been fulfilled by requiring a copy of the pertinent manual (manual sections) to be available to and considered by persons conducting the review of the procedure.

    Several procedures exist which establish the responsibilities for technical evaluation, distribution and retention of vendor manuals and updates to vendor manuals.

    The existing practice is based on pertinent vendor data being sent to the Engineering Section, either from the vendor or by inter-department transmittal. Engineering reviews this information for technical content and applicability to CR3. The relevant information is then transmitted to Document Management. Document Management controls and distributes vendor manuals per administrative procedures.

    The current practice is to use controlled vendor manuals at the job site or work area (as a reference only) to reinforce the above options; and to perform the work on safety-related equipment in accordance with the approved plant procedures and/or work instructions.

26) With regard to Section 5.3.9 of ANSI N18.7 - 1976 titled Emergency Procedures: As directed by the NRC, CR3 has developed a format for emergency procedures which is "symptom" based as opposed to "event" based as stipulated in Section 5.3.9.1.

27) With regard to Section 5.3.9.2 of ANSI N18.7 - 1976 titled Events of Potential Emergency: NRC review of the FSAR has identified all natural occurrences which affect Crystal River Unit 3. Therefore, CR3 shall interpret item (11) to mean the natural occurrences which have been evaluated in the FSAR for Crystal River Unit 3.

28) With regard to Section 5.3.9.3 of ANSI N18.7 - 1976 titled Procedures for Implementing Emergency Plan: CR3's NRC accepted Emergency Plan for Crystal River Unit 3 shall be implemented in lieu of the requirements in this Section.

29) Paragraph C.1 and Appendix A of Regulatory Guide 1.33 shall be implemented as required by the Crystal River Unit 3 PDTS 5.6.1.1.a, "Procedures." In addition, CR3 shall establish, implement, and maintain written procedures covering the activities referenced below:
    a. Refueling Operations?
    b. Surveillance and test activities of safety related equipment
    c. Process Control Program implementation

30) With regard to Section 3.4.2 of ANSI N18.7-1976 titled Requirements for the Onsite Operating Organizations: Some of CR3's technical support organizations are physically located at the CR3 site. Therefore, the second sentence of this section shall be implemented as follows: "Initial incumbents or replacements for members of the onsite operating organization and onsite or offsite technical support organizations shall have appropriate experience, training, and retraining to assure that necessary competence is maintained in accordance with the

Table 1-3
Crystal River Unit 3 Quality Program Commitments
(continued)

provisions of American National Standard for Selection and Training of Nuclear Power Plant Personnel, N18.1-1971."

For purposes of implementing Section 3.4.2 as clarified above, CR3 also defines the CR3 Relay Department Technicians who may perform safety-related activities at CR3 to be members of an offsite technical support organization.

31) With regard to Section 5.2.19.1 of ANSI N18.7-1976 titled Pre-operational Tests: This section will only apply in the event of major modification activities which are similar in nature and extent to those activities that occurred during initial construction of CR3.

32) With regard to Section 5.2.19.2 of ANSI N18.7-1976 titled Tests Prior To And During Initial Plant Operation: This period of operation has already passed and therefore this section is not applicable.

33) With regard to Section 2.2 of ANSI N18.7-1976 titled Glossary of Terms: Definitions in this Standard which are not included in ANSI N45.2.10 shall be used. All definitions which are included in ANSI N45.2.10 shall be used as clarified in CR3's commitment to Regulatory Guide 1.74. The term "onsite operating organization," defined in general terms in this Standard, shall be synonymous with "Facility Staff" which is defined in CR3's commitment to Regulatory Guide 1.74.

34) With regard to Section 5.2.15 of ANSI N18.7-1976 titled Review, Approval and Control of Procedures: CR3 provides the following alternative guidance:

Each procedure described in CR3's commitment to Regulatory Guide 1.33 as clarified above, and changes thereto, shall be reviewed and approved. The review cycle shall consist of an intra-departmental review by a Technical reviewer, and an interdisciplinary review by Impact Reviewer(s) in interfacing departments when another department's actions or procedures are impacted, and approval by the responsible Approval Authority for the procedure as specified in administrative procedures.

Non-routine procedures such as Emergency Operating Procedures (EOP), Abnormal Procedures (AP), and the Conduct of Operations During Abnormal and Emergency Events procedure shall be reviewed no less frequently than every two years to determine if changes are necessary.

Other procedures and programs (e.g., Process Control Program) shall be revised as necessary. Revisions will generally be initiated through reviews conducted by personnel during routine performance of activities. Examples of such reviews include specific assignments for non-routine procedures mentioned above; evaluations of problems encountered during performance of a procedure; evaluation of corrective actions for self-identified deficiencies or events; evaluation of events occurring at other plants; evaluation of procedure changes necessary to implement modifications; evaluation of procedure changes necessary to implement License, PDTS, or DSAR revisions; as well as evaluations of changes necessary to resolve Regulatory Issues. Such changes shall be implemented as necessary. In some situations such implementation will be completed prior to completion of the in-process activity. Guidance on the need to revise procedures shall be provided in plant administrative controls."

Table 1-3
Crystal River Unit 3 Quality Program Commitments
(continued)

NRC REGULATORY GUIDE 1.37 - "QUALITY ASSURANCE REQUIREMENTS FOR CLEANING OF FLUID SYSTEMS AND ASSOCIATED COMPONENTS OF WATER-COOLED NUCLEAR POWER PLANTS" (3/73) - ENDORSES ANSI N45.2.1 - 1973.

With the Certification of Permanent Cessation of Power Operations, DEF has permanently ceased power operations of Crystal River Unit 3 (CR3). CR3 follows this Regulatory Guide and associated Standard for the remaining important to safety activities associated with SAFSTOR, decommissioning, and the Independent Spent Fuel Storage Installation (ISFSI) with the following clarifications:

Throughout this Standard, references are made to cleanness associated with initial installation. For operations activities, qualified maintenance department personnel shall determine what items within a particular cleanness level are appropriate. As an example, the Reactor Internals are initially classified as Level B. Section 3.1.2 requires this cleanness level to have "metal clean" surfaces and either a visual inspection or a dry white-cloth wipe. After use in the Reactor Coolant System, CR3 does not intend to clean components removed for maintenance or inspection until they have a bright "metal clean" surface, nor do we necessarily commit to performance of a dry white-cloth wipe. The chemistry within the system is designed to obtain a tightly adhering metal oxide film (which reduces the luster of metal) and ALARA considerations may preclude a dry white-cloth wipe.

Components which have been removed from a system shall be considered as being the same cleanness level as the system from which they were removed. Such components need not be recleaned to meet initial installation cleanness levels prior to reinstallation. However, CR3 shall assure that measures are taken to prevent degradation of the previously established or existing cleanness state of the system.

1) For operational phase maintenance and modification activities which are comparable in nature and extent to similar activities conducted during the construction phase, CR3 shall either control these activities under this Quality Program Description or under an NRC accepted Construction QA Program. When this Quality Program Description is used, CR3 shall comply with the requirements of the Regulatory Position documented in this Guide in that QA programmatic/administrative requirements included therein (subject to the clarifications below) shall apply to these maintenance and modification activities even though such requirements may not have been in effect originally. Technical requirements associated with the maintenance and modifications shall be the original requirements or better (e.g., code requirements, material properties, design margins, manufacturing processes, and inspection requirements).

2) This Guide and Standard are applicable to those areas of the Quality Assurance Program addressing onsite cleaning of materials and components, cleanliness control, and cleaning and layup of fluid systems during the operational phase. They do not cover offsite activities unless invoked in procurement documents.

3) With regard to Paragraph C.3 of Regulatory Guide 1.37: The water quality for final flushing of fluid systems and associated components shall be at least equivalent to the quality of the operating system water for the oxygen and nitrogen content; but this does not infer that chromates or other additives, normally in the system water, will be added to the flush water.

4) With regard to Paragraph C.4 of Regulatory Guide 1.37: Expendable materials, such as inks and related products; temperature indicating sticks; tapes; gummed labels; wrapping materials (other than polyethylene); water soluble dam materials; lubricants; NDT penetrant materials, couplants, and desiccants, which contact stainless steel or nickel alloy surfaces shall not contain lead, zinc, copper, mercury, cadmium and other low melting point metals, their alloys or compounds as basic and essential chemical constituents. No more than 0.1 percent (1,000 ppm) halogens shall be allowed where such elements are leachable or where they could be released by breakdown of the compounds under expected environmental conditions.

Table 1-3
Crystal River Unit 3 Quality Program Commitments
(continued)

5) With regard to Section 1.4 of ANSI N45.2.1 - 1973 titled Definitions: Definitions in this Standard which are not included in ANSI N45.2.10 shall be used; all definitions which are included in ANSI N45.2.10 shall be used as clarified in CR3's commitment to Regulatory Guide 1.74.

6) With regard to Section 5 of ANSI N45.2.1 - 1973 titled Installation Cleaning: The recommendation that local rusting on corrosion resistant alloys be removed by mechanical methods is interpreted to mean that local rusting may be removed mechanically, but the use of other removal means is not precluded.

7) With regard to Section 9 of ANSI N45.2.1 - 1973 titled Records: CR3 shall maintain records in accordance with and to meet the requirements of Section 1.7.1.17 of the Quality Program Description and ANSI N45.2.9 as specified in Table 1-3.

Table 1-3
Crystal River Unit 3 Quality Program Commitments
(continued)

NRC REGULATORY GUIDE 1.38 - "QUALITY ASSURANCE REQUIREMENTS FOR PACKAGING, SHIPPING, RECEIVING, STORAGE, AND HANDLING OF ITEMS FOR WATER-COOLED NUCLEAR POWER PLANTS" (REV. 2, 5/77) - ENDORSES ANSI N45.2.2 - 1972.

With the Certification of Permanent Cessation of Power Operations, DEF has permanently ceased power operations of Crystal River Unit 3 (CR3). CR3 follows this Regulatory Guide and associated Standard for the remaining important to safety activities associated with SAFSTOR, decommissioning, and the Independent Spent Fuel Storage Installation (ISFSI) with the following clarifications:

1) With regard to Section 1.4 of ANSI N45.2.2 - 1972 titled Definitions: Definitions in this Standard which are not included in ANSI N45.2.10 shall be used; all definitions which are included in ANSI N45.2.10 shall be used as clarified in CR3's commitment to Regulatory Guide 1.74.

2) With regard to Section 2.1 of ANSI N45.2.2 - 1972 titled Planning: (First sentence) The specific items to be governed by the Standard shall be identified on the "Safety Listing". However, the Standard (as modified by clarifications in Table 1-3) is part of CR3's Quality Program Description and it shall therefore, be applied to those structures, systems, and components which are included in that Program.

3) With regard to Section 2.3 of ANSI N45.2.2 - 1972 titled Results: The specific methods for performing and documenting tests and inspections are given in Sections 1.7.1.10 and 11 of the Quality Program Description. These requirements in these Sections shall be implemented in lieu of the general requirements here.

4) With regard to Section 2.4 of N45.2.2 - 1972 titled Personnel Qualifications: Specific requirements for personnel qualifications and training are set forth in Section 1.7.1.2 and in the commitments to training standards in Table 1-3 of the Quality Program Description. These requirements shall be implemented in lieu of the general requirements stated in this Section.

5) With regard to Section 2.7 of ANSI N45.2.2 - 1972 titled Classification of Items: CR3 may choose not to explicitly use the four level classification system. However, the specific requirements of the Standard that are appropriate to each class are generally applied to the items suggested in each classification and to similar items.

6) With regard to Section 3.2.1 of ANSI N45.2.2 - 1972 titled Level A Items: As an alternate to the requirements for packaging and containerizing items in storage to control contaminants [items (4) and (5)], CR3 may choose a storage atmosphere which is free of harmful contaminants in concentrations that could produce damage to stored items. Similarly [for item (7)], CR3 may obviate the need for caps and plugs with an appropriate storage atmosphere, and may choose to protect weld-end preparations and threads by controlling the manner in which the items are stored. These clarifications apply whenever items (4), (5) or (7) are subsequently referenced and to Section 3.5.1 titled Caps and Plugs and Section 3.4 titled Methods of Preservation.

7) With regard to Section 3.2.2. of ANSI N45.2.2 - 1972 titled Level B Items: As an alternate to the requirements for packaging of Level B items in containers or crates, particularly large items that may not be suitable for shipment in a fully enclosed van, (reference item (5) of Section 3.2.1), CR3 may choose to use an alternate means of packaging to ensure the item is adequately protected during shipment.

8) With regard to Section 3.3 of ANSI N45.2.2 - 1972 titled Cleaning: (Third sentence) CR3 interprets "documented cleaning methods" to allow generic cleaning procedures to be written which are implemented, as necessary, by trained personnel. Each particular cleaning operation may not have an individual cleaning procedure, or which type(s) or solvent(s) may be used in a particular application.

9) With regard to Section 3.4 of ANSI N45.2.2 - 1972 titled Methods of Preservation: (First sentence) CR3 shall comply with these requirements subject to the clarifications of Section 3.2.1 (4) and (5) above, and the definition of the phrase "deleterious corrosion" to mean that corrosion which cannot be subsequently removed and which adversely affects form, fit, or function.

Table 1-3
Crystal River Unit 3 Quality Program Commitments
(continued)

10) With regard to Section 3.6 of ANSI N45.2.2 - 1972 titled Barrier and Wrap Material and Desiccants: This section requires the use of nonhalogenated materials in contact with austenitic stainless steel. Refer to Regulatory Guide 1.37 for the CR3 position.

11) With regard to Section 3.7.1 of ANSI N45.2.2 - 1972 titled Containers: Cleated, sheathed boxes may be used up to 1000 lbs. rather than 500 lbs. as specified in 3.7.1(1). This type of box is safe for, and has been tested for, loads up to 1000 lbs. Other national standards allow this (see Federal Specification PPP-B-601). Special qualification testing may be required for loads above 1000 lbs.

12) With regard to Section 3.7.2 of ANSI N45.2.2 - 1972 titled Crates and Skids: Crates and/or Skids shall normally be used for equipment in excess of 500 lbs. Skids and runners shall normally be used on boxes with a gross weight of 100 lbs. or more. Skids and runners are normally fabricated from four inch nominal lumber size and laid flat except where this is impractical because of the small dimensions of the container. If forklift handling is required, minimum floor clearance for forklift tines shall be provided.

13) With regard to Section 4.2.2 of ANSI N42.2.2 - 1972 titled Closed Carriers: The use of fully enclosed furniture vans, as recommended in (2) of this Section, is not considered a requirement. CR3 assures adequate protection from weather or other environmental conditions by a combination of vehicle enclosures and item packaging.

14) With regard to Sections 4.3, 4.4 and 4.5 of ANSI N45.2.2 - 1972 titled, respectively, Precautions During Loading and Transit, Identification and Marking, and Shipment from Countries Outside the United States: CR3 shall comply with the requirements of these Sections subject to the clarifications taken to other Sections which are referenced therein.

15) With regard to Section 5.2.1 of ANSI N45.2.2 - 1972 titled Shipping Damage Inspection: Warehouse personnel normally visually scrutinize incoming shipments for damage of the types listed in this Section; this activity is not necessarily performed prior to unloading. Since all required items receive the Item Inspection of Section 5.2.2, separate documentation of the Shipping Damage Inspection is not necessary. Release of the transport agent after unloading and the signing for receipt of the shipment may be all of the action taken to document completion of the Shipping Damage Inspection. Any nonconformance noted shall be documented and dispositioned as required by Section 1.7.1.15 of the Quality Program Description. The person performing the visual scrutiny during unloading is not considered to be performing an inspection function as defined under Regulatory Guides 1.58 and 1.74; therefore, while he shall be trained to perform this function, he may not necessarily be certified (N45.2.6) as an Inspector.

16) With regard to Section 5.2.2 of ANSI N45.2.2 - 1972 titled Item Inspection: The second division of this subsection requires six additional inspection activities if an item was not inspected or examined at the source. Nuclear Engineering and Services shall determine and document the extent of receipt inspection based on consideration of Section 5.2.2.

17) With regard to Section 6.1.2 of ANSI N45.2.2 - 1972 titled Levels of Storage: Subpart (2) is replaced with the following:

    (2) Level B items shall be stored within a fire resistant, weathertight, and well ventilated building or equivalent enclosure. This building shall be situated and constructed so that it shall not normally be subject to flooding; the floor shall be paved or equal, and well drained. If any outside waters should come in contact with stored equipment, such equipment shall be labeled or tagged nonconforming, and then the nonconformance document shall be processed and evaluated in accordance with Section 1.7.1.15. Items shall be placed on pallet for shoring or shelves to permit air circulation. The building shall be provided with heating and temperature control or its equivalent to reduce condensation or corrosion. Minimum temperature shall be 40°F and maximum temperature shall be 140°F or less if so stipulated by a manufacturer.

Table 1-3
Crystal River Unit 3 Quality Program Commitments
(continued)

18) With regard to Section 6.2.1 of ANSI N45.2.2 - 1972 titled Access to Storage Areas: Items which fall within the Level D classification of the Standard may be stored in an area which is posted to limit access, but other controls such as fencing or guards are not considered to be required.

19) With regard to Section 6.2.4 of ANSI N45.2.2 - 1972 titled Storage of Food and Associated Items: This Section is replaced with the following: "The use of food, drinks, and salt tablet dispensers in any storage area shall be controlled and shall be limited to designated areas where use or storage is not deleterious to stored items."

20) With regard to Section 6.2.5 of ANSI N45.2.2 - 1972 titled Measures to Prevent Entrance of Animals: This Section is replaced with the following: "Warehouse personnel shall be alert to detect evidence of rodents or small animals in indoor storage areas. If any evidence is detected, a survey or inspection shall be utilized to determine the extent of any possible damage or contamination. Exterminators, traps, poisons or other appropriate measures shall be used to control these animals to minimize possible undesirable effects on stored materials."

21) With regard to Section 6.3.3 of ANSI N45.2.2. - 1972 titled Storage of Hazardous Material: This Section is replaced with the following: "Hazardous chemicals, paints, solvents and similar materials shall be stored in approved cabinets which are not in close proximity to installed plant systems which are required for safe shutdown or long term cooling of the plant."

22) With regard to Section 6.4.2 of ANSI N45.2 - 1972 titled Care of Items: The following alternates are provided for indicated subparts:

    (5) "Space heaters in electrical equipment shall be energized unless a documented engineering evaluation determines that such space heaters are not required."
    (6) "Large (greater than or equal to 50HP) rotating electrical equipment shall be given insulation resistance tests on a scheduled basis unless a documented engineering evaluation determines that such tests are not required."
    (7) "Prior to being placed in storage, rotating equipment weighing over approximately 50 pounds shall be evaluated by engineering personnel to determine if shaft rotation in storage is required: The results of the evaluation shall be documented. If rotation is required, it shall be performed at specified intervals, be documented, and be conducted so that parts receive a coating of lubrication where applicable and so that the shaft does not come to rest in the same position occupied prior to rotation. For long shafts or heavy equipment subject to undesirable bowing, shaft orientation after rotation shall be specified and obtained."

23) With regard to Section 6.5 of ANSI N45.2.2 - 1972 titled Removal of Items From Storage: CR3 does not consider the last sentence of this Section to normally apply to the operational phase due to the relatively short period of time between installation and use. The first sentence of the Section is replaced with: "CR3 shall develop, issue, and implement a procedure(s) which covers the removal of items from storage. The procedure(s) assures that the status of all material issued is known, controlled, and appropriately dispositioned."

    However, when the period of time between installation and use of modifications is not short, then CR3 must define the controls which are to be implemented to assure that the quality of installed items is not degraded prior to use.

24) With regard to Section 6.6 of ANSI N45.2.2 - 1972 titled Storage Records: CR3 shall comply with the requirements of this Section with the clarification that, for record purposes, only the access of non-CR3 personnel into indoor storage areas shall be recorded. Unloading or pick-up of material shall not be considered access, nor shall inspection by NRC or other regulatory agents, nor shall tours by non-CR3 employees who are accompanied by CR3 employees.

Table 1-3
Crystal River Unit 3 Quality Program Commitments
(continued)

25) With respect to Section 6.5 of ANSI N45.2.2 - 1972 titled Removal of Items from Storage: In addition to the clarification as stated in 22) above, CR3 defines in-process controls applied to material after removal of items from warehouse storage and prior to installation or use in the plant as follows:

a) During the period between issuance and installation, measures shall be taken to assure protection at least equivalent to that which exists at the intended installation location.
b) Based upon the short in-process duration and on final acceptance following inspection and testing, rotation of equipment, meggering, energizing of space heaters, periodic inspections, access controls, and similar items primarily associated with warehouse storage are not required during the in-process period.

Consistent with overall CR3 planning objectives, safety-related material needed for maintenance and modifications should be staged (i.e., collected together as a discrete package) by warehouse personnel if practical. Such material should be maintained in the warehouse and under warehouse controls until needed for installation.

Table 1-3
Crystal River Unit 3 Quality Program Commitments
(continued)

NRC REGULATORY GUIDE 1.39 - "HOUSEKEEPING REQUIREMENTS FOR WATER-COOLED NUCLEAR POWER PLANTS" (REV. 2, 9/77) - ENDORSES ANSI N45.2.3 - 1973

With the Certification of Permanent Cessation of Power Operations, DEF has permanently ceased power operations of Crystal River Unit 3 (CR3). CR3 follows this Regulatory Guide and associated Standard for the remaining important to safety activities associated with SAFSTOR, decommissioning, and the Independent Spent Fuel Storage Installation (ISFSI) with the following clarifications:

1) For operational phase maintenance and modification activities which are comparable in nature and extent to similar activities conducted during the construction phase, CR3 shall either control these activities under the Quality Program Description or under an NRC accepted Construction QA Program. When the Quality Program Description is used, CR3 shall comply with the requirements of the Regulatory Position documented in this Guide in that QA programmatic/administrative requirements included therein (subject to the clarifications below) shall apply to these maintenance and modification activities even though such requirements may not have been in effect originally. Technical requirements associated with the maintenance and modifications shall be the original requirements or better (e.g., code requirements, material properties, design margins, manufacturing processes, and inspection requirements).

2) With regard to Section 1.4 of ANSI N45.2.3 - 1973 titled Definitions: Definitions in this Standard which are not included in ANSI N45.2.10 shall be used. All definitions which are included in ANSI N45.2.10 shall be used as clarified in CR3's commitment to Regulatory Guide 1.74.

3) With regard to Section 2.1 of ANSI N45.2.3 - 1971 titled Planning: CR3 may choose not to utilize the five-level zone designation system, but shall utilize standard janitorial and work practices to maintain a level of cleanliness commensurate with company policy in the areas of housekeeping, plant and personnel safety, and fire protection.

   Cleanliness shall be maintained, consistent with the work being performed, so as to prevent the entry of foreign material into safety-related systems. This shall include, as a minimum, documented cleanliness inspections which shall be performed prior to system closure. As necessary, (e.g., the opening is larger than the tools being used) control of personnel, tools, equipment, and supplies shall be established when major portions of the reactor system are opened for inspection, maintenance or repair.

   Additional housekeeping requirements shall be implemented as required for control of radioactive contamination.

4) With regard to Section 2.2 of ANSI N45.2.3 - 1973 titled Procedures and Instructions: Procedures and instructions shall be implemented as set forth in Sections 1.7.1.5, 10 and 11 of the Quality Program Description and by compliance with the Crystal River Unit 3 PDTS and ANSI N18.7 as set forth in Table 1-3 of that Program in lieu of the requirements set forth here.

5) With regard to Section 3.1 of ANSI N45.2.3 - 1973 titled Control of Site Area: Not applicable to the operational phase.

6) With regard to Section 3.2 of ANSI N45.2.3 - 1973 titled Control of Facilities including subsections 3.2.1 and 3.2.2: CR3 may choose not to utilize the five-level zone designation system, but shall utilize standard janitorial and work practices to maintain a level of cleanliness commensurate with company policy in the areas of housekeeping, plant and personnel safety, and fire protection.

   Cleanliness shall be maintained, consistent with the work being performed, so as to prevent the entry of foreign material into safety-related systems. This shall include, as a minimum, documented cleanliness inspections which shall be performed prior to system closure. As necessary (e.g., the opening is larger than the tools being

Table 1-3
<u>Crystal River Unit 3 Quality Program Commitments</u>
(continued)

used), control of personnel, tools, equipment, and supplies shall be established when major portions of the reactor system are opened for inspection, maintenance or repair.

Additional housekeeping requirements shall be implemented as required for control of radioactive contamination.

7) With regard to Section 3.3 of ANSI N45.2.3 - 1973 titled <u>Materials and Equipment</u>: The first paragraph in this Section is not applicable to the operational phase. Most of the items in this Section were written for construction activities. Maintenance and modification activities which are similar in nature and extent to construction activities shall be handled as set forth in Item 1 above.

8) With regard to Section 3.4 of ANSI N45.2.3 - 1973 titled <u>Construction Tools, Supplies, and Equipment</u>: Most of the items in this Section were written for construction activities. Maintenance and modification activities which are similar in nature and extent to construction activities shall be handled as set forth in Item 1 above. The portions of this Section which are considered applicable by responsible maintenance supervisory personnel shall be complied with as stated. Nuclear Oversight (NOS) Section personnel verify that appropriate controls are used through procedure reviews and surveillance activities.

9) With regard to Section 3.5 of ANSI N45.2.3 - 1973 titled <u>Surveillance, Inspection, and Examination</u>: Subparagraph (1) is not applicable to the operational phase; (2), (3), and (4) shall be implemented.

10) With regard to Section 4 of ANSI N45.2.3 - 1973 titled <u>Records</u>: CR3 shall maintain records in accordance with and to meet the requirements of Section 1.7.1.17 of the Quality Program Description and ANSI N45.2.9 as specified in Table 1-3.

11) With regard to Section 2.3 of ANSI N45.23-1973 titled <u>Results</u>: These requirements are met by implementing sections 1.7.1.10, 1.7.1.11 and 1.7.1.17 of the Quality Program Description and by compliance with ANSI N18.7 as set forth in Table 1-3 of that program in lieu of the requirements set forth here.

12) With regard to Section 2.4 of ANSI N45.2.3-1973 titled <u>Personnel Qualifications</u>: CR3 will substitute the following paragraph for this section of the standard, "All personnel working in areas with special housekeeping controls shall be familiar with the necessities and requirements for cleanness control applicable to those areas. Training programs shall be utilized for this purpose where appropriate."

Table 1-3
Crystal River Unit 3 Quality Program Commitments
(continued)

NRC REGULATORY GUIDE 1.58 - "QUALIFICATION OF NUCLEAR POWER PLANT INSPECTION, EXAMINATION AND TESTING PERSONNEL" (REV. 1, 9/80) - ENDORSES ANSI N45.2.6 - 1978.

With the Certification of Permanent Cessation of Power Operations, DEF has permanently ceased power operations of Crystal River Unit 3 (CR3). CR3 follows this Regulatory Guide and associated Standard for the remaining important to safety activities associated with SAFSTOR, decommissioning, and the Independent Spent Fuel Storage Installation (ISFSI) with the following clarifications:

1) With regard to Regulatory Position C.1 (and Section 1.2 of ANSI N45.2.6 - 1978 which it references), the qualifications of each member of the facility staff shall be as set forth in PDTS or in Items 2 or 8 below.

2) With regard to Regulatory Positions C.2, C.3 and C.6 (and Sections 1.2 and 3.5 of ANSI N45.2.6 - 1978 which they reference), personnel performing inspections, examinations or tests which are NOT covered by Item 1 above or Items 8, 9 or 12 below shall be qualified as follows:

   a. (1) Personnel performing nondestructive testing at CR3, whether CR3 or contract personnel, shall be qualified to the requirements specified within CR3's written practice and implementing procedure(s) which are based on the 1984 revision of SNT-TC-1A and/or the 1995 Edition of ANSI/ASNT CP-189, as referenced by the 2001 Edition through the 2003 Addenda of the ASME Boiler and Pressure Vessel (B&PV) Code, Section XI, which CR3 is committed to in the Repair and Replacement Program.

   (2) CR3 or contract personnel performing nondestructive testing of new systems at CR3, which are not within the scope of the Repair and Replacement Program, shall be qualified to the requirements specified within CR3's written practice and implementing procedure(s) which are based on the 1984 revision of SNT-TC-1A referenced by the 1989, 1992 and 1995 Editions and Addenda of Section III of the ASME Boiler and Pressure Vessel (B&PV) Code. The specific Edition and Addenda will be identified within the modification design package.

   (3) A non-destructive test examiner qualified under earlier Editions and Addenda of ASME Section XI shall not require requalification under later Editions and Addenda until such time as the examiner is subject to requalification.

   b. Quality Control (QC) and Non-Destructive Examination (NDE) personnel performing inspection, examination, and test activities at CR3, whether CR3 or contract personnel, shall be qualified in accordance with ANSI N45.2.6-1978. Additionally, CR3 elects not to apply the requirements of this guide to those personnel who are involved in the daily operations of surveillance, maintenance and certain technical and support services whose qualifications are controlled by PDTS or are controlled by other Quality Assurance (QA) program commitment requirements. Only personnel in the following listed categories will be required to meet ANSI N45.2.6-1978 requirements:

   (1) NDE personnel

   (2) QC inspection personnel

   (3) Receipt inspection personnel

3) With regard to Regulatory Position C.4 (and Section 1.5 of ANSI N45.2.6-1978 which it references), CR3's generic clarification at the beginning of Table 1-3 is consistent with this NRC Position.

4) With regard to Regulatory Position C.5 (and Section 3.4 of ANSI N45.2.6-1978 which it references), and ANSI N45.2.6-1978 Sections 3.2 and 3.3, CR3 reserves the right to specify activities which may be conducted by use of documented position descriptions rather than using the specific Level I, II and III designation used in the

Table 1-3
Crystal River Unit 3 Quality Program Commitments
(continued)

Standard. However, the qualifications of personnel performing similar activities shall be consistent with the qualification requirements of the Standard except as noted in Items 1 and 2 above and Item 8 below.

5) With regard to Regulatory Position C.8, CR3 has an ALARA program which is applicable to all personnel performing activities at Crystal River Unit 3 as described in the Regulatory Position.

6) Regulatory Position C.9 is not applicable to Crystal River Unit 3.

7) With regard to Regulatory Position C.10 (and Section 2.2 of ANSI N45.2.6 - 1978 which it references), CR3 shall document exceptions to the qualifications requirements as set forth in the clarification of Section 3.5. Initial and subsequent evaluation of other personnel shall be performed and documented either in required CR3 personnel evaluations or other suitable records.

8) QA personnel performing inspections, examinations and testing as part of their routine assignment as auditors or lead auditors shall be qualified as set forth in commitments to ANSI N45.2.23 elsewhere in this Table.

9) With regard to Section 1.2 of ANSI N45.2.6 - 1978 titled Applicability: The third paragraph requires that the Standard be used in conjunction with ANSI N45.2; CR3 no longer specifically commits to ANSI N45.2 in the Quality Program Description. The fourth paragraph requires that the Standard be imposed on personnel other than CR3 employees; the applicability of the Standard to suppliers shall be documented and applied, as appropriate, in the procurement documents for such suppliers.

10) With regard to Section 1.4 of ANSI N45.2.6 - 1978 titled Definitions: Definitions in this Standard which are not included in ANSI N45.2.10 shall be used; all definitions which are included in ANSI N45.2.10 shall be used as clarified in CR3's commitment to Regulatory Guide 1.74.

11) With regard to Section 2.5 of ANSI N45.2.6 - 1978 titled Physical: CR3 shall implement the requirements of this Section with the stipulation that, where no special physical characteristics are required, none shall be specified. The converse is also true: If no special physical requirements are stipulated by CR3, none are considered necessary.

12) With regard to Section 3.5 of ANSI N45.2.6 - 1978 titled Education and Experience - Recommendations: CR3 reserves the right to use personnel who do not meet all the educational and experience requirements of this Section provided: The use of personnel who do not meet these requirements shall be the exception rather than the rule and each such case shall receive a documented management evaluation and justification for the exception. An example of a documented management evaluation and justification would be one which includes objective criteria (examination, review of actual work performed) to demonstrate that equivalent competence is possessed by such an individual.

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Crystal River Unit 3 Quality Program Commitments
(continued)

NRC REGULATORY GUIDE **1.64** - "QUALITY ASSURANCE REQUIREMENTS FOR THE DESIGN OF NUCLEAR POWER PLANTS" (REV. 2, 6/76) - ENDORSES ANSI N45.2.11 - 1974.

With the Certification of Permanent Cessation of Power Operations, DEF has permanently ceased power operations of Crystal River Unit 3 (CR3). CR3 follows this Regulatory Guide and associated Standard for the remaining important to safety activities associated with SAFSTOR, decommissioning, and the Independent Spent Fuel Storage Installation (ISFSI) with the following clarifications:

1) For operational phase maintenance and modification activities which are comparable in nature and extent to similar activities conducted during the construction phase, CR3 shall either control these activities under this Quality Program Description or under an NRC accepted Construction QA Program. When this Quality Program Description is used, CR3 shall comply with the requirements of the Regulatory Position documented in this Guide in that QA programmatic/administrative requirements included therein (subject to the clarification below) shall apply to these maintenance and modification activities even though such requirements may not have been in effect originally. Technical requirements associated with the maintenance and modifications shall be the original requirements or better (e.g., code requirements, material properties, design margins, manufacturing processes, and inspection requirements).

2) With regard to Paragraph C.2(1) of Regulatory Guide 1.64: If in an exceptional circumstance the designer's immediate Supervisor is the only technically qualified individual available, this review can be conducted by the Supervisor, providing that: (a) the other requirements of the Regulatory Position are satisfied, (b) the justification is individually documented and approved by the Supervisor's management, and (c) quality assurance audits cover frequency and effectiveness of use of Supervisors as design verifiers to guard against abuse.

3) With regard to Section 1.4 of ANSI N45.2.11 - 1974 titled Definitions: Definitions in this Standard which are not included in ANSI N45.2.10 shall be used; all definitions which are included in ANSI N45.2.10 shall be used as clarified in CR3's commitment to Regulatory Guide 1.74.

4) With regard to Sections 2.1 and 2.2 of ANSI N45.2.11 - 1974 titled, respectively, Establishment and Documentation and Program Procedures: Sections 1.7.1.16 and 1.7.1.18 of the Quality Program Description and commitments to ANSI N45.2.12 as set forth in Table 1-3 of that Program shall be met in lieu of the last paragraph of Section 2.1 and items 6, 12 and 13 in Section 2.2.

5) With regard to Section 5.2.4 of ANSI N45.2.11 - 1974 titled Documentation: For the documentation of interdisciplinary design reviews, there must be documented evidence of the acceptability of design documents or portions thereof, prior to release (material, stress, physics, mechanical, electrical, concrete, etc.). The signature or initials of those who determine the acceptability of the design relative to their respective disciplinary area of concern shall be on the document or on a separate form traceable to the document. A document that indicates the reviewer's comments need not be retained.

6) With regard to Section 6.1 of ANSI N45.2.11 - 1974 titled General: The third paragraph in this Section stipulates certain requirements relative to "the results of design verification." CR3 may comply with these requirements by having the reviewer(s) sign and date an appropriate document providing the following conditions are also met:

   a) Documented engineering/design procedures are established which cover the extent of design review.

   b) The procedures identify the duties of the reviewer and the extent of his responsibility for which he attests with his signature.

   c) The procedures specify the extent of documentation necessary for the type of design verification applicable to the complexity of the design.

Table 1-3
Crystal River Unit 3 Quality Program Commitments
(continued)

d) The signature and date is affixed in accordance with the procedures.

CR3 shall also permit initials to be used in lieu of the signature required above if a file is maintained to correlate characteristic initials versus individuals such that each set of characteristic initials can be traced to an individual. This correlation must be readily available to NRC inspectors whenever the document is being used.

7) The timing of design verification is not mentioned in the Standard. CR3 shall perform verification in a timely manner. If other than by qualification testing of a prototype or lead production unit, verification should be completed prior to release for procurement, manufacturing or construction or release to another organization for use in other design activities. In those cases where this timing cannot be met, the design verification may be deferred, provided that the justification for this action is documented and the unverified portion of the design output document and all design output documents, based on the unverified data, are appropriately identified and controlled. Site activities associated with a design or design change shall not proceed without verification past the point where the installation would become irreversible without extensive demolition and rework. In all cases, the design verification shall be completed prior to relying upon the component, system, or structure to perform its safety-related function.

8) With regard to Section 10 of ANSI N45.2.11 - 1974 titled Records: CR3 shall maintain records in accordance with and to meet the requirements of Section 1.7.1.17 of the Quality Program Description and ANSI N45.2.9 as specified in Table 1-3. The additional requirements of the first sentence of the second paragraph in this Section shall also be met.

9) With regard to Section 11 (including Subsections 11.1 through 11.7) of ANSI N45.2.11 - 1974, titled Audits: The CR3's Audit Program shall be implemented in accordance with and to meet the requirements of: ANSI N18.7 and ANSI N45.2.12 as endorsed in Table 1-3 and Sections 1.7.1.16 and 1.7.1.18.1 of the Quality Program Description.

10) CR3 does not require the DSAR to be a design document. However, if the DSAR contains any design or design input that has not been previously documented, reviewed and approved in accordance with the requirements contained in the Quality Program Description or another approved design control QA program and if the DSAR is used as a design document during the design process, that specific information must be controlled and verified in accordance with the commitments here and Section 1.7.1.3 of the Quality Program Description.

Table 1-3
Crystal River Unit 3 Quality Program Commitments
(continued)

NRC REGULATORY GUIDE 1.74 - "QUALITY ASSURANCE TERMS AND DEFINITION" (2/74) - ENDORSES ANSI N45.2.10 - 1973

With the Certification of Permanent Cessation of Power Operations, DEF has permanently ceased power operations of Crystal River Unit 3 (CR3). CR3 follows this Regulatory Guide and associated Standard for the remaining important to safety activities associated with SAFSTOR, decommissioning, and the Independent Spent Fuel Storage Installation (ISFSI) with the following clarifications:

1) CR3 reserves the right to define additional words or phrases which are not included in this Standard. Such additional definitions shall be documented in appropriate procedures and/or in attachments/appendices to quality assurance procedures/manuals, or in Sections of the Quality Program Description.

2) In addition to the Standard's definition of "Inspection," CR3 shall use the following: "Inspection (when used to refer to activities that are NOT performed by quality organization personnel) - Examining, viewing closely, scrutinizing, looking over or otherwise checking activities. Personnel performing these functions are not necessarily certified to ANSI N45.2.6."

   When CR3 intends for inspections to be performed in accordance with the Quality Program Description by personnel certified as required by that Program and for activities defined by "Inspection" in ANSI N45.2.10, appropriate references to the plant quality organization which shall perform the activity or to Quality Procedures to be used for performing the activity shall be made. If such references are NOT made, inspections are to be considered under the additional definition given above.

3) In addition to the Standard's definition of "procurement documents," CR3 shall utilize the definitions given in ANSI N45.2.13 and in Regulatory Guide 1.74. The composite definition is given as follows: Procurement documents - Contractually binding documents that identify and define the requirements which items or services must meet in order to be considered acceptable by the purchaser. They include documents which authorize the seller to perform services or supply equipment, material or facilities on behalf of the purchaser [e.g., Engineering Service Agreements (agreements for engineering, construction, or consulting services)], contracts, letters of intent, work authorization (in some cases), purchase requisitions, purchase orders, or proposals and their acceptance, drawings, specifications, or instructions which define requirements for purchase.

4) "Independent Verification" - Verification by an individual other than the person who performed the operation or activity being verified that required actions have been completed. Such verification shall not require confirmation of the identical action when other indications provide assurance or indication that the prescribed activity is in fact complete. Examples include, but are not limited to: verification of a breaker opening by observing the actuation of status or indicating lights at the required panel-meter indicated value; verification that a valve has been positioned by observing the starting or stopping of flow on meter indications or by remote valve positions indicating lights.

5) "Must" - (Not defined in any ANSI Standard) - An internally auditable requirement imposed by CR3 management upon its employees, contractors, and agents - above and in excess of the legally binding requirements of the appropriate regulatory body. Such items are internally required but not externally enforceable.

6) "NRC accepted Construction QA Program" - (1) a program for design or construction which was reviewed by the QA organization of the NRC and accepted for use; (2) the revision of that NRC accepted program which is in effect at the time that CR3 authorizes commencement of work; and (3) a program which the CR3 QA organization reviews and concurs that the QA Program controls are acceptable for the activity to be performed.

Table 1-3
Crystal River Unit 3 Quality Program Commitments
(continued)

7) "Program Deficiencies" (Not defined in ANSI N45.2.10, but used and defined differently in ANSI N45.2.12) - Failure to develop, document or implement effectively any applicable element of the Quality Program Description.

8) "Quality Assurance Program Requirements" (Not defined in ANSI N45.2.10 but used and defined differently in ANSI N45.2.13) - Those individual requirements of the Quality Program Description which, when invoked in total or in part, establish the requirements of the Quality Program for the activity being controlled. Although not specifically used in the Quality Program Description, ANSI N45.2 may be imposed upon CR3's suppliers.

9) "Quality assurance records" - (Not defined in ANSI N45.2.10 and defined without "expansion" in ANSI N45.2.9 - 1974) - The definition of "quality assurance records" which is given in Section 1.4 of ANSI N45.2.9 - 1974 shall be used with the clarification that "quality assurance records" are the Lifetime Quality Assurance Records defined in Section 2.2.1 and the Nonpermanent Quality Assurance Records defined in Section 2.2.2 of the Standard as well as those records specifically required to be retained by the applicable portions of the Code of Federal Regulations, Title 10.

10) "Will" - (Not defined in any ANSI standard) - Means the same as "shall" except when context shows that it is simply being used to indicate events which are to take place in the future. In the latter case, "will" is normally followed by "be."

11) "Facility staff" - (Not defined in any ANSI standard) - Means those personnel who report to the CR3 General Manager Decommissioning. This term shall also be synonymous with the "On site Operating Organization" described (but not defined) in ANSI N18.7-1976, Section 3.4.2 and the "Unit Staff" described in the PDTS Section 5.2.2.

Table 1-3
Crystal River Unit 3 Quality Program Commitments
(continued)

NRC REGULATORY GUIDE **1.88** - "COLLECTION, STORAGE AND MAINTENANCE OF NUCLEAR POWER PLANT QUALITY ASSURANCE RECORDS" (REV. 2, 10/76) - ENDORSES ANSI N45.2.9 - 1974

With the Certification of Permanent Cessation of Power Operations, DEF has permanently ceased power operations of Crystal River Unit 3 (CR3). CR3 follows this Regulatory Guide and associated Standard for the remaining important to safety activities associated with SAFSTOR, decommissioning, and the Independent Spent Fuel Storage Installation (ISFSI) with the following clarifications:

1) With regard to Section 1.4 of ANSI N45.2.9 - 1974 titled Definitions: The only definition given in this Standard is not included in ANSI N45.2.10. The Standard's definition of "quality assurance records" shall be used with the clarification that "quality assurance records" are the Lifetime Quality Assurance Records defined in Section 2.2.1 and the Nonpermanent Quality Assurance Records defined in Section 2.2.2 as well as those records specifically required to be retained by the applicable portions of the Code of Federal Regulations, Title 10. This clarification is also given in this Table under Regulatory Guide 1.74.

2) With regard to Section 3.2.1 of ANSI N45.2.9 - 1974 titled Generation of Quality Assurance Records: The phrase "completely filled out" is clarified to mean that sufficient information is recorded to fulfill the intended purpose of the record.

3) With regard to Section 3.2.2 of ANSI N45.2.9 - 1974 titled Index: The phrase "an index" is clarified to mean a collection of documents or indices which, when taken together, supply the information attributed to "an index" in the Standard.

   The specific location of a record "within a storage area" may not always be delineated (e.g., The specific location within a computer record file may not be constant. Further, CR3 may utilize a computer assisted random access filing system where such a "location" could not be readily documented, nor would such a location be meaningful). The storage location shall be delineated, but where file locations change with time, the specific location of a record within that file may not always be documented.

4) With regard to Section 3.2.7 of ANSI N45.2.9 - 1974 titled Retention of Records: CR3 generally maintains that Lifetime QA Records as defined above in Clarification 1, are those records required to be maintained by or for Duke Energy Florida for the life of the particular item while it is installed in the plant or stored for future use. On November 30, 2016, the NRC granted an exemption request related to Record Keeping Requirements. This exemption permits the elimination of requirements to maintain records that are no longer necessary due to the permanently shut down and decommissioning status of CR3. Likewise, Non-Permanent QA Records need not be retained for the life of the item. With regard to operating phase records only, in lieu of Appendix A to ANSI N45.2.9, CR3 has established a listing of those types of operating phase records that have been assigned specific retention periods.

   The following records shall be retained for at least five years:

   a. Records and logs of facility operation covering time intervals at each power level.
   b. Records and logs of principal maintenance activities, inspections, repair and replacement of principal items of equipment related to nuclear safety.
   c. All Reportable Events submitted to the Commission.
   d. Records of surveillance activities, inspections and calibrations required by PDTS.
   e. Records of reactor tests and experiments.
   f. Records of changes made to Operating Procedures.
   g. Records of radioactive shipments.
   h. Records of sealed source and fission detector leak tests and results.

Table 1-3
Crystal River Unit 3 Quality Program Commitments
(continued)

i. Records of annual physical inventory of all sealed source material of record.

The following records shall be retained for the duration of the Facility Operating License:

a. Records and drawing changes reflecting facility design modifications made to systems and equipment described in the DSAR.
b. Records of new and irradiated fuel inventory, fuel transfers and assembly burnup histories.
c. Records of facility radiation and contamination surveys.
d. Records of radiation exposure for all individuals entering radiation control areas.
e. Records of gaseous and liquid radioactive material released to the environs.
f. Records of transient or operational cycles for facility components.
g. Records of training and qualification for current members of the plant staff.
h. Records of inservice inspection performed pursuant to PDTS.
i. Records of Quality Assurance activities required by the Quality Program Description.
j. Records of reviews performed for changes made to procedures or equipment or reviews of tests and experiments pursuant to 10 CFR 50.59.
k. Records of meetings of the PNSC.
l. Records for Environmental Qualification.
m. Records of analytical results required by the Operational Radiological Environmental Monitoring Program.
n. Records of reviews performed for changes made to the Offsite Dose Calculation Manual.
o. Records of reviews and changes to the Process Control Program. This documentation shall contain 1) sufficient information to support the change together with the appropriate analyses or evaluations justifying the change(s), and 2) a determination that the change will maintain the overall conformance of the solidified waste product to existing requirements of Federal, State, or other applicable regulations.

5) With regard to Section 4.2 of ANSI N45.2.9 - 1974 titled Timeliness: CR3's contractual agreement with its vendors, A/E, constructors, and suppliers shall constitute fulfillment of the requirements of this Section.

6) With regard to Section 5.3 of ANSI N45.2.9 - 1974 titled Storage: The first sentence is clarified by stating that an individual or group of individuals or a classification of employee shall be designated and assigned the authority to enforce written storage procedures. The term "custodian" may or may not be used as part of that designation.

7) With regard to Section 5.4 of ANSI N45.2.9 - 1974 titled Preservation: The following clarification is substituted for the current Subsection 5.4.2 "Records shall not be stored loosely. They shall be secured for storage in file cabinets or on shelving in containers." Although not a verbatim quote, this is the position taken in the latest consensus Standard [ANSI/ASME NQA-1b-1981, Supplement 17S-1, 4.2(b)].

8) With regard to Subsection 5.4.3 of ANSI N45.2.9 - 1974 titled Special Processed Records: The following clarification is substituted for the current Subsection 5.4.3: "Provisions shall be made for special processed records (such as radiographs, photographs, negatives, microfilm and magnetic media) to prevent damage from excessive light, stacking, electromagnetic fields, temperature and humidity as appropriate to the record type." This is the position taken in the latest consensus Standard [ANSI/ASME NQA-1b-1981, Supplement 17S-1, Section 4.2(c)].

Table 1-3
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(continued)

9) With regard to Section 5.5 of ANSI N45.2.9 - 1974 titled Safekeeping: Routine general office and nuclear site security systems and access controls are provided. No special security systems are required to be established for record storage areas.

10) With regard to Section 5.6 of ANSI N45.2.9 - 1974 titled Facility: This Section provides no distinction between temporary and permanent facilities. To cover temporary storage, the following clarification is added: Active records (those completed but not yet duplicated or placed on microfilm or optical disk) may be temporarily stored in one-hour fire rated file cabinets. In general, records shall not be maintained in such temporary storage for more than ninety days after completion without being duplicated (for dual storage) or being placed on microfilm or optical disk. Any exceptions to this ninety day storage shall be evaluated and approved by the Manager, Decommissioning Technical Support, a list of all such excepted records shall be maintained and available for NRC review. Open-ended documents (those revised or updated on a more-or-less continuing basis over an extended period of time (e.g., personnel qualification and training documents, equipment history cards, audit, or surveillance schedules) and those which are cumulative in nature (e.g., nonconforming item logs, control room log books, night order books) are not considered as QA records since they are not "complete." These documents are controlled by the department procedures. These types of documents shall become QA records: when they are issued as a specific revision (e.g., the audit schedule); when they are filled-up or discontinued (e.g., log books or equipment history cards); on a predefined periodic basis when the completed portion of the on-going document shall be transferred to the records storage facility as a "record" (e.g., training and qualification records). The applicable provision of Section 5.3 shall be met by QA records in temporary storage.

    Where duplicate storage is employed as permitted by the second paragraph of Section 5.6, no special construction requirements are applicable. However, the record storage locations shall be sufficiently separated from one another that they are not generally susceptible to simultaneous destruction by the same natural disaster (e.g., fire, flooding). Although not a verbatim quote, this is the position taken in the latest consensus Standard (ANSI/ASME NQA-1b-1981, Supplement 17S-1, Section 4.4.2).

    The fourth paragraph of Section 5.6, item 3 is modified to require a two-hour minimum fire rating to be consistent with the 1979 version of the Standard, NQA-1b-1981, and NRC Criteria for Record Storage Facilities (Guidance-ANSI N45.2.9, Section 5.6) issued 7/1/80.

    The fourth paragraph of Section 5.6, item 9 is clarified to read: "No pipes or penetrations except those used exclusively for fire protection, lighting, temperature/humidity control, or communications are to be located within the facility. All such penetrations shall be sealed or dampened to comply with a minimum two-hour fire protection rating." This is the position taken in the latest consensus Standard [ANSI/ASME NQA-1b-1981, Supplement 17S-1, Section 4.4.1(j)].

    With regard to Section 6.2 of ANSI N45.2.9 - 1974 titled Accessibility: The second paragraph of this Section is clarified to mean that persons authorized access shall be designated. That designation will be: 1) all designated Document Management personnel; and 2) the specific names of personnel outside that department who have been authorized to have access.

11) The CR3 program for storage of records on microfilm, dual storage or in electronic format meets the preservation requirement for the retention of QA Records.

    For management of electronic records, the appropriate controls on quality are summarized as follows:

    a) The Electronic Records Management (eRM) system does not allow deletion or modification of records. (NOTE: Authorized deletion of records per the Record Retention Rules is controlled.)
    b) The eRM system provides redundancy (i.e., system backup, dual storage, etc.).
    c) The legibility of each record is verified prior to acceptance into the eRM system.

Table 1-3
Crystal River Unit 3 Quality Program Commitments
(continued)

d) The media used by the eRM system is maintained to ensure the records are acceptably copied onto a new media before the manufacturer's certified useful life of the media is exceeded. This includes verification of the records so copied.

e) Periodic random inspections of records are performed to verify that there has been no degradation of record quality.

f) If the eRM system in use is to be replaced by a new system, the records stored on the old system are acceptably converted into the new system before the old system is taken out of service. This includes verification of the records so copied.

To implement those controls, CR3 uses the following:

- NIRMA TG 11-2011, "Authentication of Records and Media"

- NIRMA TG 15-2011, "Management of Electronic Records"

- NIRMA TG 16-2011, "Software Quality Assurance Documentation and Records"

- NIRMA TG 21-2011, "Required Records Protection, Disaster Recovery and Business Continuation"

12) On November 30, 2016, the NRC granted an exemption request related to Record Keeping requirements. This exemption permits the elimination of requirements to maintain records that are no longer necessary due to the permanently shut down and decommissioning status of CR3.

Table 1-3
Crystal River Unit 3 Quality Program Commitments
(continued)

NRC REGULATORY GUIDE 1.94 - "QUALITY ASSURANCE REQUIREMENTS FOR INSTALLATION, INSPECTION, AND TESTING OF STRUCTURAL CONCRETE AND STRUCTURAL STEEL DURING THE CONSTRUCTION PHASE OF NUCLEAR POWER PLANTS" (REV. 1, 4/76) - ENDORSES ANSI N45.2.5 - 1974.

With the Certification of Permanent Cessation of Power Operations, DEF has permanently ceased power operations of Crystal River Unit 3 (CR3). CR3 follows this Regulatory Guide and associated Standard for the remaining important to safety activities associated with SAFSTOR, decommissioning, and the Independent Spent Fuel Storage Installation (ISFSI) with the following clarifications:

1) For operational phase maintenance and modification activities which are comparable in nature and extent to similar activities conducted during the construction phase, CR3 shall either control these activities under this Quality Program Description or under an NRC accepted Construction QA Program. When this Quality Program Description is used, CR3 shall comply with the requirements of the Regulatory Position documented in this Guide in that QA programmatic/administrative requirements included therein (subject to the clarifications below) shall apply to these maintenance and modification activities even though such requirements may not have been in effect originally. Technical requirements associated with the maintenance and modifications shall be the original requirements or better (e.g., code requirements, material properties, design margins, manufacturing processes, and inspection requirements).

2) With regard to Section 1.1 of ANSI N45.2.5 - 1974 titled Scope: The last paragraph of this Section is not applicable to the operational phase.

3) With regard to Section 1.2 of ANSI N45.2.5 - 1974 titled Applicability: The first sentence in this Section is not applicable to the operational phase. CR3 shall comply with the second sentence in this Section with the clarifications that "importance of the item or service involved" is interpreted to mean those to which the Quality Program Description applies, and the extent of coverage shall be defined by supervisory maintenance personnel by the way in which they implement the other requirements of this Standard.

   In the second paragraph of this Section, CR3 shall substitute the words "maintenance and modification" for the word "construction" as the modifier of "procedures".

4) With regard to Section 1.3 of ANSI N45.2.5 - 1974 titled Responsibility: This Section's requirements are met by the definitions for positions and the organizational responsibilities outlined in Section 1.7.1.1 of the Quality Program Description, and the position descriptions for plant personnel.

5) With regard to Section 1.4 of ANSI N45.2.5 - 1974 titled Definitions: Definitions in this Standard which are not included in ANSI N45.2.10 shall be used. All definitions which are included in ANSI N45.2.10 shall be used as clarified in CR3's commitment to Regulatory Guide 1.74.

6) With regard to Section 2.1 of ANSI N45.2.5 - 1974 titled Planning: Planning requirements, when necessary, shall be incorporated into maintenance and modification procedures.

7) With regard to Section 2.2 of ANSI N45.2.5 - 1974 titled Procedures and Instructions: Procedures and instructions shall be implemented as set forth in Sections 1.7.1.5, 10, and 11 of the Quality Program Description and by compliance with ANSI N18.7 as set forth in Table 1-3 of that Program in lieu of the requirements set forth here.

8) With regard to Section 2.3 of ANSI N45.2.5 - 1974 titled Results: These requirements are met by implementing Sections 1.7.1.10, 11, and 17 of the Quality Program Description and by compliance with ANSI N18.7 as set forth in Table 1-3 of that Program in lieu of the requirements set forth here.

9) With regard to Section 2.5.2 of ANSI N45.2.5 - 1974 titled Calibration and Control: The first paragraph of this Section shall be met as set forth in Item 2 of CR3's commitment to Regulatory Guide 1.30 elsewhere in this Table.

Table 1-3
Crystal River Unit 3 Quality Program Commitments
(continued)

10) With regard to Section 4.8 of ANSI N45.2.5 - 1974 titled In-Process Tests on Concrete and Reinforcing Steel: The seventh sentence specifies the location for taking pumped concrete samples. There may be instances when the pump line discharge is inaccessible for sampling and there may be other instances when, although samples could be taken, the technician would be in such a position that he could not satisfactorily conduct the tests. In these instances, CR3 shall sample the concrete at the truck just before it enters the pump inlet hopper. When this technique is used, a correlation shall be established between truck discharge test results and pump line discharge test results.

11) With regard to Section 5.4 of ANSI N45.2.5 - 1974 titled High Strength Bolting: Item 1 of the second paragraph of this Section requires at least two threads to extend beyond the nut. CR3 shall normally meet this criterion but may accept installations which have the point of the bolt flush with or outside of the face of the nut when completely installed if a direct-tension indicator is used to tighten the bolt.

12) With regard to Section 7 of ANSI N45.2.5 - 1974 titled Records: CR3 shall maintain records in accordance with and to meet the requirements of Section 1.7.1.17 of the Quality Program Description and ANSI N45.2.9 as specified in Table 1-3.

13) With regard to Section 6.1 of ANSI N45.2.5 - 1974 titled General: Inspection and test requirements shall be implemented as defined in Sections 1.7.1.3, 10 and 11 of the Quality Program Description and CR3's commitment to Section 5.2.7 of ANSI N18.7 as described in Table 1-3 of that Program in lieu of the requirements set forth here.

14) With regard to Section 4.9.4 of ANSI N45.2.5-1974 titled Tensile Test Frequency: For testing of mechanical splices during concrete repairs when there is insufficient length of bar stub exposed at the periphery of the existing concrete to cut out a production splice for testing and have enough remaining stub to make a permanent production splice, the sample frequency below will be applied for testing sister splices only.

Frequency	Straight Bars	Curved (Horizontal) Bars
First 10 production splices	Test 3 sister splices	Test 1 sister splice
Next 90 production splices	Test 6 sister splices	Test 4 sister splices
Subsequent lots of 100 production	Test 4 sister splices	Test 3 sister splices

Table 1-3
Crystal River Unit 3 Quality Program Commitments
(continued)

NRC REGULATORY GUIDE 1.116 - "QUALITY ASSURANCE REQUIREMENTS FOR INSTALLATION, INSPECTION AND TESTING OF MECHANICAL EQUIPMENT AND SYSTEMS" (REV. O-R, 6/76) - ENDORSES ANSI N45.2.8 - 1975

With the Certification of Permanent Cessation of Power Operations, DEF has permanently ceased power operations of Crystal River Unit 3 (CR3). CR3 follows this Regulatory Guide and associated Standard for the remaining important to safety activities associated with SAFSTOR, decommissioning, and the Independent Spent Fuel Storage Installation (ISFSI) with the following clarifications:

1) For operational phase maintenance and modification activities which are comparable in nature and extent to similar activities conducted during the construction phase, CR3 shall either control these activities under this Quality Program Description or under an NRC accepted Construction QA Program. When this Quality Program Description is used, CR3 shall comply with the requirements of the Regulatory Position documented in this Guide in that QA programmatic/administrative requirements included therein (subject to the clarifications below) shall apply to these maintenance and modification activities even though such requirements may not have been in effect originally. Technical requirements associated with the maintenance and modifications shall be the original requirements or better (e.g., code requirements, material properties, design margins, manufacturing processes, and inspection requirements).

2) With regard to Section 1.1 of ANSI N45.2.8 - 1975 titled Scope: The last paragraph of this Section is not applicable to the operational phase. The applicable portions of the requirements of this Standard shall also be applied after fuel load; therefore, the last twenty-two words in the last sentence of the second paragraph under this Section are also not appropriate to the operational phase.

3) With regard to Section 1.2 of ANSI N45.2.8 - 1975 titled Applicability: The first sentence in this Section is not applicable to the operational phase. CR3 shall comply with the third sentence in this Section with the clarifications that "important mechanical items to be covered" is interpreted to mean those to which the Quality Program Description applies, and "the extent of coverage" shall be defined by supervisory maintenance personnel by the way in which they implement the other requirements of this Standard.

4) With regard to Section 1.3 of ANSI N45.2.8 - 1975 titled Responsibility: This Section's requirements are met by the definitions for positions and the organizational responsibilities outlined in Section 1.7.1.1 of the Quality Program Description, and the position descriptions for plant personnel.

5) With regard to Section 1.4 of ANSI N45.2.8 - 1975 titled Definitions: Definitions in this Standard which are not included in ANSI N45.2.10 shall be used. All definitions which are included in ANSI N45.2.10 shall be used as clarified in CR3's commitment to Regulatory Guide 1.74.

6) With regard to Section 2.1 of ANSI N45.2.8 - 1975 titled Planning: Planning requirements, when necessary, shall be incorporated into maintenance and modification procedures.

7) With regard to Section 2.2 of ANSI N45.2.8 - 1975 titled Procedures and Instructions: Procedures and instructions shall be implemented as set forth in Section 1.7.1.5, 10 and 11 of the Quality Program Description and by compliance with ANSI N18.7 as set forth in Table 1-3 of that Program in lieu of the requirements set forth here.

8) With regard to Section 2.3 of ANSI N45.2.8 - 1975 titled Results: These requirements are met by implementing Sections 1.7.1.10, 11 and 17 of the Quality Program Description and by compliance with ANSI N18.7 as set forth in Table 1-3 of that Program in lieu of the requirements set forth here.

9) With regard to Section 2.8.2 of ANSI N45.2.8 - 1975 titled Calibration and Control: The first paragraph of this Section shall be met as set forth in Item 2 of CR3's commitment to Regulatory Guide 1.30 elsewhere in this Table.

Table 1-3
Crystal River Unit 3 Quality Program Commitments
(continued)

10) With regard to Section 2.9 of ANSI N45.2.8 - 1975 titled Prerequisites: Most of the items in this Section were written for construction activities. Maintenance and modification activities which are similar in nature and extent to construction activities shall be handled as set forth in Item 1 above. The portions of this Section which are considered applicable by responsible maintenance supervisory personnel shall be complied with as stated.

11) With regard to Section 3.1 of ANSI N45.2.8 - 1975 titled Pre-Installation Verification: (Including Subsections 3.1, 3.2, 3.3, 3.4 and 3.5) Most of the items in this Section (and its Subsections) were written for construction activities. Maintenance and modification activities which are similar in nature and extent to construction activities shall be handled as set forth in Item 1 above. The portions of this Section (and its Subsections) which are considered applicable by responsible maintenance supervisory personnel shall be complied with as stated. Nuclear Oversight (NOS) personnel verify that appropriate controls are used through procedure reviews and audit activities.

12) With regard to Sections 4.1 and 4.2 of ANSI N45.2.8 - 1975 titled, respectively, General and Process and Procedure Control: CR3 meets these requirements by commitments to ANSI N18.7 as set forth elsewhere in this Table.

13) With regard to Section 4.4 of ANSI N45.2.8 - 1975 titled Inspections: The requirements of Section 4.4 shall be implemented as set forth in Section 1.7.1.10 of the Quality Program Description. The inspection program shall incorporate, as applicable, those items listed in Section 4.4.

14) With regard to Section 4.5 of ANSI N45.2.8 - 1975 titled Installation Checks: (Including Subsections 4.5.1 and 4.5.2.) The portions of this Section (and its Subsections) which are considered applicable by responsible maintenance supervisory personnel shall be complied with as stated. Such items shall be specified in appropriate maintenance or modification procedures.

15) With regard to Section 4.6 of ANSI N45.2.8 - 1975 titled Care of Items: CR3 does not consider most of this Section to be applicable to the operational phase due to the relatively short period of time between installation and use. Consistent with commitments to ANSI N45.2.3 and ANSI N18.7 as given elsewhere in this Table, CR3 shall implement appropriate housekeeping and preventive maintenance procedures to assure that all material which has been installed is appropriately protected and maintained.

16) With regard to Section 5.1 of ANSI N45.2.8 - 1975 titled General: The requirements of this Section shall be implemented as set forth in Sections 1.7.1.3 and 11 of the Quality Program Description. The test program shall consider the elements outlined in this Section, where applicable, when developing test requirements for inclusion in maintenance and modification procedures. In some cases, testing requirements may be met by post-installation surveillance testing in lieu of a special post-installation test.

17) With regard to Sections 5.2, 5.3 and 5.4 of ANSI N45.2.8 - 1975 titled, respectively, Pre-operational Testing, Cold Functional Tests, and Hot Functional Tests: Except for maintenance and modification activities which are similar in nature and extent to similar construction activities (which are handled as set forth in Item 1 above), the extensive testing program described in these Sections would not be applicable to operational phase Activities. Where certain items are applicable, as determined by supervisory maintenance personnel, they shall be included in appropriate procedures or handled as described under Item 7 above. Nuclear Oversight (NOS) personnel verify that appropriate controls are used through procedure reviews, and audit activities.

Table 1-3
<u>Crystal River Unit 3 Quality Program Commitments</u>
(continued)

18) With regard to Section 6 of ANSI N45.2.8 - 1975 titled <u>Data Analysis and Evaluation</u>: CR3 shall process and analyze inspection and test data as set forth in Sections 1.7.1.10, 11, and 17 of the Quality Program Description and our commitment to Sections 5.2.7 and 5.2.17 of ANSI N18.7 as described in Table 1-3 of the Program.

19) With regard to Section 7 of ANSI N45.2.8 - 1975 titled <u>Records</u>: CR3 shall maintain records in accordance with and to meet the requirements of Section 1.7.1.17 of the Quality Program Description and ANSI N45.2.9 as specified in Table 1-3.

Table 1-3
Crystal River Unit 3 Quality Program Commitments
(continued)

NRC REGULATORY GUIDE 1.123 - "QUALITY ASSURANCE REQUIREMENTS FOR CONTROL OF PROCUREMENT OF ITEMS AND SERVICES FOR NUCLEAR POWER PLANTS" (REV. 1, 7/77) - ENDORSES ANSI N45.2.13 - 1976.

With the Certification of Permanent Cessation of Power Operations, DEF has permanently ceased power operations of Crystal River Unit 3 (CR3). CR3 follows this Regulatory Guide and associated Standard for the remaining important to safety activities associated with SAFSTOR, decommissioning, and the Independent Spent Fuel Storage Installation (ISFSI) with the following clarifications:

1) Paragraph C.6.c of Regulatory Guide 1.123 (and Sections 10.2.a through f of ANSI N45.2.13 which it references) shall be implemented as originally written (i.e., with the verb "should" instead of the verb "shall").

2) Paragraph C.6.e of Regulatory Guide 1.123 (and Section 10.3.4 of ANSI N45.2.13 which it references) shall be implemented as originally written (i.e., with the verb "should" instead of the verb "shall"). This flexibility is necessary because CR3 may not always be able to obtain agreement with a supplier. Since CR3 retains the ultimate responsibility for performance of purchased equipment and since CR3 maintains its own Nuclear Engineering Department, CR3 should be allowed to exercise this management/engineering prerogative with respect to the final decision on post installation test requirements.

3) With regard to Section 1.3 of ANSI N45.2.13 - 1976 titled Definitions: With two exceptions (Procurement Document and Quality Assurance Program Requirements) definitions in this Standard which are not included in ANSI N45.2.10 shall be used; all definitions which are included in ANSI N45.2.10 shall be used as clarified in CR3's commitment to Regulatory Guide 1.74. The two exceptions are defined in Table 1-3 under Regulatory Guide 1.74.

4) With regard to Section 1.2.2 of ANSI N45.2.13 - 1976 titled Purchaser's Responsibilities: Item c is one of the options which may be used by CR3 to assure quality; however, any of the options given in 10 CFR 50, Appendix B, Criterion VII as implemented by Sections 1.7.1.4 and 7 of the Quality Program Description may also be used.

5) With regard to Section 3.1 of ANSI N45.2.13 - 1976 titled Procurement Document Preparation, Review and Change Control: The phrase "the same degree of control" is stipulated to mean "equivalent level of review and approval." The changed document may not always be reviewed by the originator; however, at least an equivalent level of supervision shall review and approve any changes.

6) With regard to Section 3.2.3 of ANSI N45.2.13 – For the procurement of commercial grade calibration and/or testing services, Duke Energy Corporation uses NEI 14-05A, Revision 0, "Guidelines for the Use of Accreditation In Lieu of Commercial Grade Surveys for Procurement of Laboratory Calibration and Test Services." The conditions for the use of this process are consistent with NRC Safety Evaluation dated April 1, 2016 to Union Electric Company, Callaway Plant (ADAMS Accession # ML16089A167).

7) With regard to Section 3.4 of ANSI N45.2.13 - 1976 titled Procurement Document Control: CR3 shall meet the requirements of Sections 1.7.1.4 and 7 of the Quality Program Description in lieu of the requirements specified in this Section.

8) With regard to Section 4.2 of ANSI N45.2.13 – When purchasing commercial-grade calibration services, verification that the supplier's accreditation meets NEI 14-05A, Revision 0, shall be performed and documented.

9) With regard to Section 5.3 of ANSI N45.2.13 - 1976 titled Pre-award Evaluation: CR3 shall comply with an alternate paragraph which reads: "Except in unusual circumstances (e.g., replacement parts are needed to preclude the development of some unsafe or undesirable condition at Crystal River Unit 3), a pre-award evaluation of the Supplier shall be performed as required by the Quality Program Description."

Table 1-3
Crystal River Unit 3 Quality Program Commitments
(continued)

10) With regard to Section 6.4 of ANSI N45.2.13 - 1976 titled Control of Changes in Items or Services: The phrase "the Quality Program" shall be inserted in lieu of "ANSI N45.2, Section 7."

11) With regard to Section 8.2 of ANSI N45.2.13 - 1976 titled Disposition: The third sentence of item b is revised to read: Nonconformances to the contractual procurement requirements or purchaser approved documents and which consist of one or more of the following shall be submitted to the purchaser for approval of the recommended disposition prior to shipment when the nonconformance could adversely affect the end use of a module or shippable component* relative to safety, interchangeability, operability, reliability, integrity, or maintainability:

    a) Technical or material requirement is violated;
    b) Requirement in supplier documents, which have been approved by the purchaser, is violated;
    c) Nonconformance cannot be corrected by continuation of the original manufacturing process or by rework; and/or
    d) The item does not conform to the original requirement even though the item can be restored to a condition such that the capability of the item to function is unimpaired.

    *A module is an "assembled device, instrument, or piece of equipment identified by serial number or other identification code, having been evaluated by inspection and/or test for conformance to procurement requirements regarding end use." A shippable component is a "part of a device, instrument, or piece of equipment which is shipped as an individual item and which has been evaluated by inspection and/or test for conformance to procurement requirements regarding end use."

12) With regard to Section 11 of ANSI N45.2.13 - 1976 titled Quality Assurance Records: CR3 shall maintain records in accordance with and to meet the requirements of Section 1.7.1.17 of the Quality Program Description and ANSI N45.2.9 as specified in Table 1-3.

13) With regard to Section 12 of ANSI N45.2.13 - 1976 titled Audit of Procurement Program: The CR3 audit program shall be implemented in accordance with and to meet the requirements of ANSI N45.2.12 as clarified in Table 1-3 and Sections 1.7.1.16 and 1.7.1.18.1 of the Quality Program Description.

Table 1-3
Crystal River Unit 3 Quality Program Commitments
(continued)

NRC REGULATORY GUIDE 1.144 - "AUDITING OF QUALITY ASSURANCE PROGRAMS FOR NUCLEAR POWER PLANTS" (JANUARY, 1979) - ENDORSES ANSI N45.2.12 - 1977.

With the Certification of Permanent Cessation of Power Operations, DEF has permanently ceased power operations of Crystal River Unit 3 (CR3). CR3 follows this Regulatory Guide and associated Standard for the remaining important to safety activities associated with SAFSTOR, decommissioning, and the Independent Spent Fuel Storage Installation (ISFSI) with the following clarifications:

1) CR3 will follow the requirements and recommendations of paragraphs C.1, C.2, C.3.a.2, C.3.b and C.4. The CR3 position on paragraph C.3.a.1 is as follows:

   Audits of operational phase activities shall be performed at the frequencies stated in the CR3 Quality Program Description.

   The CR3 position on paragraph C.3.b.2 is clarified as follows: When purchasing commercial-grade calibration services exceptions 6) and 8) for Regulatory Guide 1.123 apply.

2) Paragraph C.4.a of Regulatory Guide 1.144 (and Section 3.5.3.3 of ANSI N45.2.12 which it references) shall be implemented with the clarification that the Superintendent, Nuclear Oversight or his designee shall determine which reorganizations or procedure revisions are "significant."

3) Paragraph C.7 of Regulatory Guide 1.144 (and Section 5.2 of ANSI N45.2.12 which it references) shall be implemented by maintaining records in accordance with and to meet the requirements of Section 1.7.1.17 of the Quality Program Description and ANSI N45.2.9 as specified in Table 1-3. With respect to the additional audit records recommended in C.7, CR3 may retain such records, but their retention shall not be considered mandatory.

4) With regard to Section 1.4 of ANSI N45.2.12 - 1977 titled Definitions: With one exception (Program Deficiencies) the definitions in this Standard which are not included in ANSI N45.2.10 shall be used as clarified in CR3's commitment to Regulatory Guide 1.74. A clarified definition of the exception is defined in Table 1-3 under Regulatory Guide 1.74.

5) With regard to Section 2.2 of ANSI N45.2.12 - 1977 titled Personnel Qualifications: CR3 audit personnel shall be qualified to meet the requirements of ANSI N45.2.23 as endorsed in Table 1-3 and Sections 1.7.1.2 and 1.7.1.18.1 of the Quality Program Description.

6) With regard to Section 2.3 and Subsections 2.3.1 through 2.3.3 of ANSI N45.2.12 -1977 titled Training: CR3 audit personnel shall be trained as necessary to meet the requirements of ANSI N45.2.23 as endorsed in Table 1-3 and Sections 1.7.1.2 and 1.7.1.18.1 of the Quality Program Description.

7) With regard to Section 2.4 of ANSI N45.2.12 - 1977 titled Maintenance of Proficiency: CR3 audit personnel shall maintain their proficiency by meeting the requirements of ANSI N45.2.23 as endorsed in Table 1-3 and Sections 1.7.1.2 and 1.7.1.18.1 of the Quality Program Description.

8) With regard to Section 3.3 of ANSI N45.2.12 - 1977 titled Essential Elements of the Audit System: CR3 shall comply with Subsection 3.3.5 as it was originally written (Subsection 3.2.5 in ANSI N45.2.12, Draft 3, Revision 4): "Provisions for reporting on the effectiveness of the quality assurance program to the responsible management." For auditing and audited organizations within CR3, "effectiveness of the quality assurance program" is reported to responsible management by meeting the requirements of Sections 1.7.1.1, 2, 16 and 1.7.1.18.1 of the Quality Program Description and ANSI N18.7 Section 4.5 as clarified in Table 1-3. For audited organizations outside (e.g., vendors, suppliers) of CR3, transmittal of the audit report shall be deemed to meet the requirements of Subsection 3.3.5 "for reporting on the effectiveness of the quality assurance program to the responsible management."

Table 1-3
Crystal River Unit 3 Quality Program Commitments
(continued)

Subsection 3.3.6 requirements are considered to be fulfilled by compliance with the organization and reporting measures outlined in the Quality Program Description.

Subsection 3.3.7 requires verification of effective corrective action on a "timely basis." Timely basis is interpreted to mean within the framework or period of time for completion of corrective action that is accepted by the quality organization. Each finding requires response and a corrective action completion date; these dates are subject to revision (with the approval of the quality organization) and must be escalated to higher authority when there is a disagreement between the audited and the auditing organization on what constitutes "timely corrective action."

9) With regard to Section 3.5 of ANSI N45.2.12 - 1977 titled Scheduling: Subsection 3.5.3.1 is interpreted to mean that CR3 may procedurally control qualification of a contractor's or supplier's quality assurance program prior to awarding a contract or purchase order by means other than audit.

10) With regard to Section 4.3.1 of ANSI N45.2.12 - 1977 titled Pre-Audit Conference: CR3 shall comply with requirements of this Section by inserting the word "Normally" at the beginning of the first sentence. This clarification is required because, in the case of certain unannounced audits or audits of a particular operation or work activity, a pre-audit conference might interfere with the activity or with the spontaneity of the operation or activity being audited. In other cases, persons who should be present at a pre-audit conference may not always be available. Such lack of availability should not be an impediment to beginning an audit. Even in the above examples, which are not intended to be all inclusive, the material set forth in Section 4.3.1 is normally covered during the course of the audit.

11) With regard to Section 4.3.2 of ANSI N45.2.12 - 1977 titled Audit Process:

   a) Subsection 4.3.2.2 could be interpreted to limit audits to the review of only objective evidence; sometimes and for some program elements, no objective evidence may be available. CR3 shall comply with an alternate sentence which reads: "When available, objective evidence shall be examined for compliance with Quality Program requirements. If subjective evidence is used (e.g., personal interviews, direct observations by the auditor), then the audit report must indicate how the evidence was obtained."

   b) Subsection 4.3.2.4 is modified as follows to take into account the fact that some nonconformances are virtually "obvious" with respect to the needed corrective action: "When a nonconformance or Quality Program deficiency is identified as a result of an audit, unless the apparent cause, extent, and corrective action are readily evident, further investigation shall be conducted by the audited organization in an effort to identify the cause and to determine the extent of the corrective action required."

   c) Subsection 4.3.2.5 contains a recommendation which is clarified with the definition of "acknowledged by a member of the audited organization" to mean that "a member of the audited organization has been informed of the findings." Agreement or disagreement with a finding may be expressed in the response from the audited organization.

12) With regard to Section 4.3.3. of ANSI N45.2.12 - 1977 titled Post-Audit Conference: CR3 shall substitute and comply with the following paragraph: "For all external audits, a post-audit conference shall be held with management of the audited organization to present audit findings and clarify misunderstandings; where no adverse findings exist, this conference may be waived by management of the audited organization. Such waiver shall be documented in the audit report. Unless unusual operating or maintenance conditions preclude attendance by appropriate managers/supervisors, a post-audit conference shall be held with managers/supervisors for all internal audits for the same reasons as above. Again, if there are no adverse findings, the management of the internal audited organization may waive the post-audit conference. Such waiver shall be documented in the audit report."

Table 1-3
Crystal River Unit 3 Quality Program Commitments
(continued)

13) With regard to Section 4.4 of ANSI N45.2.12 - 1977 titled Reporting:

  a) This Section states that the audit report shall be signed by the audit team leader; this is not always the most expeditious route to take to assure that the audit report is issued as soon as practical. CR3 shall comply with Section 4.4 as clarified in the opening: "An audit report, which shall be signed by the audit team leader, or his supervisor in his absence, shall provide:" In cases where the audit report is not signed by the Lead Auditor due to his absence, one record copy of the report must be signed by the Lead Auditor upon his return. The report shall not require the Lead Auditor's review/concurrence/signature if the Lead Auditor is no longer employed by CR3 at the time the audit report is issued.

  b) CR3 shall comply with Subsection 4.4.3 clarified to read: "Supervisory level personnel with whom significant discussions were held during the course of pre-audit (where conducted), audit, and post-audit (where conducted) activities."

  c) Audit reports may not necessarily contain an evaluation statement regarding the effectiveness of the quality assurance program elements which were audited, as required by Subsection 4.4.4, but they shall provide a summary of the audited areas and the results.

  d) Subsection 4.4.6 requires audit reports to include recommendations for corrective actions; CR3 may choose not to comply with this requirement. Instead, CR3 auditors/lead auditors are required to document all adverse findings. The procedure for processing Audit Findings allows the auditor/lead auditor to document actions which are considered necessary to correct the finding; the auditor/lead auditor may also document actions which are reconsidered unacceptable for correcting the finding. The Audit Finding with these "Auditor Recommendations" is then transmitted to the audited organization. In addition, the auditor/lead auditor is required to review the response to the Audit Finding and determine if it is acceptable. Any disagreements must be escalated to higher management for resolution via the corrective action process.

  e) The last paragraph in Section 4.4 deals with distribution of audit reports. CR3 shall comply with these requirements after substituting the following for the last sentence: "Audit reports shall be issued within thirty working days after the last day of the audit. The last day of an audit shall be considered to be the day of the post-audit conference. If a post-audit conference is not held because it was deemed unnecessary, the last day of the audit shall be considered to be the date the post-audit conference was deemed unnecessary as documented in the audit report."

14) With regard to Section 4.5.1 of ANSI N45.2.12 - 1977 titled By Audited Organization: CR3 shall comply with the following clarification of this Section: "Management of the audited organization or activity shall review and investigate all adverse audit findings, as necessary (e.g., where the cause is not already known or another organization has not already investigated and found the cause, etc.), to determine and schedule appropriate corrective action (which includes action to prevent recurrence). They shall respond, in writing, within thirty working days after the date of issuance of the audit report. The response shall clearly state the corrective action taken or planned to prevent recurrence and the results of the investigation, if conducted. In the event that corrective action is not completed by the time the response is submitted, the audited organization's response shall include a scheduled date for completion of planned corrective action. A follow-up response shall be provided stating the corrective action taken and the date that the action was completed. If corrective actions are verified as satisfactorily completed by the quality organization prior to the scheduled completion date, no follow-up response is required. The audited organization shall take appropriate action to assure that corrective action is accomplished as scheduled. Either the Director, Audits and Programs, or their designee, may waive the requirement for a supplementary response.

Table 1-3
Crystal River Unit 3 Quality Program Commitments
(continued)

15) With regard to Section 5.1 of ANSI N45.2.12 - 1977 titled General: CR3 shall maintain records in accordance with and to meet the requirements of Section 1.7.1.17 of the Quality Program Description and ANSI N45.2.9 as specified in Table 1-3.

Table 1-3
Crystal River Unit 3 Quality Program Commitments
(continued)

NRC REGULATORY GUIDE 1.146 - "QUALIFICATION OF QUALITY ASSURANCE PROGRAM AUDIT PERSONNEL FOR NUCLEAR POWER PLANTS" (REV. 0, 8/80) - ENDORSES ANSI N45.2.23 - 1978

With the Certification of Permanent Cessation of Power Operations, DEF has permanently ceased power operations of Crystal River Unit 3 (CR3). CR3 follows this Regulatory Guide and associated Standard for the remaining important to safety activities associated with SAFSTOR, decommissioning, and the Independent Spent Fuel Storage Installation (ISFSI) with the following clarifications:

1) With regard to Section 1.4 of ANSI N45.2.23 - 1978 titled Definitions: Definitions in this Standard which are not included in ANSI N45.2.10 will be used. “AUDIT,” which is included in ANSI N45.2.10, will be used as clarified in CR3’s commitment to Regulatory Guide 1.74.

2) With regard to Section 2.2 of ANSI N45.2.23 - 1978 titled Qualification of Auditors: Subsection 2.2.1 references an ANSI B45.2 (presumed to be ANSI N45.2); therefore, CR3 shall comply with an alternate Subsection 2.2.1 which reads: "Orientation to provide a working knowledge and understanding of the Quality Program Description, including the ANSI Standards and Regulatory Guides included in Table 1-3 of that Program, and CR3's procedures for conducting audits and reporting results."

3) With regard to Section 2.3.4 of ANSI N45.2.23 – 1978 titled Audit Participation: CR3 shall substitute the following for this Section: “Prospective Lead Auditors shall demonstrate the ability to effectively implement the audit process and effectively lead an audit team. This process is described in written procedures that provide for evaluation and documentation of the results of this demonstration. In addition, the prospective Lead Auditor shall have participated in at least two Nuclear Oversight audits within the year preceding the individual’s effective date of qualification. Upon successful demonstration of the ability to effectively implement the audit process and effectively lead audits, and having met the other provisions of Section 2.3 of ANSI/ASME N45.2.23-1978, the individual may be certified as being qualified to lead audits.”

4) With regard to Section 4.1 of ANSI N45.2.23 - 1978 titled Organizational Responsibility: CR3 shall comply with this Section with the substitution of the following sentence in place of the last sentence in the Section. "Management, or the Audit Team Leader, shall, prior to commencing the audit, assign personnel who collectively have experience or training commensurate with the scope, complexity, or special nature of the activities to be audited."

5) With regard to Section 5.3 of ANSI N45.2.23 - 1978 titled Updating of Lead Auditor's Records: CR3 shall substitute the following sentence for this Section: "Records for each Lead Auditor shall be maintained and updated during the annual management assessment as defined in Section 3.2."

6) With regard to Section 5.4 of ANSI N45.2.23 - 1978 titled Records Retention: CR3 shall substitute the following sentence for this Section: "Qualification records shall be generated and maintained as required by Sections 1.7.1.2, 17, and 1.7.1.18.1 of the Quality Program Description and by commitment to ANSI N45.2.9 as set forth in Table 1-3 of that Program."

FIGURE 1-2, CRYSTAL RIVER 1, 2 & 3 SITE PLOT PLAN, CR3-G86-D

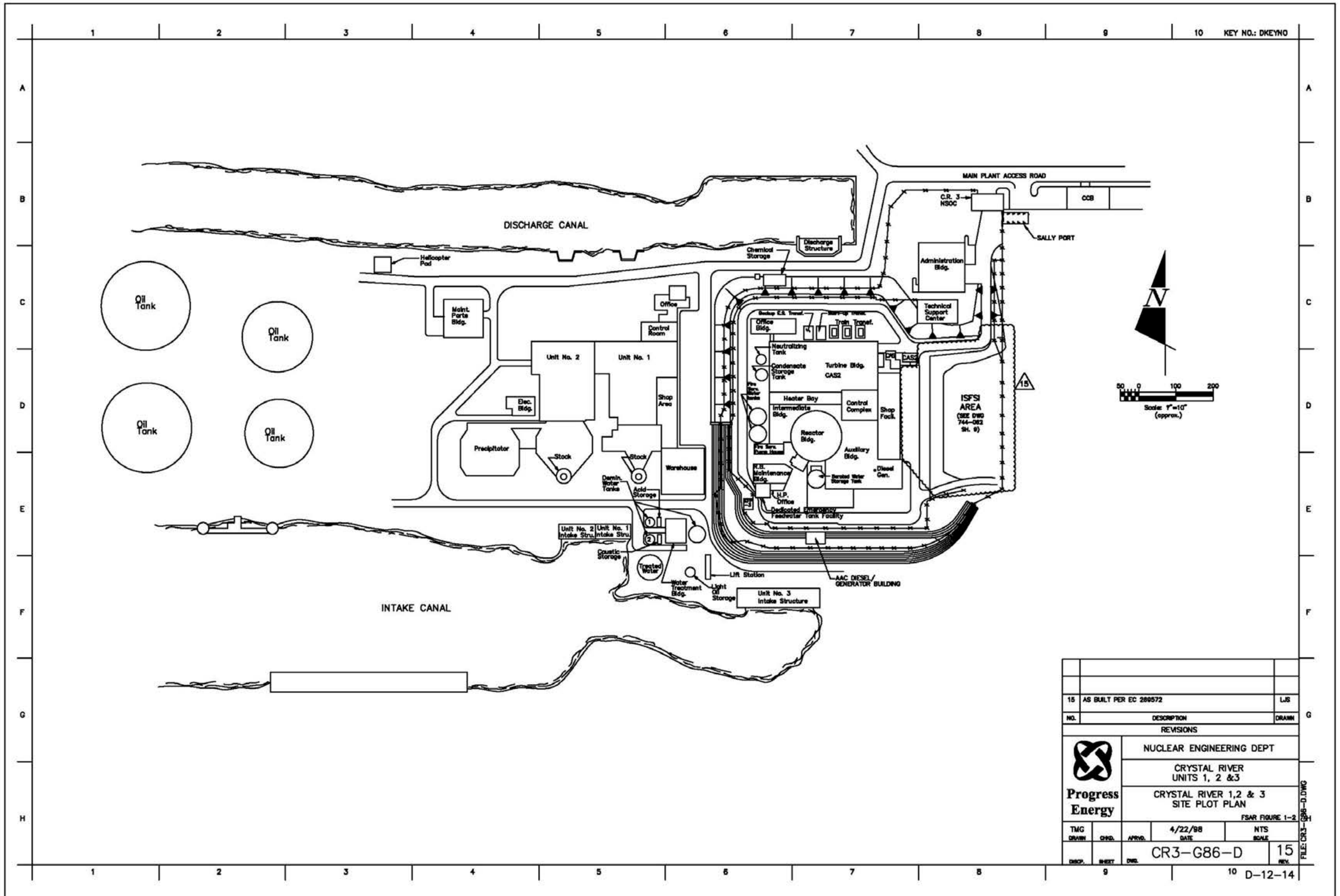


FIGURE 1-3, CRYSTAL RIVER SITE PLOT PLAN, CR3-G85-D

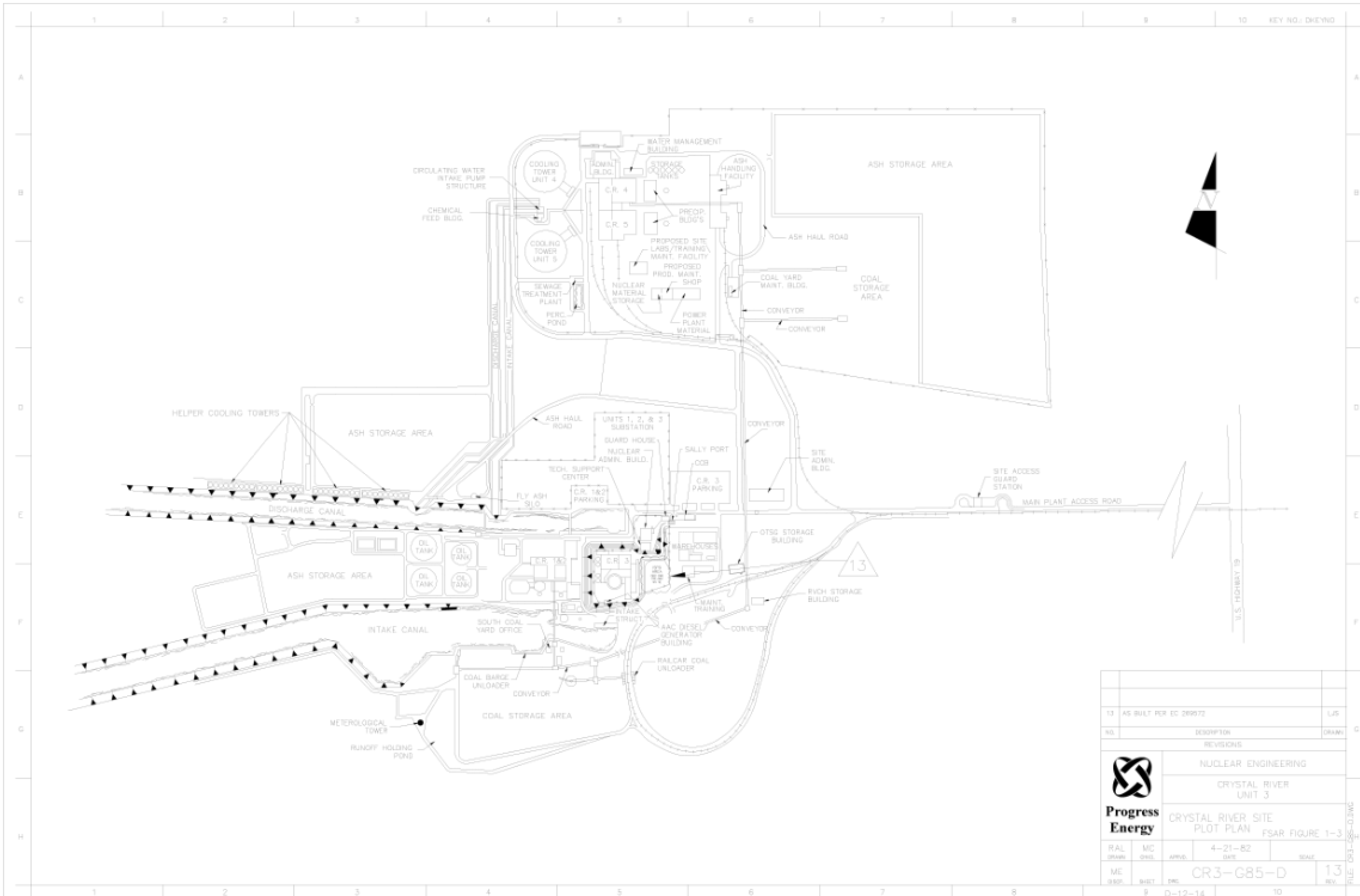


FIGURE 1-4, LAYOUT PLAN ABOVE TURBINE BLDG & HEATER BAY, BASEMENT FLOOR ELEVATION 95'-0", L-001-011

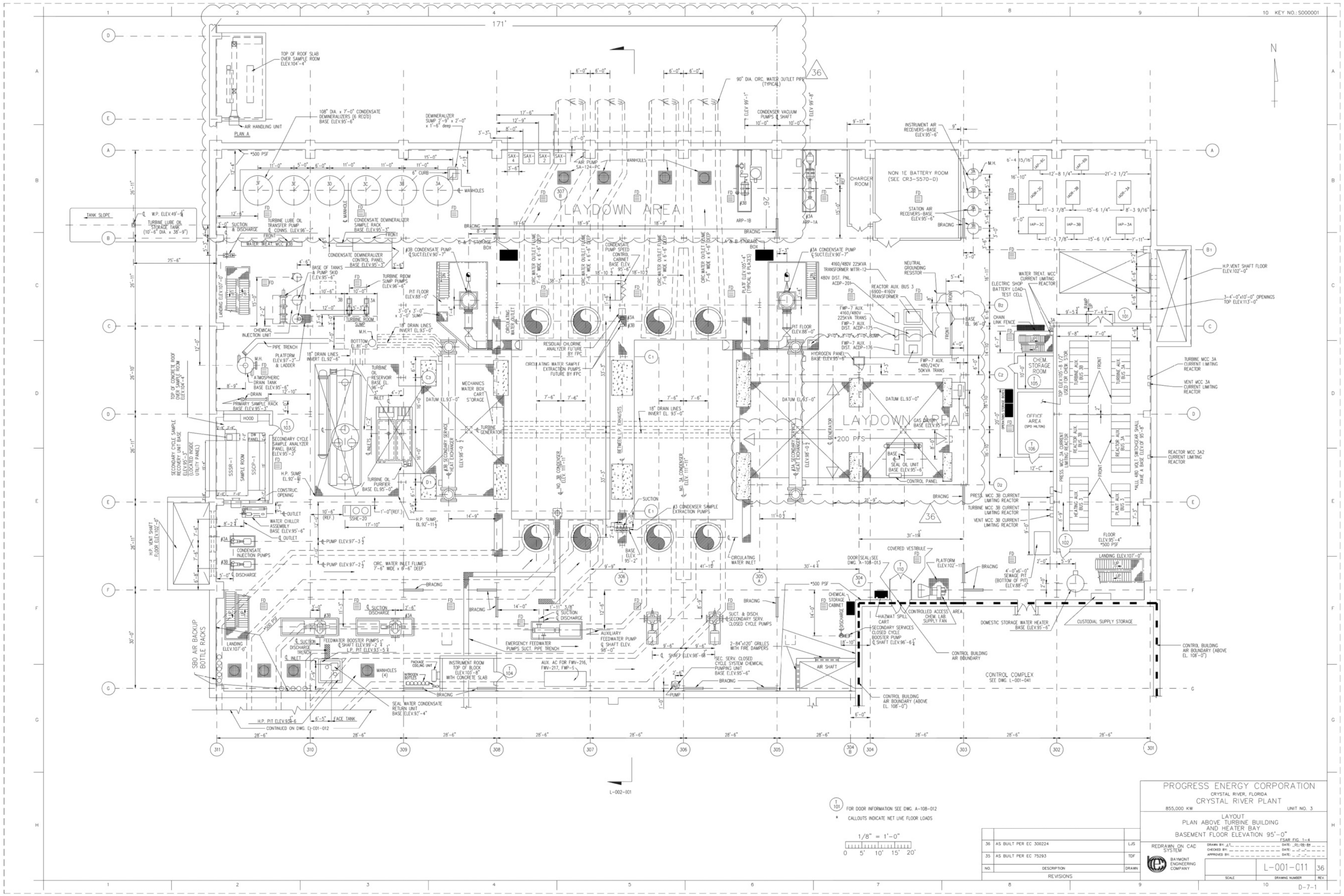


FIGURE 1-5, LAYOUT PLAN ABOVE REACTOR, AUXILIARY & INTERMEDIATE BLDGS, BASEMENT FLOOR ELEVATION 75'-0", L-001-012

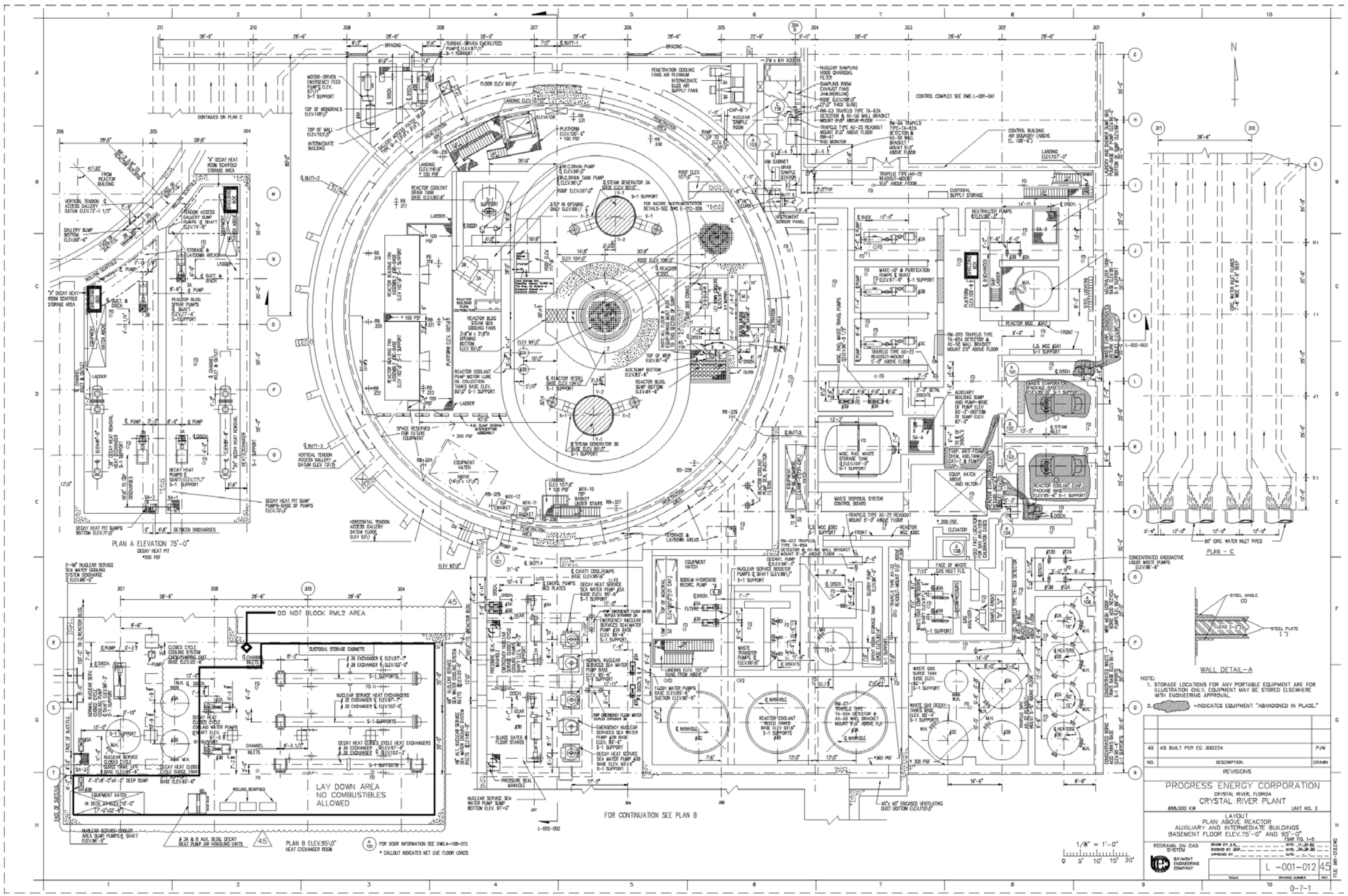


FIGURE 1-6, LAYOUT PLAN ABOVE TURBINE BLDG & HEATER BAY, MEZZANINE FLOOR ELEVATION 119'-0", L-001-021

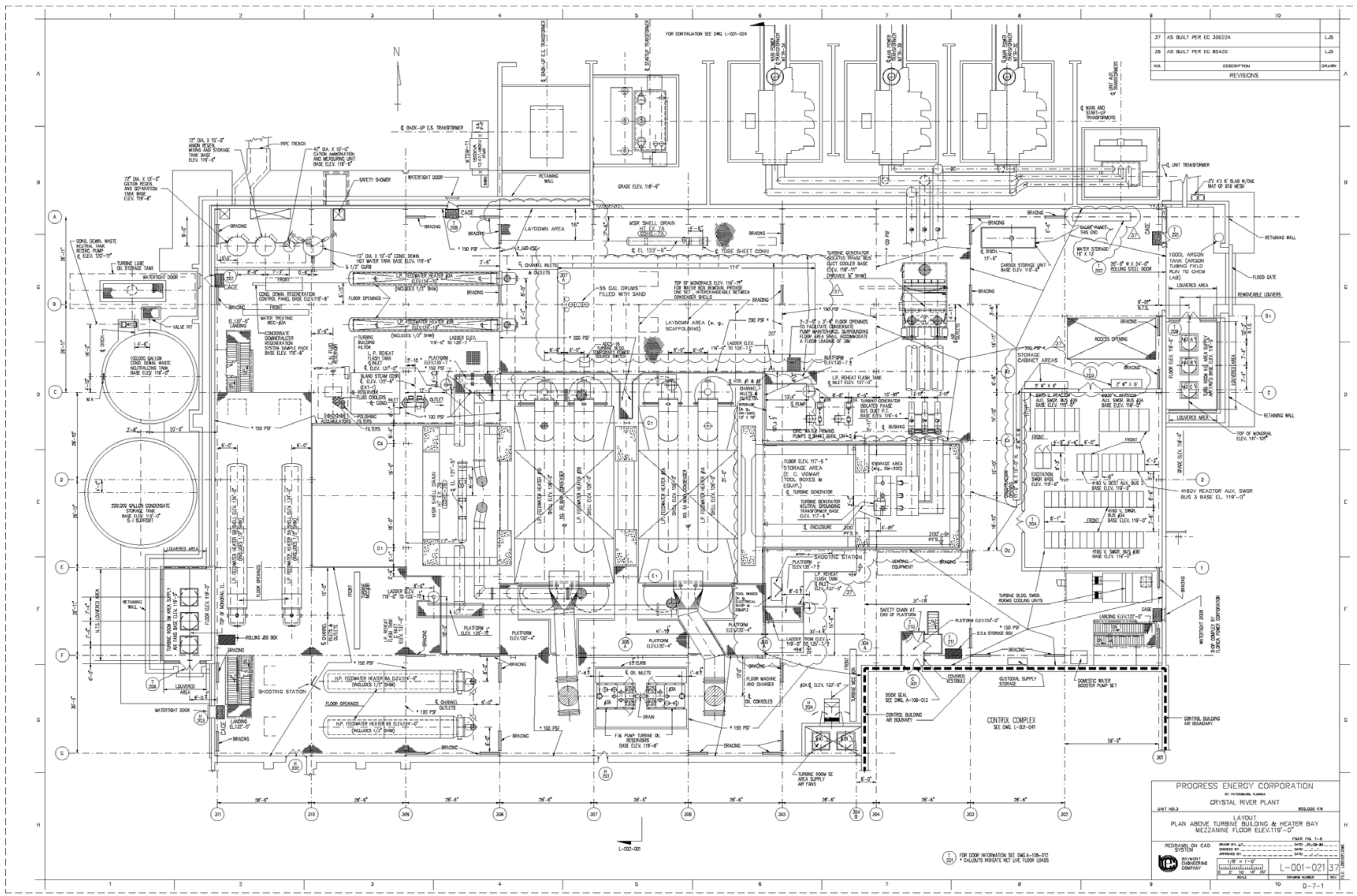


FIGURE 1-7, LAYOUT PLAN ABOVE REACTOR, AUXILIARY & INTERMEDIATE BLDGS, MEZZANINE FLOOR ELEVATION 119'-0", L-001-022

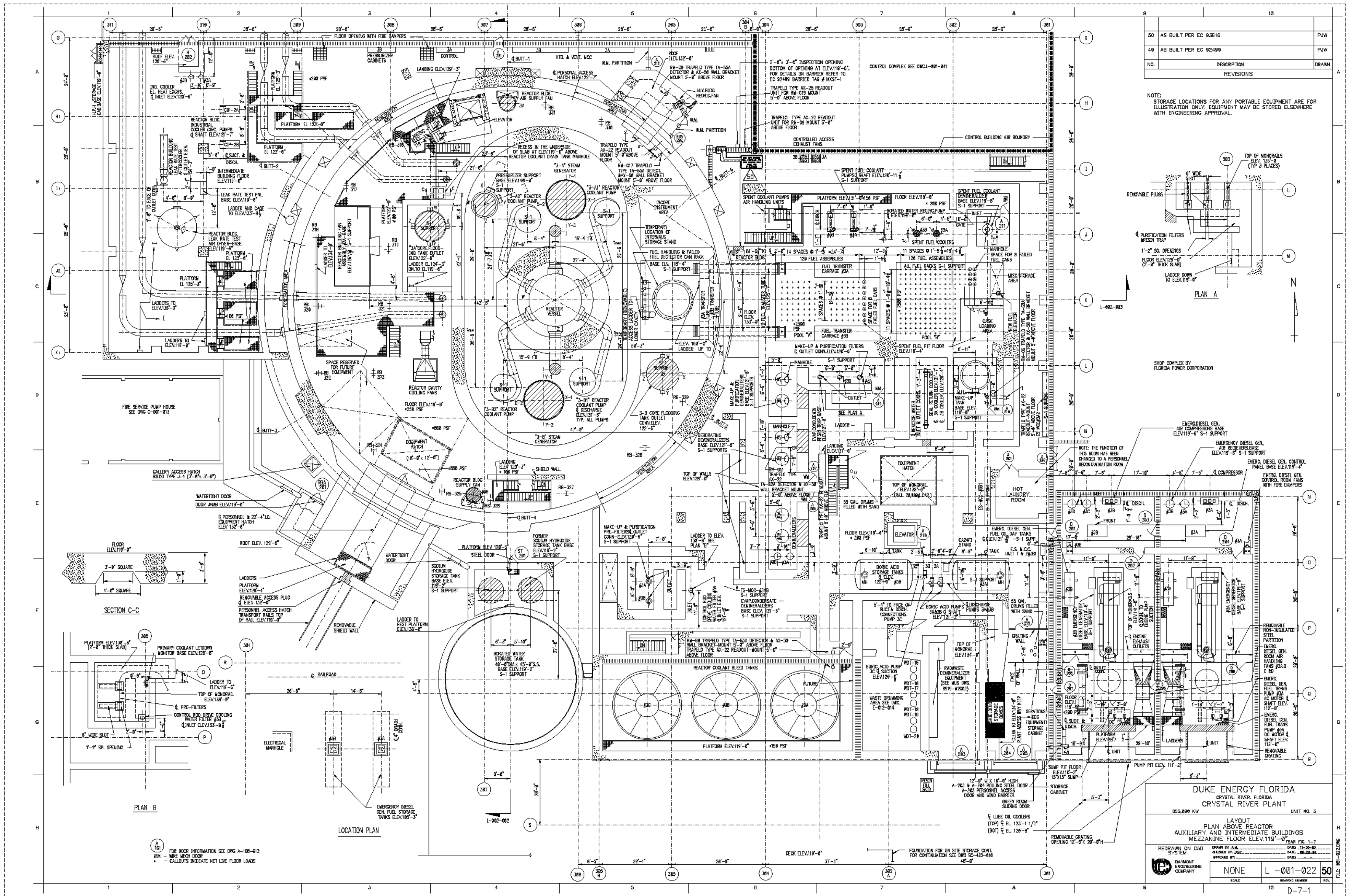


FIGURE 1-8, LAYOUT PLAN ABOVE REACTOR, AUXILIARY & INTERMEDIATE BLDGS, ELEVATION 143'-0", L-001-023

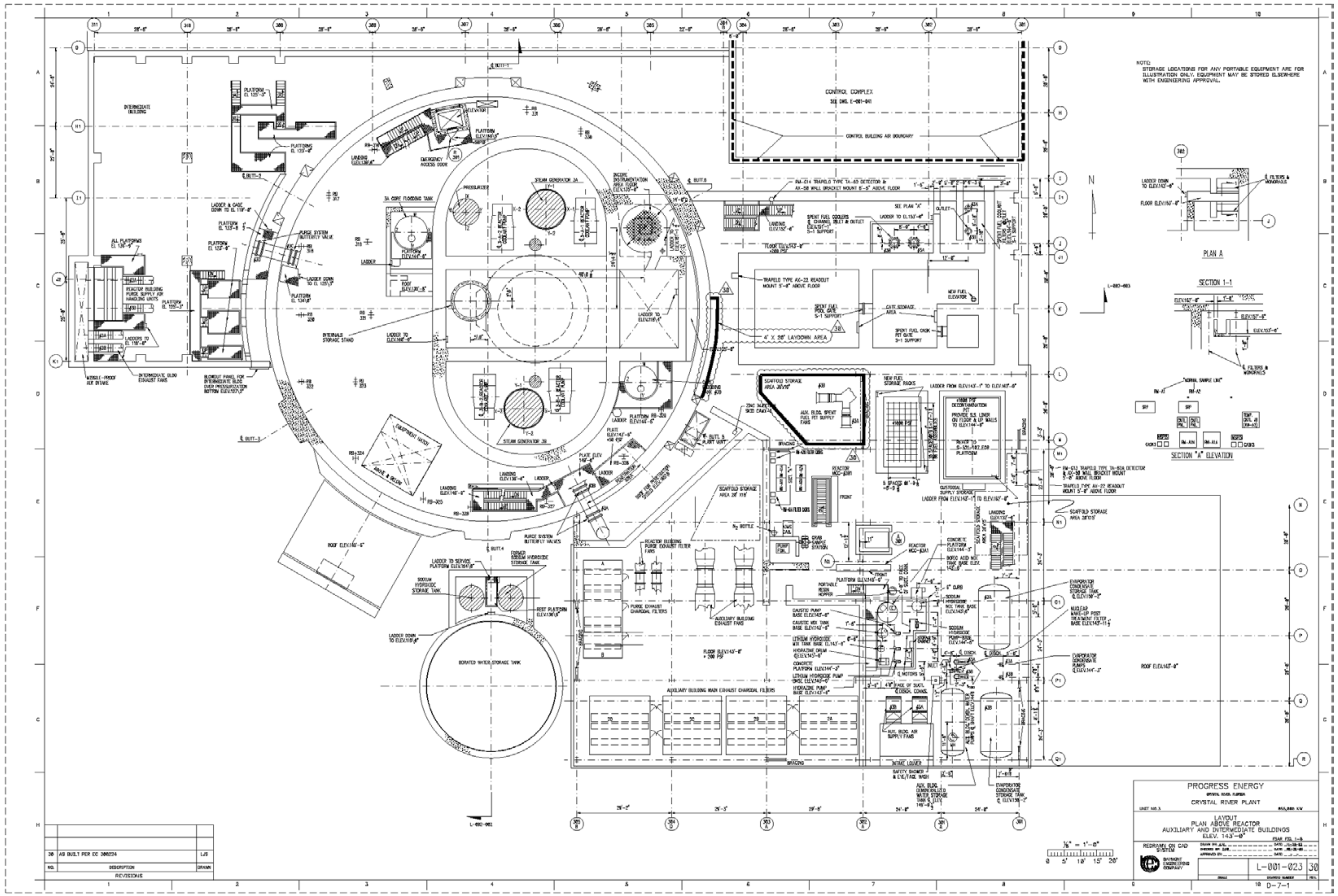


FIGURE 1-9, LAYOUT PLAN NORTH OF COLUMN LINE A, TURBINE BLDG., L-001-024

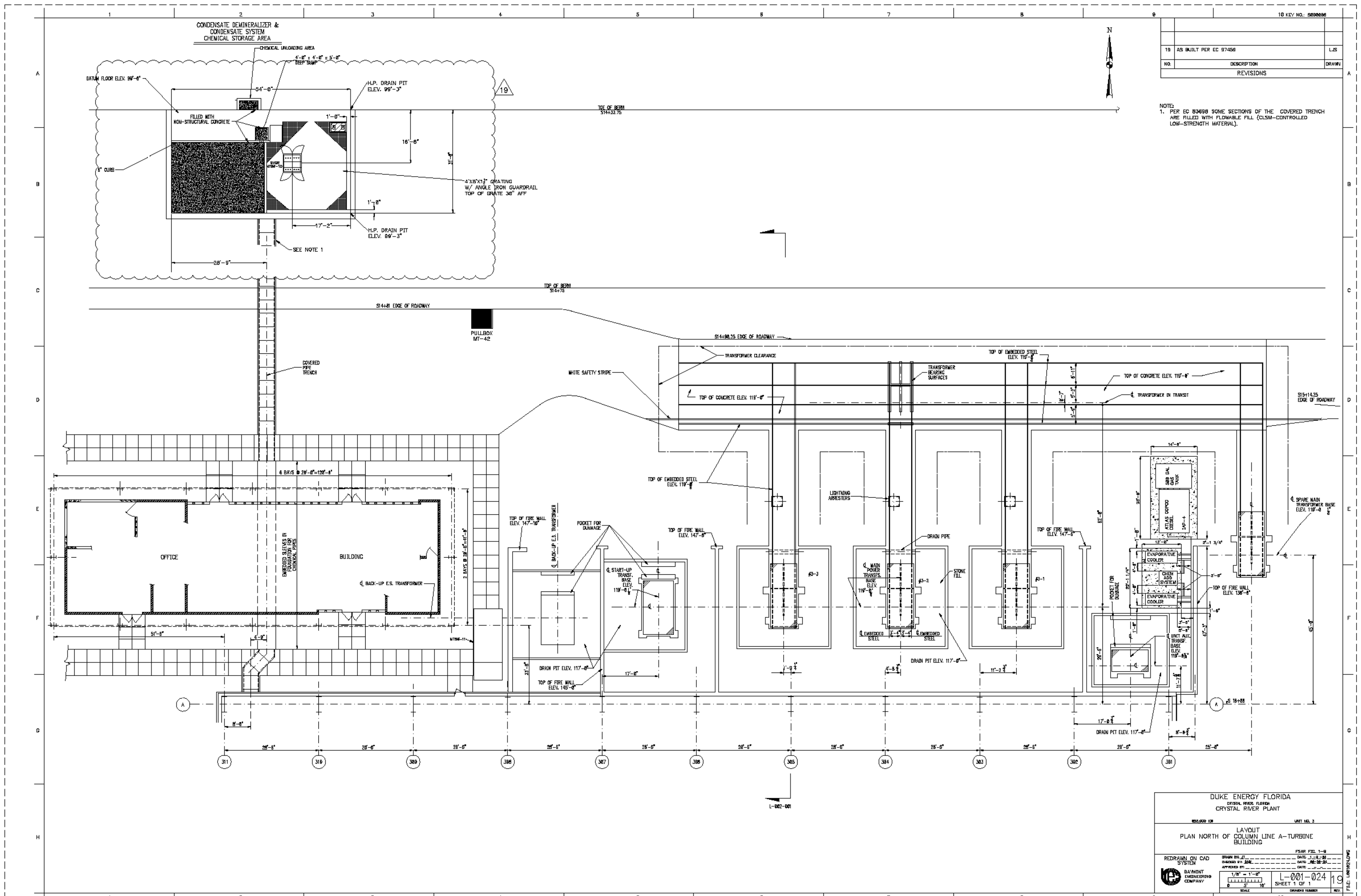


FIGURE 1-10, LAYOUT PLAN ABOVE TURBINE BLDG & HEATER BAY, OPERATING FLOOR ELEVATION 145'-0", L-001-031

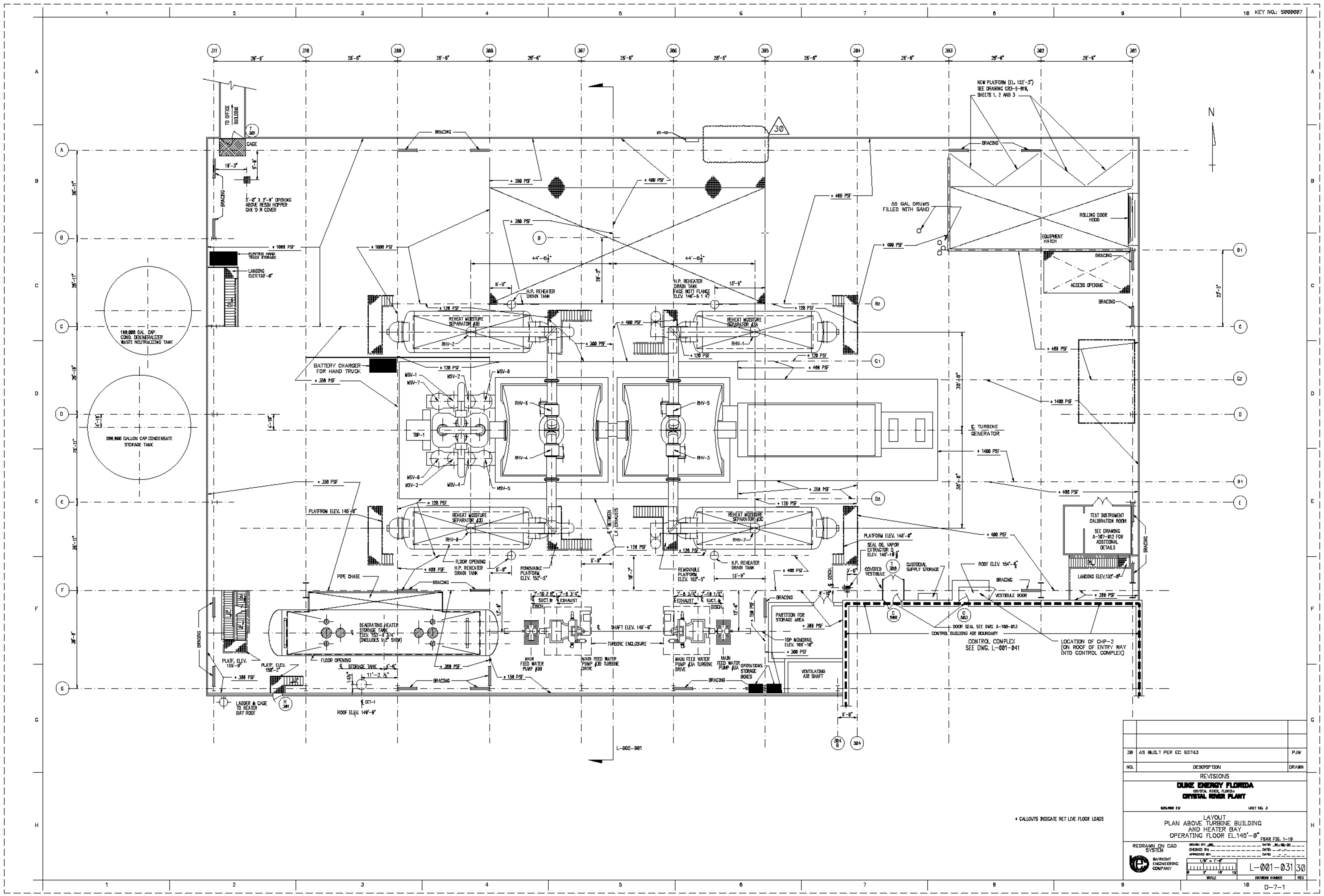


FIGURE 1-11, LAYOUT PLAN ABOVE REACTOR BUILDING OPERATING FLR ELEV 160' & AUXILIARY BLDG ELEV 162', L-001-032

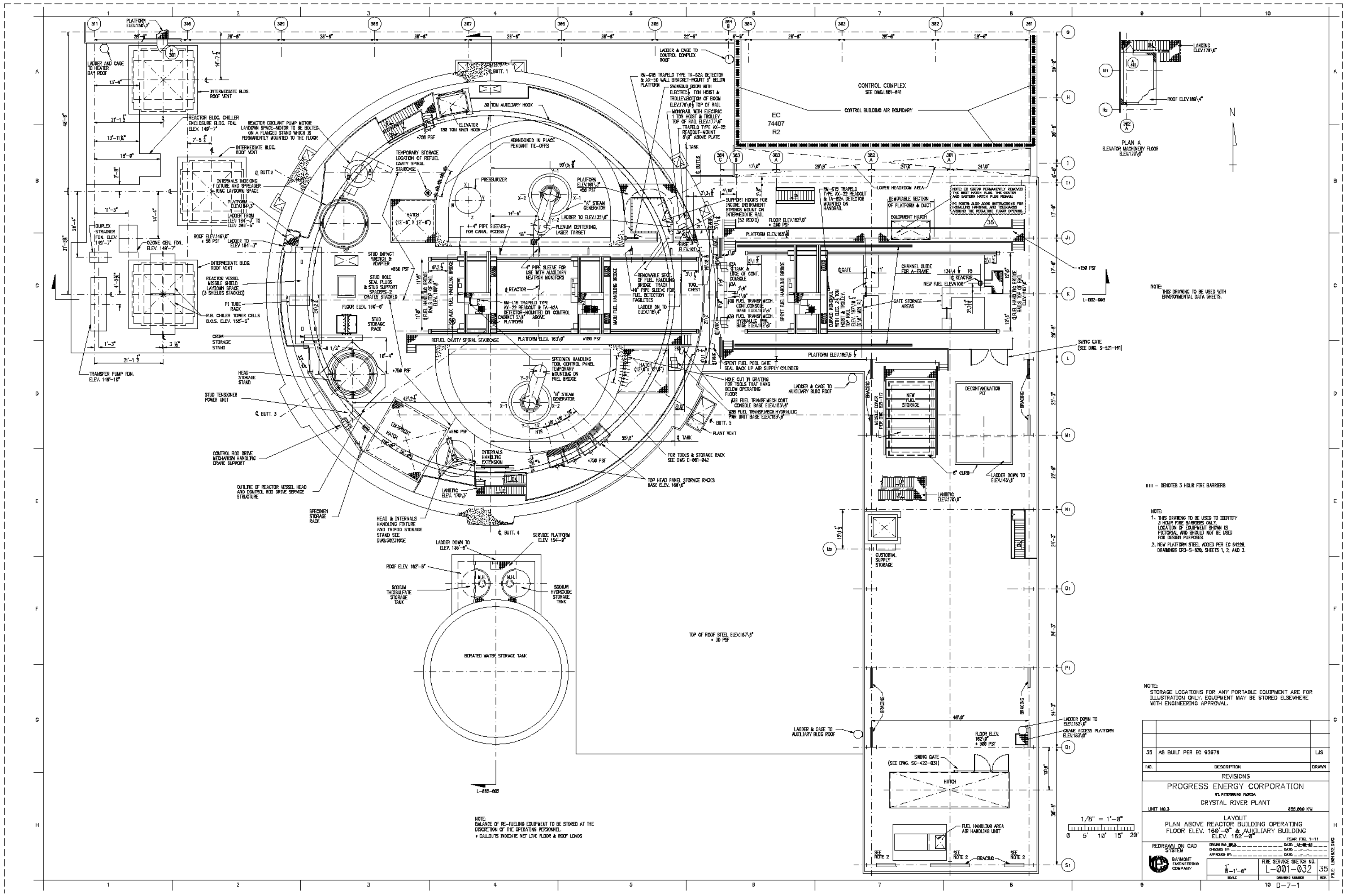


FIGURE 1-12, LAYOUT PLAN ABOVE HEATER BAY FLR ELEC 164' & 170' AND CONTROL COMPLEX ELEVATIONS 95' THRU 181', L-001-041

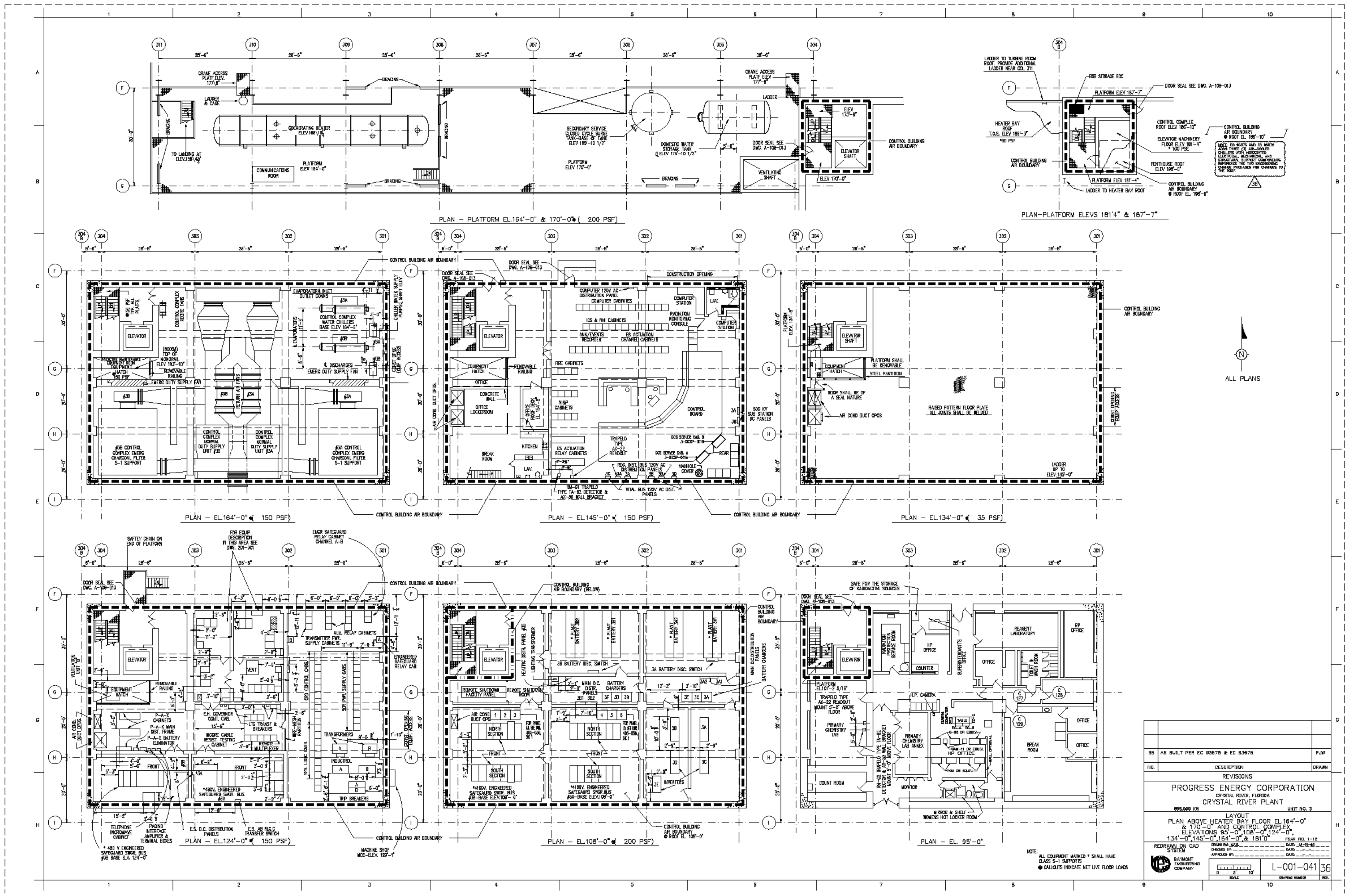


FIGURE 1-13, LAYOUT PLAN ABOVE REACTOR BUILDING, ELEV 180', L-001-042

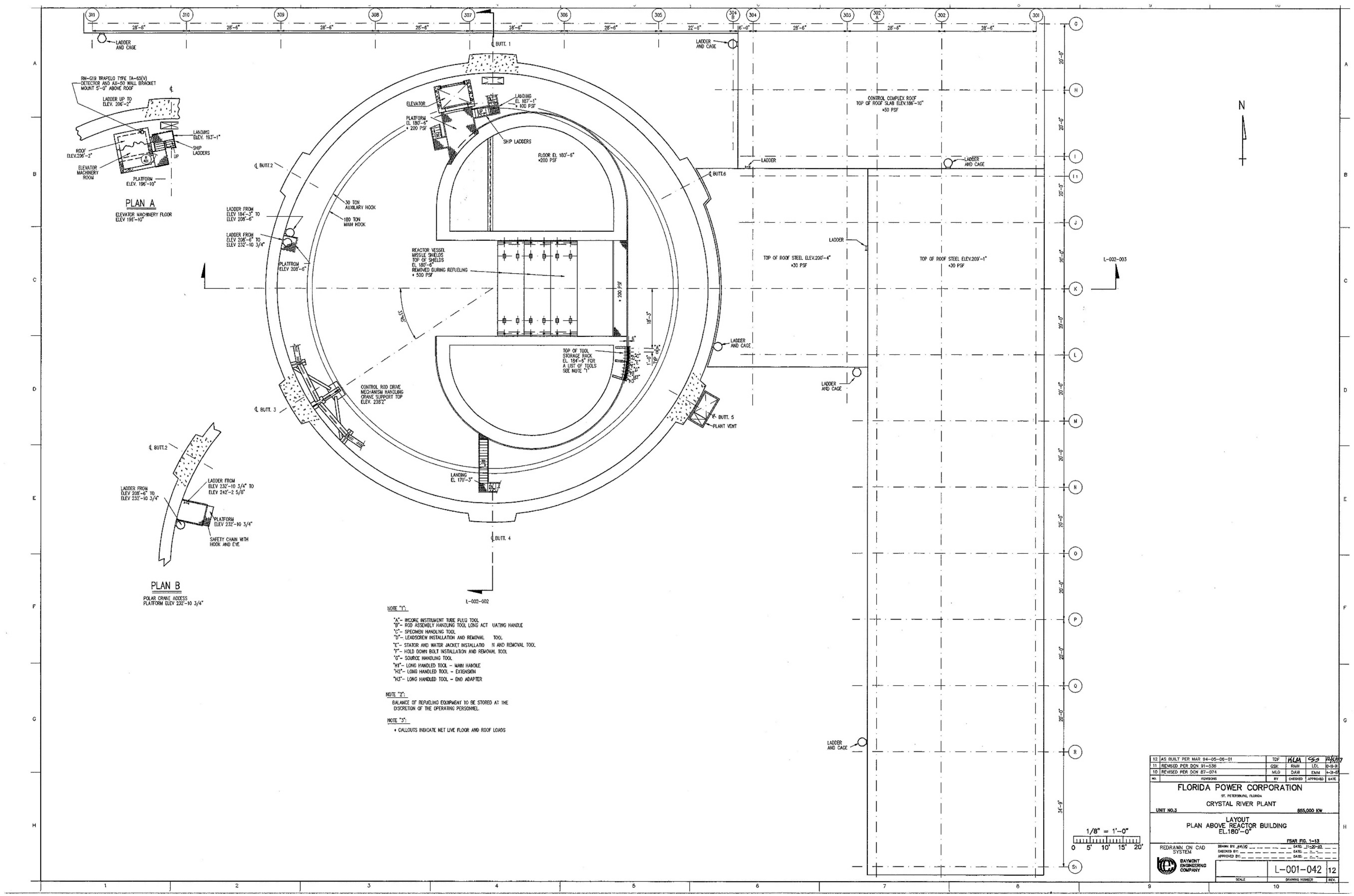


FIGURE 1-14, LAYOUT CROSS SECTION THRU TURBINE BLDG AND HEATER BAY, L-002-001

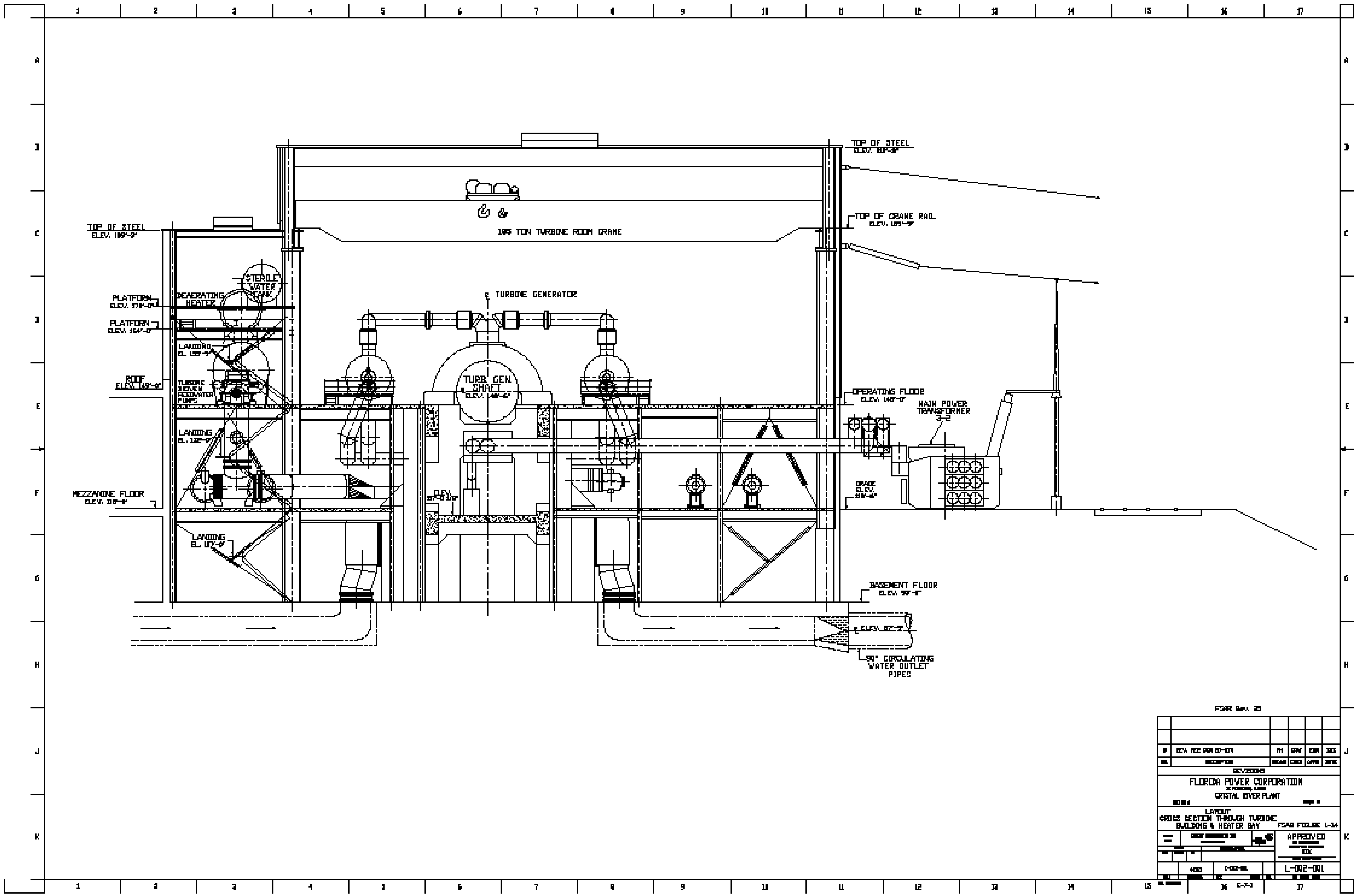


FIGURE 1-15, LAYOUT CROSS SECTION THRU REACTOR BLDG AND AUXILIARY BLDG, L-002-002

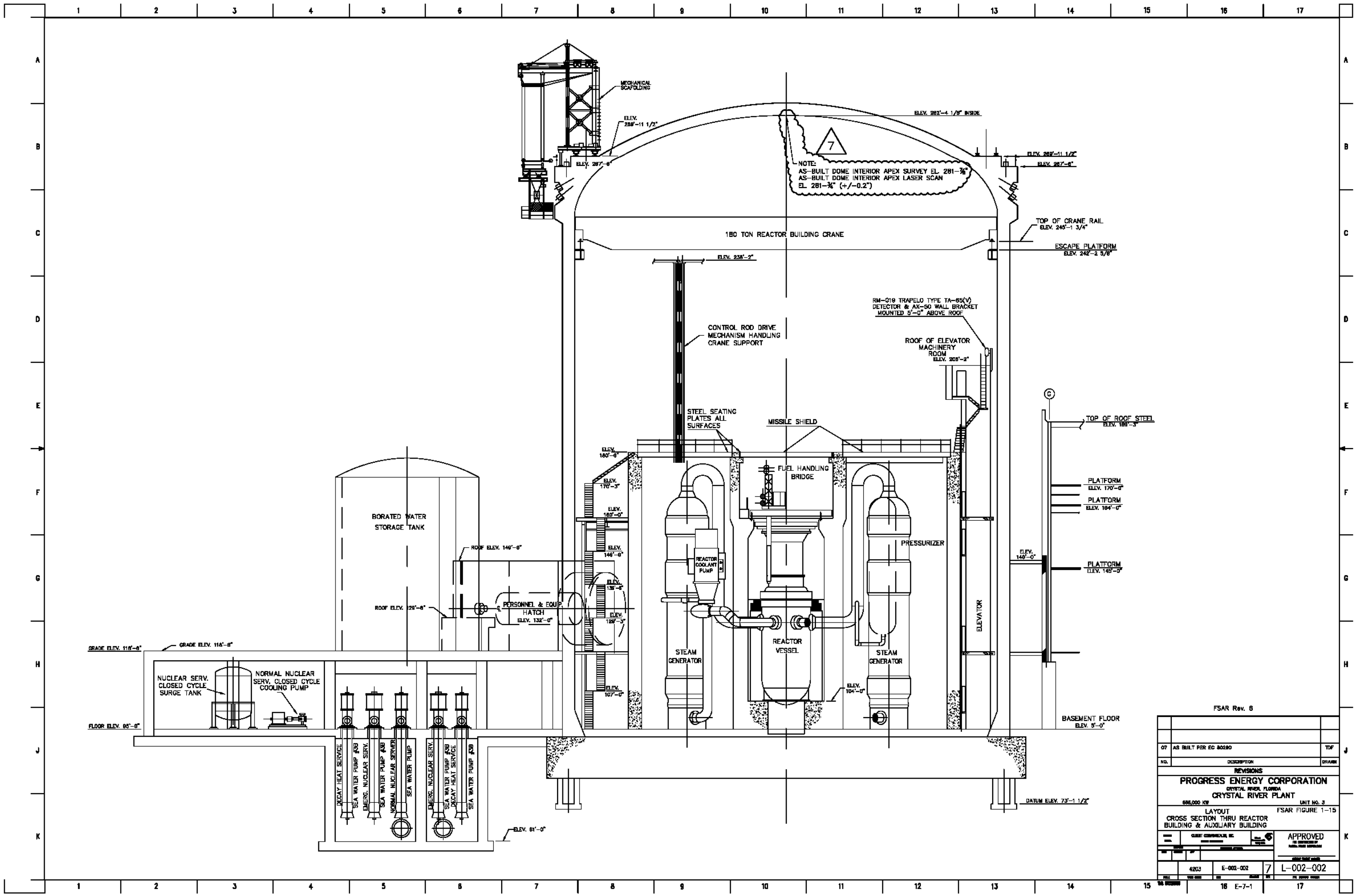
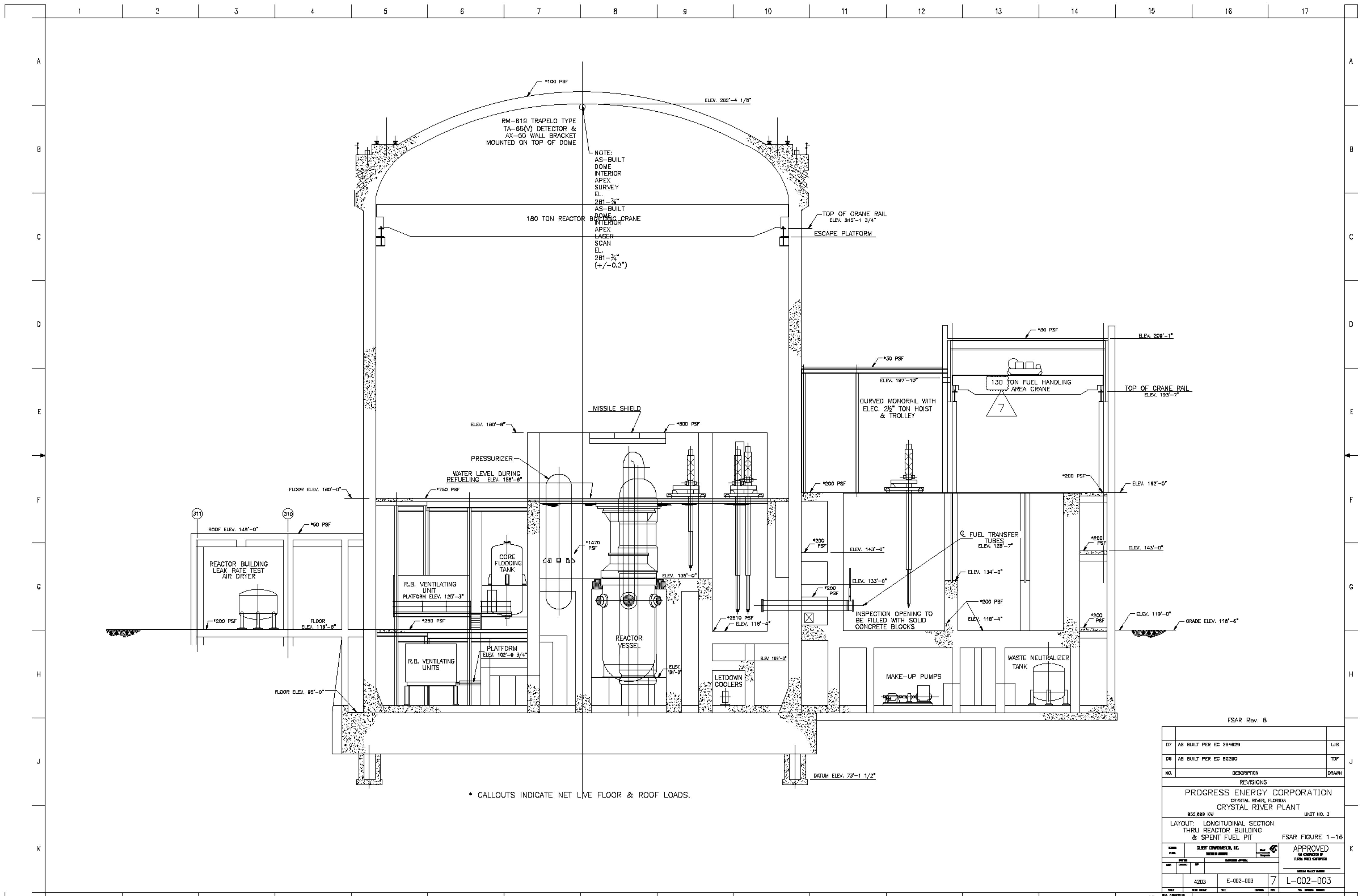


FIGURE 1-16, LAYOUT LONGITUDINAL SECTION THRU REACTOR BLDG & SPENT FUEL PIT, L-002-003



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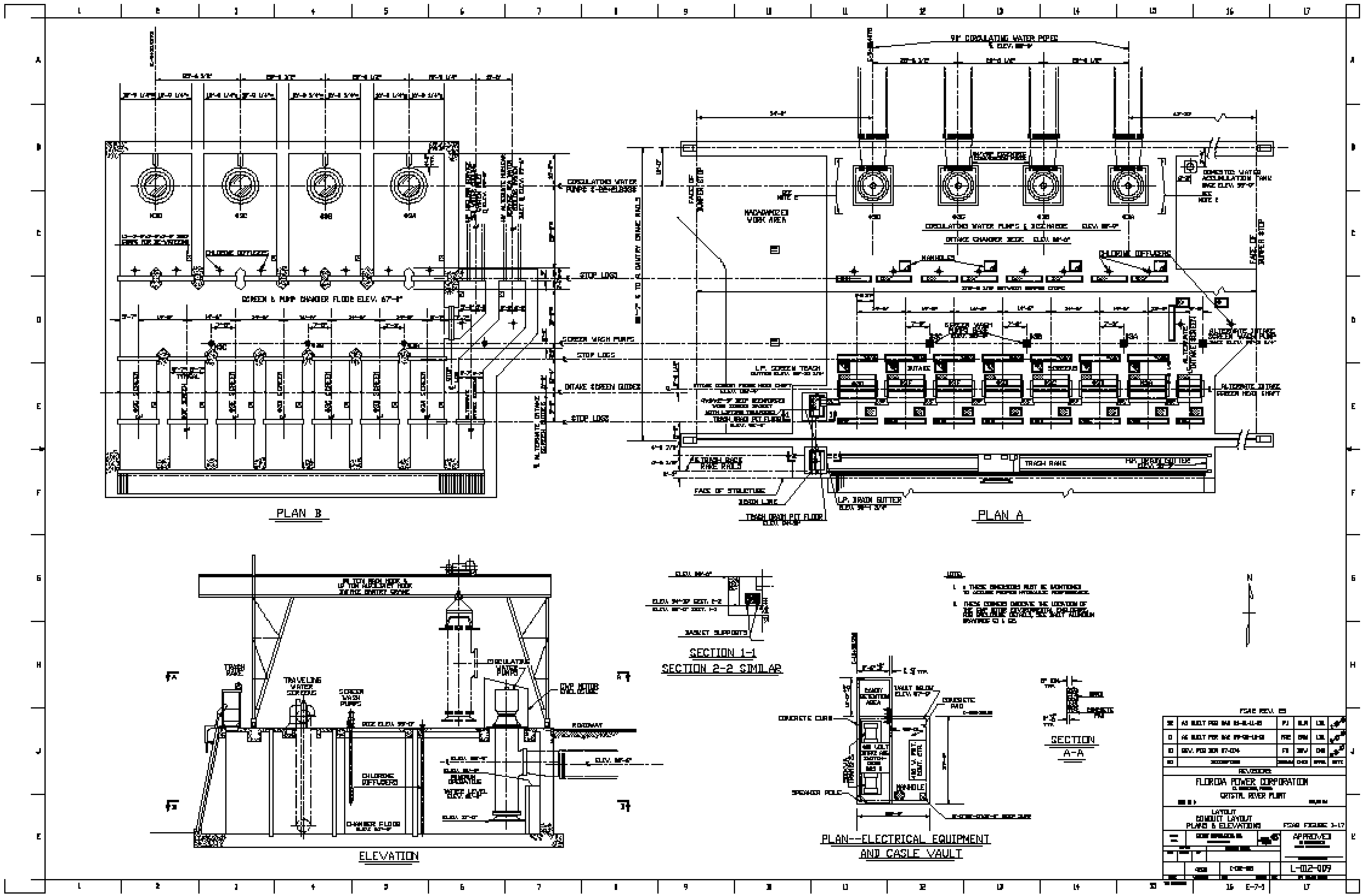
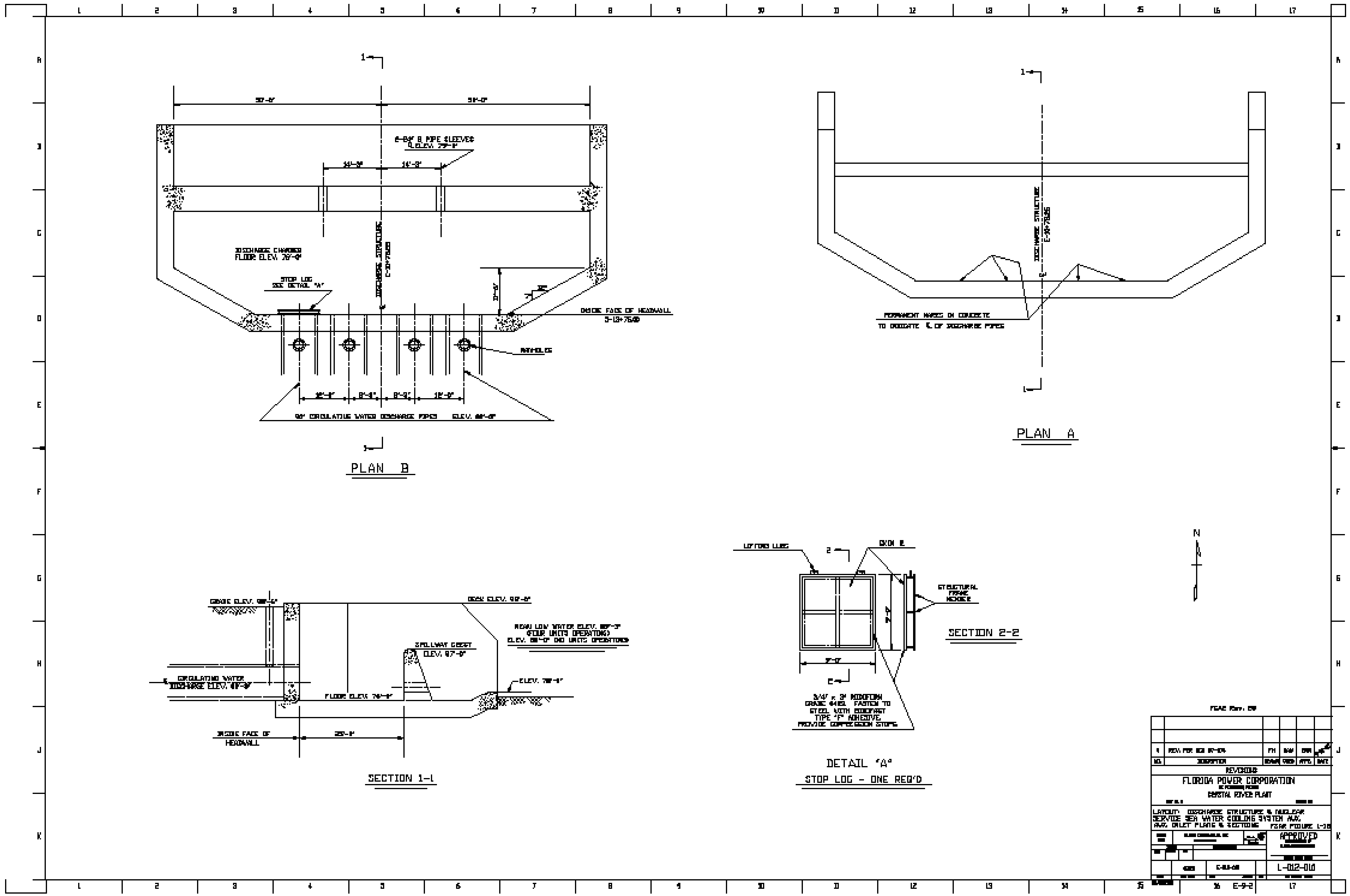


FIGURE 1-18, DISCHARGE STRUCTURE AND NUCLEAR SERVICES SEAWATER COOLING WATER SYSTEM, L-012-010



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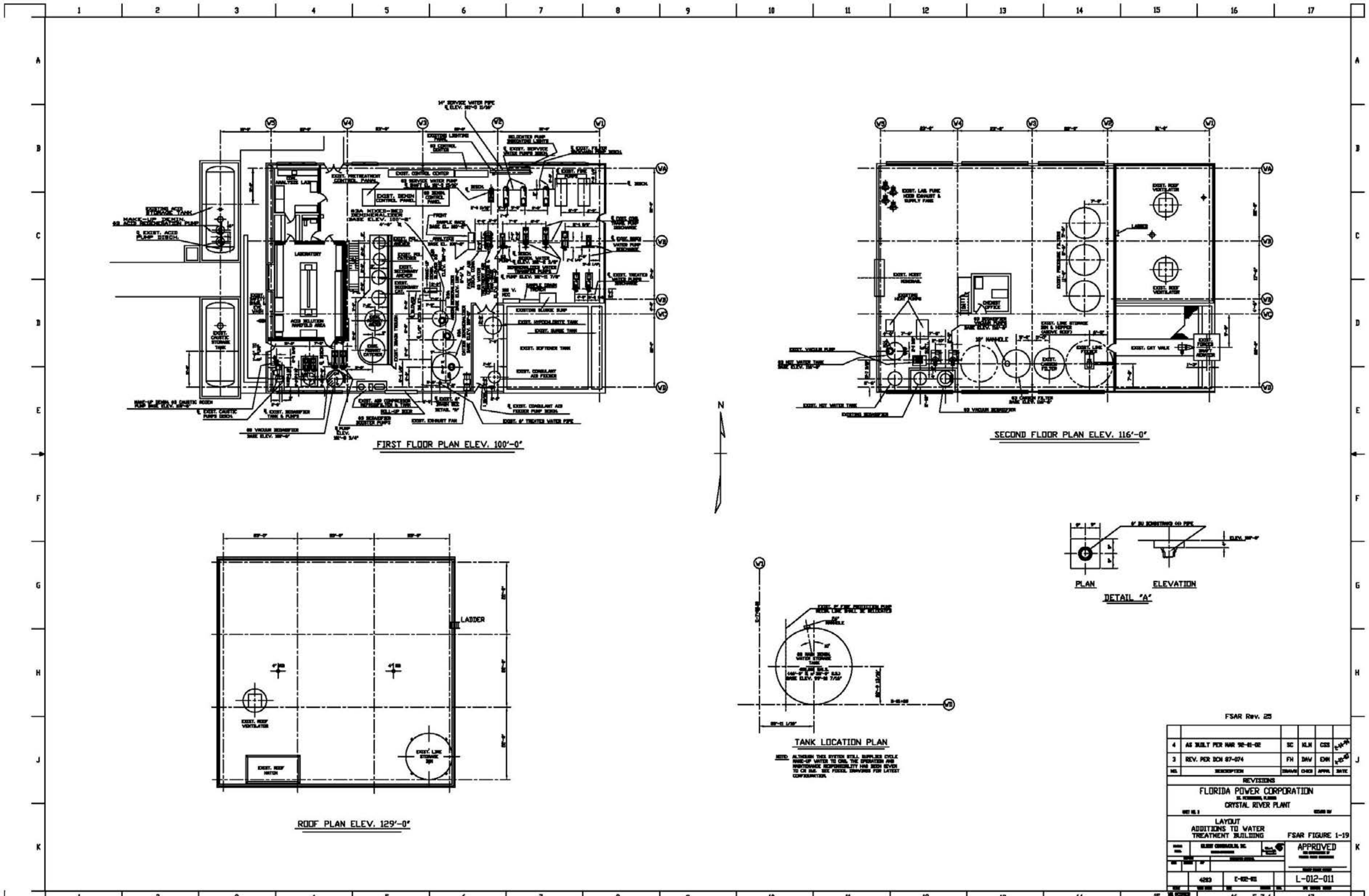

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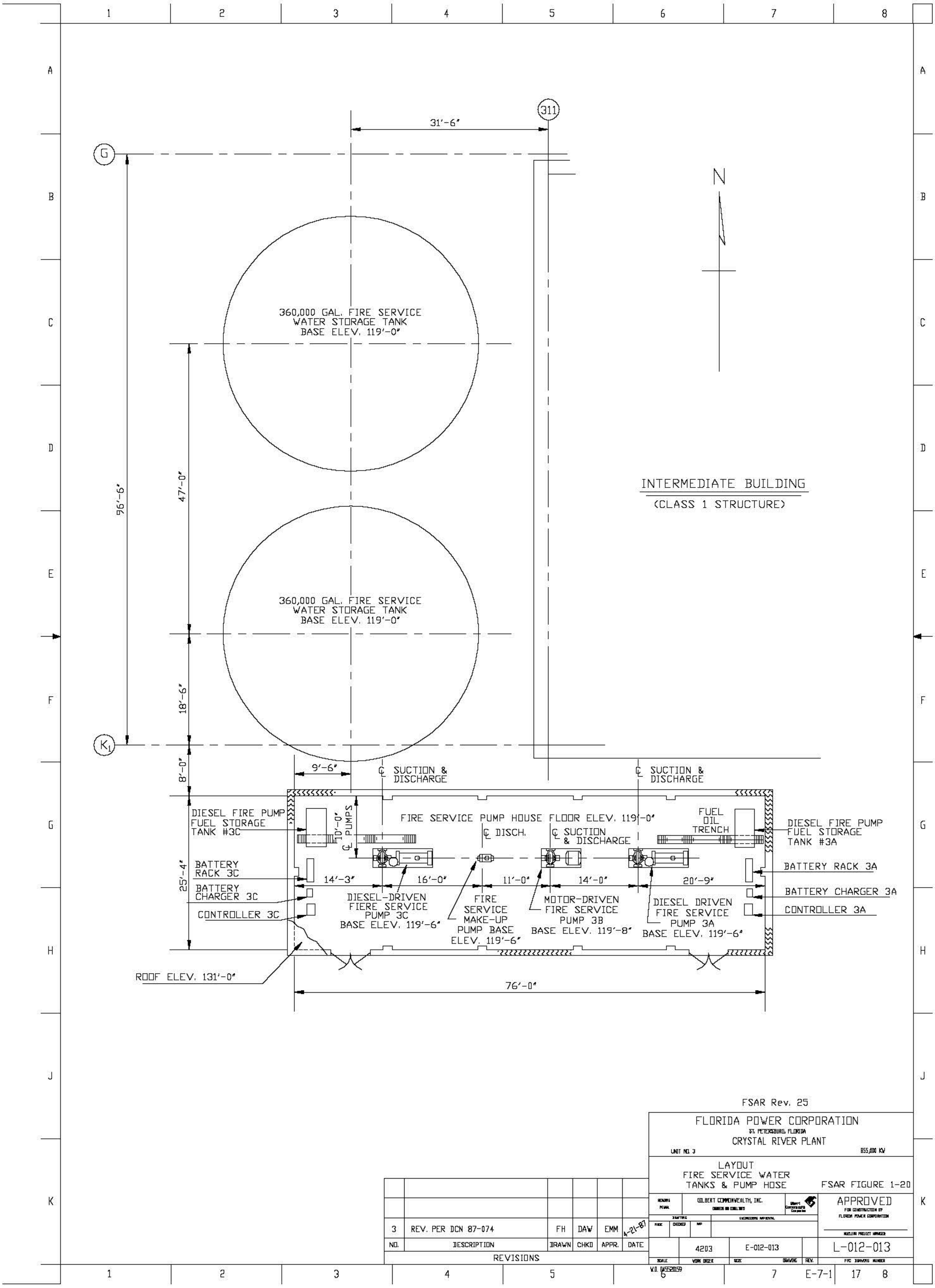


FIGURE 1-21, WASTE DRUMMING AREA ELEVATION 119'-0", L-012-014

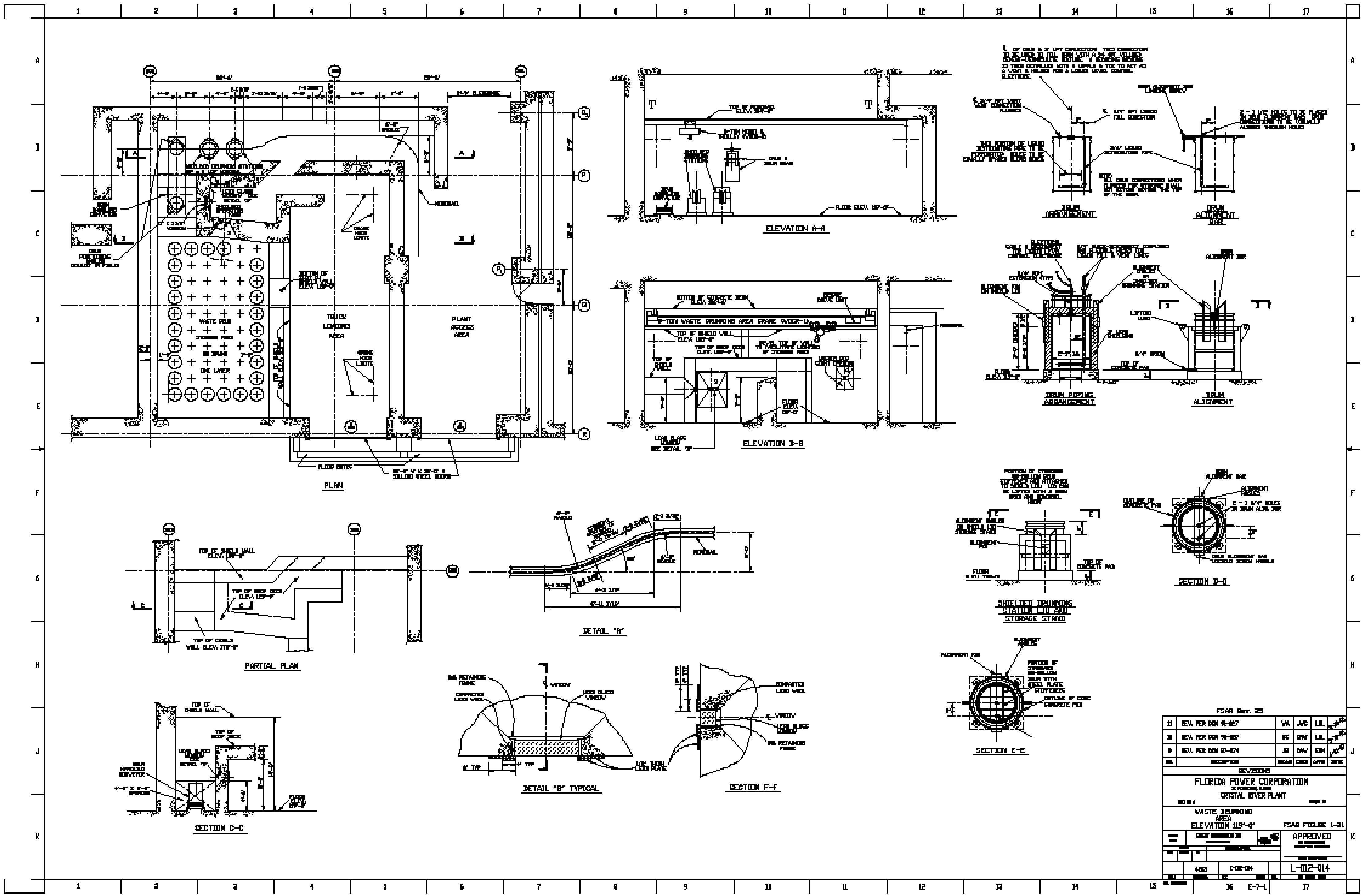


FIGURE 1-22, MARK B-9 FUEL ASSEMBLY

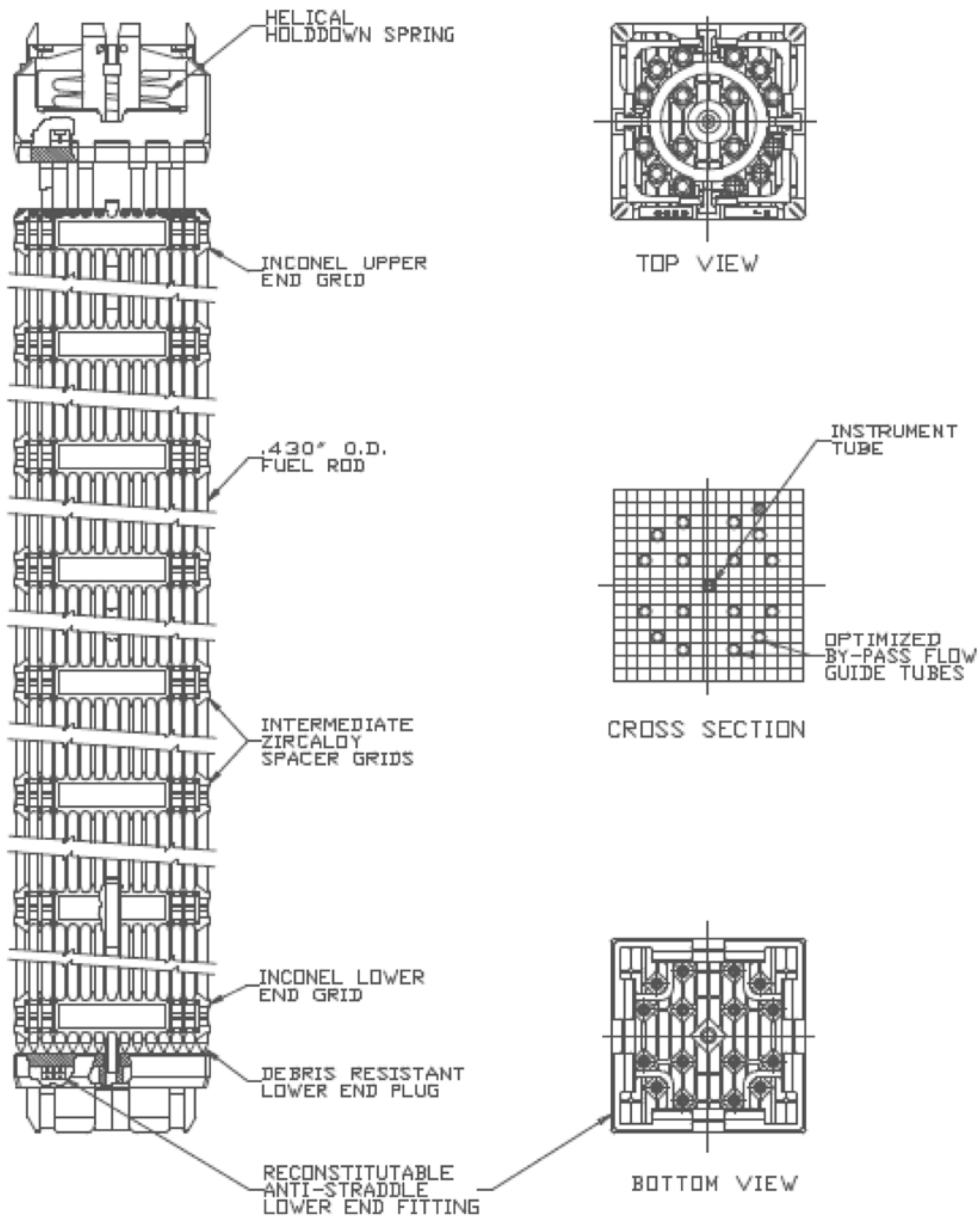


FIGURE 1-23, MARK B-10 FUEL ASSEMBLY

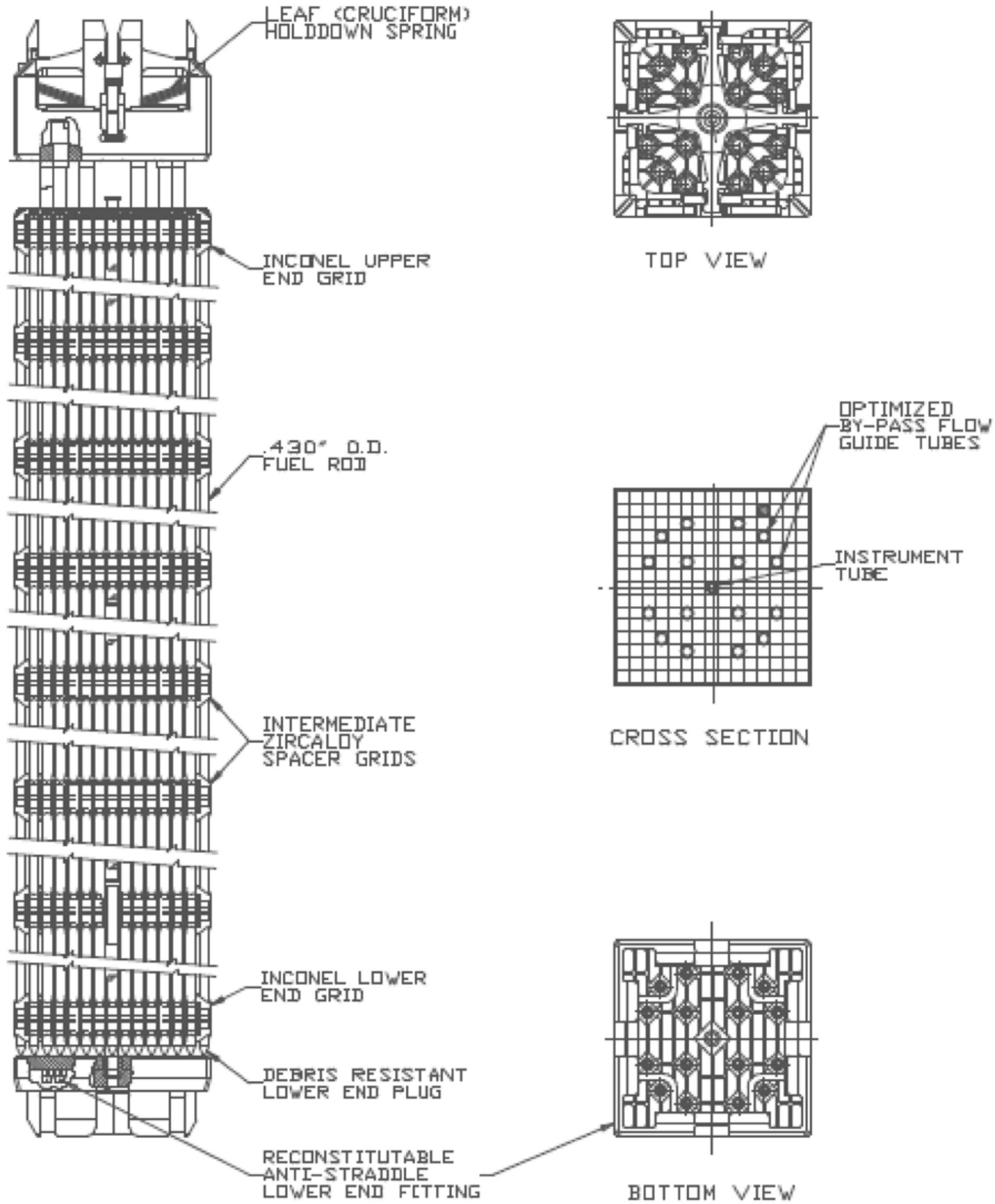


FIGURE 1-24, MARK B-HTP FUEL ASSEMBLY

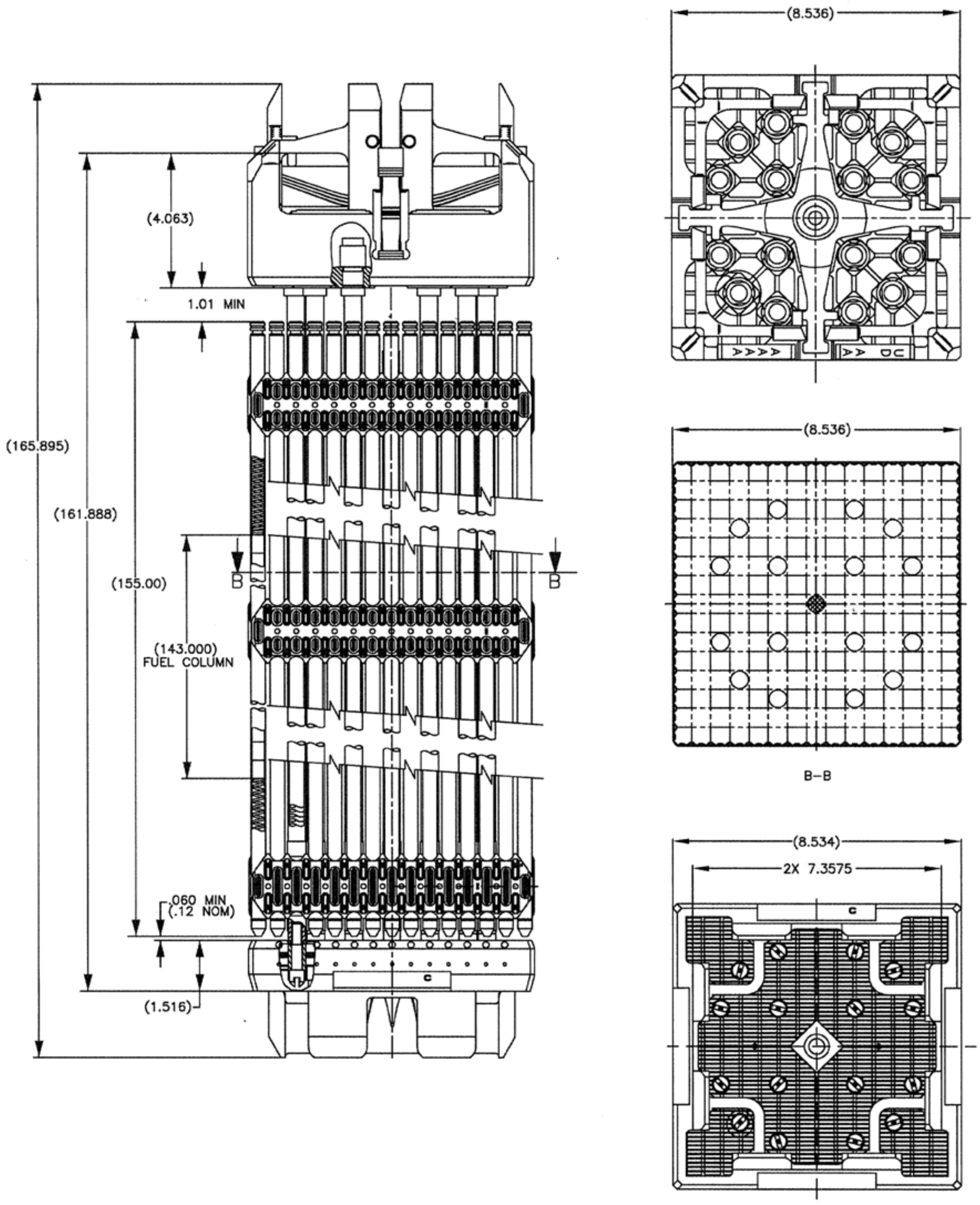
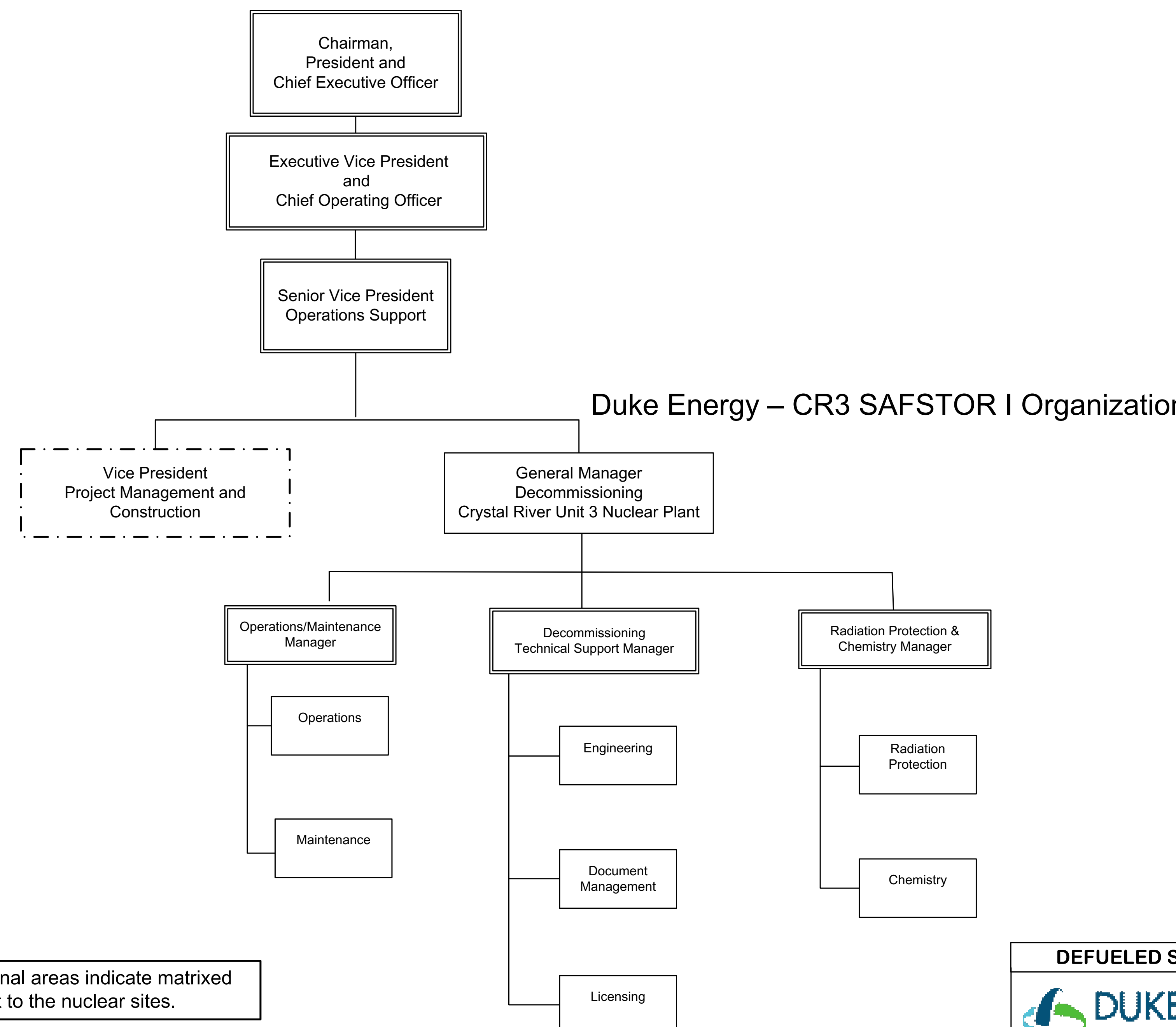


FIGURE 1-26, CR3 NUCLEAR ORGANIZATION

Duke Energy – CR3 Nuclear Organization



Note: Dashed borders around functional areas indicate matrixed organizations providing support to the nuclear sites.

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DUKE ENERGY	Duke Energy SAFSTOR I Department
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2. SITE AND ENVIRONMENT

2.1 SUMMARY

This Chapter is presented as a basis for the selection of design criteria for Crystal River Unit 3. A series of studies (geology, seismology, hydrology, meteorology, population, and land use) have been conducted.

The site is located on the Gulf of Mexico, 70 miles north of Tampa, Florida.

Dilution flow for liquid releases is drawn from and returned to the Gulf of Mexico using present intake and discharge canal.

The exclusion area has a radius of 4,400 feet within a 4,738 acre site wholly owned and controlled by Duke Energy Florida. Based on the 1973 census there is one population center of 25,000 or more within a radius of 50 miles. Ocala, Florida, with a 1973 population of 27,963 is located 36 miles ENE of the site.

The site region is predominantly agricultural in nature.

The subtropical marine climate of the site region is characterized by diurnal wind shifts from the Gulf of Mexico and frequent nocturnal inversions; however, atmospheric diffusion of waste gases is good. "Extreme mile" winds are not expected to exceed 110 mph once in 100 years. The plant is protected in accordance with the Florida Building Code for flooding and Maximum Probable Hurricane (MPH) winds.

All potable water supplies within a 20 mile radius of the site are derived from wells or springs fed by the ground water table which slopes to the Gulf of Mexico. Surface and subsurface drainage is to the Gulf of Mexico only and therefore cannot affect any potable water supplies, streams, or surface waters in the area.

The structures are founded on underlying limerock which is competent with respect to foundation conditions for Crystal River Unit 3. Florida is in a relatively aseismic zone; therefore, the Crystal River Unit 3 structures are designed conservatively to a horizontal acceleration of 0.05 gravity.

2.2 SITE AND ADJACENT AREAS

2.2.1 SITE LOCATION AND TOPOGRAPHY

The property wholly owned and controlled by Duke Energy Florida (herein referred to as the "site") is located in the northwestern portion of Citrus County, State of Florida, and lies either wholly or partly in Sections 3, 4, 5, and 9, Township 18S, Range 16E, Sections 28, 29, 30, 31, 32, 33, 34, 35, and 36, Township 17S, Range 16E, and Section 31, Township 17S, Range 17E.

This site location is approximately 7-½ miles NW of Crystal River, and 70 miles N of Tampa as shown in Figure 2-1.

Crystal River Unit 3 is located at latitude 28° 57' 25.87"N and longitude 82° 41' 55.95" W and lies in the northwest corner of Section 33, Township 17S, Range 16E. The Universal Transverse Mercator (UTM) Grid Conversion for the above latitude and longitude is N 3 204 253.13 meters and E 334 445.32 meters, respectively.

The site region is characterized by the Gulf of Mexico on the west with gradually rising terrain from mangrove swamp and marshland in the coastal areas to gently rolling hills about 16 miles to the east (see Figure 2-2).

Situated between the mouths of the Withlacoochee and Crystal Rivers, the site is primarily composed of marshland and low-lying areas, scattered with a variety of vegetation ranging from swamp grass to heavily wooded areas.

Topographic features out to a 5 mile radius are shown in Figure 2-3 and an aerial photograph of the site is shown in Figure 2-4.

2.2.2 SITE OWNERSHIP

The Crystal River site consists of 4,738 acres owned and controlled by Duke Energy Florida including a ¼ mile wide access strip provided for railroad, road, and transmission line right-of-way extending from the plant to U.S. Highway 19. This access strip is crossed by Old U.S. Highway 19 at a distance of 951 feet west of U.S. Highway 19. The owner controlled area is indicated in Figure 2-3 and Figure 2-4.

The site property extends westward to the bulkhead line. The description of the bulkhead line is as follows:

> Beginning at a point on the North boundary of Government Lot 7, Fractional Section 30, Township 17 South, Range 16 East, according to the United States Government Survey in the year l835, said point being S. 89°32'20"W., 3097.58 feet from the NE. corner of Government Lot 6 of said Section, said point being on the Government Meander Line of said Section, thence S. 89°32'20"W., along the Westerly projection of said North boundary, 1792.38 feet, thence S. 1° 27'22"E., 1045.88 feet, thence S. 66°40'17"E., 2664.67 feet, to a point on said Government Meander Line of said section, thence S. 31°06'32"E., along said Meander Line 639.41 feet, to a point on the North Boundary of Government Lot 1, Fractional Section 31, Township 17 South, Range 16 East, said point being S. 89°38'15"W., 2120.62 feet, from the NE. corner of said Lot 1, thence S. 0°22'23"E., along said Meander Line, 619.44 feet, thence S. 63°57'37"W., 1358.99 feet, thence S. 57°10'47"E., 1463.73 feet, to a point on said Meander Line, thence S. 15°46'52"E., along said Meander Line, 1073.87 feet, thence S. 20°53'46" E., along said Meander Line, 1411.51 feet, thence S. 56°06'55"E., along said Meander Line, 917.12 feet, thence S. 2°05'27"E., 310.01 feet, said point being N. 77°27'34"W., 689.30 feet from the SE. corner of Government Lot 3, of said Section 31, thence S. 44°27'17"E., 3388.52 feet, thence S. 17°54'33"E., 2138.33 feet, thence S. 63°22'02"E., 2642.19 feet, thence S. 89° 40'21"E., 1703.73 feet, thence S. 61°17'01"E., 3002.91 feet, thence S. 65°45'46"E., 1760.96 feet, to a point on the West Boundary of Fractional Section 10, Township 18 South, Range 16 East, said point being S. 0°55'35"E., 2385.01 feet from the NW. corner of said section.

Figure 2-3 shows these boundaries and the immediate adjacent areas to the plant site.

There are no public access roads to areas adjacent to the plant site except at the plant access road. Approximately four miles east of the plant, a dirt road crosses the site access road. The north and south site boundaries are bordered by woods and swamps and are generally inaccessible. The Crystal River is located due south of the site and is used for commercial fishing and pleasure craft. Directly west of the plant is the Gulf of Mexico, from which the Crystal River plant site historically received its condenser cooling water. Fishing and pleasure craft have unrestricted access to the Gulf waters. Company property extends to the Gulf of Mexico, approximately 7,000 feet beyond from CR3. Duke Energy Florida has no legal rights to any appurtenant structures which extend into the Gulf beyond the bulkhead line described previously. Small craft are prevented from entering the discharge canal by a blockade at the bulkhead line. This blockade was installed for safety concerns due to increased water turbulence caused by the mixing of reintroduced water to the canal from the helper cooling towers.

2.2.3 SITE ACTIVITY

Duke Energy Florida (DEF) operates four fossil fuel generating units (Unit 1 373 MWe, Unit 2 469 MWe, Unit 4 670 MWe, Unit 5 670 MWe) at the plant site. At present, Duke Energy Florida has approximately 375 employees assigned to these units.

The Seaboard Coast Line Railroad Co. serves the Crystal River plant over land and tracks owned by Duke Energy Florida. A spur track connecting with the DEF track approximately 3½ miles east of the plant serves a small dolomite mining and processing operation located just north of the plant site.

The railroad spur into Crystal River plant is nine miles long from the railroad company right-of-way to the plant site. Only cars consigned to the Crystal River plant are brought into the plant site over the spur.

When shipments of hazardous chemicals are brought into the plant by railroad cars, adequate safety measures will be taken to prevent release of those chemicals prior to receiving any shipment.

The Crystal River Energy Complex generating units utilize various hazardous chemicals for the treatment of the plants' cooling water systems and for the fossil units' flue gas conditioning. Additionally, the design of the Crystal River site helper cooling towers on the north bank of the discharge canal includes the occasional storage and use of chlorine and sulfur dioxide for bio-fouling control while the towers are in operation. Storage of toxic chemicals at the helper cooling towers is administratively controlled via locked valves and volume limits.

The presence of industry, transportation, or operations in the vicinity of the site does not pose any potentially significant effects on the safe operation of the nuclear facility. Since CR3 has permanently shutdown and removed spent fuel from the reactor, the risks of industry, transportation, or operations in the vicinity of CR3 are significantly reduced. The following sections address specific analyses to establish that the nuclear facility is designed to withstand safely the effects of potential accidents as a result of the presence of other facilities.

1. There is no significant manufacturing, storage or transportation of hazardous material in the 10 mile zone other than associated with the operation of the Crystal River Generating Complex and the transportation of hazardous material on public highways or railroads within the 10 mile zone.
2. A transportation accident offsite of toxic gas or other hazardous material does not provide significant potential for effects due to the four mile distance from the plant to the nearest road.
3. A natural gas pipeline crosses Route 19 (west to east side) north of the access road and approximately four miles (minimum) from CR3. Failure of this pipeline does not provide significant potential for effects due to the four mile distance and its interface with the transmission right-of-way outside the five mile distance from the plant.
4. A natural gas pipeline crosses Route 19 just north of the access road to supply gas to the Citrus County Combined Cycle Cogeneration Plant. The line terminates at the plant outside the Crystal River Owner Controlled area.
5. The Crystal River Quarry is a dolomite quarry located over four miles from the plant. The State of Florida Fire Marshall, the Internal Revenue Service, and county agencies have jurisdiction over the utilization of explosives. The Crystal River Quarry does not store explosives. The maximum amount of explosives used by the quarry is equivalent to 1,000 pounds of TNT. No hazardous effect is postulated from the use of explosives by the Crystal River Quarry, nor has any explosive shock from the blasting ever been observed at the plant site.
6. Due to the extent of the plant exclusion zone and the four mile distance of the closest approach to the plant, offsite hazardous material does not pose significant possibility for affecting plant safety.

2.2.4 POPULATION

The nearest population center of 25,000 or more is Ocala which is located 36 miles ENE of the site. Population centers of 25,000 or more within 100 miles of the site are shown in Figure 2-5 with their corresponding 1973 population estimates. For the purpose of 10 CFR 100, and the dose analyses of 10 CFR 50.67, the calculated low population distance is five miles; however, the distance based on actual population density is 41 miles. For the purpose of this application, the low population zone will be arbitrarily extended to a five mile radius.

The distribution of 1974 population within five miles is mainly in the NNW to NNE and in ESE to SSE quadrants with no known residents within a 3-½ mile radius as shown in Figure 2-6 Sheet 1. Population projections to the year 2020 are also shown in Figure 2-6 Sheet 1.

Similarly, the population distribution within a 50 mile radius for 1974 and projected for 1980, 1990, 2000, 2010, and 2020 is shown in Figure 2-6 Sheet 3.

Within the 5 mile low population zone, the only transient population are those persons who have business on or near the site or who are traveling on U.S. Highway 19 located approximately four miles east of the site.

2.2.5 MAJOR TRANSPORTATION ROUTES, WATERWAYS, AND AIRPORTS

The only major road within the five-mile radius is U.S. Highway 19, a four-lane divided highway through Citrus County. U.S. Highway 19 links St. Petersburg with Tallahassee. A railroad spur off the Seaboard Coast Line Railroad serves the DEF site. There are no airports within the five-mile limit. Boat landings are few, small, and scattered, although the Intercoastal Waterway passes within 10 miles of the site. A canal has been constructed between the Gulf and the DEF plant site for delivery of coal by barges and intake of cooling water.

No new major highways are expected to pass through the five-mile radius area. However, there will be an increase in local roads as new subdivisions and mobile home courts are constructed in the southern part of the area. If the Barge Canal is completed, canal-oriented roads and marinas can be anticipated.

The major waterways in the five-mile radius area are:

a. Crystal River Entrance Channel
b. Cross Florida Barge Canal (Only a western section has been constructed)

The only significant increase or change in the waterways system is the anticipated Intercoastal Waterway which is presently subject to environmental and economical study.

Presently, there is only one airfield and no known missile bases in the area. The airfield is located in a southeasterly direction about eight miles from the plant site. At present, there are no known plans to rebuild any airports within the five mile radius of the plant.

The present airfield has one turf runway, 2,666 feet in length, oriented in a north-south direction and one paved runway, 4,557 feet in length, oriented in an east-west direction. This field has runway lights, a rotating beacon, and a lighted wind-sox for night landings. The east-west runway is the one used by most small craft landing and taking off.

No records are currently being kept on airfield activity, thus, no data relating to aircraft type or takeoffs and landings is available.

2.3 METEOROLOGY

2.3.1 METEOROLOGICAL PROGRAM

2.3.1.1 Data Sources

Data acquired by the National Weather Service (NOAA) and summarized by the Environmental Data Service (NOAA) were utilized to determine the regional climatology pertinent to the Crystal River site.

The general climate of the region has been defined incorporating appropriate discussions from "Climates of the States - Florida" (Reference 1) and "Local Climatological Data (LCD), Annual Summary - Tampa, Florida 1975" (Reference 2).

Severe weather data were obtained from a variety of sources. Extreme wind data were obtained from studies discussed by Thom (Reference 3) and Huss (Reference 4). Tropical storm, hurricane, and tornado data were generally obtained from monthly Storm Data publications as compiled by the Environmental Data Service (EDS) (Reference 5).

Data for meteorological extremes were obtained from the 1972 LCD for Tampa and from selected substations summaries (published by EDS). These substation data were updated through 1972 from Climatological Data - Florida for: Gainesville, Cross City, Cedar Key, and Yankeetown (References 2, 6, 7, 8, 9).

Climatological data for restrictive dilution conditions were obtained from mixing level data (based on evaluation of 10 years of radiosonde upper air data) summarized by Holzworth (Reference 10). Low level inversion data compiled from radiosonde data by Hosler (Reference 11) were also utilized. Episodes of poor dilution conditions associated with stagnating anticyclones as compiled by Korshover (Reference 12) for a 30-year period were reported.

2.3.1.2 General Climate

The climate of the region around the Crystal River site is humid subtropical, which is characterized by relatively dry winters and rainy summers, a high annual percentage of sunshine, a long growing season, and high humidity. The terrain is generally flat and featureless with the Gulf of Mexico being the major climatic influence. Snowfall is virtually non-existent, but rainfall averages about 50-60 inches per year, with more than 50% of the total rainfall occurring during the months of June through September associated with thunderstorms. Temperatures in the site region (modified by the waters of the Gulf of Mexico) seldom exceed 90°F or fall below 32°F. Fog has a high frequency of occurrence at night during the winter season. Prevailing winds are somewhat erratic since the coastal regions experience frequent local circulations caused by the land-sea breeze. The coastal location of the site also results in vulnerability to tropical storms and hurricanes. In addition, tornadoes occur quite frequently in this region.

2.3.1.3 Severe Weather

2.3.1.3.1 Extreme Winds

According to Thom (Reference 3) the extreme mile wind at 30 feet above ground expected to occur once in 100 years is 110 mph; the expected 50-year extreme mile wind is 90 mph; the expected 25-year extreme mile wind is 80 mph. (The extreme mile wind speed is defined as the one-mile passage of wind with the highest speed for the day.) Based on a gustiness factor of 1.3, according to Huss (Reference 4), the extreme gust expected once in 100 years is 143 mph.

2.3.1.3.2 Tropical Storms and Hurricanes

Since 1871, when more complete weather record keeping commenced, through 1972, a total of 58 tropical storms or hurricane centers have passed within fifty miles of the Crystal River site. After 1885, weather records differentiated between tropical storms (winds less than 73 mph) and hurricanes (winds more than 73 mph). From 1886 through 1972, there have been 46 passages of tropical storms of which a maximum of 13 hurricanes were experienced within 50 miles of the site. Of these, the most destructive was probably the hurricane of October 19, 1944. However, relatively few storms have moved inland on Florida's west coast between Cedar Key and Fort Myers in the past 82 years. Most tropical storms have a tendency to recurve north and northeast off the Florida east coast, move northward parallel to the west coast, or move on a northwesterly course across the Gulf. The highest frequency of tropical storms in the site area occurs in October, with September being the month of second highest frequency.

2.3.1.3.3 Tornadoes

In the period of 1948 through 1958, more than 50% of the waterspouts reported throughout the coastal states of the United States were reported in Florida (Reference 13). Of the 1,264 reported occurrences in Florida from 1948 through 1972, 575 of these were observed on Florida's west coast. Waterspouts have occasionally caused considerable damage to shipping and have also, when crossing from water to land, become destructive tornadoes.

In the period of 1916 through 1972, a total of 776 tornadoes were reported in the state of Florida. Approximately 81 of these tornadoes were associated with the passage of tropical storms. According to statistics compiled by Thom (Reference 14), the highest frequencies of tornado occurrences are in the one-degree squares along Florida's southeast coastline, and also in the one-degree square south of Tampa. In the years 1953 through 1972, there were 20 tornadoes reported in the one-degree square in which the site is located, yielding a mean frequency of 1.0 tornado per year.

According to statistical methods proposed by Thom (Reference 14); the probability of a tornado striking a point within a given area may be estimated as follows:

$$P = \frac{\bar{z}\bar{t}}{A} \quad (2.3\text{-}1)$$

where:

P	=	the mean probability per year
$\bar{z}$	=	the geometric mean tornado path area, sq. miles
$\bar{t}$	=	the mean number of tornadoes per year
A	=	the area of concern, sq. miles

The value of t is 1.0 for the sector in which the site is located. An average path area is 2.82 mi^2, as reported by Thom for Midwest tornadoes, was conservatively used as a value for z. Using a value of A, equivalent to the total land area of the sector in which Crystal River is located, yields:

$$P = 9.0 \times 10^{-4} \text{ / year}$$

The mean recurrence interval, R = 1/P, of a tornado striking a point in any year in the 1° square in which the site is located is 1,111 years.

However, in the period from 1916 through 1972, only two tornadoes were reported in Citrus County, Florida, in which the site is located (equivalent to a probability P = 1.8 x 10^{-4} /year) and only six were reported in adjacent counties.

2.3.1.3.4 Thunderstorms

Local thunderstorms are quite frequent during the summer season. An average of more than 80 thunderstorms occur each year based on Tampa data for a 26-year period of record (January 1, 1947 - December 31, 1972). These storms are usually of short duration (1-2 hours) and occur during the afternoon. However, occasional severe thunderstorms occur associated with high winds and hail that may inflict local property damage.

There were no ice storms reported based on Tampa data for the period January 1, 1968 - December 31, 1972.

2.3.1.3.5 Meteorological Extremes

The extreme 24-hour precipitation of 38.7 inches occurred at Yankeetown during the passage of Hurricane "Easy" on September 4-6, 1950 when the storm center executed two complete loops just off the west coast of Citrus County.

2.3.1.3.6 Restrictive Dilution Conditions

Low-level inversions (the increase of temperature with height) or isothermal layers based at or below a 500 feet elevation occur approximately 32% of the total hours on an annual basis in the site region according to Hosler (Reference 11). However, these values range from 30% during the spring, 25% in summer, 37% in the winter, and 40% in the fall. These inversion conditions usually occur nocturnally, associated with cooling of the land surface and the associated cooling of the lower atmospheric layer.

The Mean Maximum Mixing Depth (MMMD) is another restriction to atmospheric dilution. The mixing layer develops throughout the day as the nocturnal inversion dissipates due to heating of the land surface and turbulent mixing within the lower atmosphere. The annual afternoon MMMD for the site region is approximately 1,400m according to Holzworth (Reference 10). Seasonal MMMD values are 1,600m (spring and summer), 1,400m (fall), and 1,100m (winter).

Stagnation episodes (periods of four days or more with persistent low wind speeds which do not exceed 8 mph, associated with poor dispersion conditions and limited mixing depths east of the Rockies for the period of 1936 to 1965) have been documented by Korshover (Reference 12). There were approximately 60 stagnation episodes in the site region involving a total of 280 days during the 30 year study period. The maximum frequency of stagnation periods occur during the spring. There were seven episodes with stagnation conditions persisting for seven or more days during 1936 to 1965.

2.3.2 LOCAL METEOROLOGY

2.3.2.1 Data Sources

The onsite meteorological data presented in this Section are for the period January 1, 1975 - December 31, 1975. The data were taken from a 195 feet guyed tower that was located approximately 2,200 feet southwest of Crystal River Unit 3.

Meteorological data are also presented for the National Weather Service (NWS) station located at the Tampa International Airport, Florida. The data are for the 10-year period January 1, 1966 - December 31, 1975, and the 1-year data period January 1, 1975 - December 31, 1975; they were obtained on magnetic tape from the National

Climatic Center. Climatological means and extremes based on local climatological summaries were also presented for the NWS stations at Tampa, Gainesville, Cross City, and Cedar Key.

2.3.2.2 Normal and Extreme Values of Meteorological Parameters

2.3.2.2.1 Average Wind Direction and Wind Speed

Monthly and annual wind roses with associated average wind speeds for the 33 and 175 foot levels, derived from onsite measurements for the period January 1, 1975 - December 31, 1975, were developed. The prevailing winds on an annual basis were from the east-northeast and east for both the 33 and 175 foot levels. These data are in good agreement with onsite wind data measured for the period January 1, 1972 - December 31, 1972.

The annual wind rose based on 1-year onsite data also compares favorably with the l0-year wind rose for Tampa.

For the period January 1, 1975 - December 31, 1975 the average wind speed for the onsite 33 and 175 foot levels was 7.9 and 11.6 mph, respectively. This compares to 7.1 mph for the 35 foot level and 10.4 mph for the 150 foot level for the period January 1, 1972 - December 31, 1972.

Calms are defined for onsite wind data as an average wind speed of less than 0.6 mph. The frequency of calms on an annual basis was 0.1% at the 33 and 175 foot levels for the period January 1, 1975 - December 31, 1975. The frequency of calms based on concurrent NWS station data for Tampa was 5.6%. The higher frequency of calms at Tampa is attributed to the higher starting threshold (2.0-2.5 mph) of the NWS wind speed sensor.

2.3.2.2.2 Wind Direction Persistence

Wind persistence is defined as a continuous air flow from a given direction or range of directions.

The maximum wind direction persistence episode for the 33 foot level was 19 hours from the north associated primarily with neutral stability conditions based on $\Delta T_{175ft\text{-}33ft}$. The maximum wind direction persistence episode for the 175 foot level was 29 hours from the east associated primarily with extremely unstable and slightly stable atmospheric conditions based on $\Delta T_{175ft\text{-}33ft}$. No persistence episode longer than 1 hour was associated with calm conditions at the 33 or 175 foot levels.

2.3.2.2.3 Atmospheric Stability

Stability classifications based on the standard deviation of wind direction fluctuations ($\sigma\theta$) and on temperature lapse rates (ΔT). Both classifications are taken from Regulatory Guide 1.23 (Reference 15).

2.3.2.2.4 Natural Fog Occurrence

The fog occurring in the Crystal River site region is predominantly of the advection type, either sea fog or tropical air fog, Byers (Reference16).

2.3.2.2.5 Ambient Temperature

The monthly mean temperatures recorded onsite are in good agreement with the average monthly temperature based on climatic data for Tampa for a 30-year period. The annual average temperature at the 33 foot level is 69.8°F. The highest temperature recorded onsite was 92.0°F; the lowest recorded was 27.5°F for the report period January 1, 1975 through December 31, 1975.

2.3.2.2.6 Precipitation

The total precipitation for the period January 1, 1975 - December 31, 1975 was 42.44 inches. The total precipitation at Tampa during the concurrent data period was 43.44 inches and is slightly less than the normal of 49.38 inches for Tampa. The maximum amount of precipitation recorded onsite during 1975 for a 1-hour period was 2.15 inches, and the maximum amount of precipitation recorded onsite for a 24-hour period was 3.23 inches. There were no cases where the onsite extreme values exceeded the maximum recorded rainfall in the site region. Precipitation in the form of snow or ice is very infrequent in the site region.

2.3.2.3 Topographical Description

The topography in the area around the site is extremely flat and featureless. The only topographical feature that could affect meteorological conditions at the site is the close proximity of the plant site to the Gulf of Mexico. This results in a local circulation known as the land-sea breeze effect. The effects of onshore offshore air flow are discussed in Reference 17.

Topographic profiles for each of 16 directions out to a distance of 5 miles were developed. These profiles indicate that the maximum difference in relief within a 5-mile radius of the plant is only 20 feet.

2.3.3 ONSITE METEOROLOGICAL MEASUREMENT PROGRAM

2.3.3.1 Meteorological Input

Crystal River has the capability to receive meteorological data from established local weather services, such as Local News and Weather television and radio stations, including the Internet. CR3 has reduced the risk of a credible accident now that it has entered decommissioning. There is no credible accident that can result in a radiological release approaching EPA Protective Action Guideline Limits off-site. Therefore, methods for assessing an off-site release are no longer warranted. Meteorological methods that provide local wind direction and wind speed data are adequate to protect on-site workers and members of the general public that may be on site. A fixed stability class representative of the site meteorology is pre-calculated.

Meteorological data can also be acquired from the two closest meteorological stations - Gainesville, Florida and Ruskin, Florida.

2.3.4 SHORT-TERM (ACCIDENT) DIFFUSION ESTIMATES

2.3.4.1 Objective

Onsite data from the Crystal River site were used to compute atmospheric diffusion factors (χ/Q) for accidental releases to the atmosphere. Calculations were performed for three receptors - the control room, the exclusion area boundary (EAB) and the low population zone (LPZ).

2.3.4.2 Models

Software program ARCON96 (Reference 52 and Reference 53) was used to estimate the control room χ/Q, and program PAVAN (Reference 54) was used to estimate χ/Qs for the exclusion area boundary and low population zone. Onsite data for the period January 1, 2003 - December 31, 2007 was used as inputs for both codes (Reference 55).

In developing the control room χ/Q, for a release from the Auxiliary Building Stack, multiple sensitivity runs were performed using various release height and receptor height combinations, and sensitivity runs were also done using

meteorological data for each year in comparison to runs using the entire 5 year dataset (Reference 56). The most limiting χ/Q from among these runs was chosen.

In developing χ/Qs for off-site releases an EAB distance of 1340 meters was used and an LPZ distance of 8047 meters was used. The EAB lies well within the site boundary for most of the 16 sectors.

An additional set of EAB and LPZ χ/Qs was developed for use with events which might occur close to containment but outside of the power block. An EAB distance of 1200 meters was used (Reference 56).

2.3.5 LONG-TERM (ROUTINE) DIFFUSION ESTIMATES

2.3.5.1 Objective

Estimates of the annual average relative concentration factors (χ/Q) were calculated for the Crystal River site based on site data collected during the period January 1, 1975 - December 31, 1975. The calculations show the maximum annual average χ/Q value of 2.5 x 10^{-6} sec/m^3 would occur in the sector to the northwest of the point of release.

2.3.5.2 Model

Annual average atmospheric dilution factors χ/Q were calculated using Equation 2.3-4 which is consistent with the guidance contained in Regulatory Guide 1.111 for determining χ/Q values for ground-level releases (Reference 23). Stability based on $\Delta T_{175ft\text{-}33ft}$ and 33 foot wind speed and wind direction data were used for the calculations. Calms were distributed based on the directional frequency of winds in the 0.6 to 1.5 mph range within the stability class associated with the calm and were assigned a wind speed of 0.25 mph.

$$\left(\frac{\chi}{Q}\right)_k = 2.032 \sum_{i=1}^{7} \sum_{j=1}^{7} \frac{n_{ijk}}{N \text{ x } \bar{u}_i \, S_{zj}(x)} \qquad (2.3\text{-}4)$$

where:

$(\chi/Q)_k$	=	concentration (χ) normalized to source strength (Q) for sector k, seconds per cubic meter
$S_{zj}(x)$	=	effective vertical dispersion parameter for stability class j at distance x, meters
n_{ijk}	=	the number of observations for joint occurrence of the i^{th} wind speed class, j^{th} stability class, and k^{th} wind direction
N	=	total of observations
$\bar{u}_i$	=	midpoint of the i^{th} wind speed class, meters per second
x	=	downwind distance, meters

An effective dispersion parameter $[S_{zj}(x)]$ is used to account for building wake effects as follows (Reference 23):

$$S_{zj}(x) = \left(\sigma_{zj}^2(x) + \frac{CH^2}{\pi}\right)^{1/2} \qquad (2.3-5)$$

with the constraint:

$$S_{zj}(x) \leq \sqrt{3}\,\sigma_{zj}(x)$$

Where:

$\sigma_{zj}(x)$	=	vertical dispersion parameter for stability class i at distance x, meters
C	=	building shape factor (0.5), dimensionless
H	=	the height of the containment, meters

The value of H in Equation 2.3-5 is 58 meters. The values of $\sigma_{zj}(x)$ are based on Reference 23.

Value for D/Q were calculated in accordance with NRC Regulatory Guide 1.111 (Reference 23) and were based on the same data set as the χ/Q analysis.

Since the straight-line airflow model does not consider the effects of spatial variations in the airflow of the site region, the χ/Q and D/Q values were adjusted as outlined in Regulatory Guide 1.111 by multiplying the right side of Equation 2.3-4 by the appropriate correction factors for open terrain. Open terrain correction factors from Regulatory Guide 1.111 were considered appropriate, since the Crystal River site is located on level terrain approximately 4,000 feet from the Gulf of Mexico. Figure 2-2 shows the general topography for a 50 mile radius, indicating there are no significant terrain characteristics that would affect onshore flow. The effects of onshore-offshore flow of air are discussed in NUS-1721 (Reference 17) and are not considered significant for onshore airflows.

2.4 HYDROLOGY

2.4.1 CHARACTERISTICS OF STREAMS IN VICINITY OF THE SITE

The major streams in the general vicinity of the site are the Withlacoochee River and the Crystal River. Withlacoochee River is the major stream, having a drainage area at its entrance into the Gulf of Mexico of approximately 2,000 square miles. The discharge of the Withlacoochee due to rain runoff is augmented by a base flow of groundwater runoff and artesian spring discharges. Crystal River is much smaller than Withlacoochee River, with its major discharge consisting of artesian spring discharges.

The plant site is located approximately 3.8 miles south of the mouth of the Withlacoochee and about the same distance north of the mouth of the Crystal River. The Cross-Florida Barge Canal which intersects with the Withlacoochee River inland meets the Gulf about one mile southeast of the mouth of the Withlacoochee River and two miles northwest of the site. The average flow from the Withlacoochee drainage basin, a portion of which enters the Gulf via the Cross-Florida Barge Canal, is approximately 1,820 cfs, with a maximum and minimum flow of 9,130 cfs and 830 cfs, respectively. The average flow of the Crystal River is approximately 600 cfs. The natural stream flows in the vicinity of the plant site have a high mineral content.

The Withlacoochee River is regular in its flow characteristics, as evidenced by the extremely flat slope of the Flow-Duration Curve, shown as Figure 2-8, and by the following summary of recorded data taken at the gauging station near Holder, where the drainage area is 1,710 square miles.

Minimum daily flow	113 cfs
Average annual discharge	1,170 cfs
Average runoff per square mile	0.68 cfs
Mean annual flood	3,500 cfs
Maximum flood of record	8,640 cfs

The flat slope of the Flow-Duration Curve illustrates the high base runoff from storage in lakes, ponds, swamps, and spring discharges from underground aquifers.

Considering the magnitudes of stream flows involved, both on a low-flow and flood basis, the streams in the vicinity can have little, if any, effect upon the site. Figure 2-9 shows the flood frequency curve, and Figure 2-10 shows the mean annual average daily discharge for the Withlacoochee River near Holder.

The present uses of the streams in the vicinity are for small commercial craft and barges, pleasure boating, commercial and sport fishing, and for other recreational purposes.

Numerous springs, lakes, and ponds exist in this section of Florida. The primary uses of these are fresh water sport fishing and water supply for cattle. Figure 2-11 shows the location of the major lakes and ponds within a 50-mile radius of the site. Figure 2-12 contains a tabulation of data for these lakes and ponds.

Water for all public supplies in the area of Crystal River, and most of the water used by municipalities and industries in the area of the Withlacoochee River are obtained from wells drilled into the Floridian Aquifer. Figure 2-13 and Figure 2-14 shows the location of public water supplies within a 20-mile radius of the site, none of which are affected by site originated water.

Based on information received from the Southwest Florida Water Management District and on Potentiometric Surface, Florida Aquifer-Atlas HA-440, groundwater has an extremely low probability of being pumped from beneath the plant site, even under adverse conditions. Such adverse conditions based on variable rainfall and groundwater conditions could include some localized flow toward the east. The groundwater flows in the Crystal River area generally are to the southwest as seen in Figure 2-15. This figure clearly indicates that, assuming groundwater flow perpendicular to the potentiometric contours, the groundwater flow is to the west and southwest in the site area and is directed out toward the Gulf and away from any potential users of groundwater. This fact is further confirmed by the three test wells near the entrance road and monitored by the U.S. Geological Survey that the water level slopes downward toward the Gulf at a rate of two feet per mile on an east-west line. The salt-water intrusion line is approximately three miles east of the site. There are no wells within several miles of the site. These conditions, coupled with the evidence shown in the test wells on the entrance road, indicate that there is essentially no eastward flow of groundwater (or variation in chloride content) beyond about two miles east of the plant. This gives assurance that there is essentially no conceivable means that any potential groundwater users will be able to use existing groundwater located beneath the plant site. The nature of the site environs gives assurance that future groundwater users will not be affected.

2.4.2 FLOOD STUDIES AND HURRICANE EFFECTS

The plant site is located on the Gulf Coast of Florida, immediately adjacent to the bulkhead line, and is within an area exposed to maximum intensity hurricanes. The most extensive flooding to be expected at the site will be produced by abnormally high tides resulting from hurricanes blowing onshore, and augmented by the forces and run-up characteristics of waves occurring at the height of the hurricane tide.

Class I equipment and structures, which are required to protect the health and safety of the public, are designed to withstand the effects of a Probable Maximum Hurricane (PMH). The characteristics and effects of the PMH were computed from procedures outlined in the following publications:

a. Memoranda HUR 7-97 and 7-97A, prepared by the Hydrometeorological Branch of the National Oceanic and Atmospheric Administration (NOAA).

b. Technical Memorandum No. 35, Storm Surge on the Open Coast: Fundamentals and Simplified Prediction, prepared by the U.S. Army Corps of Engineers, Coastal Engineering Research Center.

c. Estimation of Hurricane Surge Hydrographs, by G. Marinos and J. W. Woodward, ASCE Paper 5945, May 1968, WW2.

d. Technical Report No. 4, 3rd Edition, Shore Protection Planning and Design, prepared by U.S. Army Coastal Engineering Research Center.

The general concept for hurricane protection included the following considerations:

a. Full protection against hurricane tides and wave action for all components which must operate for a safe and orderly shutdown of the nuclear unit. This protection will be provided for any intensity hurricane up to and including the Probable Maximum Hurricane (PMH). Due to the decommissioning status of CR3, almost all Systems, Structures, and Components (SSC) have been reclassified and are no longer considered as Class I.

Class III non-safety related SSCs are designed to flood levels based on FEMA's Flood Insurance Rate Map (FIRM). Designing to FIRM levels complies with Citrus County and Florida Building Codes. Derivation of the flood level is contained in EC 299162.

2.4.2.1 Maximum Hurricane Surge Level

The critical approach path for a hurricane with onshore winds is from the southwest, tracking on a northeasterly course, as shown on Figure 2-21. However, the approach path of wave trains that will produce the maximum run-up levels at the site is along a north-south section with waves approaching the plant from the south, as shown on Figure 2-16. The maximum tidal set-up, however, must be produced by winds blowing onshore to the east. This concept of maximum wave action occurring perpendicular to the hurricane winds is considered to be a conservative assumption for the maximum set-up condition.

The path of approach for a PMH to produce maximum surge heights is from the southwest travelling forward toward the site in a N 63° E (True) direction. The center of the PMH would pass north of the plant site at a distance that results in the maximum winds passing directly over the site area. Surge calculations were calculated along a traverse line intersecting the site at a bearing of N 63° E. Since this traverse is approximately normal to the offshore bottom contours, the surge computations based on the bathystrophic storm tide theory gave maximum surge heights.

The primary parameters for the PMH, as a result of maximizing to obtain the highest surge level, are as follows:

Central Pressure Index (CPI)	26.70 in. Hg
Asymptotic Pressure	31.25 in. Hg
Radius of Maximum Wind (R)	24 nautical miles
Forward Translational Speed (T)	20 knots
Maximum Wind Speed (V_x)	149 mph
Astronomical Tide	4.3 feet

The maximum storm tide level computed from the above parameters by the AEC and presented in their letter of October 12, 1973 to Florida Power Corporation is at elevation 121.4 feet (MLW is at elevation 88.0). This determination considered the topography of the site along the approach path and the critical section shown on Figure 2-17. The maximum storm tide level also considered a two feet reduction due to the effects of backwater

storage resulting from the extensive flooding of the surrounding countryside some five hours prior to the peak of the surge hydrograph and run-off into peripheral areas not directly affected by the hurricane surge. The site in this extreme circumstance becomes analogous to a small island in a mountain of water. The 30-foot land contour (based on the USGS datum where sea level is 0 feet) is located about six nautical miles inland from the site. For a description of this site see Reference 46 "Report - Verification Study of Dames and Moore's Hurricane Storm Surge Model with Application to Crystal River Unit No. 3 Nuclear Plant - Crystal River, Florida - for Florida Power Corporation."

For Class III SSCs, the flood level is shown on FEMA's Flood Insurance Rate Maps. For CR3, the maximum flood height is EL 107' (plant datum), as evaluated in EC 299162.

2.4.2.2 Hurricane Wave Action and Run-up

Figure 2-18 summarizes the relationships between the stillwater level, wind-generated wave height, breaking wave height, and wave run-up. The maximum (highest one percent of the waves) and significant (average of the highest one-third) wave heights were determined by wind vectors normal to the coast along the traverse, and were calculated using the storm surge computer output. Figure 2-20 presents a summary of this analysis for wind-generated waves approaching the site. On Figure 2-18, the intersection of the breaking wave curve and the generated wave height curves show that with the highest one percent of the waves breaking, the maximum height of the waves that can travel across the fill approaching the plant is 18.0 feet. With the average of the highest 33% of the waves breaking, the maximum height of the waves that can reach the protective embankment without breaking is 15.5 feet (Reference 46).

Maximum tidal set-up will be produced by winds blowing onshore along a traverse normal to the offshore bottom contours. Consequently, the critical approach path for a hurricane is from the southwest, tracking on a northeasterly course. However, the approach path of wave trains that will produce maximum run-up at the site is along a north-south section with the waves approaching the plant from the south. This critical traverse of the wave train is across a reach of natural ground about one mile wide, then over 600 feet of compacted fill (elevation 98), and against an embankment slope (berm) rising to a top elevation of 118.5 feet, which protects the plant. This concept of maximum wave action occurring perpendicular to the hurricane winds is considered to be a conservative assumption for this already extremely severe condition.

The effects of breaking waves and wave run-up on the embankment slope were evaluated from the model tests previously conducted at the University of Florida. A detailed report study is included in Reference 33. The section tested on the approach path includes, progressively: a storage area at elevation 98, the intake canal with bottom elevation 67, the intake structure at top elevation 100, a 30 foot wide embankment behind the intake at top elevation 98, an embankment sloped at 2:1 from elevation 98 to 112.5 to a 46.5 foot wide berm at this elevation, then a 2:1 embankment to elevation 118.5 which is grade elevation around the plant. The main building is set back about 100 feet from the top edge of the embankment. Since the model accurately represents the characteristics of topography, structures, and the protective embankment, the test results include the local effects unique to this site which are not reflected in generalized analytical relationships, but which increase in importance as the waves approach the plant site.

Before performing the run-up tests, experiments were conducted to determine the most adverse test conditions (i.e., the combination of wave period and height which caused the maximum run-up over the total range of interest). From these pre-test experiments and periodic checks during the tests, it was found that the maximum run-up occurred with a prototype wave period of 5.4 seconds, and prototype wave amplitudes of 10 to 15 feet, the specific height depending on the test case. The wave action testing was conducted at prototype tide levels from elevation 104 to 120 feet in two-feet increments. Unlike the spectrum of wind-generated waves, the model tests were conducted with waves essentially uniform in height.

Initial tests were conducted on a smooth slope embankment. Subsequent tests simulated the run-up effects against the stepped embankment. The test results indicated no overtopping of the smooth slope below tide levels of

110 feet, with occasional overtopping starting at a tide level of 112 feet and becoming continuous above elevation 114. Tests on the stepped slope revealed no overtopping below tide elevation 112 feet, slight overtopping at 114 feet, and more continuous overtopping above elevation 116 feet. Pertinent results of the model tests are shown on Figure 2 and Figure 3 and on Table 1, Table 2 and Table 3 in Reference 47.

The applicability of the model tests for the 33.4 surge level is illustrated by Figure 3 and amplified by two letters from Dr. R. G. Dean dated April 3, 1972, and April 11, 1973, contained in the Attachments in Reference 47. As indicated by Figure 2-19, the median and maximum run-up elevations occurring at a tide level of 121.4 feet are 126.0 and 127.0, respectively, on Profile 5, the profile corresponding to the stepped slope that is constructed at the Crystal River site.

For a slope of unlimited height, the maximum and median run-up corresponding to the indicated stillwater hydrograph were developed using Figure 2-19 and are shown in Figure 2-18. When the height of the wind-generated waves reaching the protective embankment becomes 10 to 15 feet (the range of wave heights found from the model tests to cause the greatest run-up), the results of the model tests become applicable. Until that time, the wind-generated waves would produce less run-up than indicated; the run-up from the test results is therefore shown as a dotted line. Employing the conservative assumption that the wave height in the model tests was the "maximum" generated wave height (i.e., assuming that all of the waves attacking the embankment are "maximum" waves), Figure 2-18 shows that the model results become applicable at 22.0 hours after the center of the hurricane crosses the continental shelf. This is also the time of maximum stillwater level.

Hurricane wave action and run-up for the Emergency Feedwater Pump (EFP-3) Building are evaluated separately in DEF Calculation S98-0312, Revision 0, "CR3 Determination of Wave Effects on EFP-3 Building".

For Class III SSCs, the flood level evaluated in EC 299162 is based on FEMA's Flood Insurance Rate Map. The base flood elevation shown on the flood maps includes the effects of waves and wave run-up.

2.4.2.3 Minimum Tide Hurricane

Minimum tides and water levels are not critical during SAFSTOR conditions.

2.4.2.4 Facilities Required for Flood Protection

The following equipment (component) is required to remain functional during the postulated hurricane to prevent water intrusion into and out of the plant buildings where radiological materials may exist. These barriers/seals prevent potential transfer of radiological material through unmonitored pathways during a flooding event.

a. Water-tight seals for all underground electrical conduit entering the building below grade.
b. Auxiliary Building Sea Water Room Raw Water pit manhole covers and level test flanges are maintained in accordance with procedure EM-220 (Violent Weather).

Pump Location	Tag Number	Number	Capacity (gpm)	Discharge To
Turbine Room	SDP-1A/1B	2	500	Condensate Demineralizers Regeneration Neutralization Tank
Nuclear Service Area Sump	SDP-2A/2B	2	250	Turbine Room Sump
Condensate Pump Pit	SDP-5A/5B	2	50	Turbine Room Sump
Tendon Access Gallery	SDP-3A/3B	2	50	Nuclear Service Area Sump
Auxiliary Building	WDP-4A/4B	2	125	Misc. Waste Storage Tank

Pump Location	Tag Number	Number	Capacity (gpm)	Discharge To
Reactor Building	WDP-2A/2B	2	100	Misc. Waste Storage Tank
Decay Heat Pit	WDP-3A/3B	2	30	Misc. Waste Storage Tank
Laundry and Shower	WDP-22A/22B	2	40	Laundry/Hot Shower Monitoring Tank
Turbine Room	SEP-1A/1B	2	125	Sewage System
Diesel Generator	SDP-9A/9B	2	20	Turbine Room Sump

In the event hurricane winds are capable of blowing the siding off the turbine building and the associated rain is driven into the turbine building, the pumping system within the building is capable of removing an 8 inch per hour rainfall continuously without accumulation. For the unlikely situation where rainfall may exceed this rate for a short period of time, there remains a reserve sump capacity of about 30,000 gallons.

Outside the structure, the site drainage system has been designed to preclude ponding even during the PMH. The design utilized the Rational Method with the following parameters:

Rainfall intensity	10 inches per hour
Run-off coefficients	0.95 for roof surfaces 0.80 for paved areas 0.40 for soil

The greatest overland distance that run-off must travel to reach a catch basin is only 200 feet, at a minimum ground slope of 0.5%.

Roof drains discharge directly into the Storm Drainage System and are designed to accommodate a rainfall intensity of 6 inches per hour. For this design capacity, no roof ponding will occur up to a 1,000 year rainfall. In the event of the PMH, ponding could occur up to a maximum of three inches around the eaves of the structures, and beyond this level would overflow the eaves. The roof structures are adequately designed to support the ponded water.

2.4.2.4.1 Operational Requirements

Procedures exist to monitor water level of the Gulf and obtain meteorological forecasts when a Hurricane Watch or Hurricane Warning is in effect. [TSA149]

2.4.2.5 Embankment Slope Protection

Due to revised hurricane criteria and analyses, the surge level has changed from an original elevation of 112.6 feet to elevation 121.4 feet. Wave heights have correspondingly increased from an initial height of 11.4 feet to 18.0 feet. To provide greater resistance against the increased wave forces, an armor covering of 3,000 psi reinforced concrete is provided. The reinforced concrete provides resistance to erosion and dynamic impact. 1,500 psi unreinforced concrete is placed at the toe of the embankment and at the top of the embankment to prevent possible undermining of the slope armor.

TSA149 *This paragraph was added when Technical Specification Amendment 149 was issued. Obtain Licensing & Regulatory Affairs concurrence before changing.*

Stability analyses were undertaken for the embankment in order to establish the high degree of stability the embankment possesses against the hypothetical failure along a circular arc passing through both the foundation and the embankment. Using the following strength parameters, the minimum factor of safety against failure is 4.3.

The foundation material upon which the embankment is constructed was placed in 1964 from construction excavations onsite. This material has had nine years to consolidate, with considerable construction activity surcharge. No significant settlement is anticipated from this material or from the Zone III material placed in the embankment under 95% of Maximum Modified Proctor density compaction criteria required by Specification SP-5901 (Reference 34). Any potential settlement would occur prior to the placement of the concrete cover since the full weight of the embankment is imposed on the foundation first during construction.

The general characteristics of Zone III fill are such that when compacted to 95% Maximum Modified Proctor density, as required by Specifications SP-5901, Zone III becomes a very dense and stable material.

The compacted Zone III material has a low permeability of 1×10^{-5} cm/sec and it was noted in conducting triaxial shear tests that difficulty was experienced in attempting to saturate the samples. This characteristic is beneficial in preventing potential uplift forces from developing beneath the surface materials. The embankment mass does not respond quickly to saturation from increasing tide levels that have a duration of only 10 hours since several weeks would be required to totally saturate the mass.

Although soil-cement was not used on this site, the original design of the soil-cement section was based upon the results of its use and documentation in Bonny and Merritt Dams by the Bureau of Reclamation, and also the results of three design-mix programs conducted by Pittsburgh Testing Laboratory.

The use of concrete as protection is quite common. Concrete has been used to pave dams and levee slopes, used as shore protections along coastlines, used as dams to contain lakes and reservoirs, and used as spillways which are subjected to a degree of abuse which could never be equaled in its use as the embankment protection at Crystal River Unit 3.

The most conservative design features of the embankment protection structure involves the expectation that it will never lose its engineered fill base anywhere. However, for the purpose of structural analysis, the assumption was made that an improbable loss of support through wash-out might occur on both sides of expansion joints and along the lower and upper toe lines of the armored structure. A superimposed vertical static load of 32 psi was used as the equivalent of the dynamic forces acting during a PMH. The reinforced concrete structure was designed to sustain this load on freely spanning cantilevers within the working stress range of reinforcing steel and concrete allowed by the ACI-318-63 design code.

Since the chances of a wash-out are unlikely, the armor will always be continuously supported. The only induced stresses, therefore, will be purely compressional. This, then, indicates that the armor is very conservatively designed.

The assumption of improbable loss of support through wash-out, and the superimposed vertical static load of 32 psi during a PMH (as mentioned above), were conservative design features considered for the embankment protection structure as a whole. However, for the addition of the Diesel Driven Emergency Feedwater Pump (EFP-3) Building at the 112.5 feet elevation on the wave steps, the loss of support through wash-out was not considered a credible design condition.

Since the Diesel Driven Emergency Feedwater Pump Building uses an isolated portion of the concrete embankment structure as the building basemat, loads from this building could exceed those considered in the previous embankment analysis. As an alternate justification, geotechnical services and slope stability analyses were performed for the specific wave step panel affected. Results are documented in Reference 50 (FTI Document No. 38-1247584-00, "Geotechnical Engineering Services Report for Diesel Driven Emergency Feedwater Pump Project") and indicate a minimum safety factor of 1.7 for the worst case loading conditions.

2.5 ENGINEERING GEOLOGY AND FOUNDATION CONSIDERATIONS

2.5.1 GENERAL

The general geology and regional tectonics of the Florida peninsula were studied to determine the history of diastrophism and the stability of the plant site as influenced by regional and local tectonic elements. One structural feature, the Ocala Uplift, was disclosed as providing substance for considering the effect it and its related faulting would have upon the structural competence of the site.

Having established the geologic setting, field studies, including exploratory drilling, field and laboratory testing of subsurface elements, a seismic refraction survey, and photogeologic studies were performed to determine the subsurface conditions of the proposed site to establish its capability to support the nuclear powered generating unit.

The site is located on gently southwesterly dipping biogenic carbonate (limestone and dolomite) rocks which have been differentially dissolved along the most pervious zones of the rock, resulting in a network of general vertically oriented dissolved zones (solution channels). The bedrock solution process was studied, and the results are summarized herein. The natural solution process is not considered to present any future threat to the soundness of the rock system.

A detailed study of the location of the solutioning features was not initiated, as the whole of Citrus County is underlain by limestone and other calcareous deposits which have been subjected to solutioning with occasional resultant sinks, vertical tubes and cavities. The closest observable feature of this type is approximately 1,800 feet north of the structures. It is approximately 15 feet in diameter and water filled to about 5 feet from the surface. There is no evidence of general subsurface subsidence and it may be a solution feature more aptly called a "solution pipe." Since no subsurface subsidence was noted to occur at the plant site, it is felt that these features have no effect on the plant foundations.

The site is located in an area where groundwater exists under water table conditions, as opposed to quite general artesian conditions throughout most of the State. The groundwater table occurs at a depth of approximately 10.0 feet below ground surface. Based on a ground datum of 100.0 feet, groundwater levels were recorded to rise approximately 1.5 feet at peaks of high tides. Differential lags in groundwater fluctuations in response to tidal changes indicate variable transmissibilities of the limestone.

The nearest faulting occurs at a distance of three miles to the east of the site. Stratigraphic correlation and continuity of seismic refraction profiles negate the possible existence of subsurface faults at the site.

The seismic study was performed by Weston Geophysical Research, Inc., who conducted a thorough literature search and examined available seismograms to determine the epicentral distribution of earthquakes in the southeastern United States, their focal depth, intensity and/or magnitude, and attenuation characteristics (Reference 36). Regional tectonics and fault patterns were investigated for possible association with known epicenters. This study indicated that the State of Florida is seismically inactive, and that the closest area to the site of significant seismic activity is Charleston, South Carolina. Attenuation data available for this area indicates that the site experienced an observed intensity no higher than Intensity V (Modified Mercalli). The maximum ground motion at the site due to this earthquake probably did not exceed 0.025g. For design purposes the maximum ground acceleration was assumed to be 0.05g. Response spectra were developed for the site normalized for a maximum ground motion of 0.05g and based upon a large earthquake in the Charleston, South Carolina region and moderate earthquakes in Florida.

From the results of site subsurface and regional tectonic investigations it has been concluded that the foundation mass system can competently support the generating plant and that inactive regional tectonic elements present no threat to the structural integrity of the installation.

2.5.2 REGIONAL TECTONICS

A geologic history of the Crystal River area was conducted to outline the general geology of the area, the history of diastrophism, and the influence of these elements upon the structural integrity of the geology at the Crystal River site. The following Sections summarize briefly surface and subsurface geologic elements.

2.5.2.1 Surface Geology

The immediate surface at most places in the State is underlain by Pleistocene deposits, of which two principal kinds are recognized. The most widely distributed is a series of ". . .littoral, sublittoral and estuarine sandy formations corresponding to . . . difference stages of sea level." "The other kind, which underlies the east coast and the southern part of the State is divisible into three contemporaneous marine formations . . ." (Cooke, 1945); these are composed of marine sediments constituted by a coquina, an oolite, and a coral reef facies.

Interspersed throughout the Pleistocene formations are Recent marine sediments along the coasts which are composed of quartz sand with local broken shell admixtures as far south as Cape Romano on the west coast and Miami on the east coast. Windblown sand is added to a white limy ooze which accumulates along the remainder of the coastline. Continental deposits of silt and sand are being deposited in tidal areas and along rivers. Muck and peat deposits are accumulating in shallow lakes, ponds, and swampy areas. "A kind of travertine or caliche locally forms at the surface in southern Florida" (Cooke, 1945).

2.5.2.2 Subsurface Geology

If the surface were denuded of these Recent and Pleistocene deposits, the whole of Florida would be represented by Tertiary Formations, the oldest of which would be the Avon Park Limestone of late Middle Eocene Age (Cooke, 1945). This oldest lithologic unit is exposed in Citrus and Levy Counties just a few miles from the Crystal River site.

All of these Tertiary sedimentary rocks of Peninsular Florida are predominantly allochemical limestones which are in part dolomitized. North of a line drawn between Levy and Nassau Counties the sequence of sedimentary rocks is constituted basically by clastic sediments (Pressler, 1947). The plant site is located in the zone of allochemical carbonates.

2.5.3 SITE GEOLOGY

The existing facilities are located in the NW corner of Section 33, Township 17 S, Range 16 E, Citrus County, Florida. The Duke Energy Florida property adjoins U.S. Route 19/98 near Red Level, approximately 7½ miles NW of Crystal River.

The site is adjacent to the Gulf of Mexico in a former marsh area that was reclaimed for plant site development. The entire area is one of very low relief (originally two to five feet above mean sea level) and is located within the Terraced Coastal Lowlands of the Coastal Plain of West Florida. For the sake of continuity with the major portion of this report, all elevations hereafter mentioned are referenced to the Duke Energy Florida's Plant Datum (mean Gulf low water level equals plant datum 88 feet).

Bedrock at this site is approximately 20 feet beneath the present ground surface which is characterized by surface fills. The thickness of this surface fill is approximately three to five feet in the area studied. The natural soil cover at the site consists of Recent deposits of thinly laminated, organic sandy silts and clays, interspersed with a Pleistocene marine deposit known as the Pamlico Terrace Formation. These deposits blanket the site and have a variable but average thickness of approximately 4 feet. Beneath these soils is the residual limy soil unit derived from the decomposition of the underlying bedrock.

The bedrock examined during this subsurface investigation consists of biogenic carbonates of Tertiary Age. Two distinct Eocene formations were identified during the test boring program. The upper-most member is the Inglis Member of the Moodys Branch Formation. This unit overlies an unconformity consisting of very dense silt, sands, and organic clays of variable thickness which represent a formerly exposed erosional surface known as the Jackson-Claiborne Unconformity. The materials comprising this surface are derived, in part, from reworked residual soils, formed from the underlying sequence of carbonates known as the Avon Park Formation, and from fine sands and organic clays deposited on the formerly exposed surface in a sub-tidal flat environment such as exists today at the present ground surface from the plant site to the coast. The configuration of the unconformity can be represented as an undulatory surface in the area investigated, ranging from an elevation of -10 feet to an elevation of +20 feet.

2.5.3.1 Faulting

Prior to 1951, faulting was not recognized in Florida, but upon Vernon's publication of Bulletin 33 of the Florida Geological Survey, a well-defined system of high angle faulting was mapped along the east side of Citrus and Levy Counties.

The western-most extension of the mapped faulting lies 3 miles east of the plant site. All faulting common to Levy and Citrus Counties is oriented along steeply dipping (greater than 60°) northwest trending fault planes which have produced displacements of 20 to 160 feet.

Supplemented by photogeologic studies, subsurface explorations in the form of test borings and a seismic refraction survey eliminate the possibility that faulting exists beneath the plant site because:

1. Correlation of marker horizons delineated by rock cores negates any inference that vertical displacement exists in the subsurface. Marker beds exist at expected elevations in points of investigation (drill holes).
2. Seismic profiles, likewise, showed no anomalous changes in elevations of velocity layers.

2.5.3.2 Bedrock Solution Studies

The near surface marine carbonate rocks constituting the germane portion of the stratigraphic column of the Crystal River area belong to the Inglis Limestone Member of the Moodys Branch Formation of Jackson Age and the Avon Park Limestone of Claiborne Age. These two lithostromes are separated by a well-marked unconformity. These limy sediments are quite porous and have high interstitial permeabilities which have been secondarily augmented by fracturing in response to the minor diastrophism responsible for the Ocala Uplift. In the vicinity of the plant site, the regional dip of these strata is southwesterly at approximately five feet per mile off of the axis of the Ocala Uplift.

Since these rocks are inherently pervious and are broken by high angle fractures, infiltration and recharge to the groundwater table is rapid. Fresh water entering the underground moves rapidly down gradient (toward the Gulf of Mexico) and attacks limy sediments. The result of this natural process is the destructive alteration of the carbonate rock leaving a labyrinth of channels throughout the rock mass. The purpose of this study was to determine the rate at which this solutioning process takes place and to establish the effect such a deleterious process will have upon the foundation of the generating station during its life.

Well-documented subsurface data obtained from exploration and grouting of the foundation for Crystal River Unit 2 showed that the solutioning process has been most intense in the first 100 feet of section below the existing ground surface. Curtain and consolidation grouting carried out to closure on final order holes spaced on a maximum of 8-foot centers successfully injected 7% of grout over the total volume of rock. This figure represents the volume of voids existing in the subsurface, but does not equal the total volume of solution channels because certain solution channels have been infilled by secondary depositional processes. However, the results of the exploratory drilling and the grout hole drilling indicate that the volume of solution channels probably does not exceed 15% of rock mass.

Based upon accepted geochronology of the area, the age of the solutioned sediments is considered to be 40 million years. Carbon 14 dating of organic material developed on and included in the unconformity between the Inglis and

the Avon Park Members of the Moodys Branch Formation revealed the rock to be older than the limits of the dating process (40,000 years). It is therefore certain that the age of the rock is more than 40,000 years, and it is accepted as being 40 million years old.

Assuming the law of uniformitarianism to be true, the statement can be made that 15% of the rock mass has been dissolved in a period of 40 million years and definitely in more than 40,000 years. On this basis, therefore, the solution rate of the limestone is 15% per 40,000,000 years or approximately 3.75×10^{-7} percent per year. In the life of a 40 year plant, an additional 1.5×10^{-5} percent could be expected to be dissolved.

Considering the most extreme case under this line of reasoning, all solutioning has occurred during the Recent Epoch (the last 10,000 years) or since the Pleistocene Ice Age when base level was established as it essentially is today, the calculations produce the maximum solution rate.

Assuming that only 10,000 years have been required for 15% of the rock mass to dissolve, the solution rate is 1.5×10^{-3} percent per year. In a forty year life of the plant only 6.0×10^{-2} percent of the total volume would be dissolved. Such a small percentage of solutioning is insignificant to the stability of the rock mass.

Based upon verbal communication with Mr. R. D. Cherry (U.S. Geologic Survey), recent studies indicate that within an area of infiltration of 720 square miles, including the area of the plant site, a total of 243 tons per day of solids is being dissolved by the solutioning effect of groundwater.

This represents a total of 764 pounds per day per square mile. Considering that the area of the generating facilities covers approximately 230,400 square feet or 0.0082 square miles, the expected quantity of dissolved solids removed from beneath the plant area daily would be about 6.3 pounds. Then assuming that all of the solutioning will occur in the first 100 feet of depth beneath ground surface and that the unit weight of the limestone is 100 pounds per cubic feet, it follows that 0.063 cubic feet per day or 23 cubic feet per year are dissolved from 23,040,000 cubic feet of rock. This figure represents 1×10^{-4} percent per year.

In the life of a 40 year plant an additional 4×10^{-3} percent could be expected to be dissolved. If one considers the volumes involved, 4×10^{-3} percent represents only 920 cubic feet of material dissolved during the entire life of the plant.

Comparing the figures obtained by the above two methods (excluding the most extreme case), the percent of the rock dissolved over the life of the plant ranges from 1.5×10^{-5} to 4×10^{-3} percent. Dissolved volumes calculated by either method represent insignificant amounts of deleterious action. Further, the grouting process used in the foundation of Crystal River Unit 2 and 3 reduced the permeability of the carbonate rocks from a figure in excess of 65,500 feet per year to less than 2,000 feet per year. With the permeability decreased by more than 30 times, exposure of the limestone to potential solvent groundwater will be effectively reduced by the same factor.

It is concluded that the natural solution process will not affect the structural integrity of the foundation of the power plant.

2.5.3.3 Groundwater

Throughout west central Florida, groundwater occurs under both water table and artesian conditions. Water table conditions occur as shallow aquifers composed of recent sediments and as pervious Tertiary Age limestones. Artesian conditions develop where pervious limestone beds are overlain by effectively impervious layers which are recharged when they intersect the ground surface at points of higher elevation.

The Inglis Limestone Member of the Moodys Branch Formation and the Avon Park limestone are the stratigraphic units which occur at the surface (beneath a thin veneer of Pleistocene and recent sediments) in the vicinity of the plant site. These two stratigraphic units compose a part of the Floridian aquifer which supplies most of the water in the State. Except in Citrus and Levy Counties, these rock units are buried beneath more impervious units creating

artesian conditions. In Citrus and Levy Counties where these units are exposed they contain water which occurs under water table conditions, and these zones occur along the Gulf Coast.

Hydraulic gradients slope seaward and groundwater eventually discharges into the Gulf of Mexico. Surface expressions of artesian conditions exist in the form of springs at numerous places throughout the area. Recharge to the groundwater table occurs as a result of approximately 55 inches of annual rainfall, most of which occurs during the summer months.

Intrusion of salt water exists along the Gulf Coast, which indicates that the chloride concentration should be in excess of 250 ppm at the plant site. Due to the density differences between seawater and fresh water, chloride concentrations can be expected to increase with depth near the coast. According to "Florida Land and Water Resources - Southwest Florida (1966)" at a distance of 10 miles inland, the salt water - fresh water interface occurs at a depth of 300 feet while it occurs near the surface on the coast.

At the plant site proper, groundwater occurs under water table conditions with groundwater levels at or near elevation 90 feet (plant datum), assuming mean low tide to be at elevation 88 feet. Fluctuations of the water levels in the test boring holes were monitored with changing tides to observe the effect of the tides at various points (drill holes) at the plant site.

A tidal rise of about 2.2 feet (to Elev. 92.2 feet) produced an average rise of about 1.5 feet in the groundwater levels in the plant site. A time lag between peaks of high tides and peaks of water levels in drill holes varied from less than one hour to as long as four hours. Average lags ranged between one and two hours. The significance of such empirical observations is that differential attenuation times between drill holes indicate that variation in the transmissibility of the limestone do exist. These variations are attributed to the presence of solution channels.

The permeability of the bedrock has been measured at a value in excess of 65,500 feet/year (limit of test equipment). Laboratory permeability tests which were made on representative core samples obtained from Crystal River Unit 2 give values of primary or interstitial permeabilities of from 7.4×10^{-4} cm per sec. to 3.4×10^{-2} cm per sec. In general, permeability coefficients were slightly higher in a horizontal direction (i.e., parallel to bedding).

Infiltration of surface waters into the groundwater table will occur at a rapid rate by virtue of the 10^{-3} cm per sec. permeability coefficient. Adding the effect of vertical jointing, the permeability is greatly increased. For all practical purposes infiltration of surface water into the ground can be considered to be almost instantaneous.

Chemical analyses of groundwater at the site indicate the water contains more than 350 ppm chlorides with a pH range of 7.0 to 7.1 and a conductivity of greater than 2,000 micromhos/cm.

2.5.4 SEISMOLOGY

The seismicity analysis and response spectra were conducted by Weston Geophysical Research, Inc. (Reference 36). Reverend Daniel Linehan, S.J., Director of Weston Observatory, acted as a consultant on the seismicity analysis. The response spectra was completed by Dr. C. Allin Cornell, Department of Civil Engineering, Massachusetts Institute of Technology.

2.5.4.1 Seismicity Analysis

The State of Florida is an area which is considered seismically inactive. In a 300-year history, only eight earthquakes of Intensity IV (Modified Mercalli) or greater have had their epicenters located within the State. No earthquake is known to have occurred within 50 miles of the plant site. Only one tsunami, or seismic sea wave, has ever been noted along the Gulf Coast of the United States. This wave was caused by the Puerto Rican earthquake of October 11, 1918, and was very small as recorded on the tide gauge at Galveston, Texas. There is no record of a tsunami or seismic sea wave ever having affected the Crystal River area.

The two strongest earthquakes to have affected the site area in north central Florida, were the Northern Florida earthquake of January 12, 1879, listed as Modified Mercalli VI; and the Charleston, South Carolina earthquake of 1885 which had an epicentral Intensity X, Modified Mercalli. There is no evidence that seismic activity in the southern Appalachians or in the Greater Antilles Islands of the West Indies had any effect on the Crystal River site.

An attenuation curve of earthquake intensity with distance for the Atlantic and Gulf Coastal Plains indicates rather slow attenuation of intensity with distance, due apparently to the deep Cretaceous sediment areas of the Coastal Plain regions. Based upon this attenuation data, the Florida earthquake of 1879 would have produced an intensity no higher than the Modified Mercalli IV at the site; and the Charleston earthquake of 1885 would have had an intensity no higher than V at the Crystal River site.

Based upon the relationship between earthquake intensity and ground acceleration given in Nuclear Reactors and Earthquakes TID-7024, United States Atomic Energy Commission, the Charleston, South Carolina earthquake would have resulted in a ground acceleration of approximately 0.025g at the site. A design value of 0.05g, or double the estimated acceleration from the history of the site, is considered conservative.

2.5.4.2 Response Spectra

The complete design response spectra was developed for the maximum ground acceleration of 0.05g, and considered earthquake magnitude and duration, epicentral distance and focal depth, the intervening material through which the seismic waves propagate, and geologic conditions local to the site.

The shapes of the spectra were estimated by two methods, one adjusting the average strong motion spectra for moderate distances computed by Housner, and another by application of a method developed by Estere and Rosenblueth (Reference 36). The resulting design response spectra represent the estimated spectra coinciding with established ground acceleration level of 0.05g, and are shown on Figure 2-22 and Figure 2-23. Seismic design spectra for maximum hypothetical earthquake is shown in Figure 2-24.

2.5.4.3 Seismic Hazard Evaluation

Probabilistic seismic hazard computations were made for nuclear power plant sites in the central and eastern United States. These computations were made for the Seismicity Owners Group (SOG) using the EQHAZARD computer package developed by the SOG and the Electric Power Research Institute (EPRI) and input parameters defined by six earth-science teams who participated in the SOG-EPRI program. Both the methodology and the input to the methodology were reviewed by the USNRC and were found acceptable (NRC letter dated September 20, 1988 to Seismicity Owners Group) for calculating probabilistic seismic hazard at nuclear power plants in the central and eastern U.S.

2.5.4.3.1 Seismic Hazard Results

Six descriptors of ground motion were calculated: Peak Ground Acceleration (PGA) and spectral velocity (pseudo-relative velocity) for 5% of critical damping at 25 Hz, 10 Hz, 5 Hz, 2.5 Hz, and 1 Hz. These ground motion estimates were used to construct uniform hazard spectra. Table 2-4 shows the annual probability of exceedance for PGA for ground acceleration ranging from 0.005g to 1.0g at CR3. The table shows the mean probability and 15%, 50% and 85% confidence level probabilities of exceedance.

These results are from application of a general methodology and interpretations of tectonic conditions, ground motion estimates, and soil response for the entire central and eastern United States. These general interpretations do not incorporate site unique conditions that may exist. For this reason, these results should be considered conservative and may be refined when specific site conditions are considered.

2.5.5 FIELD INVESTIGATION

In April 1967, a Phase I subsurface exploration program was initiated consisting of 71 borings taken at the study area on the site. The location of the initial borings is shown in Figure 2-25. This was supplemented by extensive field and laboratory testing, aerial photograph interpretation, and a regional surface geologic investigation.

In general, 2-¾ inch core samples were recovered, along with 2-inch split barrel drive samples, to an average depth of 100 feet, with three deep holes to 200 feet. Recovered samples were examined and classified to determine:

a. Soil types and configuration
b. Engineering properties of soil and rock
c. Extent and degree of bedrock-weathering
d. Extent and magnitude of solution channels and secondary infilling

Data on engineering properties of soil and rock were obtained from in situ testing including pressure-meter testing, seismic refraction, shear wave propagation, and plate load testing. The load testing was conducted in a cement grouted test area at various levels within the foundation materials.

Laboratory testing of core samples was performed to aid in an investigation of the strength and compressibility of the foundation materials. Laboratory testing was also conducted to investigate the supporting characteristics of load-bearing fill, both in a compacted (Zone II) and grouted (Zone I) state.

Water level observations were recorded over the study area and compared with tidal observations to measure groundwater response to tidal fluctuations.

The data collected in all of the above aspects of the field investigation were incorporated into the interpretation of the engineering geology of the site and the foundation systems analysis, prepared by Woodward-Clyde & Associates (Reference 37).

A Phase II exploration program conducted between May 20, 1968 and May 3, 1970, included Boring Series A, B, AB, AS, CE, NSS, TA, and SW. These borings were drilled to further study the variations in the foundation rock mass as revealed by previous exploration and by observation and investigation during the first stages of excavation.

The above holes were generally spaced on twenty-foot centers in the fuel handling, diesel generator and south auxiliary building areas, or as shown in Figure 2-25.

Drive sampling of materials not sufficiently intact to yield core samples was conducted by the standard penetration resistance method. Diamond-core sampling using a 2-¾ inch core barrel was used where the penetration resistance exceeded 50 blows per 6 inch penetration.

Twelve additional drill holes (Series CWL, NSL, & PH) were drilled between February 3, 1969 and March 5, 1969 to explore foundation conditions beneath the intake structure and along the route of the nuclear service lines and the cooling water lines.

Series TB and C borings in the turbine building area and the diesel generator area, were drilled between November 23, 1969 and December 18, 1969.

Phase III drilling, which included Boring Series CI, X, PH, ESP, and PGE, was conducted during construction using standard penetration and core sampling techniques.

The CI boring series was drilled to investigate a suspect foundation area detected north of the nuclear service seawater pump structure. This area was subsequently treated with chemical grout.

The X-series of holes was for the purpose of substantiating the adequacy of the foundation rock system beneath the decay heat pit and reactor building. The TB series of holes was for further exploration under the turbine building and the PH holes for the pump house foundation. The ESP & PGE holes were post-grouting holes to verify the condition of the foundation materials in two selected areas.

2.5.6 FOUNDATION ANALYSIS

2.5.6.1 Loading Conditions

Class I structures are constructed to bear at various elevations ranging from 56.33 feet, in the nuclear service seawater pump pit area of the auxiliary building, to 91 feet in the turbine generator building area, to 112.5 feet for the diesel driven emergency feedwater pump building. The reactor building comprises the most heavily loaded plant unit, being supported by a 12½ foot thick, 147 foot diameter foundation mat, bearing at elevation 80.5 feet.

The average unit loading of the reactor building under operating conditions is reported to be about 7.8 ksf. Contact pressures were computed for the following static loading cases:

a. Dead load + prestress
b. Dead load + prestress + 1.5 loss-of-coolant accident pressure (1.50P)

For these cases the maximum contact pressures were 10.3 and 23.4 ksf, respectively.

The average unit pressures imposed by other plant units generally range between 2.5 and 7 ksf. The nuclear service seawater pump pit area which has been carried down to a base elevation of 56.33 feet imposes a gross unit loading of 8.3 ksf although the net imposed pressures are significantly less due to the considerable excavation unload.

2.5.6.2 Foundation Analysis

The bearing capacity of the foundation materials was analyzed to evaluate the deep crushing potential of the least competent foundation member within the Inglis Member - the Differentially Cemented Limerock. The analysis consisted of a "worst case" approach, considering that the entire foundation system above the dolarenite will respond as a weakly-cemented sand, containing discontinuities in the form of very loose zones of infill and/or cavities, of limited horizontal extent. The analysis investigated the required shear strength, with depth, to produce an adequate safety factor against local shear failure under operating loads imposed by the reactor building foundation system.

Comparison of the imposed loading with the conservatively estimated shearing strength of the foundation materials indicated that an adequate factor of safety against a bearing capacity failure would be achieved under the most unfavorable conditions which could be reasonably postulated. This conclusion, however, was predicated on the assumption that all significant voids occurring above elevation +30 feet would be filled so as to minimize local overstressing and possible future progressive failure.

Two basic criteria were used to establish the fact that all voids were adequately filled by consolidation grouting. They are:

a. Unit take of closure holes
b. Permeability tests

Permeability tests are used as a post grouting testing procedure.

2.5.6.3 Foundation Treatment

To assure the continuity and integrity of the solutioned limestone within a specified depth extending downward from the bearing level of foundation elements, consolidation grouting using a cement base grout was accomplished subsequent to the removal of unsuitable surficial bearing materials.

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45. Florida Power Corporation, Crystal River Unit 3 Applicant's Environmental Report, Operating License Stage, January 4, 1972.
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48. Chart 11408, "United States Gulf Coast. Florida, Crystal River to Horseshoe Pt.," National Ocean Survey, NOAA, July 17, 1982.
49. Deleted.
50. FTI Document No. 38-1247584-00, "Geotechnical Engineering Services Report for Diesel Driven Emergency Feedwater Pump Project."
51. Deleted.
52. NRC Regulatory Guide 1.194, “Atmospheric Relative Concentrations for Control Room Radiological Habitability Assessments at Nuclear Power Plants”
53. NUREG/CR-6331, Rev.1, “Atmospheric Relative Concentrations in Building Wakes”
54. NUREG/CR-2858, “PAVAN: An Atmospheric Dispersion Program for Evaluating Design Basis Accidental Releases of Radioactive Materials from Nuclear Power Stations”
55. Murray & Trettel Doc Number MT-CR3-7222008, “Meteorological Data Review” (Record Numbers 3918211 and 3918213) with Supplements (Record Numbers 515057, 5152966, and 5310468)
56. Calculation N12-0002, Rev. 1, “Calculation of Atmospheric Relative Concentration, X/Q”
57. EC 294154, System Reclassification Resulting from Decommissioning
58. EC 299162, Revise Flood Design Basis

TABLE 2-1 - CRYSTAL RIVER ACCIDENT X/Q (SEC/M3) VALUES

	X/Q Values					
Time Period / Receptor Location	0 – 2 hours	0 – 8 hours[1]	8 – 24 hours	1 – 4 days	4 – 30 days	Distance in meters
Exclusion Area Boundary	1.54E-04	---	---	---	---	1340
	1.78E-04	---	---	---	---	1200
Low Population Zone	2.70E-05	1.15E-05	7.54E-06	2.99E-06	7.94E-07	8047
	2.75E-05	1.18E-05	7.68E-06	3.05E-06	8.12E-07	7907
Control Room	5.14E-03	4.07E-03	1.75E-03	1.34E-03	1.04E-03	32

Note 1: ARCON96 calculates the Control Room value for a 2 – 8 hour time period.

TABLE 2-2 -
CRYSTAL RIVER ANNUAL AVERAGE X/Q AND D/Q VALUES AT THE SITE BOUNDARY

All X/Q and D/Q values were adjusted for open level terrain correction factors according to Regulatory Guide 1.111 (Reference 23) .
Values in parentheses are powers of 10.

(January 1, 1975 - December 31, 1975)			
Receptor Direction	**Receptor Distance (m)**	**X/Q (sec/m^3)**	**D/Q (m^{-2})**
NNE	1376	1.2 (-6)	1.9 (-8)
NE	1502	1.2 (-6)	1.6 (-8)
ENE	1361	1.3 (-6)	1.9 (-8)
E	2161	5.5 (-7)	9.7 (-9)
ESE	2114	4.4 (-7)	7.4 (-9)
SE	2523	3.1 (-7)	3.5 (-9)
SSE	2994	2.8 (-7)	1.6 (-9)
S	2758	7.0 (-7)	4.0 (-9)
SSW	2381	8.6 (-7)	4.5 (-9)
SW	2240	1.6 (-6)	1.1 (-8)
WSW	2271	2.2 (-6)	1.3 (-8)
W	2507	2.3 (-6)	9.3 (-9)
WNW	2475	1.4 (-6)	5.5 (-9)
NW	1628	2.5 (-6)	1.5 (-8)
NNW	1392	1.3 (-6)	1.3 (-8)
N	1340	1.4 (-6)	1.9 (-8)

TABLE 2-3 - GEOLOGIC GLOSSARY

Allochemica	Components formed by chemical precipitations within the basin of deposition, but which have suffered some later transport or can be differentiated from "Normal" chemical precipitates. (Shells, oolites, pellets, etc.)
Alluviation	The deposition of mechanical sediments by rivers anywhere along their course.
Anticline	Rock strata that are arched so as to slope away from the plane of axis on either side.
Bedrock	Solid rock in-situ, not transported.
Bioclastic	Rocks which owe their fragmentation to the activity of organisms.
Biogenic	Pertaining to a deposit resulting from the physiological activity of organisms
Biopelcalcarenite	Composite word; "bio" for biogenic type, "pel" for pellet rock, calcarenite if allochems lie within 0.0625 and 1mm size range.(Carbonate rocks)
Brachyanticline	Along narrow anticline.
Calcilutite	When allochems size is less than 0.0625mm (Carbonate rocks).
Clastic	Rock composed principally of detritus, transported mechanically into its place of deposition
Concretions	Nodular concentration developed by localized deposition of material from solution, generally about a nucleus.
Diagenic	Alteration of a bed or beds before consolidation or while in environment of deposition.
Diastrophism	The process or processes by which the crust of the earth is deformed
Dip	Angle at which a stratum is inclined from the horizontal.
Doloarenite	When size of dolomite allochems lie within 0.0625 and 1mm range
Dolomite	A carbonate rock consisting dominantly of $CaMg(CO_3)2$.
Eustatic	Pertaining to simultaneous, world-wide changes in sea level
Fault	A fracture along which there has been displacement of the two sides relative to one another and parallel to the fracture.
Flexure	A bend in strata or any planar structure (fold).
Graben	A block that has been downthrown along faults relative to the rocks on either side.
Horst	A block that has been uplifted along faults relative to the rocks on either side.
Joint	A fracture or parting which interrupts abruptly the physical continuity of a rock mass.
Karrenfield (Karrenfelder)	An erosional furrowed surface topography resulting from differential solution of limestone and removal of residual lime stone soil.
Karst	Marked by sink, or karst, holes interspersed with abrupt ridges and irregular protuberant rocks, usually underlain by caverns and underground streams.
Lithification	The complex of processes that converts a newly deposited sediment into an indurated rock.

TABLE 2-3 - GEOLOGIC GLOSSARY (continued)

Lithology	Composition and texture of rocks.
Lithostrome	A lithostratigraphic layer consisting of one or more beds of essentially uniform or uniformly heterogenous lithologiccharacter.
Littoral	Taking place on or near the shore.
Neritic	Of or pertaining to the marine environment which extends from low tide to a depth of 600 feet, or the lower limit of effective penetration of the radiant enery of the sun.
Pellet	Tiny ellipsoidal particles of carbonate, usually less than 1mm in size.
Penecon-temporaneous	Implies that in a cherty limestone or a concretionary shale, that the chert of concertion was formed at almost the same time as the deposition of the material of the surrounding rock.
Quaquaversal	Dipping outward in all directions from a central point, as a dome in stratified rock.
Residual	Material resulting from the decomposition of rocks and remaining essentially in place.
Saddle	A low point on a ridge or crestline or a structural feature created by the sagging of an anticline.
Scarp	An escarpment, cliff or steep slope of some extent along the margin of a plateau, mesa, terrace, or beach.
Strand	Synonymous with beach--that portion of the shore between low and high water
Sublittoral	A zone from a depth of 40 to 60 meters to about 200 meters or the edge of the continental shelf
Superjacent	Lying above or upon.
Syncline	A fold in the rocks in which the strata dip inward from both sides toward the axis.
Tectonics	A study of the structure of the earth in response to earth movements.
Terrace	Relatively flat, horizontal, orgently inclined surfaces which are bounded by a steeper ascending slope on one side and a steeper descending slope on the opposite side
Unconformity	A surface of erosion or non-deposition that separates younger strata from older rock
Upwarp	An area that has been up lifted, generally used for broad anticlines

TABLE 2-4 - ANNUAL PROBABILITY OF EXCEEDANCE FOR PEAK GROUND ACCELERATION (PGA)

PGA (g)	Mean	15%	Fractiles 50%	85%
0.005	1.1×10^{-3}	3.1×10^{-4}	7.7×10^{-4}	2.7×10^{-3}
0.050	5.2×10^{-5}	1.7×10^{-5}	4.8×10^{-5}	1.0×10^{-4}
0.100	1.6×10^{-5}	5.4×10^{-6}	1.5×10^{-5}	3.1×10^{-5}
0.250	2.0×10^{-6}	5.4×10^{-7}	1.7×10^{-6}	3.8×10^{-6}
0.500	2.2×10^{-7}	2.8×10^{-8}	1.1×10^{-7}	4.7×10^{-7}
0.700	6.1×10^{-8}	3.8×10^{-9}	1.9×10^{-8}	1.1×10^{-7}
1.000	1.5×10^{-8}	4.1×10^{-10}	2.2×10^{-9}	1.9×10^{-8}

FIGURE 2-1, GENERAL AREA MAP

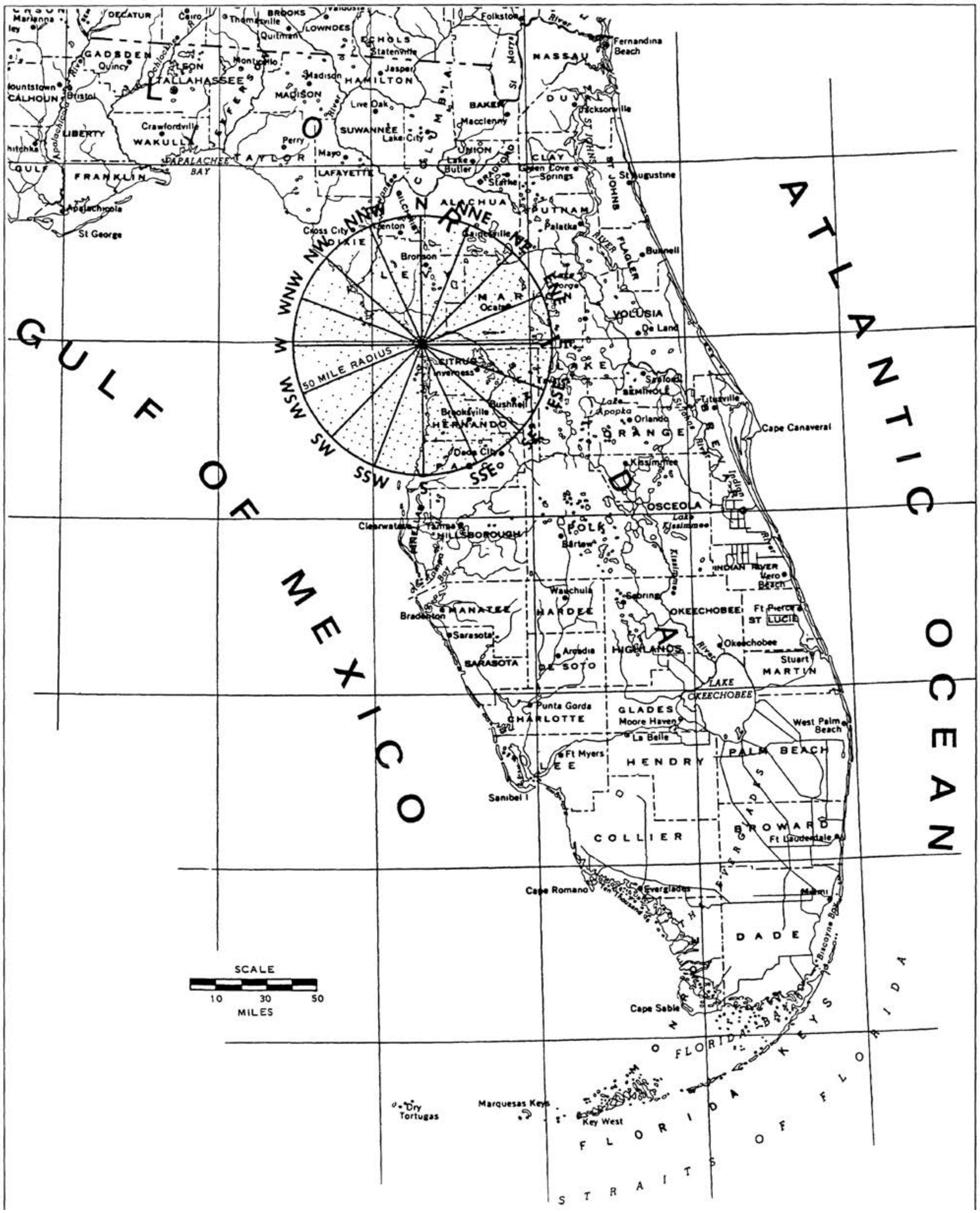


FIGURE 2-2, SITE TOPOGRAPHY WITHIN A 50 MILE RADIUS

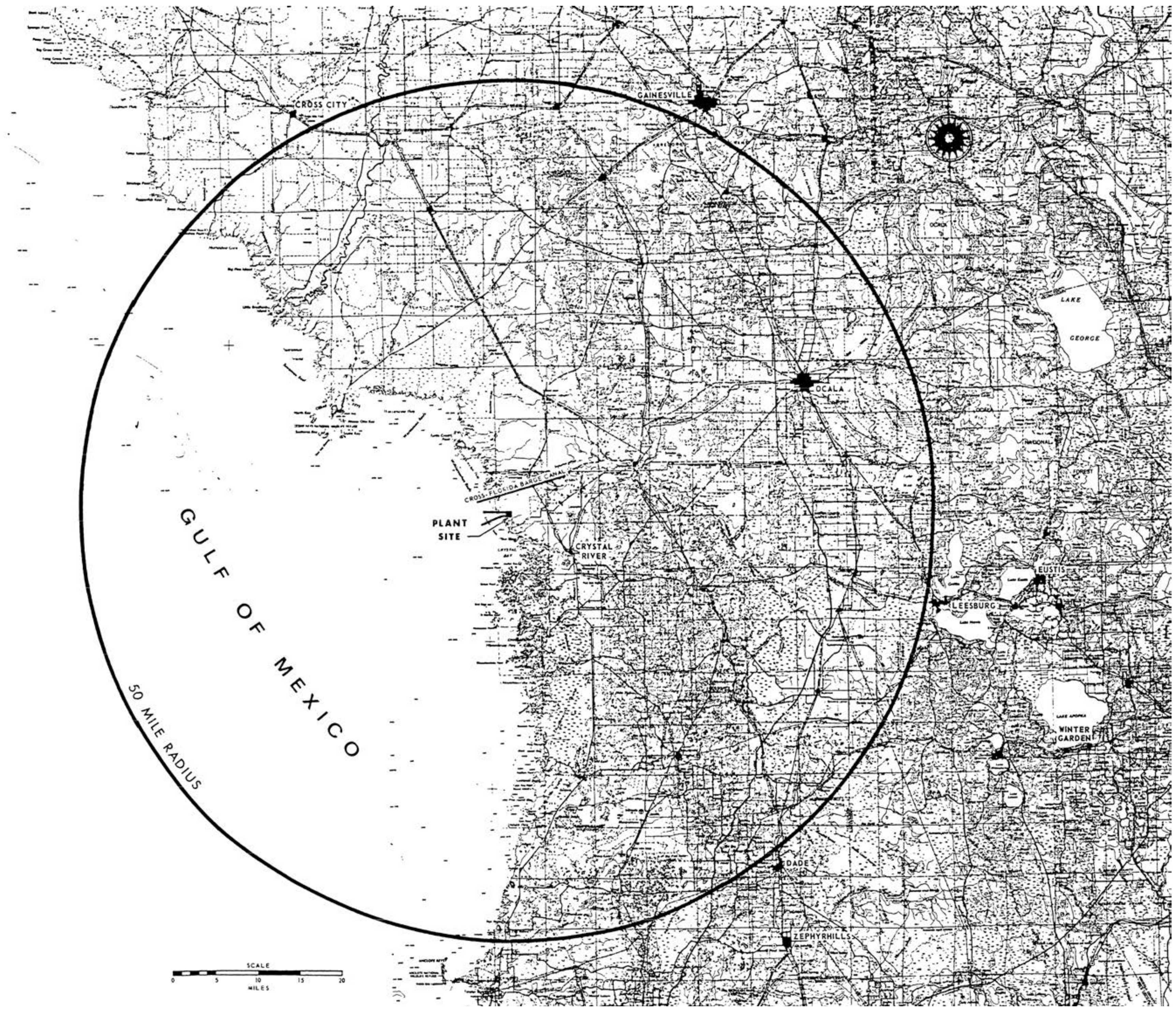


FIGURE 2-3, SITE TOPOGRAPHY WITHIN A 5 MILE RADIUS

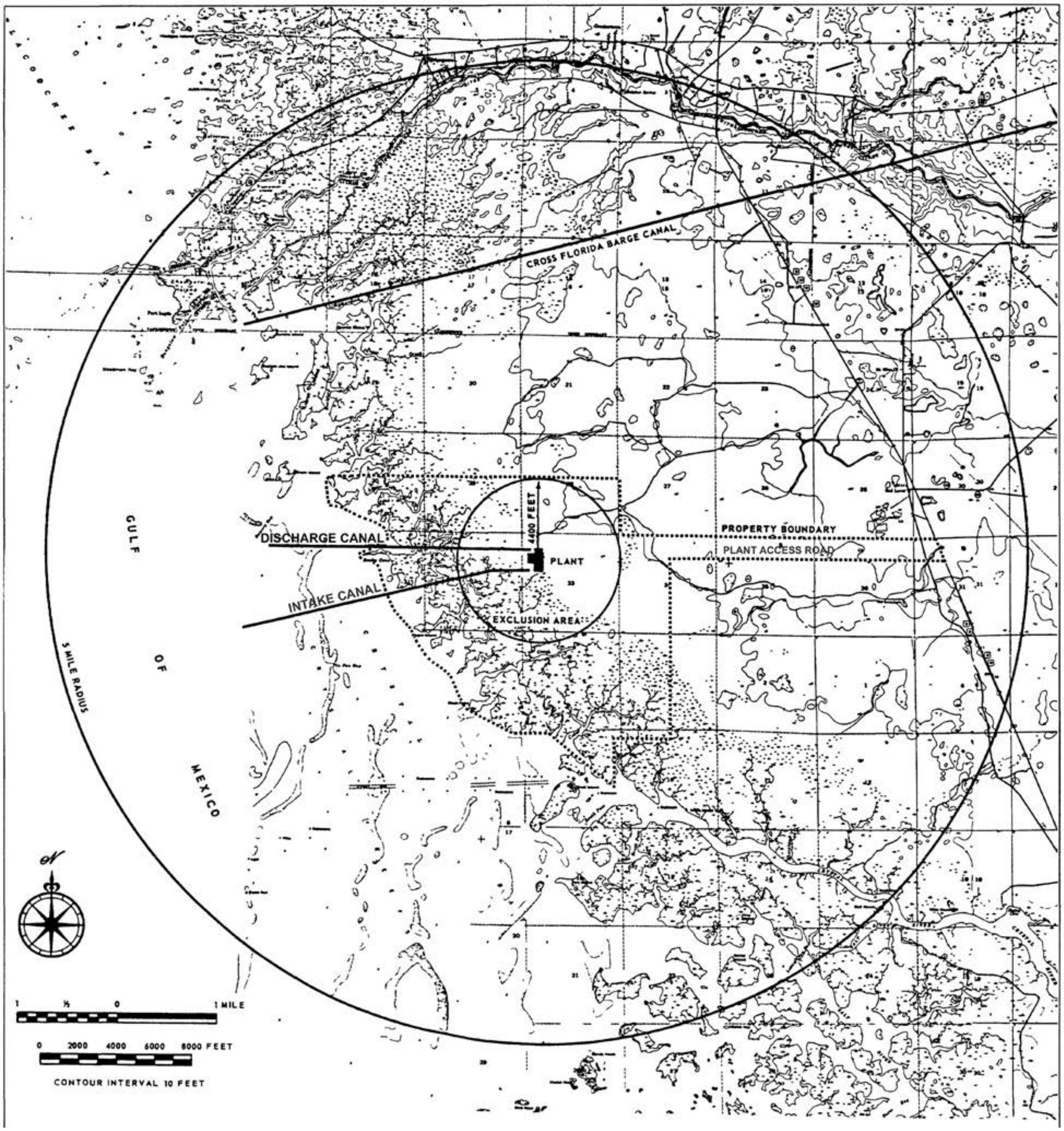


FIGURE 2-4, TOPOGRAPHY PLAN AND LOCAL AERIAL PHOTOGRAPH

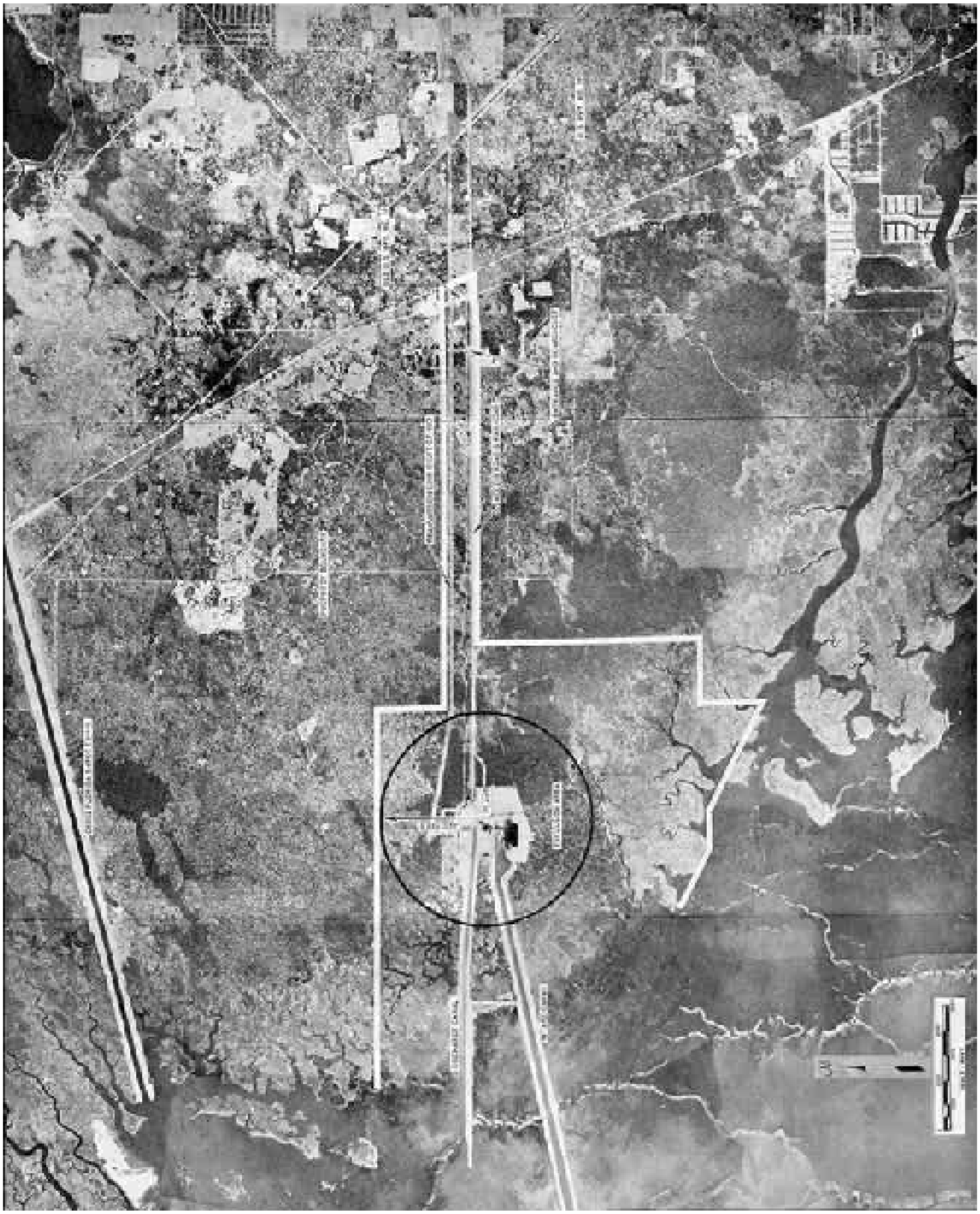


FIGURE 2-5, POPULATION CENTERS OF 25,000 OR MORE WITHIN 100 MILES OF PLANT SITE

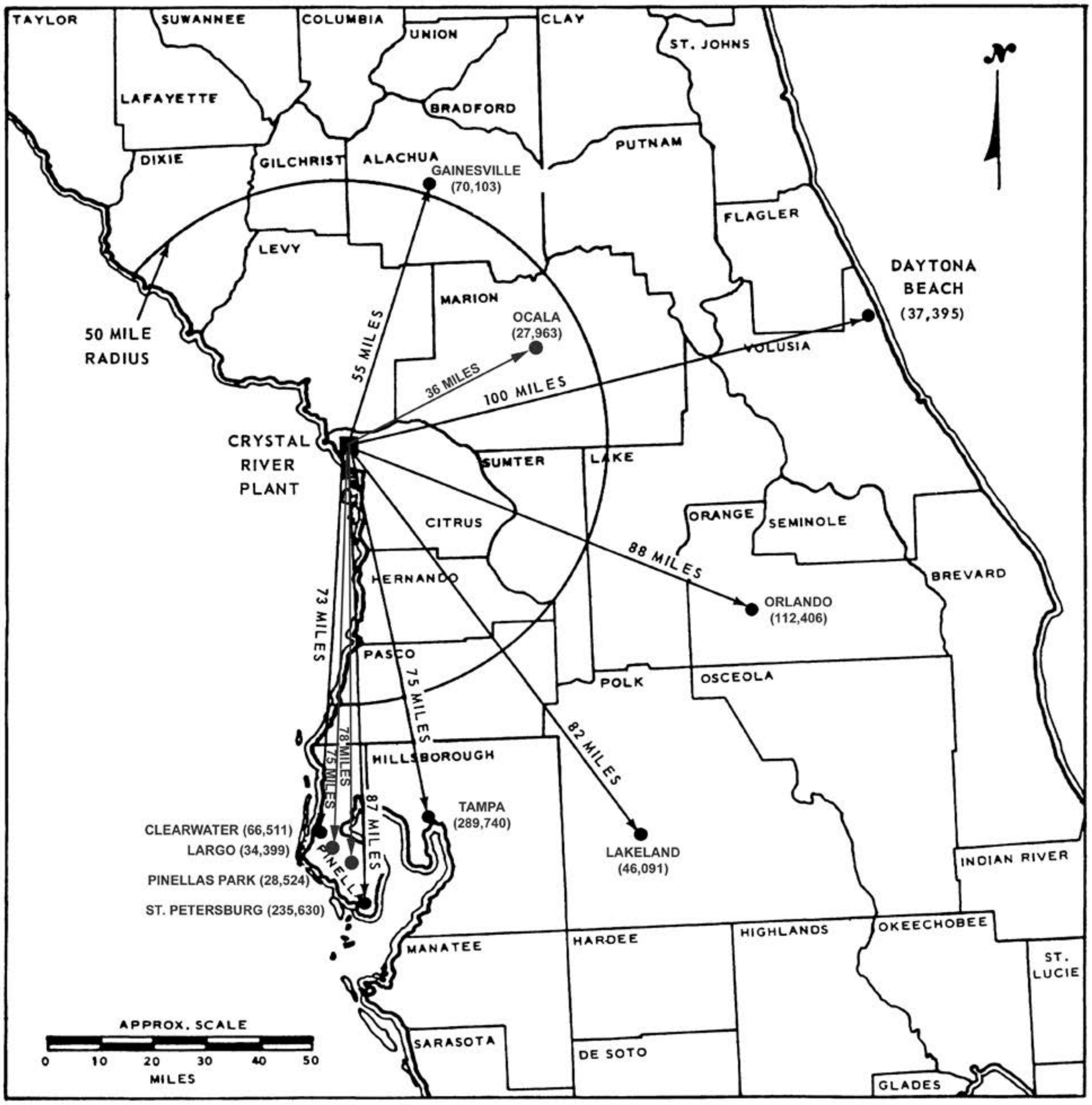


FIGURE 2-6 SHEET 1, POPULATION DISTRIBUTION

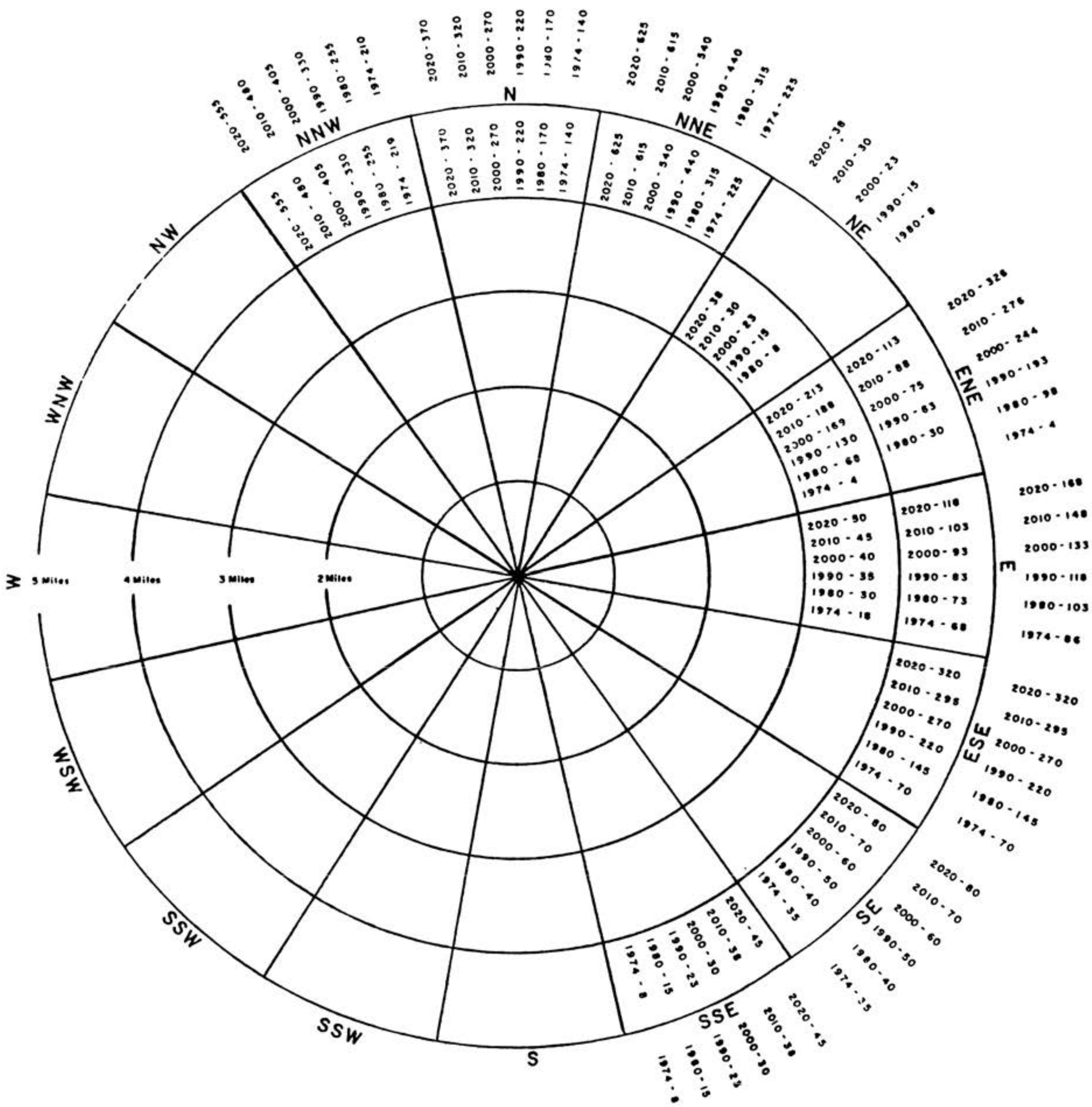


FIGURE 2-6 SHEET 2, POPULATION DISTRIBUTION

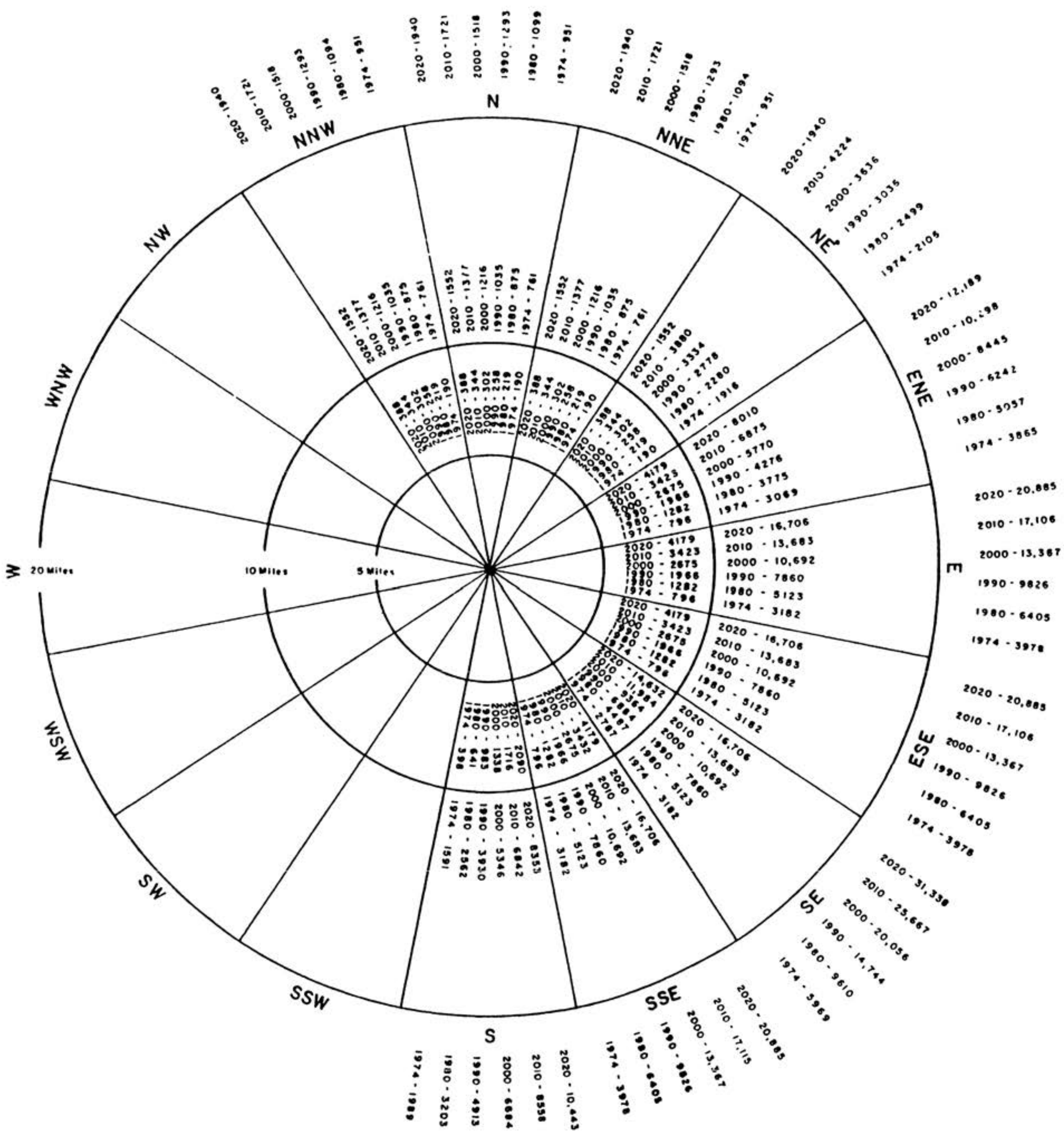


FIGURE 2-6 SHEET 3, POPULATION DISTRIBUTION

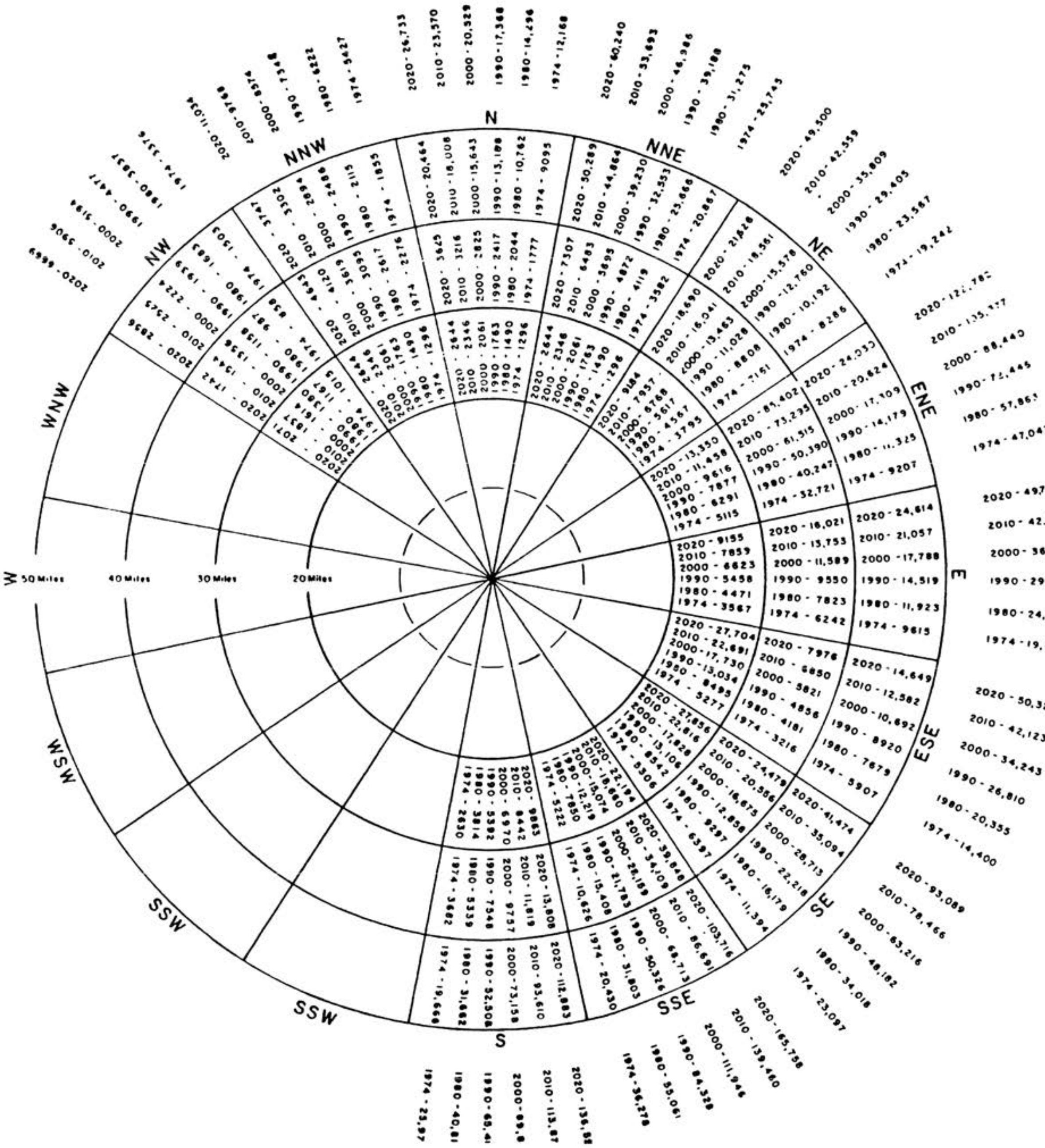
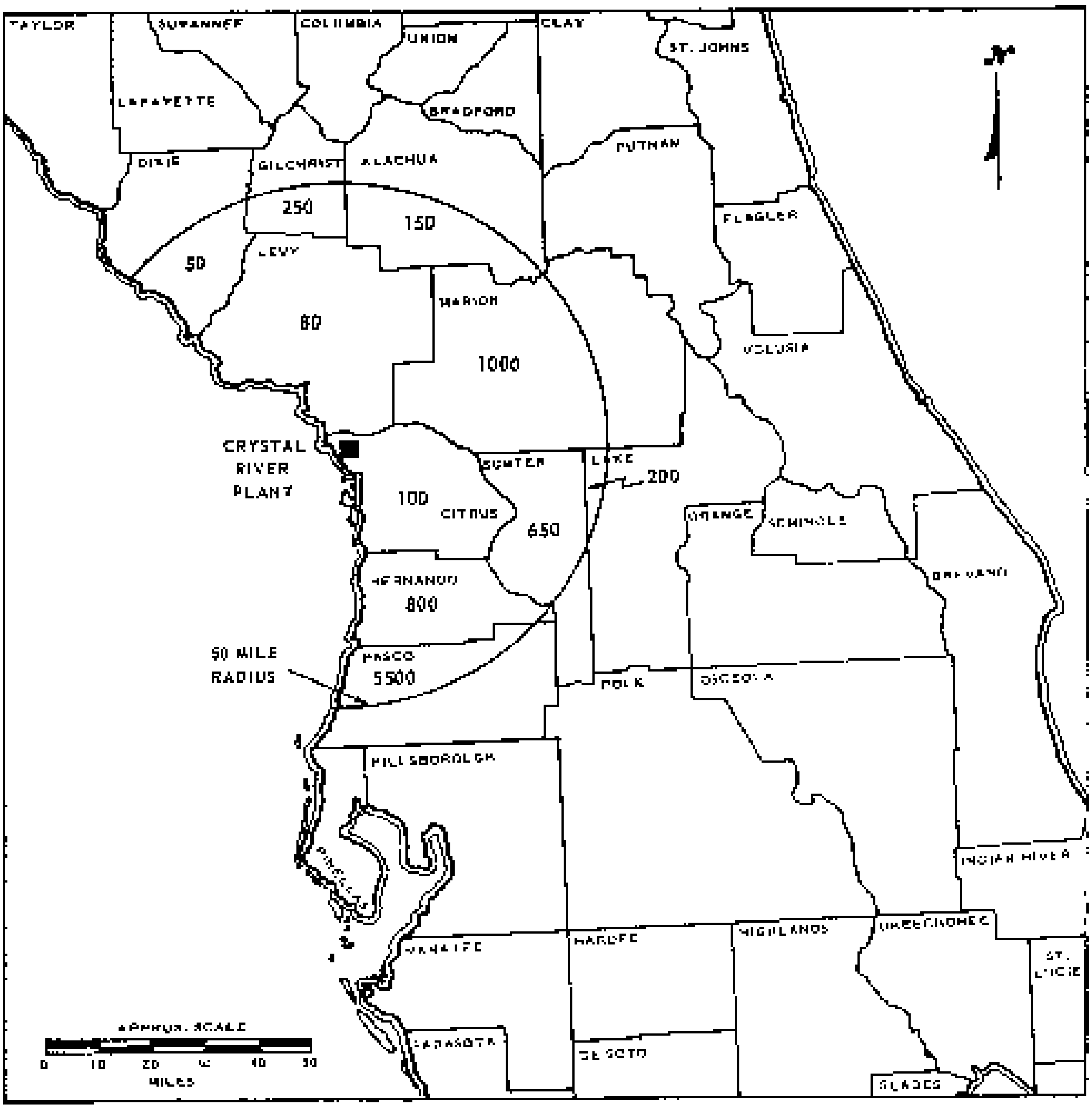


FIGURE 2-7, DAIRY ANIMALS WITHIN A 50 MILE RADIUS



SOURCE: 1964 CENSUS OF AGRICULTURE ADJUSTED TO 1966 BY C. W. REAVES, DEPT. OF DAIRY SCIENCE.
UNIV. OF FLA., GAINESVILLE

FIGURE 2-8, FLOW DURATION CURVE - WITHLACOOCHEE RIVER, NEAR HOLDER

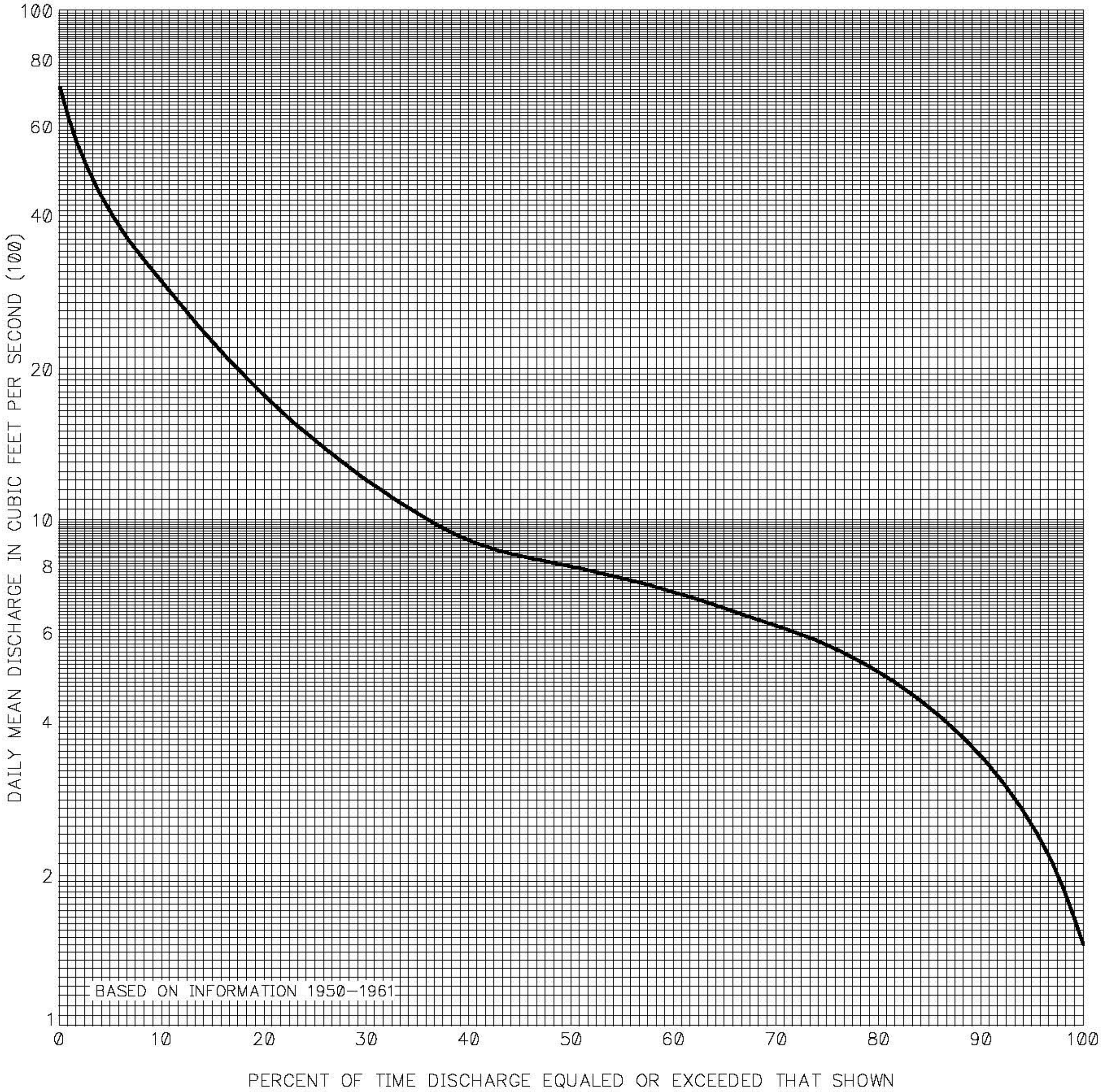


FIGURE 2-9, FLOOD FREQUENCY - WITHLACOOCHEE RIVER, AVERAGE MONTHLY DISCHARGE

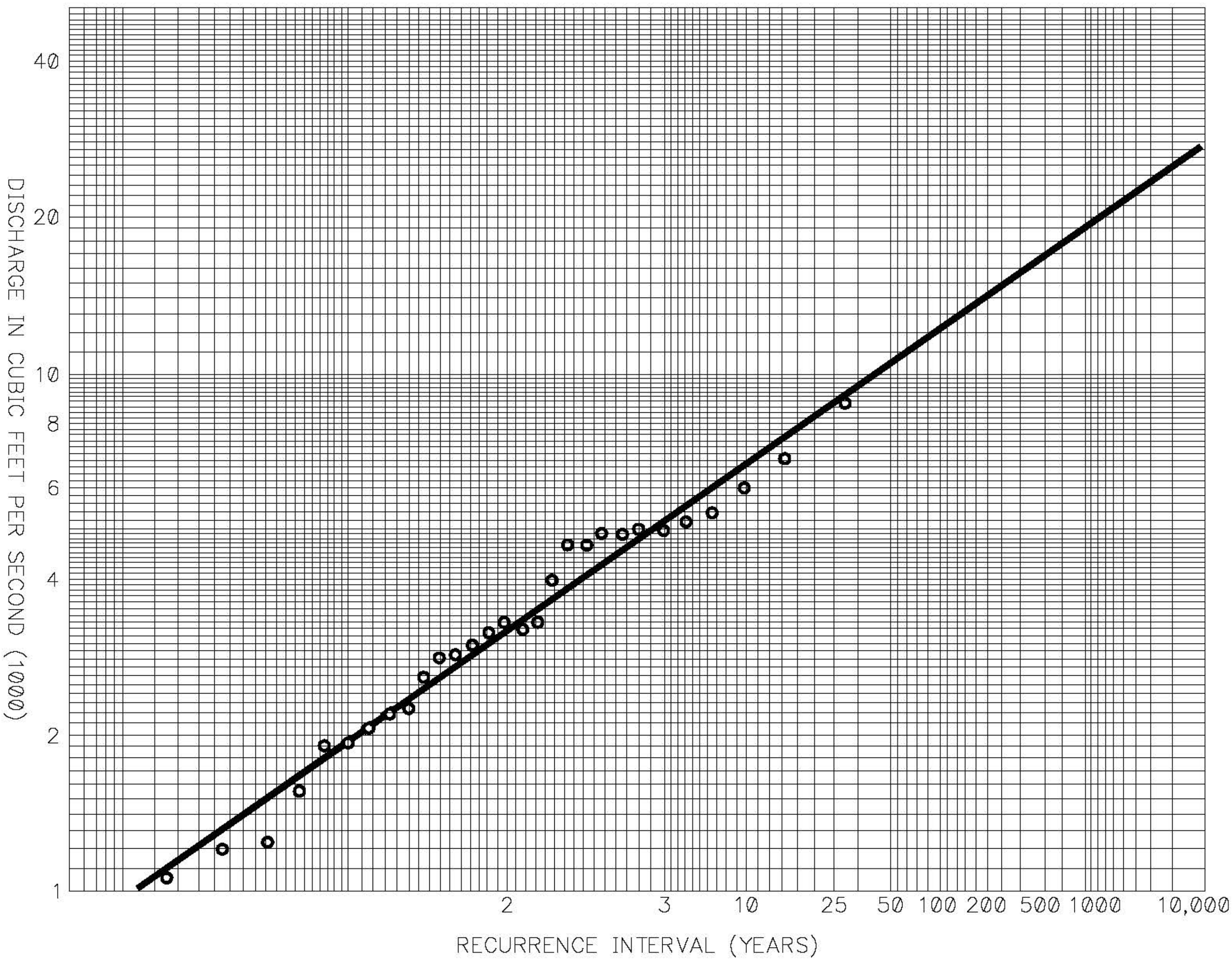


FIGURE 2-10, MEAN FLOW CHART - WITHLACOOCHEE RIVER AVERAGE MONTHLY DISCHARGE

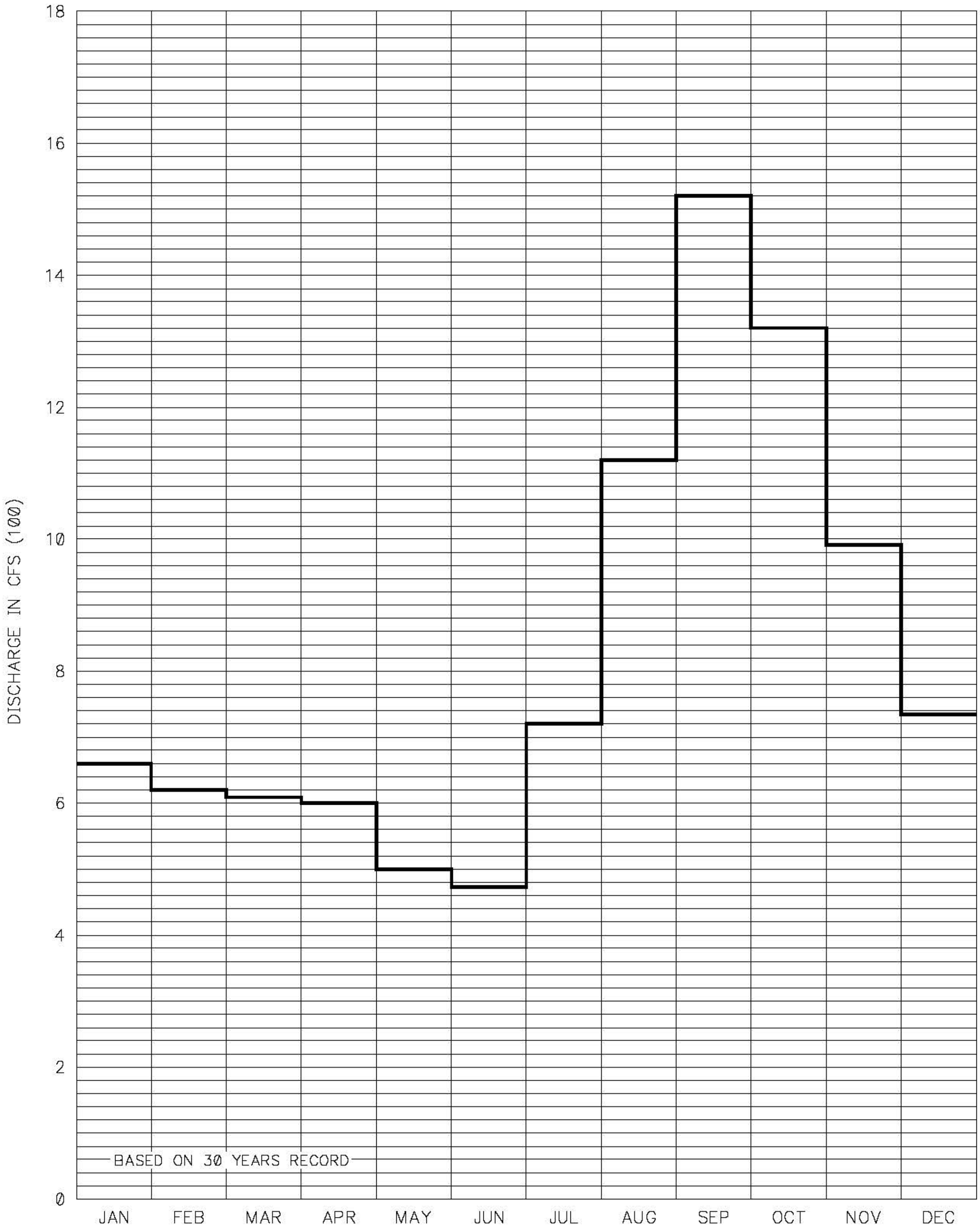


FIGURE 2-11, MAJOR LAKES & PONDS WITHIN A 50 MILE RADIUS

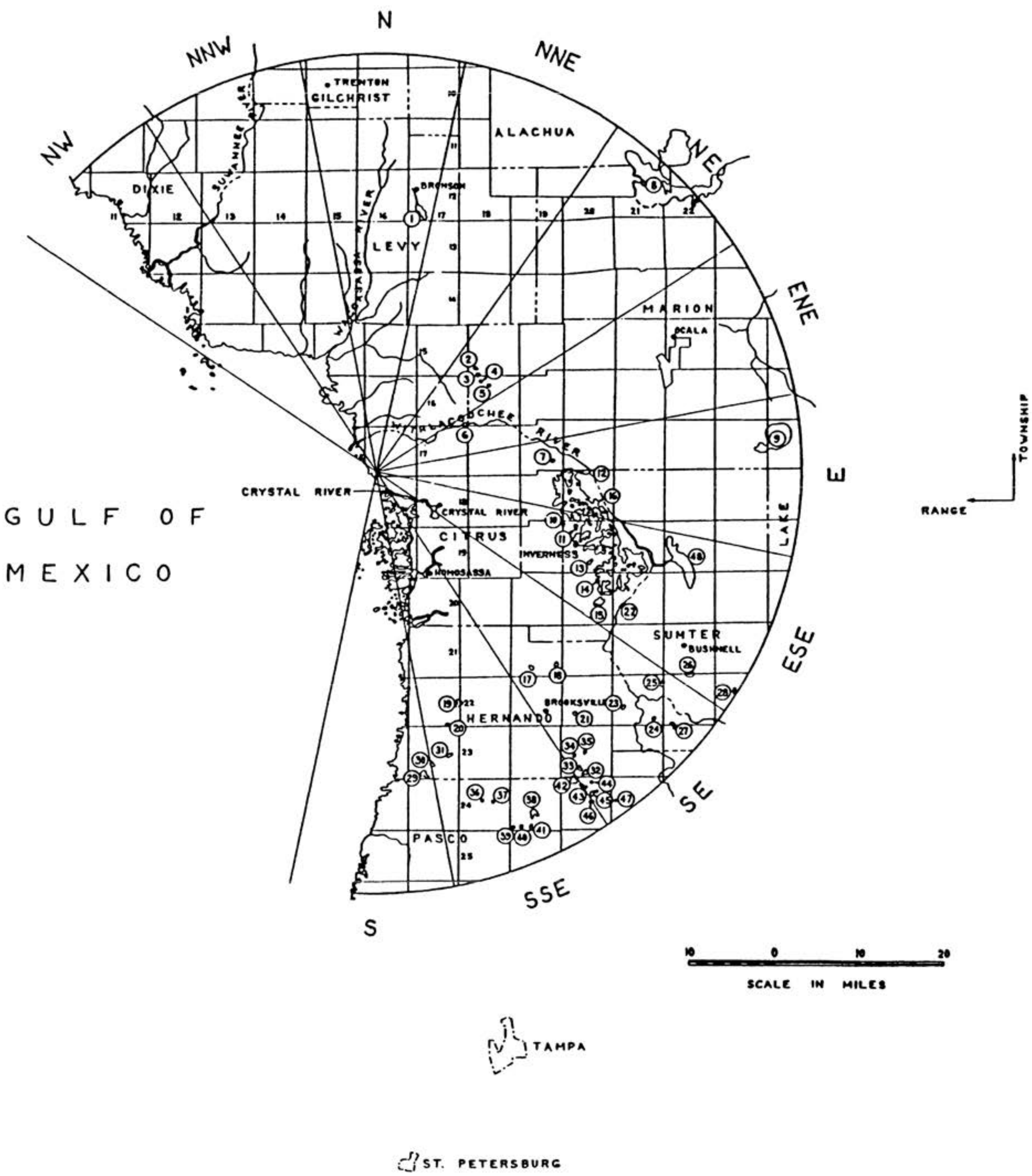


FIGURE 2-12, TABULATION OF DATA FOR LAKES AND PONDS WITHIN A 50 MILE RADIUS

	NAME OF LAKE	SURFACE (acres)	DRAINAGE (sq. mi.)	ELEVATION	*TYPE	COUNTY	LOCATION SECTION	LOCATION TOWNSHIP SOUTH	LOCATION RANGE EAST
1.	Chunky Pond	663		55	3	Levy	29	12	17
2.	Little Bonable Lake	157		55	1	Marion	30	15	18
3.	Bonable Lake	211		63	3	Marion	31	15	18
4.	Lindsay Lake	176		60	3	Marion	6	16	18
5.	Turner Lake	171		60	3	Marion	9	16	18
6.	Withlacoochee Reservoir	3,657	2,000.0	27	3	Levy	36	16	17
7.	Twomile Prairie Lake	106		35	2	Citrus	26	17	19
8.	Orange Lake	16.470	1.100.0	55.5		Alachua		T11 & 12S	R21 & 22E
9.	Lake Weir	5,450	50.0	57		Marion	NW 1/4	17	24
10.	Connell Lake	173		33	4	Citrus	6	19	20
11.	Cato Lake	103		39	4	Citrus	5	19	20
12.	Tsala Apopka Lake	20.770	100.0	39	3	Citrus		19	20
13.	Fort Cooper Lake	156		33	4	Citrus	27	19	20
14.	Unnamed Lake	123		42	4	Citrus	2	20	20
15.	Bradley Lake	590		41	2	Citrus	23	20	20
16.	Unnamed Lake	198		38	4	Citrus	22	18	20
17.	Simmons Prairie Lake	153		75	4	Hernando	27	21	19
18.	Lake Lindsay	137	3.0	65	4	Hernando	25	21	19
19.	Tooke Lake	236		19	4	Hernando	13	22	17
20.	Grear Hope Pond	123		17	4	Hernando	35	22	17
21.	Bistre Lake	307		72	1	Hernando	29	22	20
22.	Bonnet Lake	100		43	3	Citrus	20	20	21
23.	Oriole Lake	106		47	4	Hernando	20	22	21
24.	Long Lake	100		62	3	Hernando	35	22	21
25.	Unnamed Lake	259		68	3	Sumter	7	22	22
26.	Matchett Lake	150		74	3	Sumter	33	21	22
27.	Unnamed Lake	107		72	4	Hernando	32	22	22
28.	Merritt Pond	170		91	4	Sumter	21	22	23
29.	Hunter's Lake	302	2.4	19	4	Hernando	32	23	17
30.	Hog Pond	122		15	4	Hernando	28	23	17
31.	Weekiwachee Prairie Pond	289		23	4	Hernando	23	23	17
32.	St. Clair Lake	164		113	2	Hernando	33	23	20
33.	Nicka Lake	155		113	2	Hernando	28	23	20
34.	Neff Lake	226	5.0	101	1	Hernando	20	23	20
35.	Mountain Lake	127		105	3	Hernando	16	23	20
36.	Crews Lake	693	136	58	3	Pasco	16	24	18
37.	Unnamed Lake	110		60	4	Pasco	15	24	18
38.	Big Fish Lake	270		76	4	Pasco	21	24	19
39.	Unnamed Lake	122		78	4	Pasco	31	24	19
40.	New River Pond	106		77	4	Pasco	32	24	19
41.	Unnamed Lake	175		77	4	Pasco	33	24	19
42.	Hancock Lake	519		108	4	Pasco	5	24	20
43.	Middle Lake	215		107	1	Pasco	4	24	20
44.	Mud Lake	266		108	4	Pasco	3	24	20
45.	Moody Lake	205		108	3	Pasco	10	24	20
46.	Lake Iola	107	0.50	147	2	Pasco	15	24	20
47.	Blanton Lake	126		93	4	Pasco	16	24	21
48.	Lake Panasoffkee	4,463	320	41		Sumter		T18 & 20S	22
49.	Deer Lake	156		127	4	Pasco	1	25	20
50.	Long Pond	200		24	–	Levy	16	12	15
51.	Lake Levy	4,556		63	–	Alachua	24	11	19

*Type 1. Lake with stream inlet
2. Lake with stream outlet
3. Lake with stream inlet and outlet
4. Lake with no stream inlet or outlet

Data supplied by Florida Board of Conservation and the U.S. Department of the Interior, Geological Survey

FIGURE 2-13, PUBLIC WATER SUPPLIES WITHIN A 20 MILE RADIUS

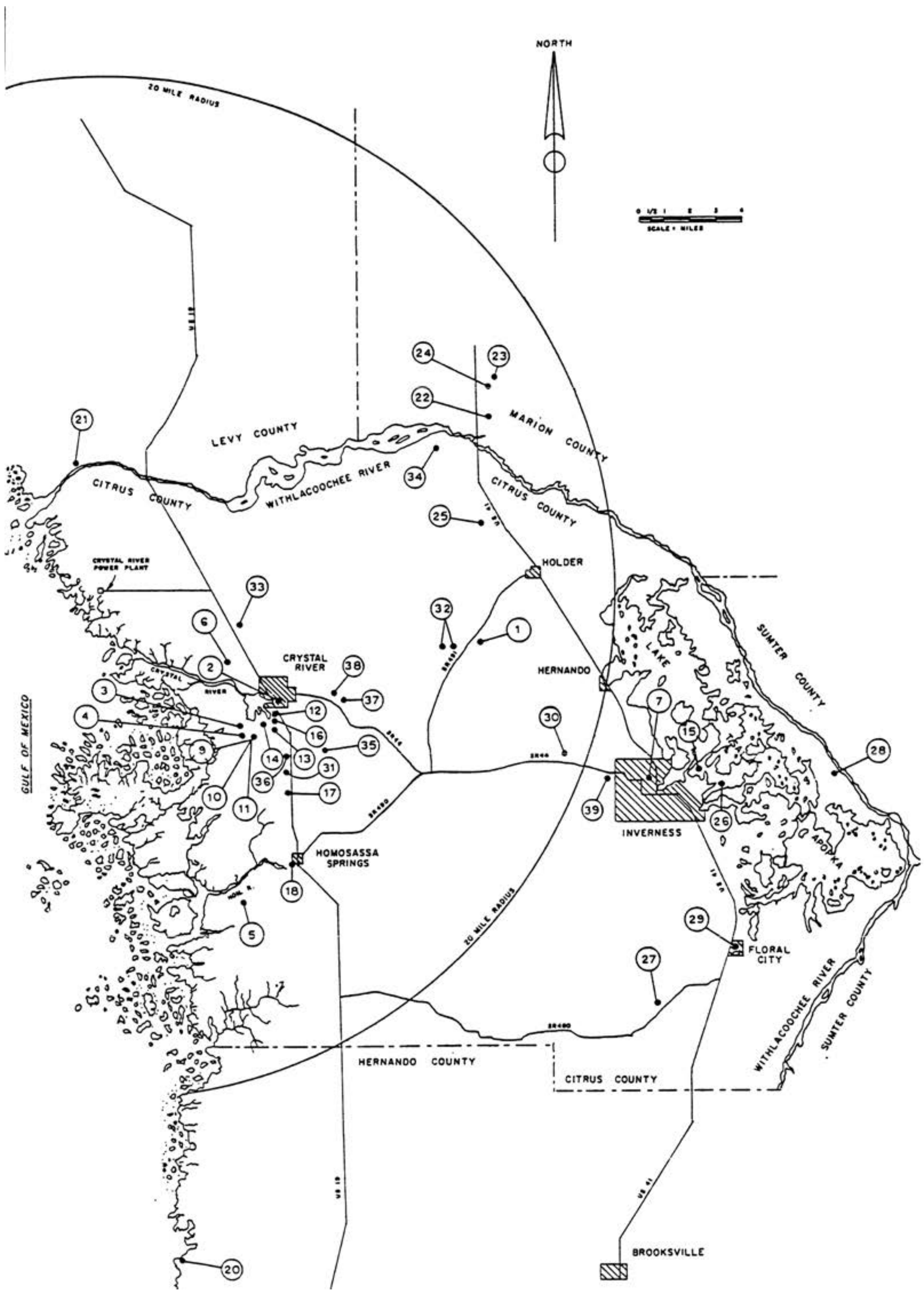


FIGURE 2-14, LOCATION OF PUBLIC WATER SUPPLIES WITHIN A 20 MILE RADIUS

	Name	County	City	Address	Population Served	Number of Services	Source	Well Depth Ft.	Well Dia. In.	Pump Cap'y GPM
1.	Beverly Hills S/D	Citrus	S/D	6 mi. north of Lecanto	42	27	2 Wells	280 290	10 10	400 400
2.	Crystal River	Citrus	Crystal River		1100	300	Well	130	10	600
3.	Crystal Shores Estates	Citrus	Crystal River	Route 41 Crystal River		23	Well			
4.	Dixie Shores S/D Suncoast City	Citrus	Crystal River	Rt. 40 Crystal River		9	Well	44	4	15
5.	Homosassa District	Citrus	Homosassa				Well	80	8	550
6.	Indian Waters S/D	Citrus	Homosassa Springs				Well	113	6	200
7.	Inverness	Citrus	Inverness		2000	550	Wells	105	10	400
8.	Mayfair Garden Acres	Citrus	Crystal River	State Road 44			Well	62	4	61
9.	Palm Springs S/D	Citrus	Crystal River	Rt. 40 Crystal River Box 50		15	2 Wells	30 80	3 3	
10.	Palm Springs Villas	Citrus	Crystal River	Rt. # 1 Crystal River		6	Well	83	6	
11.	Palm Springs Villas	Citrus	Crystal River	U.S. 19 South		10	5 Wells			
12.	Paradise Gardens	Citrus	Crystal River	Rt. #19 South			Well		6	57
13.	Plantation Hotel	Citrus	Crystal River	Crystal River		200	Well		4	
14.	Pleasure Isles	Citrus	Crystal River	Rt. # 1 Crystal River Box 117		3	Spring			
15.	Point O' Woods	Citrus	Inverness	U.S. 41 South		10	Well	57	6	
16.	Port Paradise	Citrus	Crystal River	Village of Picardy		20	2 Wells	100 100	4 4	
17.	Rene's Mobile Home Park	Citrus	Crystal River	Rt. # 1 Crystal River Box 14			Well	65	2	50
18.	Spring Village Trailer Park	Citrus	Homosassa Springs	P.O. Box 436			Well	100	4	25
19.	Tropic Terrace	Citrus	Crystal River	4800 73rd Ave. North Pinellas Park			Well	180	6	50
20.	Smith System	Hernando	Bayport	Bennet Pond Road Brooksville		10	Well	46	4	
21.	Yankeetown	Levy	Yankeetown			233	3 Wells	70 55	4 4	50 100
22.	Dunnellon Water Dept.	Marion	Dunnellon	City Hall	1300	440	2 Wells	88	8	500
23.	Rainbow Lakes Estate	Marion	Dunnellon	Rt. #1 Box 400A-1		6	Well	472	8	500
24.	Rainbow Lakes Estate Club House	Marion	Dunnellon	Rt. #1 Box 400		1			4	
25.	Citrus Springs	Citrus	Dunnellon							
26.	Connon's Trailer Park (Edgewater Mobile Resort)	Citrus								
27.	Camp E. Nenahnezee	Citrus	Floral City							
28.	Boy Scouts of America	Citrus	Inverness							
29.	Floral City	Citrus	Floral City							
30.	Golden Terrace Estates	Citrus								
31.	Ozello, Inc.	Citrus	Ozello							
32.	Pine Ridge Development	Citrus	Pine Ridge							
33.	Thunderbird Mobile Park	Citrus	Crystal River							
34.	Rush Trailer Park	Citrus	Crystal River							
35.	Seven Pines Estates	Citrus	Crystal River							
36.	Unger Trailer Park	Citrus	Crystal River							
37.	Yates Scenic Gardens	Citrus	Crystal River							
38.	Yates Mobile Home Park	Citrus	Crystal River							
39.	Vocational High School	Citrus	Inverness							

Note: A careful check was made of all public water sources here indicated. There have been 15 new public water supplies placed in service since the previous list was made, according to the Citrus County Health Department. Two public water services were discontinued: MayFair Garden Acres and Tropic Terrace. In Figure 2-23, all public water supplies are indicated by the numbers shown on this figure.

Data from Florida State Board of Health

FIGURE 2-15, POTENTIOMETRIC SURFACE - MAY, 1969 - FLORIDIAN AQUIFER, SOUTHWEST FLORIDA WATER MANAGEMENT DISTRICT, FLORIDA

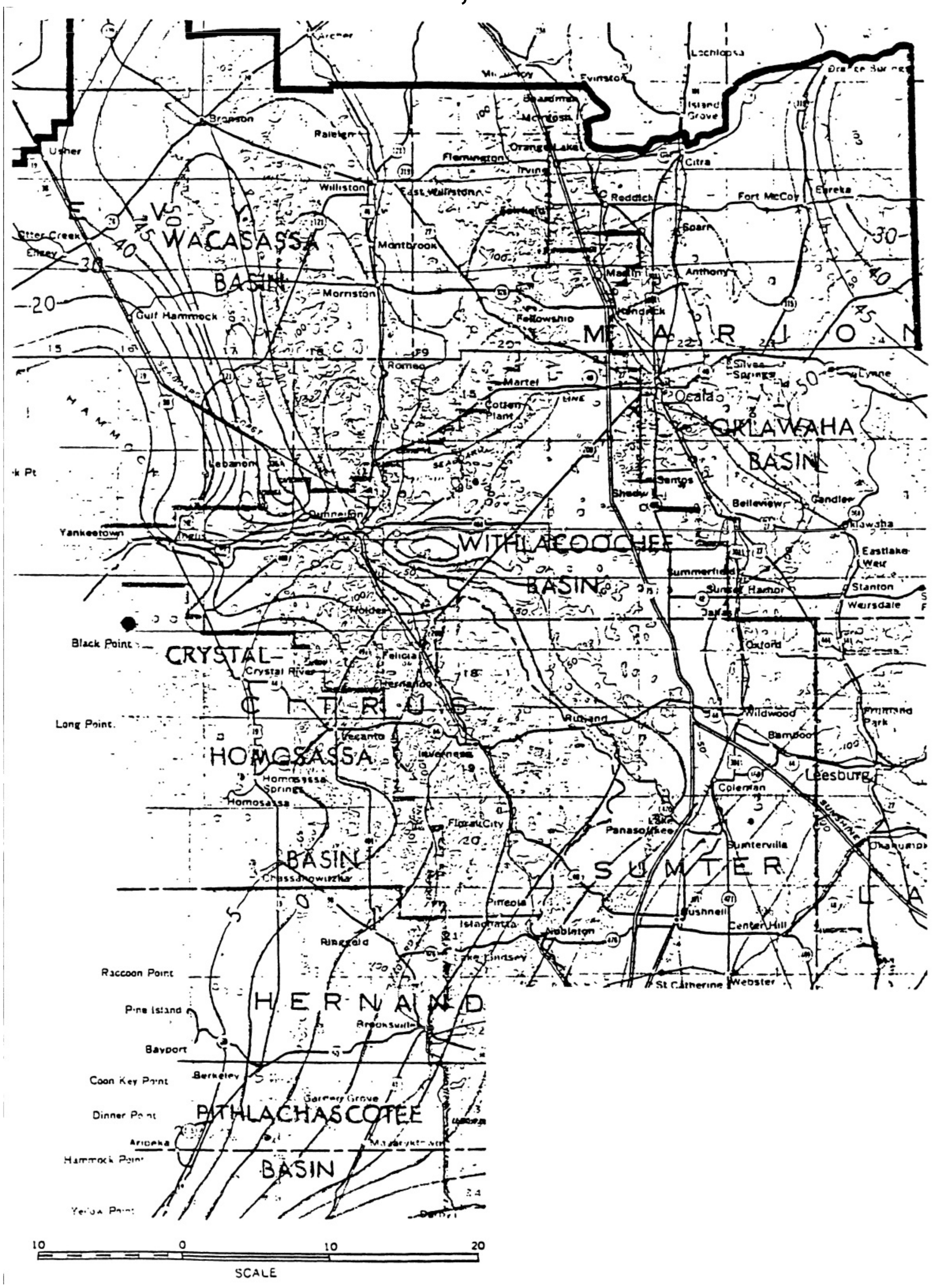


FIGURE 2-16, WAVE ACTION STUDY, GENERAL PLANT AREA

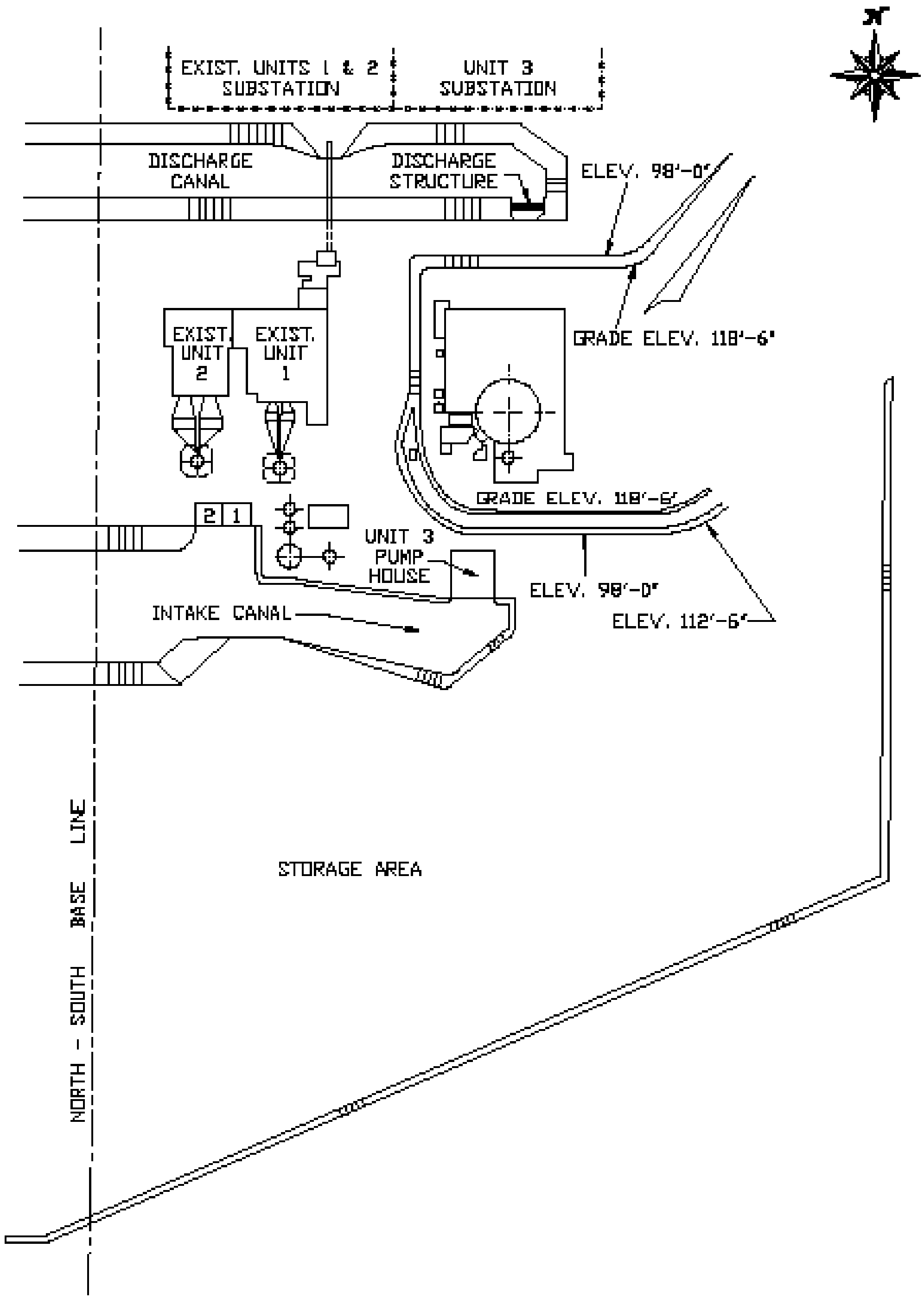
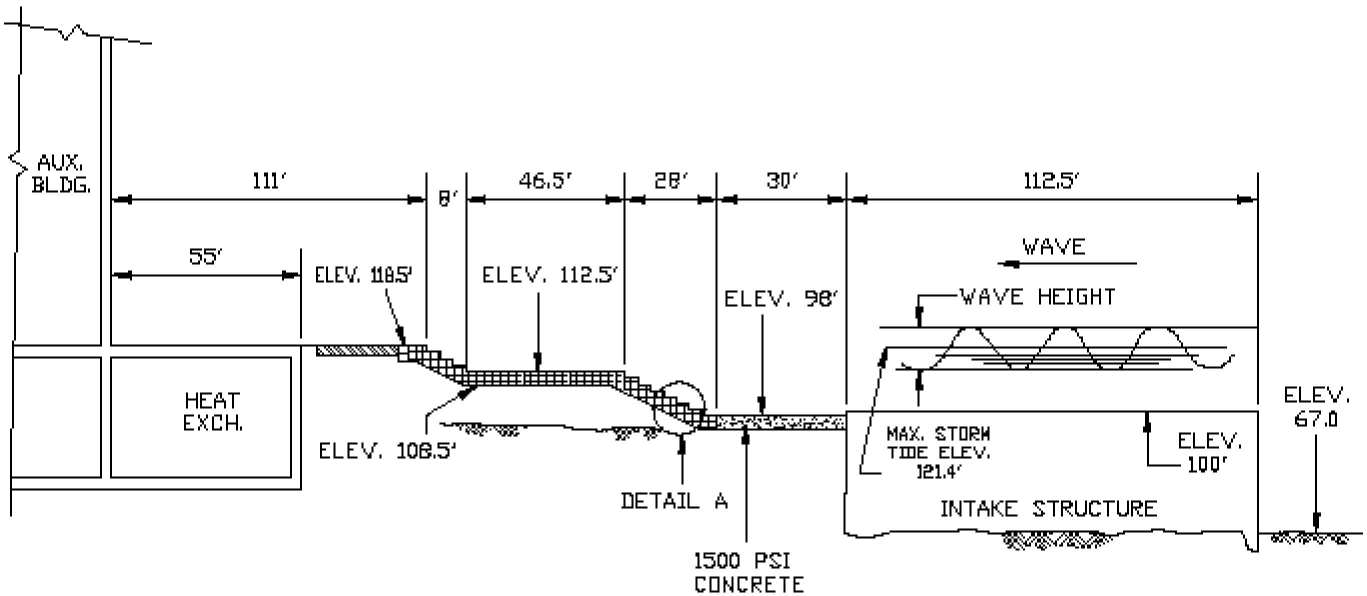


FIGURE 2-17, WAVE ACTION STUDY, SECTION OF MAXIMUM WAVE ATTACK



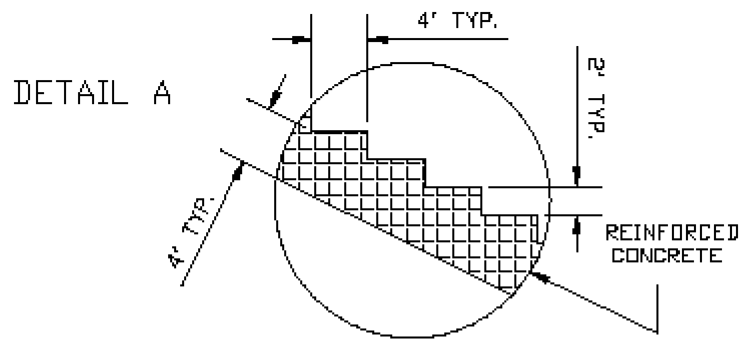


FIGURE 2-18, DESIGN WAVES AND WATER LEVELS VS. TIME

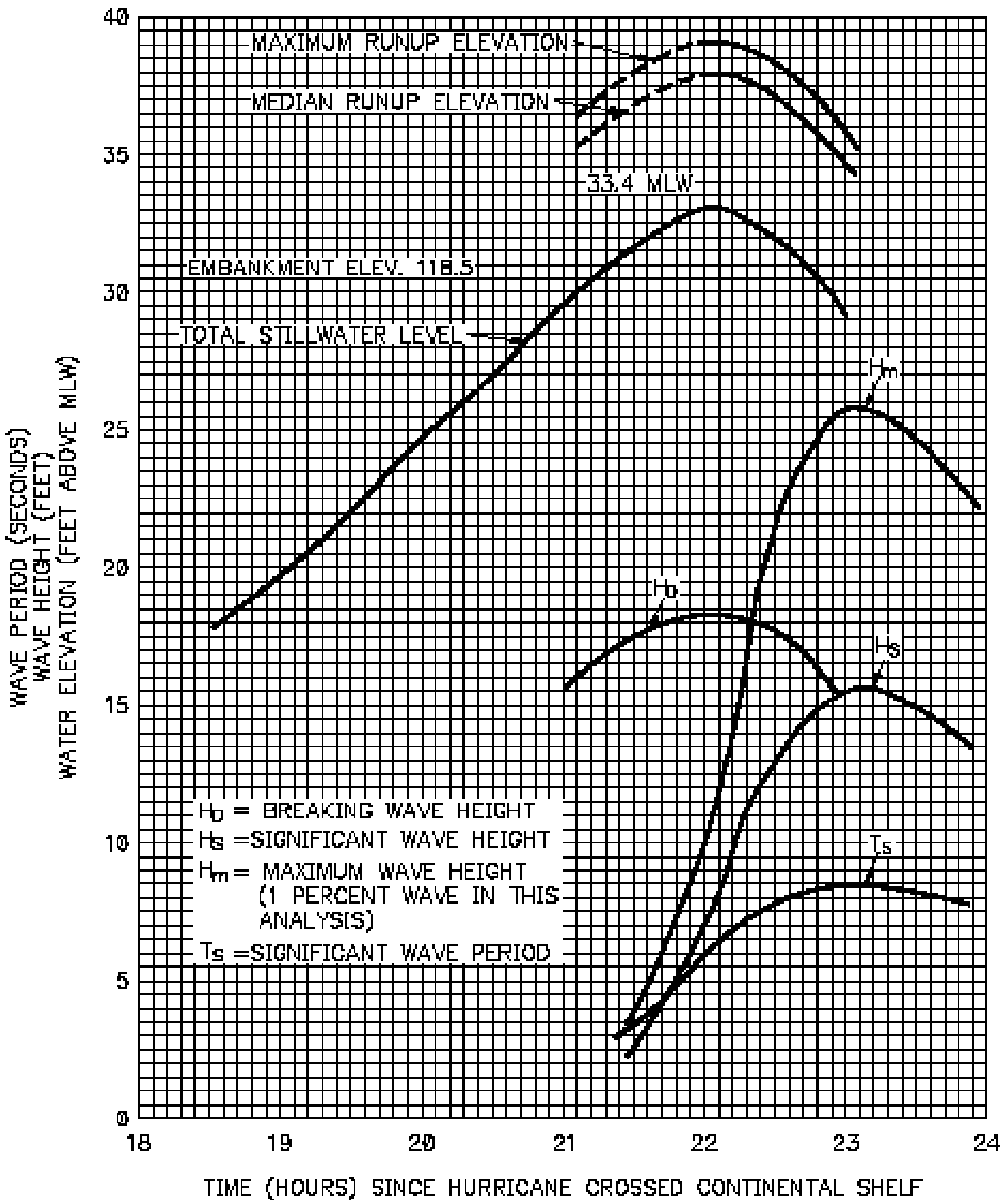


FIGURE 2-19, RUNUP AND OVERTOPPING VS. TIDE LEVEL

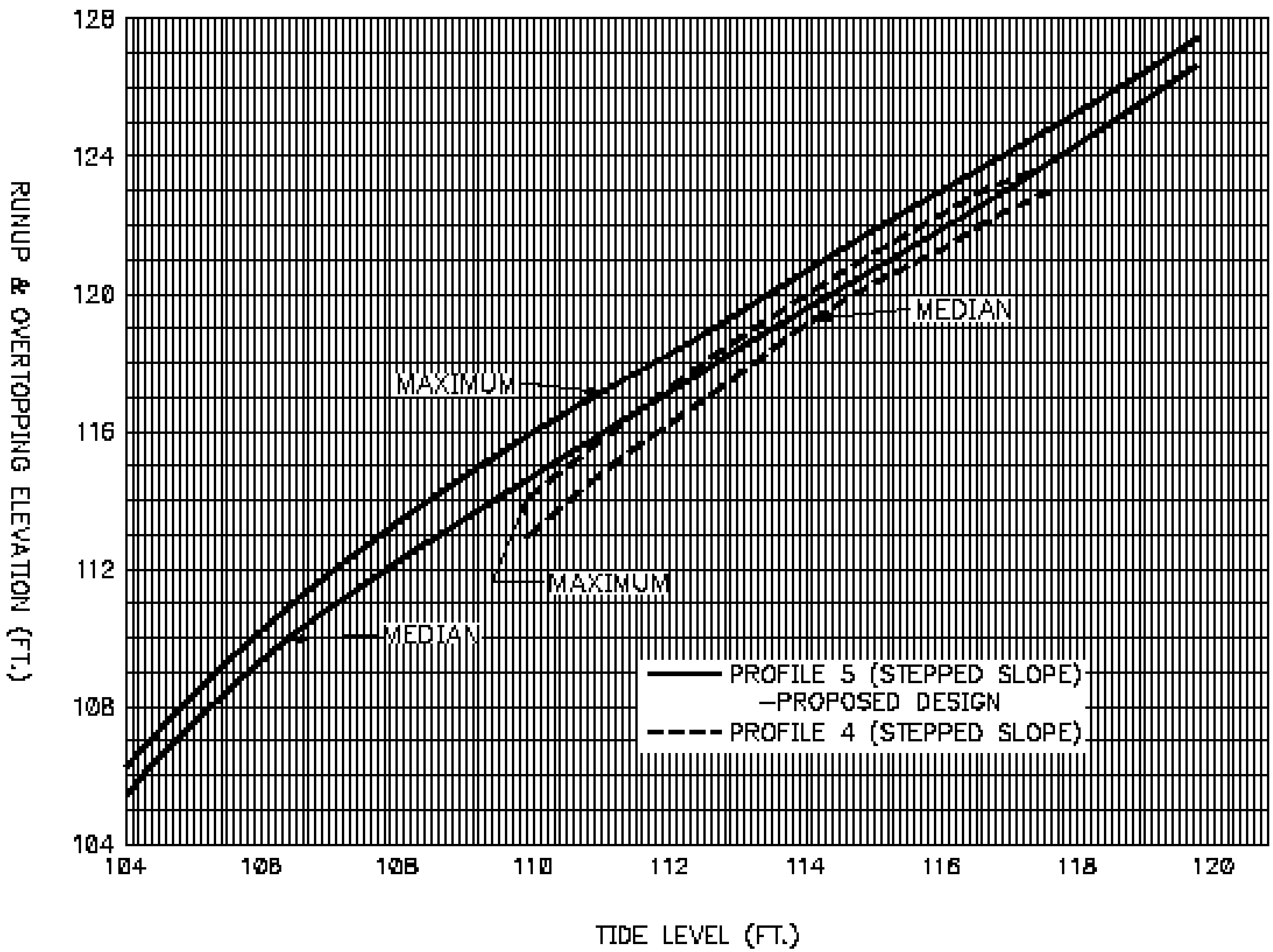


FIGURE 2-20, WIND AND WAVE CHARACTERISTICS VS. TIME

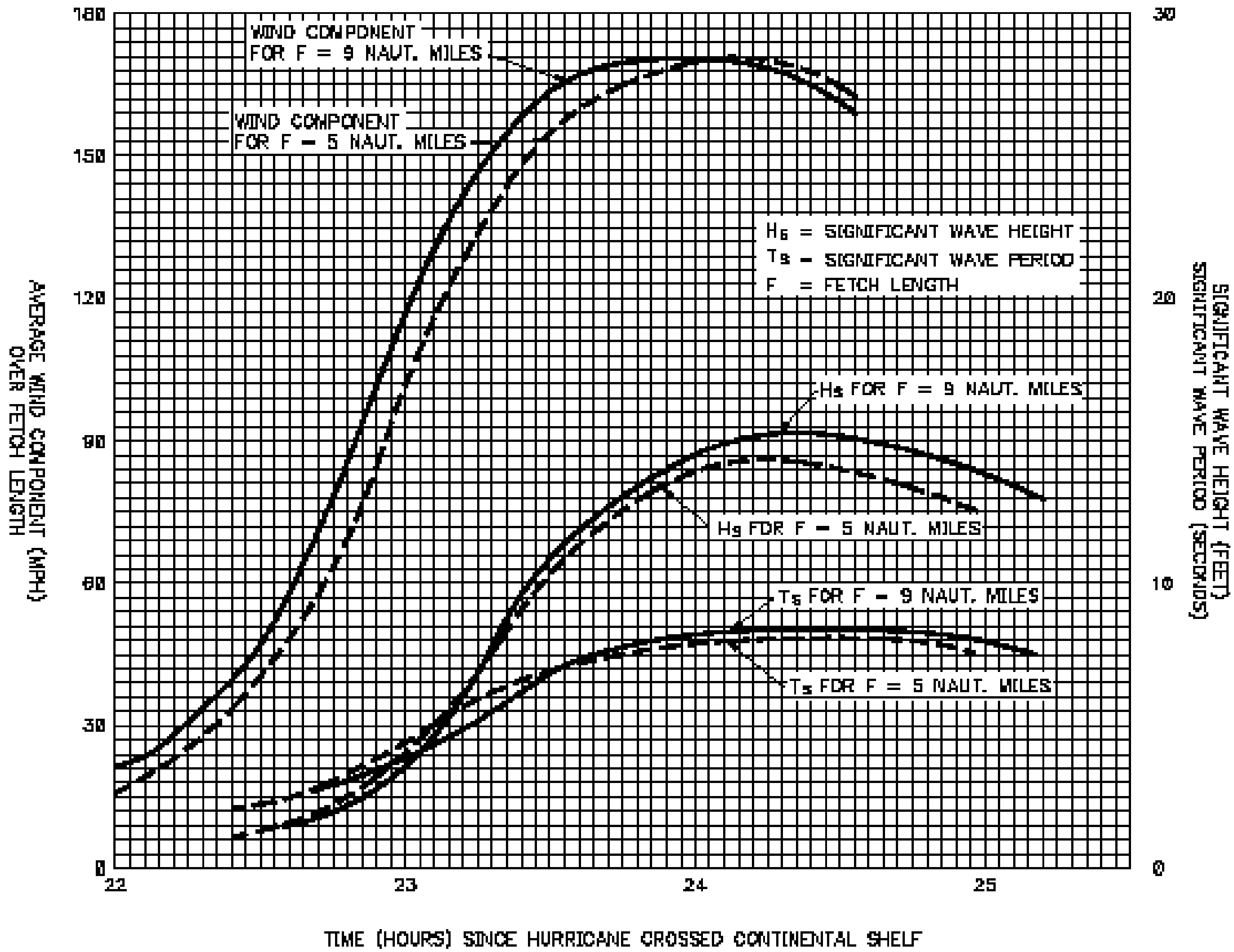


FIGURE 2-21, DESIGN HURRICANE 1, ON-SHORE WINDS

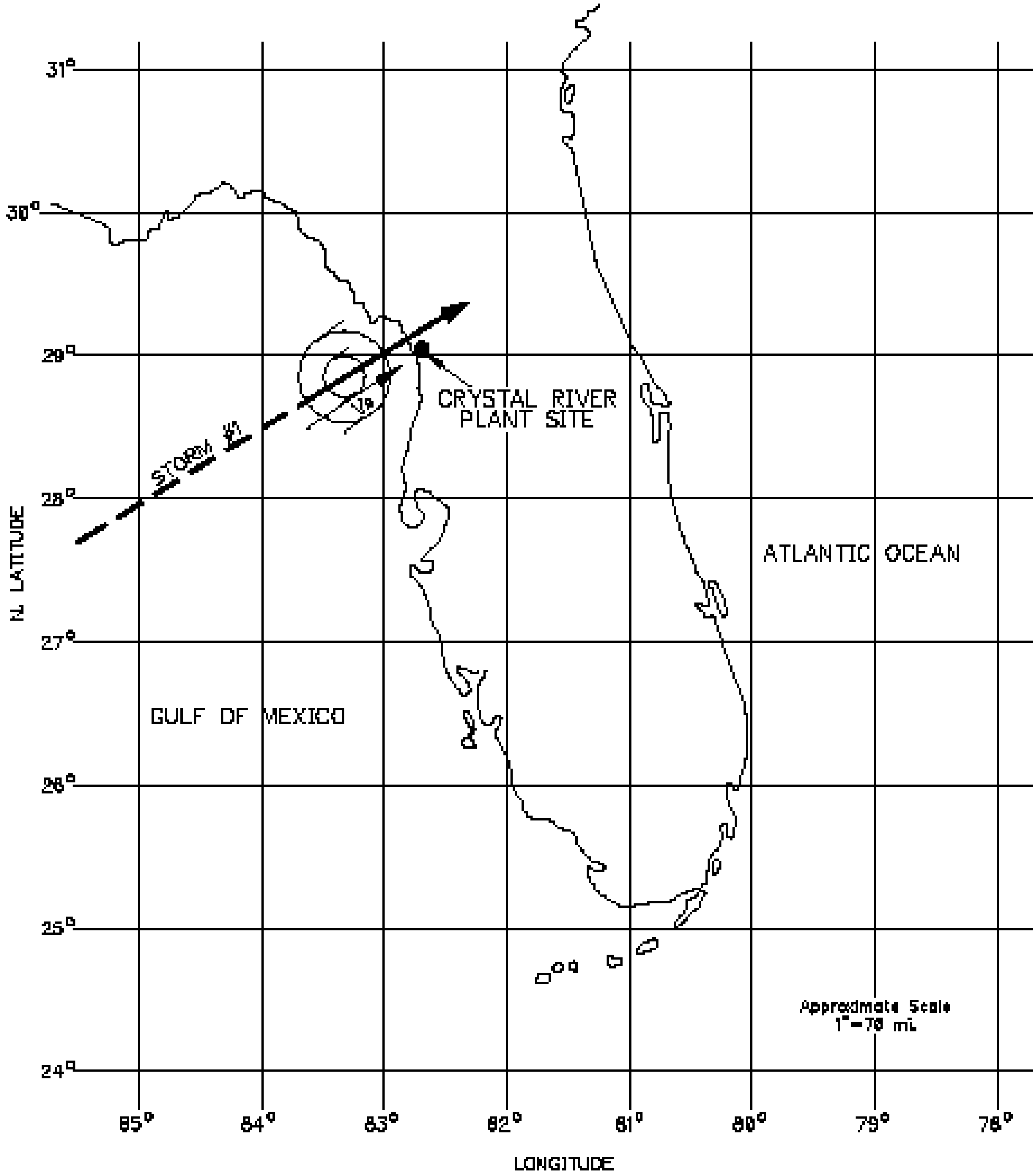


FIGURE 2-22, DESIGN ACCELERATION SPECTRA

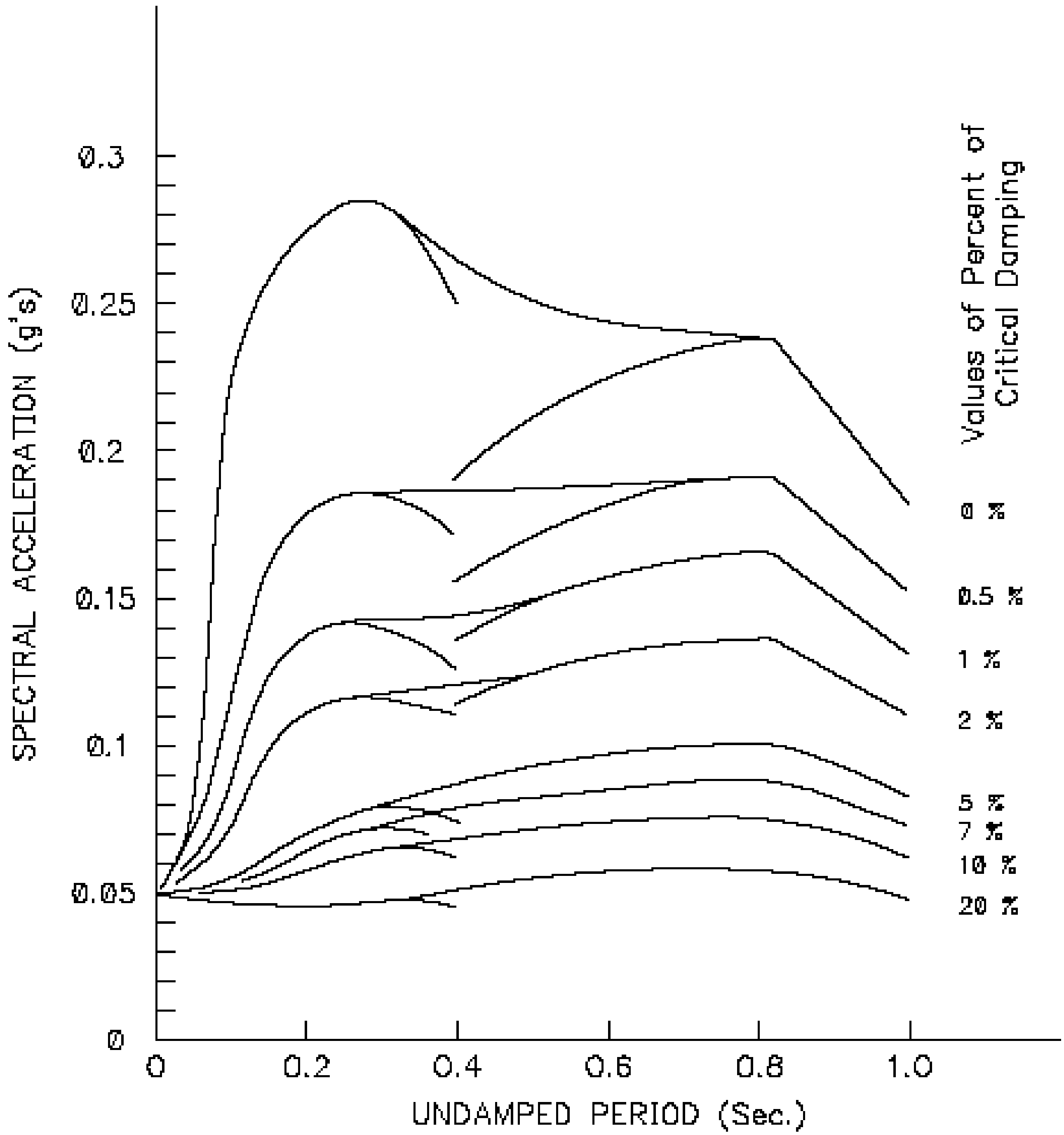


FIGURE 2-23, MAXIMUM HYPOTHETICAL ACCELERATION SPECTRA

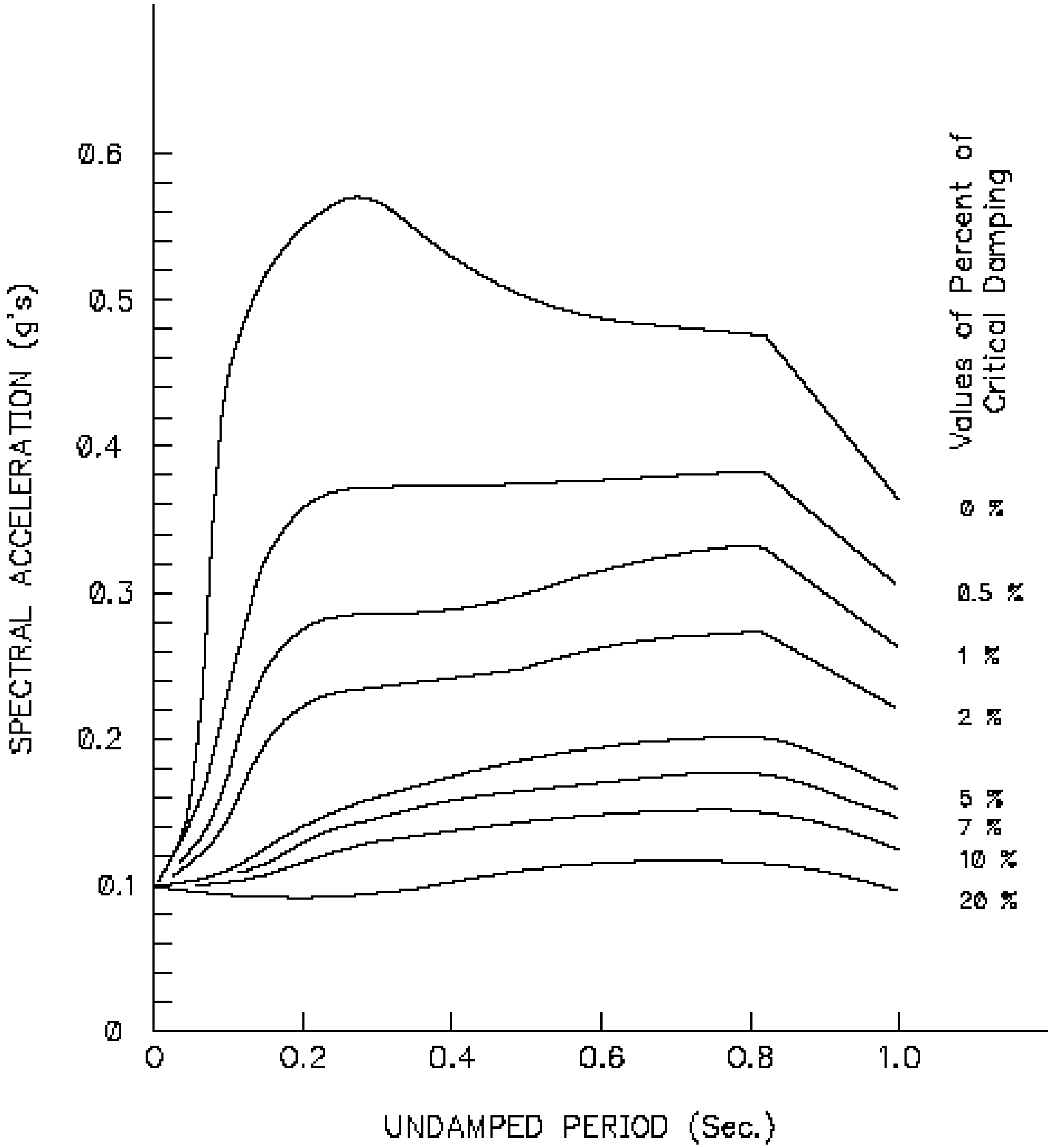


FIGURE 2-24, RESPONSE SPECTRUM FOR DESIGN

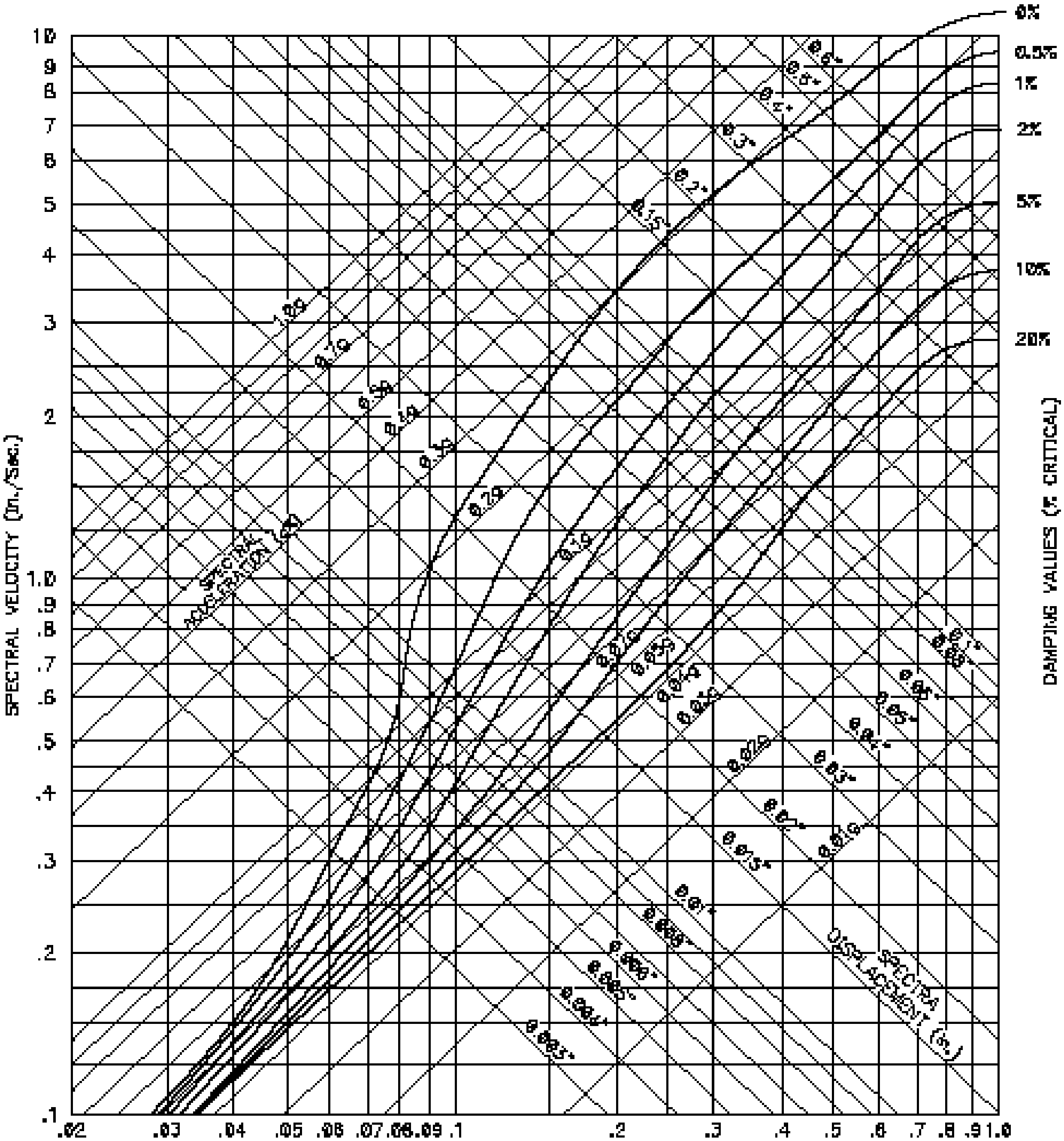


FIGURE 2-25, TEST BORING LOCATIONS

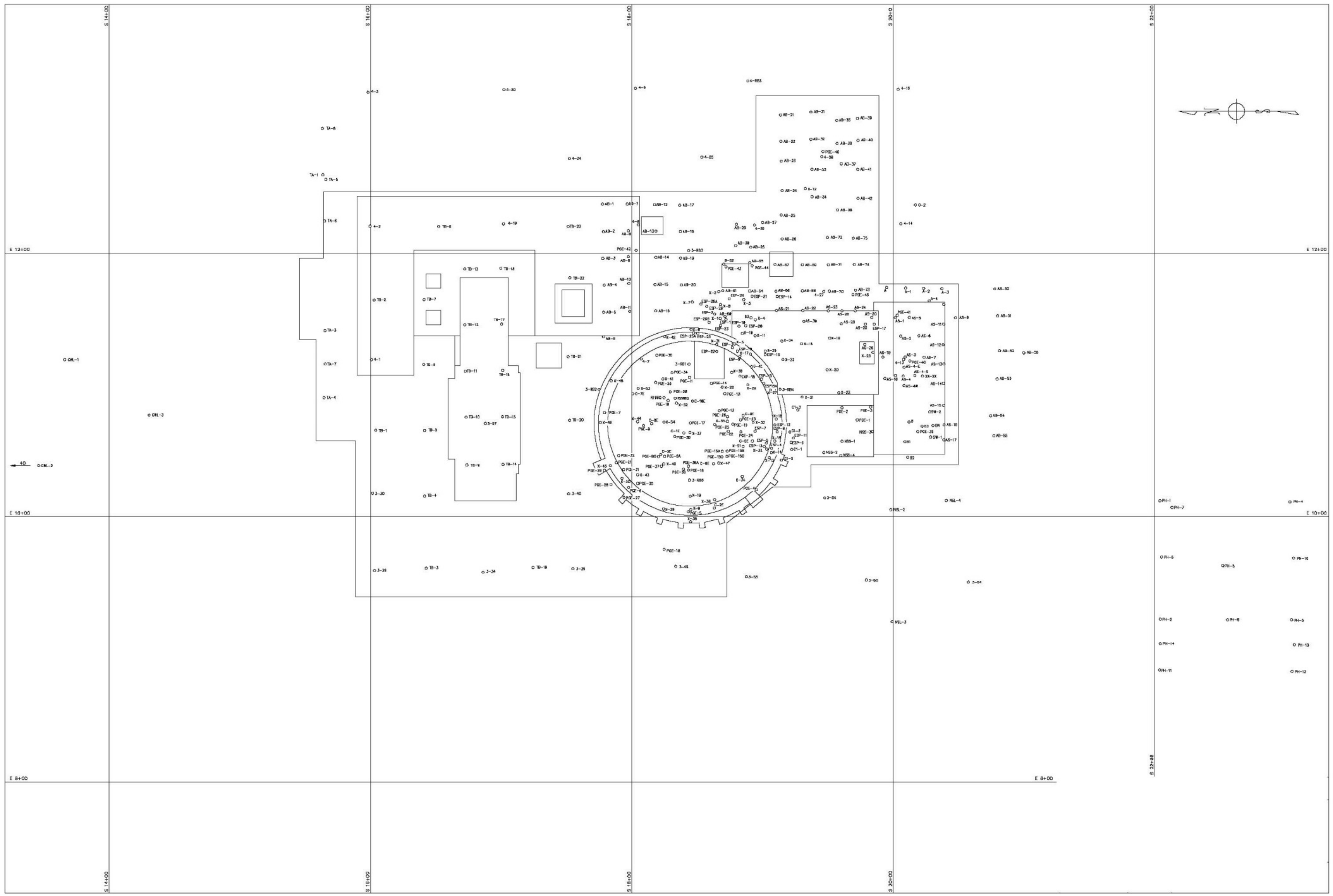


FIGURE 2-26, GEOLOGIC SECTION A-A

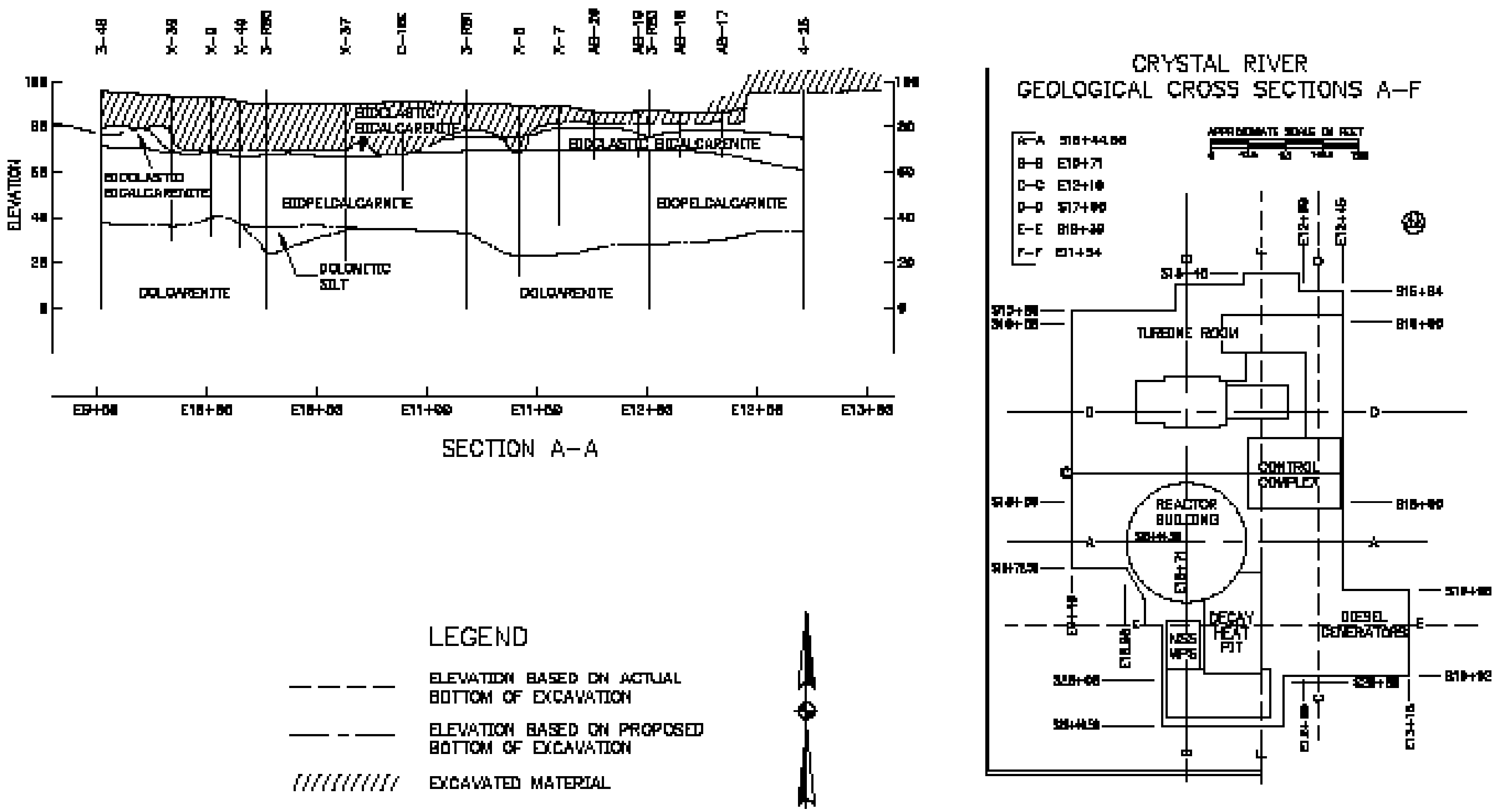


FIGURE 2-27, GEOLOGIC SECTION B-B

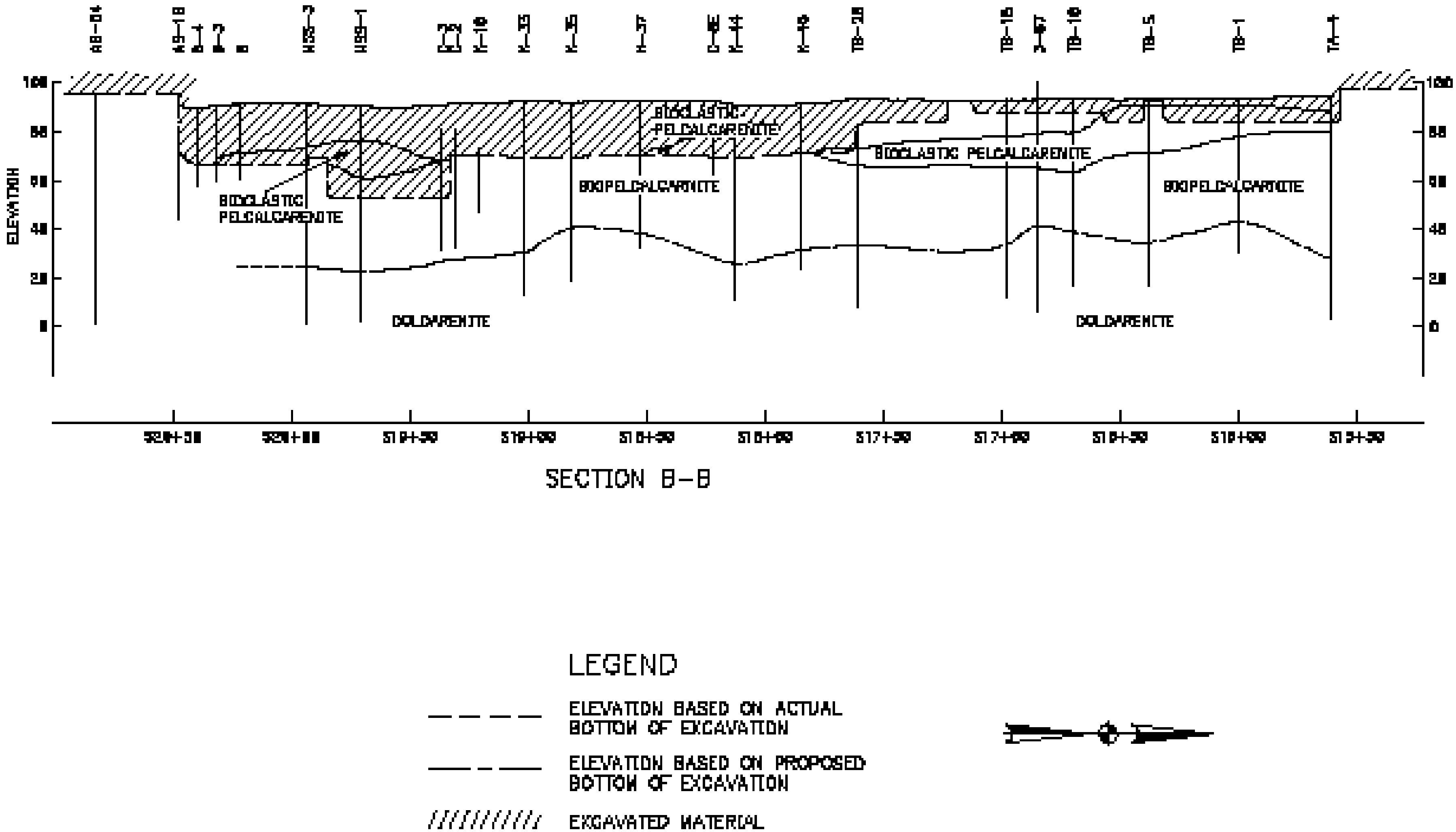


FIGURE 2-28, GEOLOGIC SECTION C-C

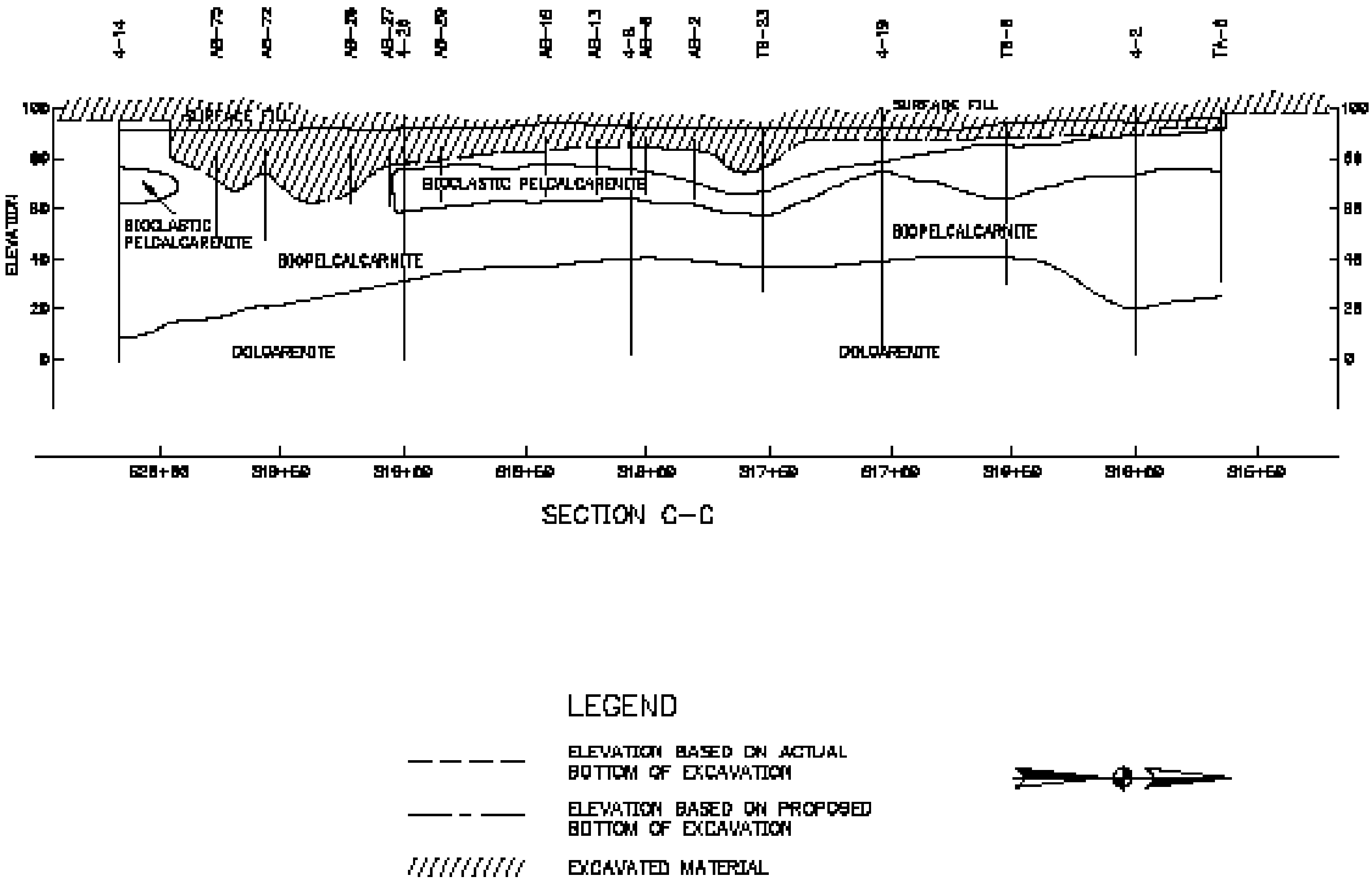


FIGURE 2-29, GEOLOGIC SECTION D-D

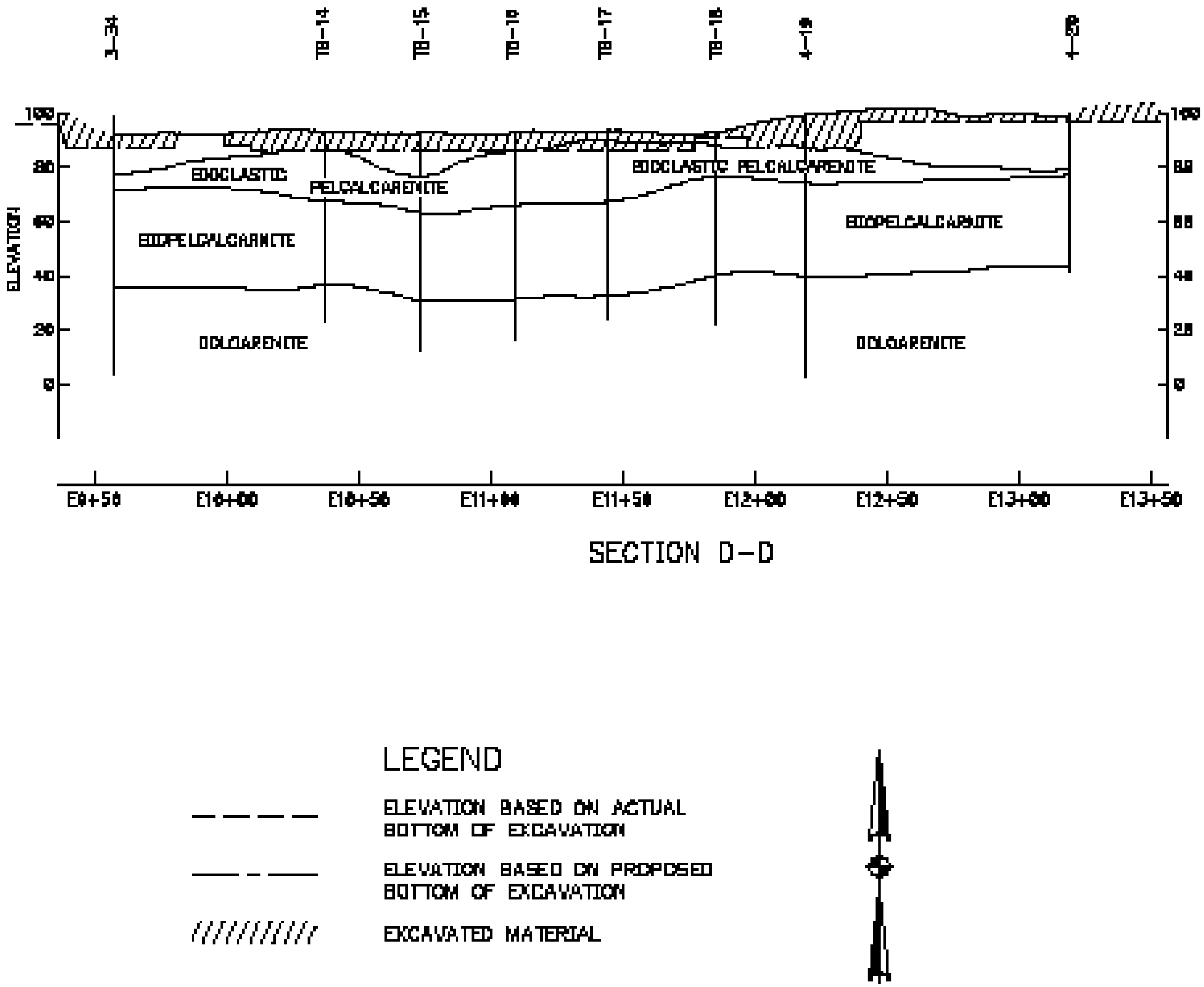


FIGURE 2-30, GEOLOGIC SECTION E-E

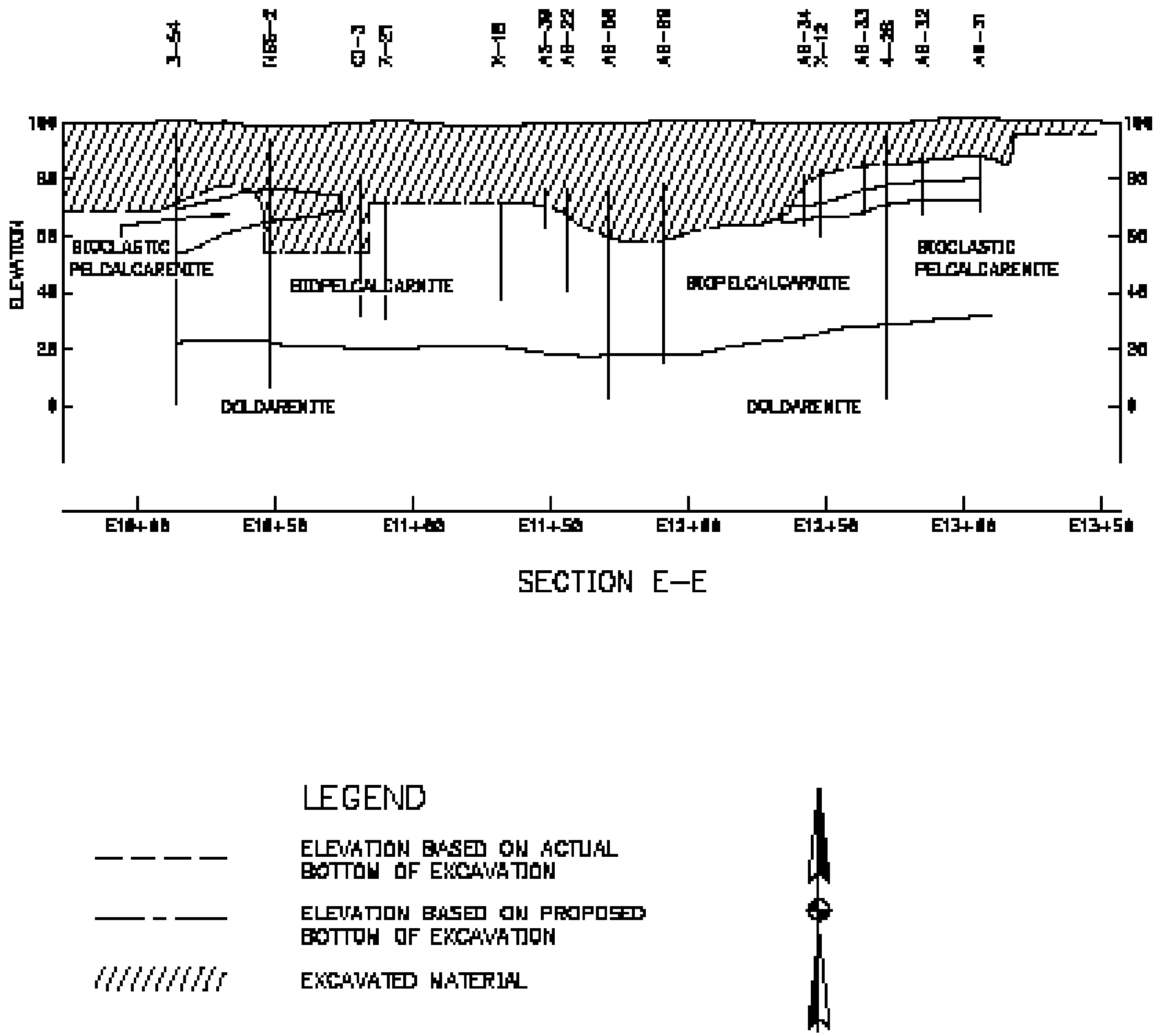


FIGURE 2-31, GEOLOGIC SECTION F-F

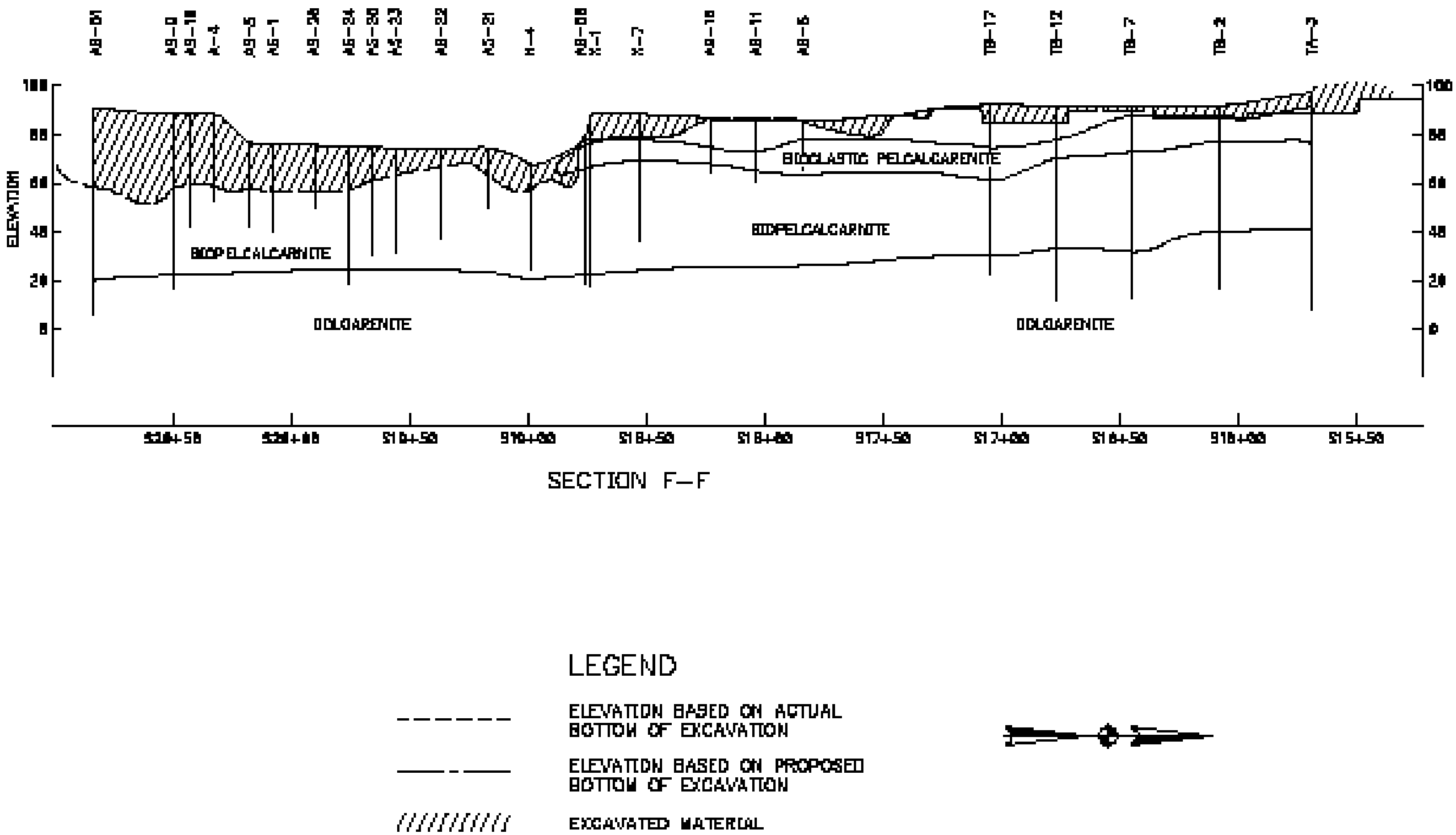


FIGURE 2-32, GEOLOGY OF FOUNDATION ROCK SYSTEM SHOWING CONTOURS OF COMPETENT BEARING MATERIAL, FRACTURE (JOINT) SYSTEM AND SURFICIAL INFILL MATERIAL

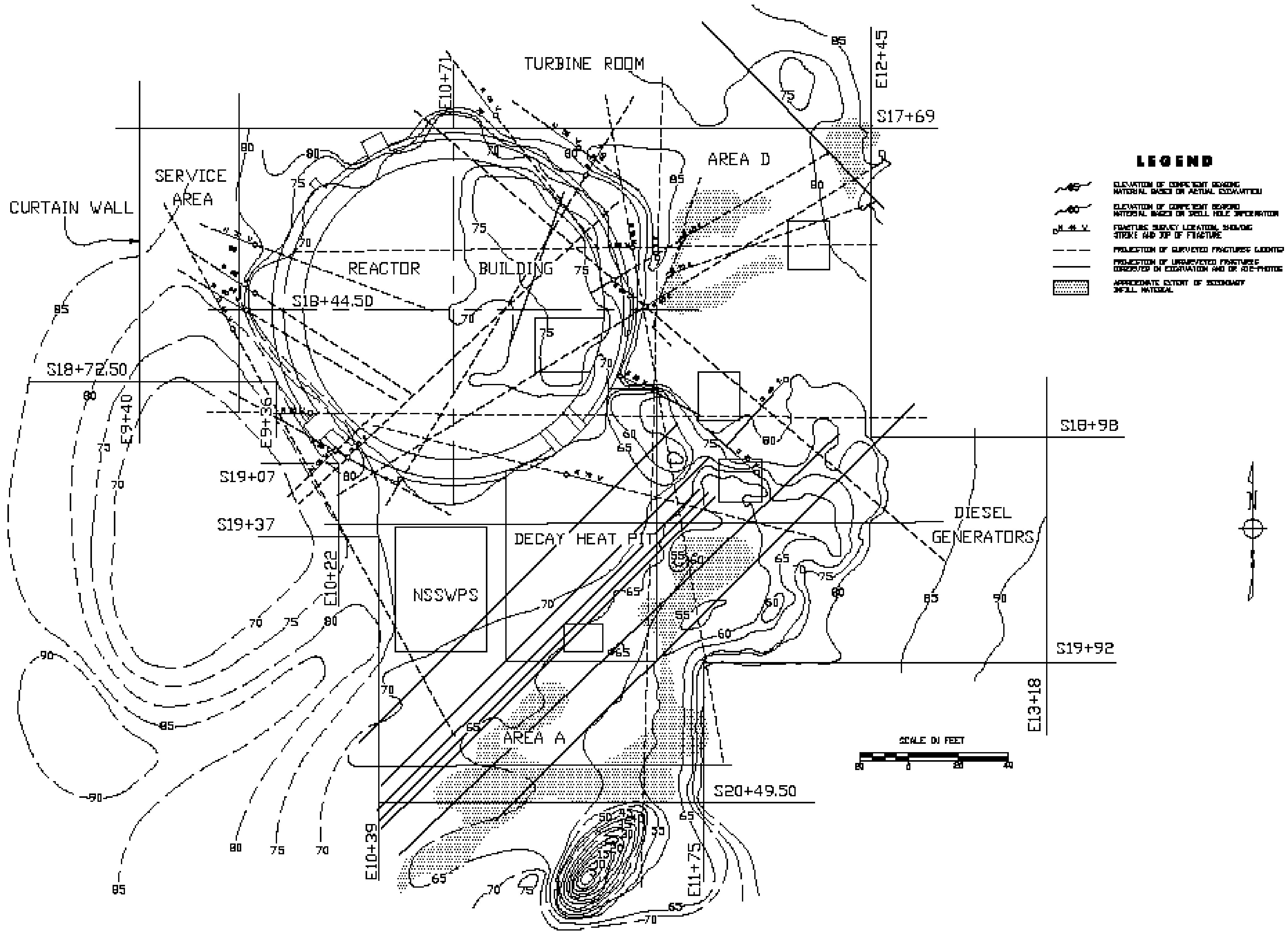


FIGURE 2-33, LOCATION OF CONE PENETROMETER TESTS

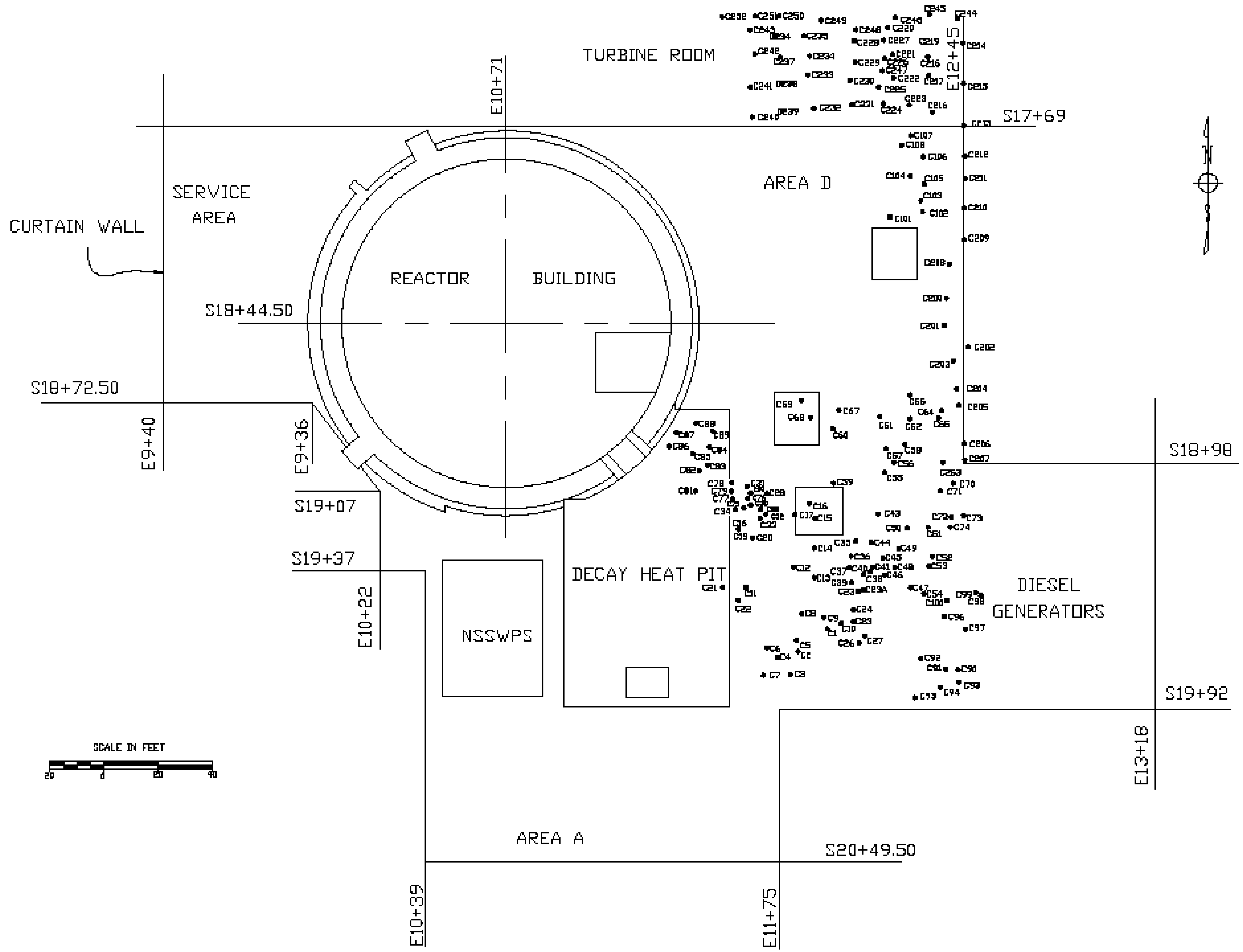


FIGURE 2-34, LOCATION OF PLUG SAMPLER TESTS

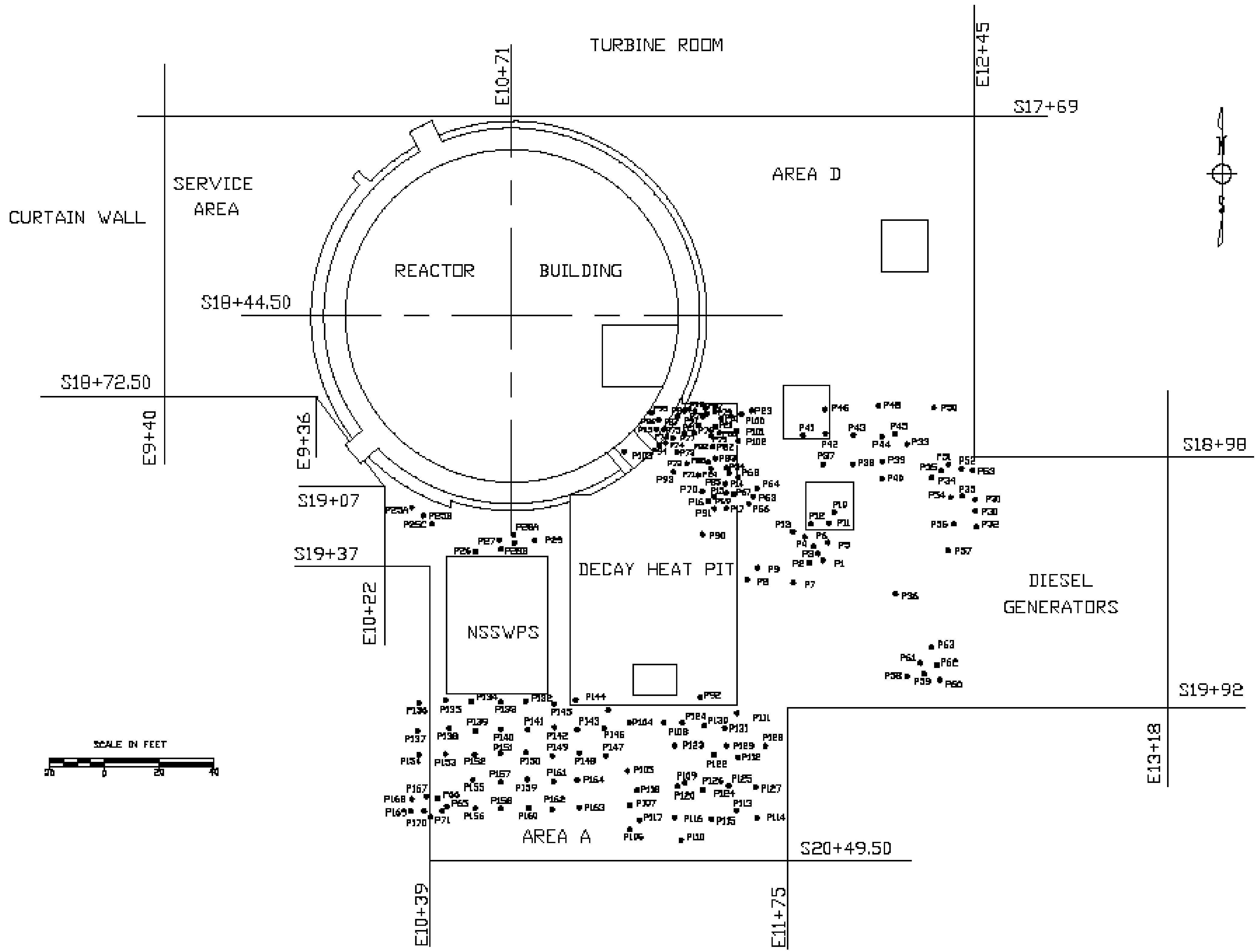
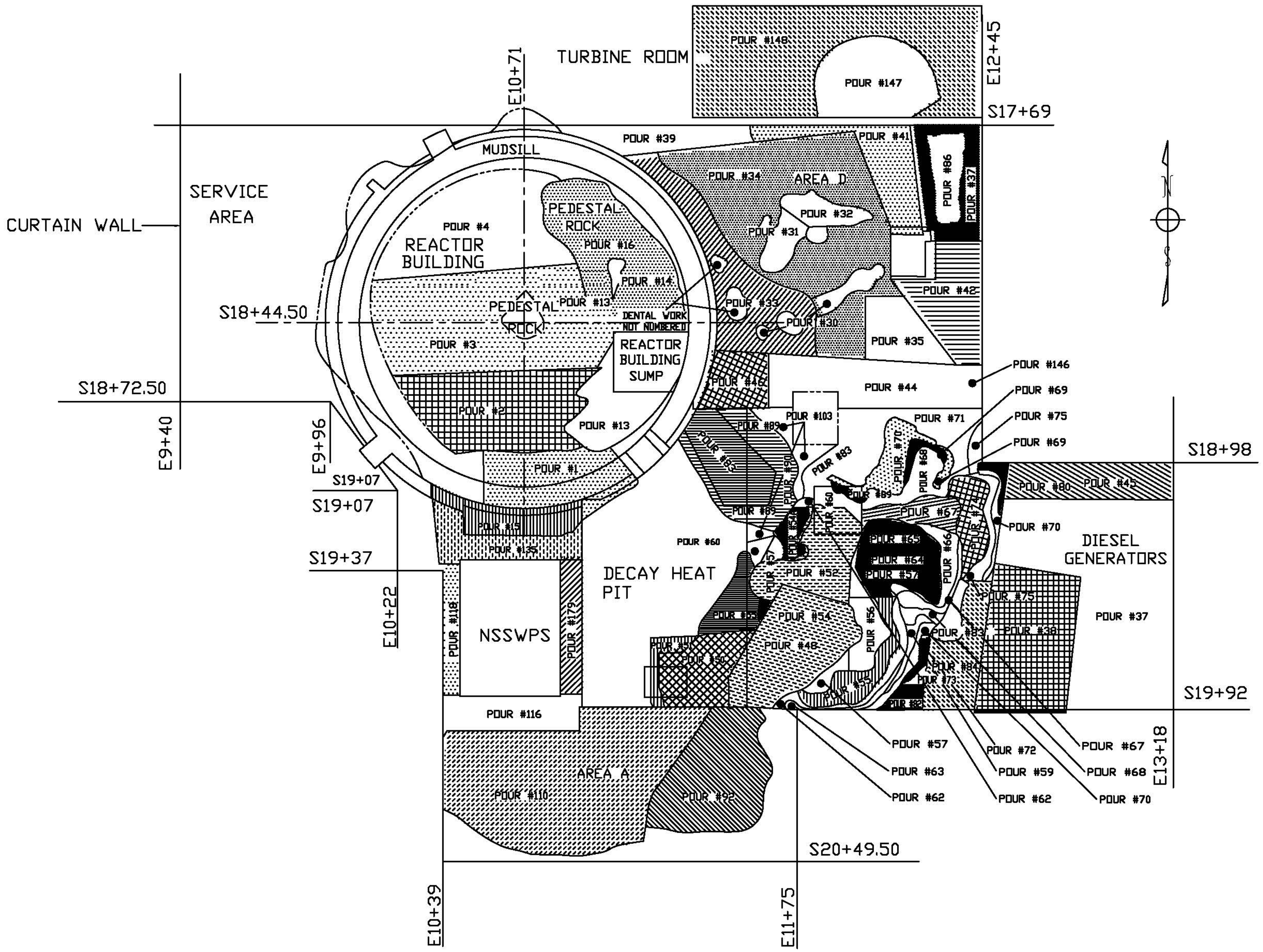


FIGURE 2-35, DELINEATION OF BACKFILL MATERIALS PLACED AGAINST IN-SITU FOUNDATION MATERIAL



GENERAL NOTES ON BACKFILL

REACTOR BUILDING

POUR NO. 1 POUR NO. 2 POUR NO. 3 POUR NO. 4	1500 PSI CONCRETE PLACED ON GROUTABLE LIMEROCK AGGREGATE BLANKET COVERING DIFFERENTIALLY CEMENTED LIMEROCK THIS ALSO APPLIES TO MUDSILL NORTH AND WEST OF PEDESTAL ROCK REMAINDER OF MUDSILL PLACED ON BARE COMPETENT MUDSILL LIMEROCK
POUR NO. 13	1500 PSI CONCRETE PLACED OVER CAP ROCK & PREVIOUS PLACED CONCRETE
POUR NO. 15	1500 PSI CONCRETE PLACED PARTIALLY ON POUR NO. 1: PARTIALLY ON BARE COMPETENT LIMEROCK AND PARTIALLY ON AGGREGATE POCKETS
POUR NO. 14	30:70 GROUT MIXTURE USED FOR DENTAL WORK
POUR NO. 16	1500 PSI CONCRETE PLACED OVER CAP ROCK
PEDESTAL ROCK,	BARE COMPETENT LIMESTONE COVERED WITH 1500 PSI CONCRETE POUR NUMBERS 9, 10, 13, 14, & 16 (NOT SHOWN)
REACTOR BLDG. SUMP	BARE NATIVE LIMEROCK COVERED WITH COMMON ACCESS FILL

AREA D

DENTAL WORK	
UNNUMBERED	30:70 GROUT MIXTURE
POUR NO. 30	30:70 GROUT AND 1500 PSI CONCRETE
POUR NO. 31	30:70 GROUT MIXTURE
POUR NO. 32	1500 PSI CONCRETE - SMALL AGGREGATE POCKET

CONCRETE BACKFILL

POUR NO. 33 POUR NO. 34 POUR NO. 35 POUR NO. 39 POUR NO. 41 POUR NO. 42 POUR NO. 44 POUR NO. 46 POUR NO. 82 POUR NO. 88 POUR NO. 103 POUR NO. 146	1500 PSI POURED AGAINST DENSE TO VERY DENSE MATERIAL AND COMPETENT LIMEROCK
POUR NO. 48 POUR NO. 50 POUR NO. 86 POUR NO. 60 POUR NO. 62 POUR NO. 63 POUR NO. 65 POUR NO. 66 POUR NO. 67	1500 PSI CONCRETE POURED PARTLY OVER DENSE TO VERY DENSE LIMEROCK AND PARTLY OVER PREVIOUSLY PLACED CONCRETE POUR NO. 68 POUR NO. 77 POUR NO. 148 POUR NO. 69 POUR NO. 79 POUR NO. 70 POUR NO. 80 POUR NO. 74 POUR NO. 84 POUR NO. 72 POUR NO. 87 POUR NO. 73 POUR NO. 89 POUR NO. 75 POUR NO. 90
POUR NO. 52 POUR NO. 54 POUR NO. 56 POUR NO. 59 POUR NO. 64 POUR NO. 74 POUR NO. 83	1500 PSI CONCRETE POURED OVER DENSE TO VERY DENSE LIMEROCK PARTLY OVER PREVIOUSLY PLACED CONCRETE & PARTLY OVER AGGREGATE BACKFILL

SCALE IN FEET

20 0 20 40

DIESEL GENERATOR AREA

POUR NO. 37	1500 PSI CONCRETE POURED AGAINST DENSE TO VERY DENSE MATERIAL AGGREGATE POCKET PLACE IN SOUTHEAST CORNER
POUR NO. 38	AS ABOVE: NO AGGREGATE POCKET
POUR NO. 45	1500 PSI CONCRETE PLACED AGAINST DENSE TO VERY DENSE MATERIAL AGGREGATE DRAIN PLACED IN NORTH EDGE

NSSWPS AREA

CONCRETE BACKFILL

POUR NO. 116 POUR NO. 118 POUR NO. 129 POUR NO. 135	5000 PSI CONCRETE POURED OVER DENSE TO VERY DENSE LIMEROCK

AREA "A"

CONCRETE BACKFILL

POUR NO. 92 POUR NO. 110	1500 PSI CONCRETE PLACED PARTLY OVER AGGREGATE BACKFILL

FIGURE 2-36, LOCATION OF ZONE I STRUCTURAL FILL

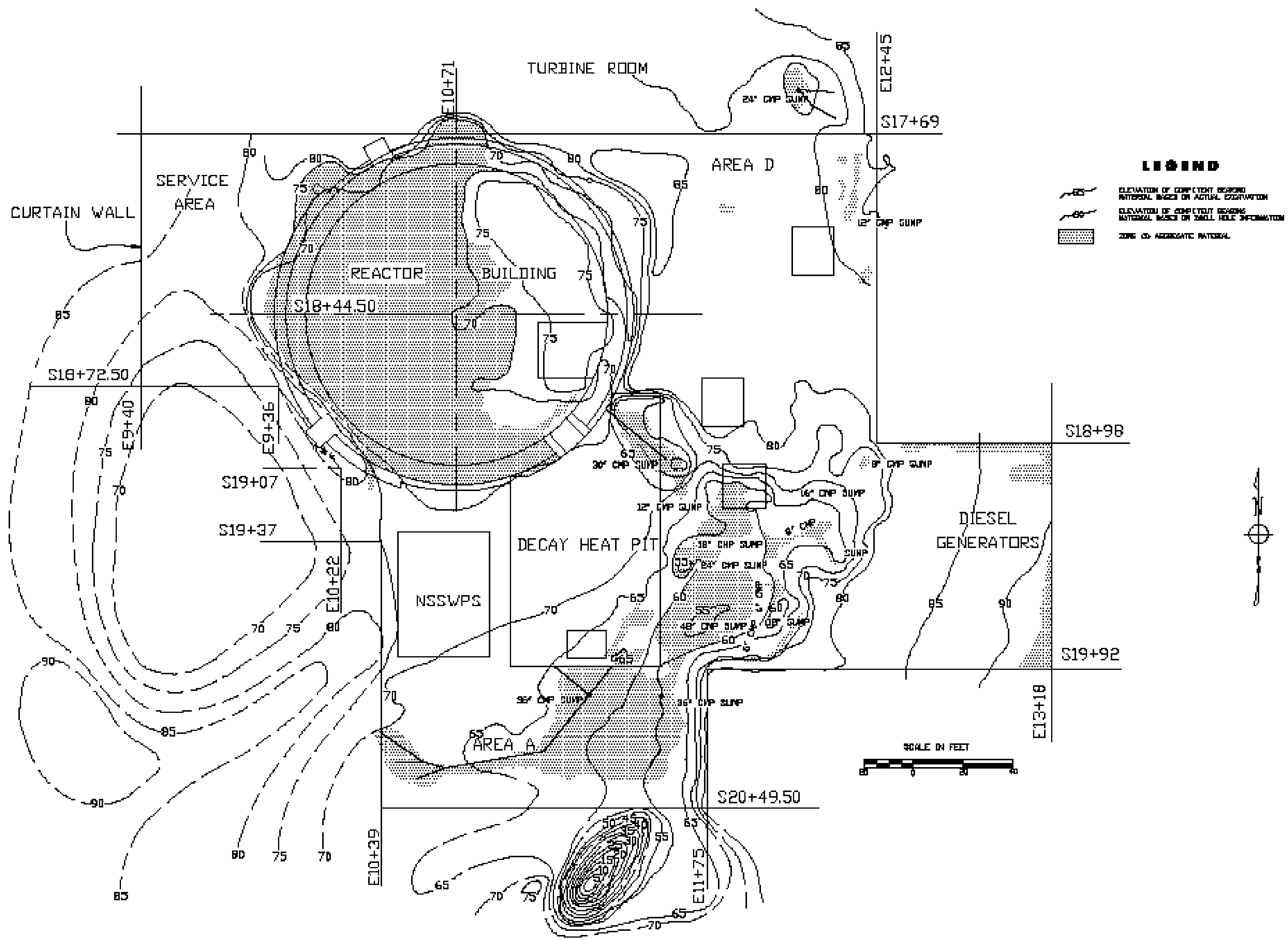


FIGURE 2-37, OUTLINE OF CHEMICAL GROUTING AREAS

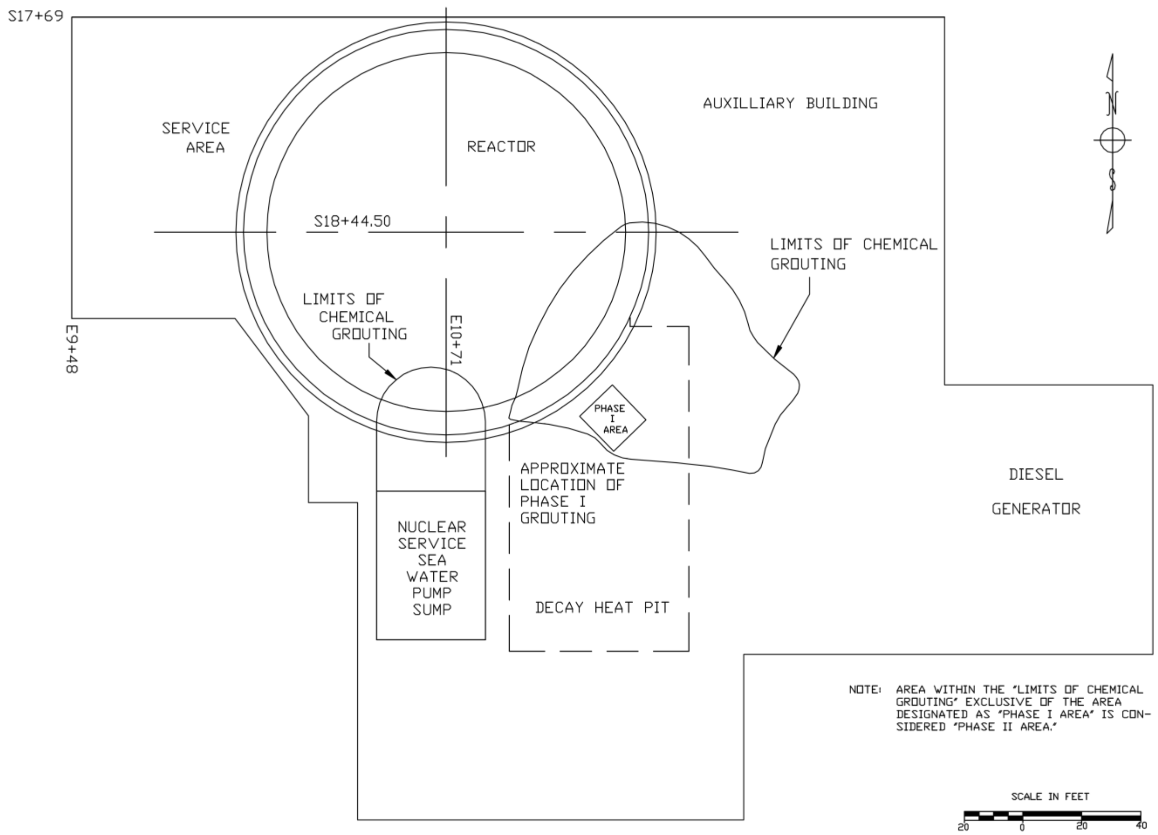


FIGURE 2-38, CUMULATIVE CHEMICAL GROUT TAKE PROFILE TEST AREA 1

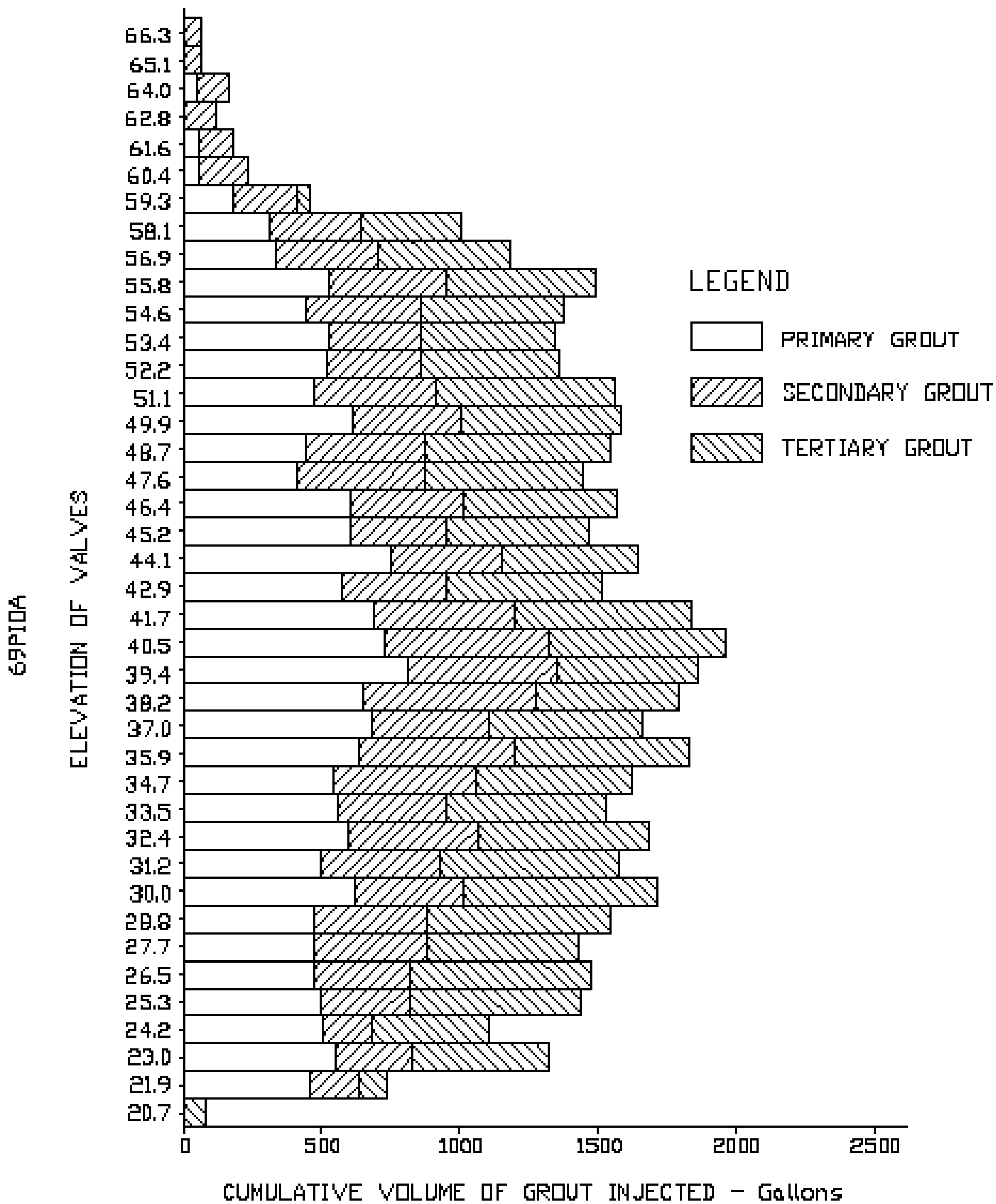


FIGURE 2-39, CUMULATIVE CHEMICAL GROUT TAKE PROFILE TEST AREA 2

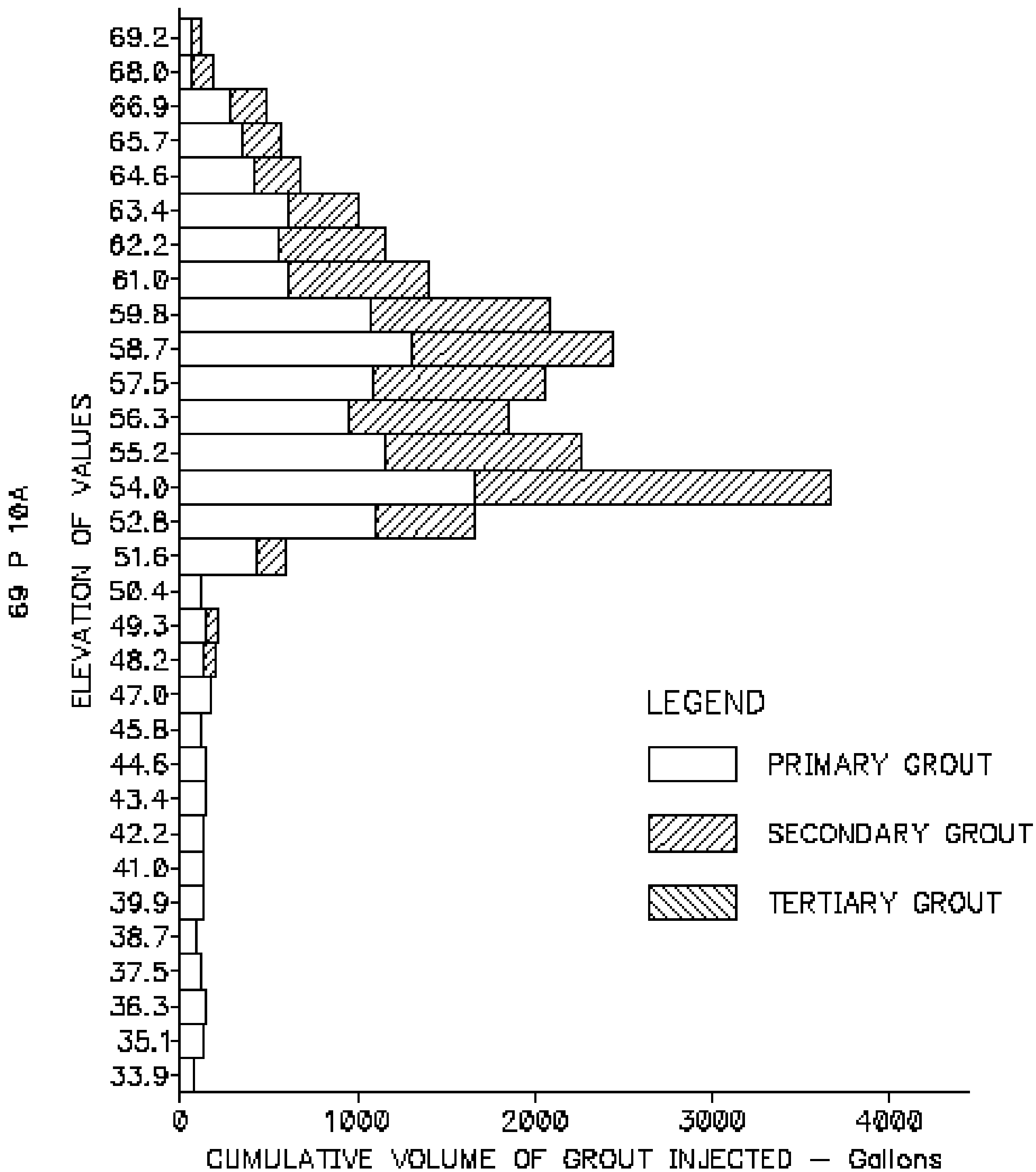


FIGURE 2-40, CUMULATIVE CHEMICAL GROUT TAKE PROFILE PRODUCTION AREA A

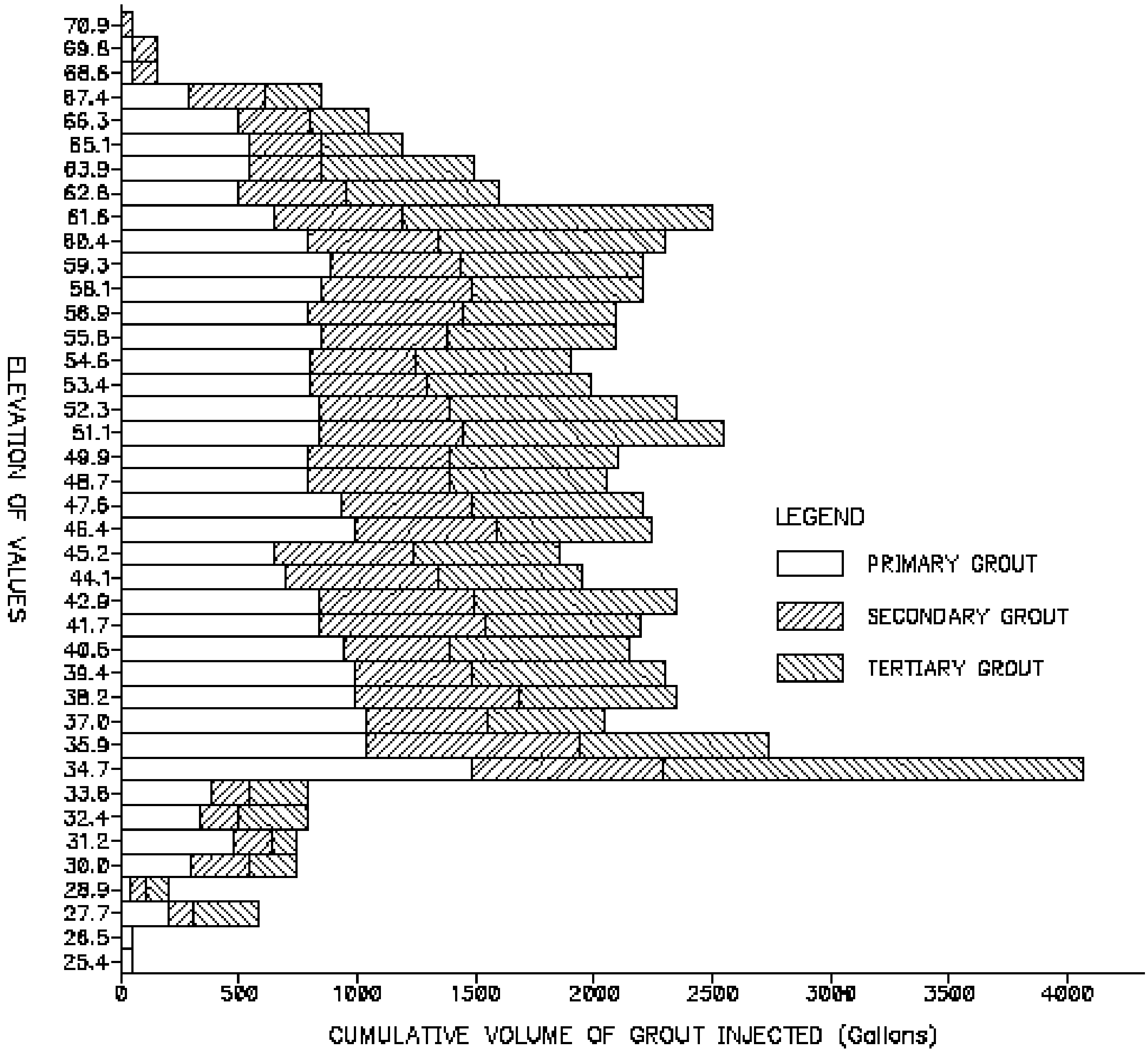
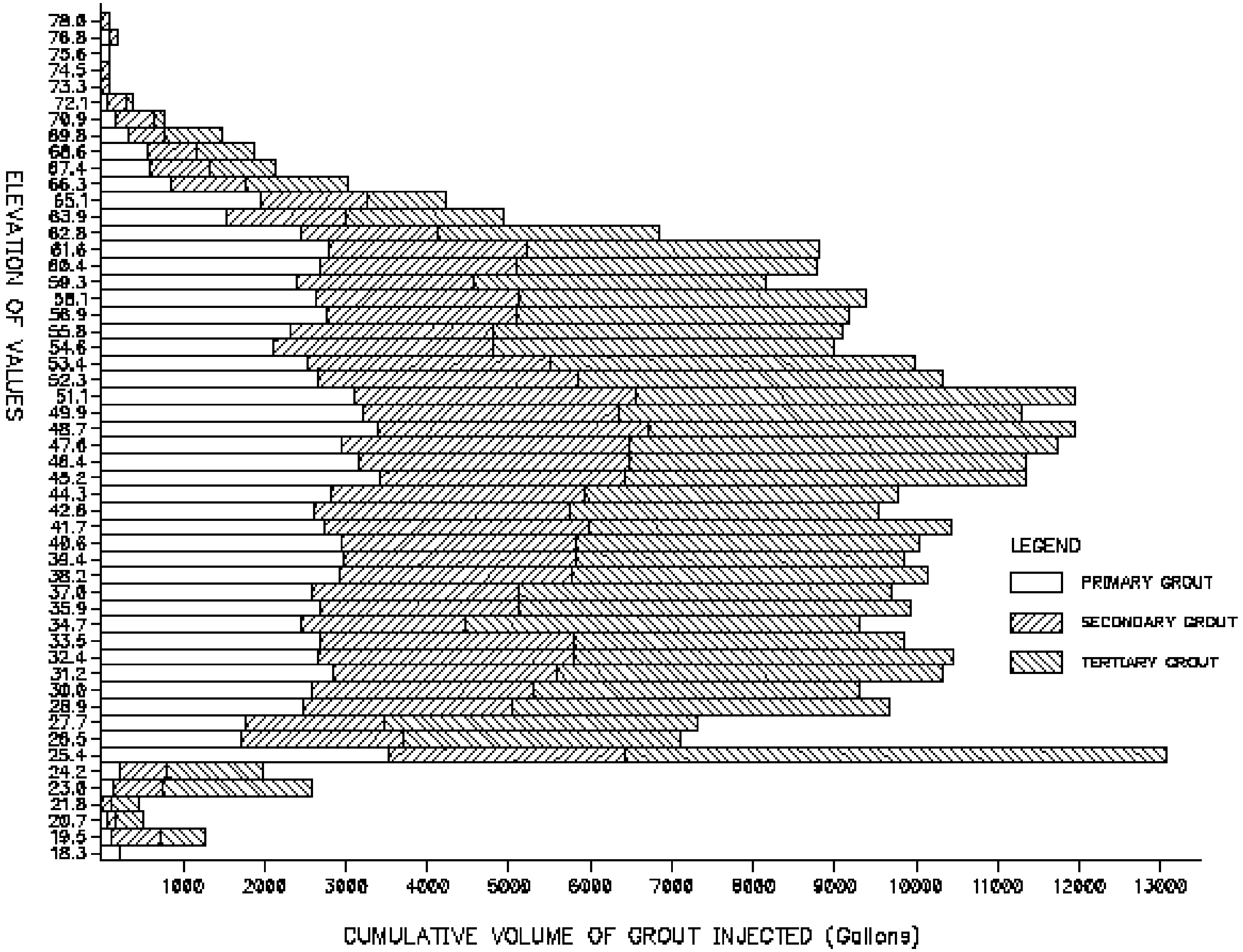


FIGURE 2-41, CUMULATIVE CHEMICAL GROUT TAKE PROFILE PRODUCTION AREA



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3. FACILITY DESIGN

This section describes the CR3 Facility Design. Included are Seismic Classifications and Design for Systems, Structures, and Components, as well as more detailed discussions of all systems determined to be important to the permanently defueled condition with all spent fuel in dry storage (ISFSI) and no new fuel on site.

The following codes and standards were used, as applicable, in the design, fabrication, and testing of components and structures associated with the auxiliary systems:

a. ASME Boiler and Pressure Vessel Code, Section II, Material Specification.
b. ASME Boiler and Pressure Vessel Code, Section III, Nuclear Vessels.
c. ASME Boiler and Pressure Vessel Code, Section VIII, Unfired Pressure Vessels and ASME Nuclear Case Interpretations.
d. ASME Boiler and Pressure Vessel Code, Section IX, Welding Qualifications.
e. Standards of the American Society for Testing Materials.
f. USAS, B31.1 and USA Standard Code for Nuclear Power Piping USAS 31.7.
g. USASI, C50.20-1954 Test Code for Polyphase Induction Motors and Generators.
h. USASI, C50.2-1955 for Alternating Current Motors, Induction Machines, and General and Universal Motors.
i. Standards of the Institute of Electrical and Electronics Engineers.
j. Standards of the National Electrical Manufacturers Association.
k. Hydraulic Institute Standards.
l. Heating, Ventilating, and Air Conditioning Guide; American Society of Heating, Refrigerating and Air Conditioning Engineers.
m. Standards of Tubular Exchanger Manufacturers Association.
n. Air Moving and Conditioning Association.
o. Manufacturers Standardization Society.
p. American Standards Association.
q. Standards of the American Water Works Association.

To assist in review of the system drawings, a standard set of symbols and abbreviations has been used and is summarized in Figure 3-17a.

3.1 STRUCTURAL DESIGN CLASSIFICATION

Containment for Crystal River Unit 3 was provided by the reactor building, including its steel liner and the Reactor Building Isolation Systems.

Other special structures are those structures which house equipment which was vital to monitoring containment integrity, vital to safe shutdown of the reactor, or contain significant quantities of radioactive materials.

The plant structures, components, and systems were originally designed and classified according to their function and the degree of integrity required to protect the public. Due to decommissioning and the transfer of all spent fuel to dry storage, the function of all plant structures, systems, and components has changed.

3.1.1 CLASSES OF STRUCTURES AND SYSTEMS

3.1.1.1 Class I

Those structures, components, and systems, including instruments and controls, whose failure might have caused an uncontrolled release of radioactivity are designated Class I. When a system as a whole is referred to as Class I, certain portions not associated with loss-of-function of the systems may have been designated Class II or III, as appropriate. A listing of Class I structures, components, and systems follows:

<u>Class I - Structures, Components, and Systems</u>

a. Building and Structures

   ◊ None

b. Miscellaneous Systems and Components

   ◊ None

3.1.1.1.1 Class I*

Structures, components, and systems designated as Class I* have been designed in accordance with the criteria in Section 3.2.1.1.6 to prevent seismic falldown only. These structures, components, and systems no longer serve an active safety function but must not fail and cause damage to Class I SSC.

3.1.1.2 Class II

Those structures, components, and systems which are important to reactor operation but not essential to safe shutdown and isolation of the reactor, and whose failure would not result in the release of substantial amounts of radioactivity, are designated Class II.

3.1.1.3 Class III

The balance of structures, components, and systems are designated Class III.

3.1.2 SEISMIC DESIGN BASES

3.1.2.1 Class I Design Bases

Structures, components, and systems designated as Class I have been designed in accordance with the criteria in Section 3.2.1.1.6.

3.1.2.2 Class I* Design Bases

Structures, components, and systems designated as Class I* have been designed in accordance with the criteria in Section 3.2.1.1.6 to prevent falldown only.

3.1.2.3 Class II Design Bases

Structures, components, and systems designated as Class II have been designed for the ground accelerations as described in Section 3.2.1.1.6.a, or in accordance with the recommendations of the Uniform Building Code (Reference 1).

3.2 REACTOR BUILDING (HISTORICAL)

The reactor building is a concrete structure with a cylindrical wall, a flat foundation mat, and a shallow dome roof. The foundation slab is reinforced with conventional mild-steel reinforcing. The cylinder wall is prestressed with a post-tensioning system in the vertical and horizontal directions. The dome roof is prestressed utilizing a three-way post-tensioning system. The inside surface of the reactor building is lined with a carbon steel liner to ensure a high degree of leak tightness during operating and accident conditions. Nominal liner plate thickness is 3/8 inch for the cylinder and dome and 1/4 inch for the base.

The foundation mat is bearing on competent bearing material and is 12-½ feet thick with a 2 feet thick concrete slab above the bottom liner plate. The cylinder portion has an inside diameter of 130 feet, wall thickness of 3 feet 6 inches, and a height of 157 feet from the top of the foundation mat to the spring line. The shallow dome roof has a large radius of 110 feet, a transition radius of 20 feet 6 inches, and a thickness of 3 feet.

The reactor building has been designed to maintain structural integrity and not fail in any manner that would damage spent fuel. The liner has been anchored to the concrete so as to ensure composite action with the concrete shell.

The design and construction of the reactor building has been given a thorough re-evaluation subsequent to the discovery on April 14, 1976, of a delaminated condition in the dome. The upper part (approximately 12 inches thick) of the 3 feet design concrete thickness separated from the lower part of the dome structure parallel to the membrane over an approximate diameter of 105 inches. Extensive analytical and field investigations were conducted to establish an acceptable repair program. This repair program included removal of the upper part of the dome, placement of non-prestressed reinforcing steel mats, installation of radial reinforcement, and placement of concrete to restore the dome to a thickness of 3 feet. Details of the delaminated condition of the dome, re-evaluations of the dome, and the dome repair program are described in the report: "Final Report - Reactor Building Dome Delamination," December 10, 1976.

The current condition of the containment building shell is the result of a sequence of events outlined in the below paragraphs. The evaluation of the current condition is discussed below (Section 3.2.4).

In support of the steam generator replacement (SGR) project (Fall 2009) a temporary access opening was created in the post-tensioned, reinforced concrete wall and the interior steel liner plate at azimuth 150 degrees (Bay 34). The concrete opening measured approximately 25’-0” wide x 27’-0” high, and the liner plate opening measured 23’-6” wide x 24’-9” high. The bottom of the concrete opening was located at elevation 183’0”, and the bottom of the liner plate at elevation 184’0”. Creation of the SGR Opening required the removal of the concrete, rebar, tendons, tendon sheaths and liner plate within the boundaries of the Opening, and detensioning of selected vertical and horizontal tendons adjacent to the Opening. During concrete removal an area of delamination was identified located at a depth of approximately 10” from the outer surface of the 42” thick containment wall. Subsequent field investigations determined that the delamination occurred only in Bay 34. Repair of the delamination required detensioning additional hoop and vertical tendons and the removal of additional concrete from Bay 34. Additional detensioning resulted in vertical cracks in other areas of the structure as discussed in NCR 00395843 (Reference 93). Final restoration required installing new non-pre-stressed reinforcing steel mats and radial reinforcement in Bay 34, and placement of concrete in lifts to restore the containment wall to its as designed configuration. Additional details on the repair of Bay 34 is contained in EC 275220 (Reference 85).

Following restoration of Bay 34, crews began re-tensioning the tendons with a prescribed sequence as outlined in EC 275221 (Reference 86). On March 14, 2011, shortly after completion of the 100th horizontal tendon re-tensioning step, a delamination occurred in Bay 56. This event is documented in NCR 453318 (Reference 94).

Re-tensioning was suspended at that point to examine the condition of the building (via Impulse Response testing and core bore inspections) and allow repair plans to be developed. In addition to the major delamination in Bay 56, several localized areas of concern were discovered in the other bays.

In early July, 2011, the delamination in Bay 56 expanded to a lower elevation as documented in NCR 475123 (Reference 95), and on July 26, 2011, Bay 12 also delaminated as documented in NCR 478650 (Reference 96). This later delamination resulted in concrete spalling off and falling inside the Intermediate Building.

In addition to these two major delamination events, additional NDE scans performed in Bay 23, Bay 61, and Bay 45 showed the extent of cracking in these bays was expanding between the scans performed after the March event. The extent of condition of these other Bays is documented in EC 288572 (Reference 97).

In August, 2011, EC 282247 (Reference 87) was issued to detension all vertical tendons to approximately 75% of Design Load where the design load was between 70 and 74% of Guaranteed Ultimate Tensile Strength (GUTS).

3.2.1 STRUCTURAL DESIGN PARAMETERS

The reactor building is to be reclassified as a Class I* structure. Due consideration was given to the dead load, live load, hurricane wind loading and seismic loadings.

The design criteria for the reactor building are covered in Section 3.2.3.

3.2.1.1 Design Loads

The design loads for the reactor building have been determined based on operating and Seismic I* requirements. See Section 3.2.4 for additional discussion on the design loads used for the current condition of the containment building shell.

3.2.1.1.1 Dead Load

The dead load consists of the weight of the complete structure.

3.2.1.1.2 Prestress Load

The concrete shell has been left in a partially prestressed condition sufficient to ensure stability and reduce the risk of damage to spent fuel. The current condition of the containment building shell is outlined in Section 3.2.4. below.

3.2.1.1.3 Live Load

Applicable loads on the reactor building shell due to factored loads are:

		Factored Loads
a.	Pipe penetration	Pipe break & earthquake
b.	Pipe supports	Pipe break & earthquake
c.	Polar crane	Earthquake

3.2.1.1.4 Wind Load

The wind velocity, as a function of height and drag coefficients, has been established on the basis of ASCE Paper No. 3269 (Reference 2).

The basic wind velocity (wind 30 feet above grade) has been based on the fastest mile of wind with a 100 year period of recurrence.

3.2.1.1.5 Tornado Load

Class I structures have been designed to withstand short term tornado loadings, including tornado generated missiles. The tornado design requirements are:

a. Tangential wind velocity of 300 mph.
b. An external pressure drop of 3 psig.
c. Missile equivalent to a utility pole 35 feet long, 14 inches in diameter, density of 50 lb./cu.ft. and traveling at 150 mph.
d. Missile equivalent to a one ton automobile traveling at 150 mph.

A 300 mph wind has been applied in accordance with standard wind design practice and utilizing applicable pressures, shape factors, and drag coefficients from ASCE Paper No. 3269. The pressure drop of 3 psig is conservative considering that most measured pressure drops have been in the order of magnitude of 1.5 psig.

The effect of the following tornado-borne missiles was also analyzed:

a. A 4 inch by 12 inch by 12 feet long wooden plank traveling end-on at 300 mph.
b. A missile equivalent to a 3 inch diameter schedule 40 pipe, 10 feet long, traveling end-on at 100 mph.

It is obvious from the following comparison data that they do not have significant effect on the structural integrity of the Class I structures. The spectrum of potential tornado missiles with its kinetic energy and penetration depth is shown in the following data:

Potential Tornado Missiles - Comparison of Kinetic Energy and Penetration					
Tornado Missile	**<u>Geometric Properties</u>** **L = Length** **D = Diameter** **A = Minimum Cross Sectional Area**	**Density P(lb/ft^3) or Weight W(lb)**	**Velocity (mph)**	**Kinetic Energy (ft.-lb)**	**Penetration Depth Into Concrete (inches)**
1. Utility Pole	L = 35 feet D = 14 inch	P = 50	150	1,415,000	2.5
2. Compact Auto	A = 6.25 ft^2	W = 2000	150	1,505,000	6.2
3. 3 inch Schedule 40 pipe (piece)	L=10 feet	W = 75.8	100	25,300	0.3
4. Wood Plank	4 in x 12 in L = 12 feet	W = 108	300	325,000	1.3

A comparison of values in the above data indicates that two missiles are critical for design. The 14 inch diameter utility pole is critical because of smaller end dimension with fairly large kinetic energy. The compact auto is critical because of possessing the highest kinetic energy and high penetration depth.

It is predicted that maximum penetration depth for the worst probable missile is 6.2 inches against the exterior concrete wall thickness of safety related structures which is 24 inch minimum. As such, it is concluded that secondary missiles will not be generated within the structure which could damage the safety related equipment and systems.

3.2.1.1.6 Earthquake Load

The site seismology and response spectra are described in Chapter 2.

The seismic design of Class I and Class I* structures is based on the response to a ground acceleration as described below. Class II structures are typically designed to Section 3.2.1.1.6.a only.

a. Primary steady state stresses, when combined with the seismic stress resulting from the response to a ground acceleration of 0.05g acting horizontally and 0.033g acting vertically and occurring simultaneously, have been maintained within the allowable working stress limits accepted as good practice and, where applicable, set forth in the appropriate design standards, e.g., ASME Boiler and Pressure Vessel Code (Reference 3), ACI 318-63 (Reference 4), AISC Specification for the Design and Erection of Structural Steel for Buildings (Reference 5), and USAS (ANSI) B31.1 (Reference 6).
b. Primary steady state stress, when combined with the seismic stress resulting from the response to a ground acceleration of 0.10g acting horizontally and 0.067g acting vertically and occurring simultaneously, has been limited so that the function of the structure is not impaired in protecting spent fuel or spent fuel cooling SSCs.

The respective vertical and horizontal seismic components at any point on the shell have been added by summing the absolute values of the response (i.e., stress, shear, moment, or deflection) of each contributing frequency due to vertical motion to the corresponding absolute values of the response of each contributing frequency due to horizontal motion.

3.2.1.1.7 Groundwater and Floods

The foundation slab design took into consideration groundwater pressure. Fluctuations in the groundwater due to flood and normal variation have been given due consideration in designing the foundation mat (see Section 2.4 and Section 2.5).

3.2.2 MATERIAL SPECIFICATIONS AND QUALITY CONTROL

3.2.2.1 Corrosion Protection

The reactor building is protected against external corrosive influences by the following means:

a. The tendon access gallery at the base of the cylinder includes a drainage system.
b. The reactor building is completely surrounded by either a circular retaining wall or adjacent structures extending from the top of the foundation to at least elevation 119 feet. The retaining wall provides access space for stressing and inspecting tendons. At the base of the retaining wall between the wall and the reactor building, a drainage system was provided to keep water from accumulating in the access space. Therefore, waterproofing of the lower portion of the cylinder

was not required. The retaining wall and adjacent structures preclude contact of groundwater with the shell.

c. A cover of concrete over reinforcement was provided in excess of that for normal construction as exemplified by code requirements.

d. A water-tight galvanized steel conduit for tendons was used with the added precaution of a thicker walled, rigid steel conduit in the base mat and extending immediately above into the cylinder.

e. An inboard-oriented haunch results in only nominal tensile stresses occurring at the outer fibers. They are within the capacity of concrete in tension, thus eliminating potential cracking that may allow ingress of water.

The outer surface of the steel is in direct contact with the concrete, which provides adequate corrosion protection because of the alkaline properties of the concrete.

The tendons were inserted in galvanized steel conduit embedded in the concrete which provided an additional water barrier, as well as an electrical shield against stray currents. The inner surface of the steel conduit, as well as the tendons including anchorages, was protected with a heavy wax base corrosion inhibitor. The end anchorages were covered with a gasketed metal container. The retaining wall and drainage system around the reactor building provide excellent protection for the liner and tendons against corrosion. Metallic components including the liner plate and tendon conduit were electrically connected to prevent stray current corrosion.

3.2.3 STRUCTURAL DESIGN CRITERIA

In its current condition the CR3 containment structure is not qualified to the ACI 318-63 code. All un-delaminated areas of the structure are considered acceptable; however, presumed intact concrete in delaminated regions is experiencing compression in excess of working stress design code allowables. The analysis of the containment building shell in the current condition is discussed below in Section 3.2.4.

3.2.4 METHOD OF ANALYSIS

3.2.4.1 Reactor Building Concrete Shell

The current condition of the reactor building shell is outlined in Section 3.2. The current condition of the reactor building shell essentially contains the following tendon prestress condition:

- The dome has 18 tendons that were left at 50% Design Load (35% GUTS) following the dome repair in 1976. The remaining 105 tendons were left at 100% Design Load (70% GUTS).
- All of the vertical tendons are currently at 75% Design Load (52.5% GUTS).
- Thirty horizontal (hoop) tendons are at 50% Design Load (35% GUTS).
- The remaining hoop tendons are at 100% Design Load (70% GUTS) or were tensioned during original construction and have not been re-tensioned since.

The overall evaluation of the current condition consists of several analysis/calculations. These calculation use a combination of ANSYS modeling and more traditional analysis techniques to show that the damaged sections of the containment shell are not code compliant but, this is considered acceptable for the current shut down condition of the plant. The analysis discussed above uses OBE seismic loading (Section 3.2.1.1.6.a) compared to Working Stress Allowable.

3.2.4.2 Reactor Building Liner

The reactor building liner anchors are vertical angles and are spaced horizontally at 18 inches center-to-center. For the analysis of the current condition of the reactor building shell, the liner is not specifically modeled as a structural element but included in the dead weight calculation.

3.2.5 RESULTS OF ANALYSIS AND DESIGN

3.2.5.1 Results of Analysis

The results of the analysis discussed in Section 3.2.4 indicate the containment shell is damaged but will not fail in a manner that would impact spent fuel.

3.2.5.2 Design

3.2.5.2.1 Reactor Building Concrete Shell

The Reactor Building Shell has been checked for the factored loads and load combinations as outlined in Analysis/Calculations S13-0009 (Reference 98) and S13-0010 (Reference 99). These analysis show the containment is not code qualified.

In summary, the analysis described in the above referenced analysis/calculations show that while the containment shell is damaged it will not fail in a manner that impacts equipment within the building.

3.2.5.2.1.1 Polar Crane

The structural components of the polar crane are Class I*.

In order to ensure stability during an earthquake, the crane trolley and bridge are restrained at all times. The containment polar crane is prevented from dislodging from the crane rails by the method indicated in Figure 3-3.

The standard WF-section used as a crane runway girder bracket has been welded directly to a continuous bent plate ring that is thicker than the liner (see Figure 3-4). The wheel loads from the polar crane are carried by the runway girder and transferred to the brackets. The bracket moment and shear due to the wheel loads are resisted by two built-up tees embedded in the wall. The shear is transferred from the bracket web plate, into the stiffener plates, and resisted by bearing on the tee web plates.

The horizontal load components are transferred directly, at the elevation of the top flange of the runway girder, to structural tees embedded in the wall.

3.2.5.2.2 Reactor Building Liner

The liner has been designed to support dead load and wind loads during the erection period.

The technical conditions applicable to the design of the reactor building liner are as follows:

Inside diameter	130 ft
Cylinder height	157 ft
Ellipsoidal dome a) Short radius b) Long radius	 20.5 ft 110 ft

Operating temperature, range a) Inside b) Outside	 +90 to 110°F +25 to 100°F

The steel plate for the main shell, excluding specially reinforced areas, is 3/8 inches thick.

The steel plate for the base liner, including the pits and sumps, has a thickness of ¼ inch.

The liner was reinforced about openings in accordance with the Pressure Vessels Code (i.e., by replacing cut out area of 3/8 inch liner plate).

3.3 OTHER CLASS I STRUCTURES AND SYSTEMS

Other Class I structures are listed in Section 3.1.1.1.

3.3.1 METHODS OF ANALYSIS

3.3.1.1 Seismic Analysis of Structures

The containment was analyzed using ANSYS as described in Section 3.2.3. The other structures which are not shells of revolution were analyzed by response spectrum method.

The vertical component of ground motion is assumed to be 2/3 of the horizontal one. The criterion used was to combine the responses due to vertical and horizontal input by the absolute sum. Since Florida is in the low seismicity zone, the structural response due to horizontal input is very small compared with allowable and does not control the design. Hence, the structural response due to vertical input was assumed to be insignificant. Stresses and deflection resulting from the combined influence of the normal loads and the additional loads from the 0.05g earthquake were calculated and checked against the limits imposed by the design standard or code. The combined influence due to 0.1g earthquake was checked to verify that deflections do not prevent functioning and that stresses do not produce rupture of excessive distortion.

3.4 NOT USED

3.5 NOT USED

3.6 NOT USED

3.7 VENTILATION AND PURGE SYSTEMS

3.7.1 DESIGN BASES

3.7.1.1 System Functions

This system is scheduled for future abandonment.

The Reactor Building (RB) Ventilation and Purge Systems consist of the RB area fans subsystem (AH-XB) and the RB purge subsystem (AH-XC). These subsystems are designed to perform the following functions:

a. Recirculate air within the RB (AH-XB).

b. Supply unfiltered air from the Intermediate Building to the RB and provide an exhaust path from the RB to the Auxiliary Building (AH-XC).

Additional information on the Reactor Building Ventilation and Purge Systems is provided in Section 3.8.

3.7.2 DESCRIPTION

The Reactor Building Ventilation and Purge Systems are shown in flow diagram Figure 3-5. Forced air supply to the RB no longer exists, unfiltered ambient air from the Intermediate Building is drawn into the RB. Air inside the RB can be circulated with fans (AHF-3A/3B) before air is drawn out of the RB via the purge exhaust flow path. The purge exhaust flow path is open to the Auxiliary Building (AB) which is normally kept at a slightly negative pressure. The AB negative pressure provides the motive force to move air through the RB.

3.7.3 OPERATING MODES

3.7.3.1 Normal Operation

The Reactor Building Ventilation and Purge systems may be operated when needed for recirculation and purging.

The Industrial Cooling (CI) System is an abandoned containment cooling system. CI system plate heat exchanger CIHE-10 originally served as a radiological boundary between the seawater room and the cooling towers on the South Berm. This heat exchanger has been abandoned and is now isolated to the South Berm by system valves. These valves are not abandoned because they now serve as the radiological barrier as well as flood barriers for the seawater room.

3.8 PLANT VENTILATION SYSTEMS

3.8.1 DESIGN BASES

Ventilating systems have been provided throughout the plant areas. These systems maintain suitable ambient conditions for personnel and equipment during all operating periods. A more detailed description of the functions of the systems is included under operational limits.

3.8.2 SYSTEM DESCRIPTION AND EVALUATION

These systems are scheduled for future abandonment.

The schematic flow diagrams for the systems are indicated in Figure 3-5, Figure 3-19, Figure 3-20, Figure 3-21, Figure 3-22 and Figure 3-29. The main and subsystems for the various buildings are as follows:

a. Reactor Building Cooling Systems (AH-XB)

1. Reactor Building Air Supply System (AH-XB), two units (AHF-3A, AHF-3B).

b. Reactor Building Purge Systems (AH-XC)

1. Two throttled open valves (AHV-1A, AHV-1B) that provide a path for air flow from the Reactor Building to the Auxiliary Building. Air flow into the Reactor Building is through the personnel air lock. This flow path maintains some RB ventilation during SAFSTOR.

c. Auxiliary Building Systems (AH-XD, AH-XJ, AH-XE, AH-XH)

1. Auxiliary Building Supply System (AH-XD), two 50% capacity fans (AHF-11A, AHF-11B) for various auxiliary building areas.
2. Fuel Handling Area System (AH-XE), one 100% capacity fan (AHF-10) for the fuel handling area.
3. Auxiliary Building Exhaust System (AH-XJ), four 50% capacity fans (AHF-14A, AHF-14B, AHF-14C, AHF-14D), four 25% capacity filter plenums (AHFL-2A, AHFL-2B, AHFL-2C, AHFL-2D).
4. Spent Fuel Pit Supply System (AH-XH), two 100% capacity fans (AHF-23A, AHF-23B) for the spent fuel pit surface area.

d. Control Complex Systems (AH-XK, AH-XS, CH)

1. Normal Duty Supply System (AH-XK), two 100% capacity fans (AHF-17A, AHF-17B) and two 100% capacity cooling coils (AHHE-5A, AHHE-5B).
2. Return Air System (AH-XK), two 100% capacity fans (AHF-19A, AHF-19B).
3. Control Complex Chilled Water System (CH-CC), one 100% capacity chiller (CHHE-4C) and two 100% capacity pumps (CHP-6A, CHP-6B).
4. EFIC Room HVAC System (AH-XS), one 100% capacity Air Handling Unit with Filters (AHF-54A). During normal operation, the Control Complex Chilled Water System supplies cooling water to AHHE-43 and uses AHF-54A fan.

e. Miscellaneous Systems (AH-XU)

1. 1. Turbine Building Switchgear Rooms, two 100% capacity Filter and Cooling Supply Systems (AHHE-10A, AHHE-10B).
2. Miscellaneous support buildings.

3.8.2.1 Modes of Operation

Normal modes of operation of the Plant Ventilation Systems are as follows:

a. AHF-3A/3B is being maintained as a method of circulating air within the Reactor Building during SAFSTOR. Maintenance on these fans will be minimized to prevent unnecessary entries in to the Reactor Building.

b. The Reactor Building Purge System is being utilized to provide an air flow path into and out of the Reactor Building during SAFSTOR. Two throttled open valves (AHV-1A, AHV-1B) provide a path for air flow from the Reactor Building to the Auxiliary Building. Air flow into the Reactor Building is through the personnel air lock.

c. In the auxiliary building, the supply fans and two of the four exhaust fans from the auxiliary building operate continuously during normal plant operation to maintain a negative internal building pressure.

Where redundant fans have been provided, the inactive fan is isolated by automatic control dampers. All auxiliary and fuel handling building systems are operated from the control room. Heating coils in the supply systems are automatically controlled.

Safety devices include: high temperature devices in the discharges from fans to stop fans and alarm in the control room on indication of high temperature and flow switches to indicate loss of air flow.

d. 1. The Control Complex Normal Duty Supply System is operated from the control room and runs continuously. This system consists of the Control Complex normal duty supply fan units AHF-17A and AHF-17B and the Control Complex return fans AHF-19A and AHF-19B. Outside air intake dampers AHD-1C and AHD-1E are open, atmospheric relief discharge to outside dampers AHD-2E is open and AHD-2C is closed, and relief damper AHD-3 is throttled. The dampers supplying conditioned air to the Chemistry and Health Physics area on the 95 ft. elevation, AHD-12 and AHD-12D, are open. The dampers for the standby fan are closed.

The chiller (CHHE-4C) and associated pumps (CHP-6A, CHP-6B) provide supplemental cooling to maintain the Control Complex suitable for equipment and personnel comfort during normal and emergency conditions. This chiller is air-cooled and rejects the heat from the Control Complex to the atmosphere. The Control Complex Chilled Water System is shown on Figure 3-23 and Figure 3-32.

e. Miscellaneous systems provide cooling or heating as required by control systems to maintain personnel comfort or ambient condition suitable for equipment operation.

1. Turbine Building Switchgear Area - During these periods, when maintenance is being performed, cooling to the switchgear rooms may be provided by opening the doors to the switchgear rooms and providing cooling air from the ventilated open floor areas of the Turbine Building to the switchgear rooms using portable fans.

2. The EFIC Room HVAC System is comprised of one 100% air handling unit, AHF-54A.

f. The battery rooms are exhausted through separate duct runs into a common exhaust main outside either battery room. The supply air for the battery rooms is furnished from a single main, but each battery room is supplied from a separate runout from this main so that none of the supply ductwork is common to both rooms at the battery room elevation. Physical isolation of one battery room from the adjacent battery room is provided by fire and explosion resistant construction in the common wall between the rooms.

3.8.2.2 Reliability Considerations

The functions of all heating and ventilating systems are assured by redundancy of equipment when required, by instrumentation and control as required for operation and monitoring, and by equipment arrangement for optimum maintenance and service availability.

3.8.2.3 Operational Requirements

The systems provide air circulation, filtration, and maintain temperatures as noted below:

a. In the Auxiliary Building, the supply units furnish filtered [85% efficiency] and tempered outside air. Supply and Exhaust are arranged to direct this air from areas of low to higher activity eventually directing it to the Main Exhaust Filter System and from there through fans to the exhaust vent. The main exhaust filters include roughing and HEPA filters.

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All the Exhaust System equipment for the Fuel Handling and Auxiliary Building areas have a Seismic Class II rating.

1. The normal and emergency total system exhaust flow rate with two fans operating is a nominal 156,680 cfm.
2. Failure of either the Supply or Return System is indicated by flow switches in the supply and exhaust mains and by an automatic trip alarm at the fan circuit breaker. If failure is indicated the redundant unit would be manually started from the control room or from a local station.
3. The exhaust fans are axial flow type, each nominally sized at 78,340 cfm, at 11 inches total pressure.
4. The total nominal filter capacity provided is 160,000 cfm. These filter systems include roughing and HEPA filters. Four 25% capacity units were considered adequate because:
   (a) The system has many exhaust branches varying in exhaust priority. If necessary, some of the less important exhaust branches could be temporarily closed while available exhaust filter capacity was decreased.
5. Filter replacement - The affected plenum could be closed so that the exhaust flow rate approximately matched the remaining nominal filter capacity. If no branches were isolated, the remaining filters would filter at a flow rate higher than the nominal rating with a resulting lower residence time, as noted under paragraph 4, above. The system was not designed to accommodate the loss of half the filter capacity as suggested because the out-of-service time required for filter replacement is short and any plenum repair period is expected to be similarly short. Also, a single filter failure would involve removal of the affected cell and the installation of a replacement cell or baffle plate and this time period would be short.
6. The fuel handling area and fuel pool are not isolated portions of the Auxiliary Building and the ambient is common to all. The system is designed to maintain a slight negative pressure in the Auxiliary Building relative to the outside. This pressure is common to both fuel and Auxiliary Building areas. Other areas such as the Control Complex and penetration areas have separate systems independent of the Auxiliary Building System with no provision to maintain pressure differential between their ambient and outside. Thus, any leakage between these areas and the Fuel and Auxiliary Buildings would be in the direction of the Fuel and Auxiliary Buildings.

b. In the Control Complex during normal operating periods, in areas supplied by this system, air is recirculated, a minimum amount of outside air is added, and ambient is maintained at approximately 75°F.

3.8.2.4 Control Building Duct Arrangement

Refer to Figure 3-26, Figure 3-27 and Figure 3-28. The fresh air intake (outside air) is located in Figure 3-27 at column line I between columns 301 and 302. Also, refer to Section 31 in Figure 3-28.

3.8.2.5 Emergency Feedwater Initiation and Control Cooling Duct Arrangement

Refer to Figure 3-30. This figure depicts the ductwork arrangement for cooling the EFIC rooms which are located on elevation 124'0" of the Control Complex building.

3.9 PLANT FIRE PROTECTION PROGRAM

3.9.1 INTRODUCTION

The Fire Protection Program at Crystal River Unit 3 (CR3) consists of activities and functions that are performed to minimize the probability and consequences of a postulated fire. In the event of a fire, the program and system designs ensure the requirements of 10CFR50.48(f) are satisfied.

The Fire Protection Program has been formulated in accordance with governing documents listed in Table 3-9.

A modification to the fire protection system is in progress. The modification when complete, will install three new sections of piping. A yard main loop is being installed around the north, south, and west portions of the berm which is connected to Fire Service Tank 1A (FST-1A). Two new dry manual standpipes are also being installed, one standpipe system for the Turbine Building and one standpipe system for the Auxiliary Building. The new yard main loop around the berm provides connection points where a Fire Department/Fire Truck can connect and use the water stored in FST-1A. The Fire Department/Fire Truck also has the ability to connect to either of the dry manual standpipe systems. These standpipe systems supply water to different elevations in the Turbine and Auxiliary Buildings.

3.9.2 DESCRIPTION

The Fire Protection Program uses the Defense-In-Depth concept to achieve a high degree of fire safety at CR3. The Defense-In-Depth concepts: (1) Reasonably prevent fires from occurring; (2) Rapidly detect, control, and extinguish fires that do occur and that could result in radiological hazard; and (3) Ensure that the risk of fire-induced radiological hazards to the public, environment and plant personnel is minimized.

Details of each element of the CR3 Fire Protection Program are provided in the Crystal River Unit 3 Fire Protection Plan. Elements of the fire protection program include the following:

a. Organization; Responsibility, Authority and Qualifications
b. Administrative Controls and Procedures
c. Fire Brigade; Composition, Responsibility, and Training
d. Fire and Emergency Response Activities
e. Fixed Fire Protection Features with Controls and Compensatory Measures
f. Reducing risk of fire-induced radiological hazard to the public, environment and plant personnel.
g. Fire Protection Quality Assurance Program

3.9.3 ORGANIZATION; RESPONSIBILITY, AUTHORITY AND QUALIFICATION

The line of authority for those positions with responsibility for the fire protection program is defined in the CR3 Fire Protection Plan with detailed responsibilities and qualifications.

3.9.4 ADMINISTRATIVE CONTROLS AND PROCEDURES

Administrative controls covering CR3's Fire Protection Program are provided by the Fire Protection Plan and the Plant Operating Manual. Controls that are established include administrative restraints of:

a. In Situ and transient combustibles
b. Use of temporary structures

c. Ignition sources
d. Smoking
e. Leak testing
f. Design, maintenance and plant modification processes
g. Surveillance of installed systems

3.9.5 FIRE BRIGADE; COMPOSITION, RESPONSIBILITY, AND TRAINING

An organized fire brigade of CR3 personnel is trained and equipped to respond and combat fire related emergencies at CR3. The fire brigade provides a source of qualified individuals from which a two (2) member Fire Brigade Team is formed for each shift. Each Fire Brigade Team is led by an individual who is qualified to assume command of a fire emergency and direct team actions.

Fire Brigade members are required to be physically fit and adequately trained before they may serve on the brigade. Initial training and periodic requalification of Fire Brigade members is conducted by qualified training instructors.

3.9.6 FIRE AND EMERGENCY RESPONSE ACTIVITIES

In the event of a fire incident at CR3, individual and team response actions are defined by plant procedures contained in the Plant Operating Manual (POM).

When required by plant procedures, notification and instructions are provided to plant personnel over the plant PA system. Radios are used by Fire Brigade Team members for direct communications with the Control Room and the Fire Team Leader during emergencies. Fire Brigade Team members responding to an emergency are provided communications capability and lighting. Tactics and strategies for combating fire in all areas of the plant are provided in the CR3 Pre-Fire Plans.

Local area fire departments provide assistance to the CR3 Nuclear Plant Fire Brigade. This assistance is defined under a Memorandum of Understanding (MOU) and responders remain under the direction of the CR3 Fire Brigade Leader.

3.9.7 FIXED FIRE PROTECTION FEATURES WITH CONTROLS AND COMPENSATORY MEASURES

Defense-In-Depth is accomplished through fire prevention, fire detection and suppression, and compartmentalization. Design and administrative controls ensure that fire protection features are installed and maintained to perform their intended function.

In situ fire protection at CR3 includes (but is not limited to) a water supply system, fire detection systems, automatic fire suppression systems, manual fire suppression systems and fire barriers. These installed features provide safety of both personnel and plant property.

3.9.7.1 Water Supply System

The fire service water for CR3 originates from well fields to the east of the nuclear site, kept in storage tanks at Units 1 & 2 and is pumped through a 12 inch main by two 1000 gpm pumps to the site fire service water system.

There is one storage tank (FST-1A) containing 300,000 gallons surveilled volume dedicated to fire service water storage. Level monitor annunciates in the Control Room if the level drops 12 inches below full.

Two fire service pumps, one diesel-driven and one electric motor-driven, each rated for 2000 gpm at 125 psi provide operating pressure under system use. A 30 gpm motor-driven pressure maintenance (jockey) pump is provided to

maintain a minimum of 105 psi in the fire service water system under no-use conditions. The jockey pump may be set to shut off at a selected pressure below 130 psi. Operating characteristics are described in the CR3 Fire Protection Plan. The fire service pumps are located in a pump house which is separate from other plant buildings and structures. The pump house and pumps are protected by a wet pipe sprinkler system. The fire service yard main loop completely surrounds the plant and is sectionalized by post indicator valves for isolation and maintenance purposes. These post indicator valves are protected with steel and concrete posts if subject to mechanical damage. Each of the two fire service pumps feed into the main loop through isolation and check valves. Isolation valves are provided between the three points at which the pumps feed the loop so that any failure of the loop can be isolated for maintenance.

Headers from the main loop, which can be isolated by post indicator valves, supply fire service water to sprinkler systems and manual hose stations located in plant buildings.

3.9.7.2 Fire Detection Systems

Fire detection systems employed at CR3 use ionization and thermal devices. These fire detection systems are installed throughout CR3 to:

- ◊ Provide early warning of fire through local visual alarms and audio alarms in selected areas of the powerblock. Remote audio and visual alarms are also provided at the three core loop panels.
- ◊ Provide initiation signals to automatic suppression systems

The Auxiliary Building, Intermediate Building, Control Complex, Turbine Building Switchgear Rooms and the two (2) Central Alarm Stations (CAS and CAS2) are monitored by ionization or thermal fire detectors. All of the aforementioned systems provide only audio and visual alarms, except that:

- A signal from any two detectors in the cable spreading room is required to actuate the Halon suppression system in the Control Complex, 134 ft El, Cable Spreading Room.

Separate fire detection capability is provided by detectors associated with the ventilation system for the plant. These detectors initiate alarms and/or closure of fire dampers or shutdown ventilation equipment and provide annunciation in the Control Room. Both ionization and thermal detectors are used and are located either in ventilation ducts or in exposed locations in the room they monitor.

3.9.7.3 Automatic Fire Suppression Systems

a. Automatic Sprinkler Systems

   Automatic sprinkler systems are installed at CR3.

   1) Three types of Automatic Sprinkler Systems are installed:
      Automatic Wet Pipe, Automatic Pre-Action, and Automatic Dry Pipe.

      A) Automatic Wet Pipe Systems protect;

         - ◊ Fire Pump House
         - ◊ Auxiliary Building, Elevation 95' & 119'
         - ◊ Control Complex, Elevation 95' & 124'
         - ◊ Intermediate Building, Elevation 95' & 119'
         - ◊ Turbine Building, Elevation 95' & 119'

B) An Automatic Pre-Action Sprinkler System is installed to protect the Standby Diesel Generator Room. A pre-action valve is installed in the system to initiate a local visual trouble alarm and audible and visual trouble and fire alarms in the control room.

C) An Automatic Dry Pipe Sprinkler System is installed to protect the RB Maintenance Support Building.

b. Halon 1301 System

A fixed, automatic, Halon 1301 Fire Suppression System consisting of 11 spheres is installed to protect the Cable Spreading Room in the Control Complex. A reserve Halon supply can be initiated through a "main-reserve" transfer switch. The Halon 1301 system is activated by an ionization detection system which automatically and simultaneously closes the air duct fire dampers, starts Halon mixing fans, initiates local audible and visual alarms, and initiates audible and visual alarms in the Control Room. The entrance is fitted with a self-closing door to ensure containment of the necessary 5% concentration of Halon 1301 to effect extinguishment.

c. FM-200 (heptafluoropropane)

An FM-200 Fire Suppression System is installed in the CAS and CAS2 buildings. All chemical for suppressing a postulated fire in the buildings is stored in dedicated tanks internal to the buildings. The CAS contains a stand-alone system, whereas the CAS2 system is controlled by the CR3 Control Room; however, both buildings initiate visual and audible alarms in the Control Room.

3.9.7.4 Manual Fire Suppression Systems

Manual fire suppression entails the use of fire protection equipment intended for use by trained personnel. Such equipment at CR3 includes:

a. Fire Extinguishers

Portable fire extinguishers are located throughout the site area. The types of fire extinguishers in use are listed in the Fire Protection Plan.

b. Standpipes and Hose Stations

The standpipes and hose station systems installed at CR3 are Class II with the following exception:

◊ Reactor Building - Class III

c. Fire Hydrants

Eight (8) fire hydrants, each connected to the yard loop and provided with isolation valves are installed around the berm. Additionally, one (1) hydrant is located at the base of the berm east of the TSC building and two (2) hydrants are North and West of the Nuclear Administration Building. Fire hydrants within the CR3 Protected Area are protected from physical damage as appropriate.

d. Fire Carts

Two (2) fire carts are located in the plant. These fire carts contain firefighting equipment necessary to support the fire brigade in response to a fire.

e. Foam Carts

One (1) foam cart (wheeled container for 30 gallons of fire suppression agent with hose and nozzle assembly) is provided at the plant for fire brigade use.

3.9.7.5 Fire Barriers and Penetrations

a. Fire Barriers

Fire barriers are utilized at CR3 to create the compartmentalization element of fire protection defense-in-depth. Fire barriers take the form of fire rated walls, floors, or ceilings.

b. Penetrations

Fire barrier penetrations consist of fire doors, fire dampers and through penetrations of electrical and mechanical components. Through penetrations are sealed to prevent passage of fire through those openings and fire doors and dampers are required to remain operable (i.e., maintain the ability to close and seal the opening in the event of a fire).

3.9.7.6 Controls and Compensatory Measures

The CR3 fire protection program is designed to assure that adequate levels of protection are provided at all times. This is accomplished by establishing controls and compensatory measures through surveillance procedures, as well as requiring reporting and documenting of noncompliances when systems, barriers, and other fire protection features are determined to be inoperable.

3.9.8 FIRE PROTECTION QUALITY ASSURANCE PROGRAM

Branch Technical Position CMEB 9.5-1, Revision 0, 1975, and Appendix A establish Quality Assurance criteria for ensuring that the guidelines for design, procurement, installation, testing, and the administrative control for fire protection systems are satisfied. The Fire Protection Quality Assurance Program is detailed in the CR3 Fire Protection Plan.

3.10 ELECTRICAL SYSTEMS

3.10.1 DESIGN BASES

The design of the Electrical Systems for Crystal River Unit 3 provides a reliable power source and equipment to ensure continued operation of important to the defueled condition (ITDC) equipment and other SSCs that may be used at CR3.

A modification to the electrical distribution system is in process. The new electrical distribution system will provide power to required components in the plant after the existing in-plant electrical distribution system is de-energized.

3.11 ELECTRICAL SYSTEM DESIGN

3.11.1 NETWORK INTERCONNECTIONS

Electrical Power is provided to Crystal River Unit 3 from the Duke Energy Florida (DEF) Transmission Network through the Crystal River 230 kV Substation via the Crystal River Unit 3 12 kV distribution line.

3.11.1.1 Single Line Diagrams

Figure 3-8 presents single line diagrams of the 230 kV substation Electrical System.

Figure 3-9 presents an “Electrical One Line Diagram Composite” of the CR3 electrical distribution system.

3.11.1.2 Reliability Considerations

Reliability considerations to minimize the probability of power failure due to faults in the network interconnections and the associated switching are as follows:

a. The 230 kV Switchyard has primary and backup breaker relaying to minimize the probability of a power failure due to faults.

3.11.2 PLANT DISTRIBUTION SYSTEM

The Plant Distribution System consists of the various Auxiliary Electrical Systems designed to provide reliable electrical power. The systems have been designed with a reliable power source to accomplish this.

3.11.2.1 Single Line Diagrams

Figure 3-9, Figure 3-10, Figure 3-11, Figure 3-12, Figure 3-13, and Figure 3-14 are single line diagrams of the Crystal River Unit 3 Distribution System.

3.11.2.2 4160 Volt Auxiliary System

The 4160 Volt Auxiliary System has four bus sections. Two of these bus sections, ES Buses 3A and 3B, comprise the 4160 volt switchgear of the two former redundant Class 1E electrical systems. Another two bus sections, designated as Unit Buses 3A and 3B, have balance of plant oriented loads. ITDC equipment can be a formerly safety related or a balance of plant SSC.

Unit Buses 3A and 3B are powered from the Crystal River Unit 3 12 kV distribution line through a step down transformer (MTTR-7) and the ES Buses are powered from Unit Buses 3A and/or 3B through the direct cable connection.

3.11.2.3 480 Volt Auxiliary System

The 480 Volt Auxiliary System has six power centers consisting of a 4160/480 volt transformer and its associated 480 volt switchgear (bus).

Five of the power centers are normally powered from the 4160 volt Unit Buses 3A and 3B. The 480V Plant Auxiliary Bus 3 is normally powered from the 4160 volt ES Bus "B".

The 480 volt auxiliary system also has one power center, consisting of a 12,470/480 volt transformer and its associated 480 volt bus powered from the Crystal River Unit 3 12 kV distribution line.

The 480 volt auxiliary system also has two power centers that are identified as 480V ES Bus 3A and 480V ES Bus 3B (formerly part of the Class 1E system). The 480V ES Buses 3A and 3B are normally powered from 4160V ES Buses 3A and 3B, respectively.

The 480V ES Bus 3A and 3B can be connected together. Interlocks are provided in the tie breaker controls to prevent paralleling 4160V ES Buses 3A and 3B.

The 480V ES Bus 3A may provide an alternate power feed to 480V Reactor Auxiliary Bus 3A. The 480V Plant Auxiliary Bus 3 may provide alternate power feed to four of the power centers (including 480V Reactor Auxiliary Bus 3B) using a tie breaker.

ES Motor Control Centers, ES 3A-1, 3A-2, 3A-3, 3AB, 3B-1, 3B-2 and 3B-3 provide power and control for associated equipment. The 3A and 3B Motor Control Centers are powered from 480V ES Buses 3A and 3B, respectively. The 3AB Motor Control Center is powered from either the 480V ES Bus 3A or 3B, using a non-automatic transfer switch. The transfer switch has controls located locally and in the main control room.

3.11.2.4 250/125 Volt DC System

The 250/125 Volt DC System provides a source of reliable continuous power for motors, control, and instrumentation. In general, DC motors are rated 240 volts and control circuits 125 volts DC.

The 250/125 Volt DC System consists of two main bus sections, supplied by a battery and battery chargers. A spare 125 volt DC battery charger is provided for backup.

The battery chargers normally supply the DC System load and the float charge to the battery. The chargers are also capable of supplying a 24 hour equalize charge. The chargers are supplied with a high/low voltage alarm relay to monitor the DC System. A high voltage alarm is provided to protect against battery overcharging during normal operation.

Hydrogen evolution from all batteries located in the Control Complex (including the Battery Rooms), and the subsequent hydrogen buildup in the Control Complex and in the Battery Rooms, was determined for various modes and operating conditions. Many factors influence the hydrogen generation rate and concentration of hydrogen (e.g., ambient temperature, cell voltage, air dilution flow rates, room/building volumes, system operating mode, etc.).

Hydrogen reaches flammable levels at 4% by volume in air. NFPA69 recommends that concentrations of flammable gasses be limited to 25% of their flammable limit in air. This corresponds to 1% for hydrogen.

During normal recirculation mode, with batteries fully charged and on the float, Control Complex hydrogen concentration lags hydrogen concentration at the primary source (in the Battery Rooms). Assuming zero in-leakage of fresh dilution air, the Control Complex could remain in the emergency recirculation mode for at least 30 days before the 1% limit is reached in the Battery Rooms. The Control Complex as a whole could reach 1% over a somewhat longer period.

A worst case scenario permits hydrogen to reach 1% in the Battery Room in approximately 15 hours. This requires: (1) worst case room temperature, (2) cell voltage just below the alarm setpoint [2.34 VPC], and (3) a total loss of Battery Room HVAC. This worst case scenario envelops less challenging conditions, such as, a loss of Battery Room HVAC during normal or emergency recirculation.

Battery discharge is monitored by contact making ammeters mounted in the main DC panels. This provides a remote alarm when the battery is supplying power to the system.

The electrolyte level, specific gravity, and cell voltage of each battery cell is checked on a periodic basis.

The arrangement and number of chargers and DC distribution panel boards are as shown on Figure 3-13 and Figure 3-14. The output of spare battery chargers may be fed to either half of their corresponding 250/125 Volt DC System.

One battery can supply all necessary components. The station battery can also supply emergency lighting in the Control Room and the access and egress corridor thereto.

The battery has been sized to have an 8 hour rating of 1708 ampere hours, based on the use of 116 cell batteries and discharge to 1.81 volts per cell, 210/105 volts across the battery at 77°F.

The battery system is provided with sufficient battery charger capacity (not including the spare battery charger of the system) to fully recharge the associated battery in less than 24 hours while supplying the maximum steady state DC load connected to the system.

Each battery charger has been sized at 200 amperes continuous capacity.

A single line diagram of the DC System showing essential loads is given in Figure 3-13 and Figure 3-14.

The 250/125 VDC distribution panels are provided with loss of voltage relays and status lights that indicate "POWER AVAIL" at the COF panel for the operator.

3.11.2.5 120 Volt AC Vital Power System

The 120 Volt Vital Power System provides power, instrumentation, and control loads.

The vital power system consists of four bus sections from 480 Volt ES motor control centers through transformers. Indicating lights provide the status of each circuit.

3.11.2.6 120 Volt AC Regulated Power System

The 120 Volt AC Regulated Power System supplies instrumentation, control, and power loads requiring regulated 120 volts AC power. Three distribution panels distribute 120 VAC power received from reactor auxiliary motor control centers via regulating transformers.

3.11.2.7 120Y208 Volt AC Power System

A low voltage 120Y208 Volt AC Power System supplies instrumentation, control, and power loads requiring unregulated 120Y208 volt AC power. It consists of distribution panels and transformers fed from motor control centers.

3.11.2.8 Evaluation of the Physical Layout - Electrical Distribution System Equipment

The Electrical Distribution System equipment has been located to minimize the vulnerability of vital circuits to physical damage. The locations are as follows:

a. The Crystal River Unit 3 12 kV distribution line is normally powered from the 230 kV Switchyard and is stepped down to 4160 Volt by transformer MTTR-7 on the South berm of CR3.

b. The unit auxiliary 4160 volt switchgear, and 480 volt switchgear are located to minimize exposure to mechanical, fire and water damage. This equipment is protected by overcurrent devices to permit safe operation under normal and short circuit conditions.

c. The ES 4160 volt switchgear and 480 volt switchgear are physically separated from each other and from unit auxiliary switchgear and located in a Class I structure to further minimize exposure to mechanical, fire, and water damage. This equipment is protected by overcurrent devices to permit safe operation under normal and short circuit conditions.

d. The 480 volt motor control centers are located in the areas of electrical load concentration. ES motor control centers are located in Seismic Class I areas.

e. The plant DC system battery and associated chargers are located in separate rooms within the control complex (a Class I structure), to minimize vulnerability to damage. The DC system battery room has a separate supply air duct and is exhausted by a common duct. The supply and exhaust ducts in the battery room can be automatically isolated by activating isolation dampers in the ducts. The isolation dampers are activated by a temperature switch located in the room. In addition, the room has a smoke detector which actuates an annunciator alarm on the main control board.

f. Cables, trays, conduit, and electrical equipment are routed to help ensure the separation of power, control, and instrument circuits by function.

    1. All cable trays have their own unique number affixed to them.

2. All cables have their circuit identifying number permanently affixed to each end and wherever they leave their assigned cable tray.

3.11.2.9 General Cable Considerations

a. The application and routing of control, instrumentation, and power cables are such as to minimize their vulnerability to damage from any source. All cables, including interlocked armor and rubber insulated, are designed using conservative margins with respect to their current carrying capacities, insulation properties, and mechanical construction. Power and control cable insulation rating is typically 90°C. Appropriate instrumentation cables are shielded to minimize induced voltage and magnetic interference.

b. Power circuit cables are sized on the basis of the maximum ambient temperature expected, the current requirements of the respective equipment, and the designed cable tray loading. The reference used for cable selection is the CR3 Electrical Design Criteria - Cable Ampacity Sizing or the National Electric Code (NEC) may be used for analyses/designs post cessation of power operation.

c. An ambient temperature of 50°C within the reactor, auxiliary, and intermediate buildings, and an ambient temperature of 40°C in other plant areas was the original design basis ambient for all power cable ratings. Alternatively, actual or greater than ambient temperatures may be used.

d. In general, motor and transformer feeder cables are rated at 125% of full load current. In some cases, the 125% of full load current rating is not met. However, as a minimum, the cable will have a rating of 115% of full load current. This provides for motor and equipment operation at service factor ratings.

e. A small number of existing power cables were found prior to cessation of power operations that have ratings in specific locations which do not meet the CR3 Electrical Design Criteria – Cable Ampacity Sizing. These cables were evaluated individually to determine if they were suitable for continued use. The identification of these cables and the evaluations for continued use are documented in Engineering Calculations.

f. Fire seals are used at cable trays and cable runs where they enter or leave Class I areas, enter or leave the control and auxiliary buildings, and where vertical trays pass through floor openings.

g. When trays containing power cables are enclosed with fire rated material, ampacity derating factors are applied in accordance with the CR3 Electrical Design Criteria – Cable Ampacity Sizing. The derating factors in these Design Criteria are in accordance with the NRC Safety Evaluation Report, "Crystal River Unit 3 – Revision 1 to Safety Evaluation Report Addressing Thermo-Lag Related Ampacity Derating Issues," dated January 21, 2000 (3N0100-07).

h. Power and control cable trays are ladder type. Where there are horizontal trays passing under grating or hatches, the top tray has a solid cover which is spaced above the tray for ventilation. Where a tray has a vertical rise near a walkway, through or above the floor, covers are installed for protection.

i. The general purpose power and control cable used for the original construction was determined to be flame retardant using a draft of the test defined in IEEE Standard 383. Specifically, these cables were found not to propagate a fire while ignited by a burlap igniter. The general purpose instrument cables were exposed to the vertical flame test described in Section 6.19.6 of IPCEA S-19-81 and, in all cases, there was not significant flame travel and no continued burning after the fifth application of flame. General purpose power, control and instrument cable installed subsequent to the original construction has been purchased as flame retardant, in accordance with the requirements of IEEE Standard 383 or equivalent.

   Some special purpose cable such as alarms and indications is not certified as flame retardant. The amount of this cable that is not certified as flame retardant is considered to be not significant.

3.11.2.10 Separation of Circuits

a. General Separation Requirements:

1. Power and control cable rated at 600 volts or below are not placed in wireways with cable rated above 600 volts.
2. Low level analog signal cables are not routed in wireways containing power or control cables.
3. Wireways are identified using permanent markings. The purpose of such markings is to facilitate cable routing identification for future modifications or additions.
4. Permanent identification of cables and conductors is made at all terminal points.

3.11.2.11 Cable Tray Loading and Separation

a. 4160 Volt Power Tray:

1. No other type of cable is routed in the same tray with 4160 volt power cable.
2. There is only one layer of cable in a tray.
3. For fire rated material applications refer to Section 3.11.2.9.g.

b. 480 Volt and DC Power Tray:

1. No other type of cable other than 600V or 1 kV rated cable are mixed in the same tray carrying 480 Volt, 125 Volt AC, and 250/125 Volt DC power circuit cables.
2. Tray loading of 50% physical fill is the design objective. However, in certain areas where physical limitations govern, the tray fill exceeds 50%. For fire rated material applications refer to Section 3.11.2.9.g.

c. Control Tray:

1. In general, control tray 120 VAC or 125 VDC is reserved for those cables providing control paths for electrically controlled plant equipment. The most common control wire size used at CR3 is #14 AWG insulated to a minimum of 600 Volts. In some instances, 480 Volt, 120 Volt AC, and 125 Volt DC power cables size #8 AWG and smaller are allowed to be routed in the control cable trays to facilitate installation.
2. In general, control cable tray loading of 50% physical fill is the design objective. However, in certain areas where physical limitations govern, the cable fill may exceed 50%. In all cases, however, thermal loading has been considered. Certain power cable with conductor sizes larger than #8 AWG are installed in control trays. These exceptions have been evaluated and approved for physical compatibility by analysis/calculation, in accordance with CR3 Nuclear Engineering procedures.

d. Instrument Tray:

1. In general, instrument cable tray loading of 50% physical fill is the design objective. However, in certain areas where physical limitations govern, the cable fill may exceed 50%.
2. There are no other types of cables mixed with instrumentation cabling except alarm, telephone, and low level paging circuits. Certain control cables are installed in instrumentation trays, and certain instrument circuits are installed in control trays. These exceptions have been evaluated and approved for resistance to electrical noise by an analysis/calculation, in accordance with CR3 Nuclear Engineering procedures.

e. Dual Use Tray:

1. Dual use trays contain control and instrument cables routed to the Main Control Board from individual control and instrument trays in the Cable Spreading Room. Due to physical limitations preventing the use of separate raceways, trays with a dual classification are utilized for both control and instrument circuits in this location. Instrument circuits routed in dual use trays consist of cable that is noise-resistant (such as shielded twisted pairs) and has been evaluated and approved for resistance to electrical noise by analysis/calculation, per CR3 Nuclear Engineering procedures.
2. No power circuits are routed in dual use trays.

3.11.3 SOURCES OF AUXILIARY POWER

3.11.3.1 Description of Power Sources

Each source has various degrees of redundancy and reliability as outlined below:

3.11.3.1.1 Crystal River Unit 3 12 kV Distribution Line

The Crystal River Unit 3 12 kV distribution line originates in the Crystal River 230 kV Substation. The distribution line is normally powered from a 230 kV to 13.09 kV step-down substation transformer with a tap-changing automatic voltage regulator. The line can also be powered from other sources using manual switching. The distribution line is routed directly to Unit 3 partially as an overhead line and partially as surface mounted conduit on the Unit 3 berm. The line serves various low voltage loads around the perimeter of Unit 3 and supplies a 13.09 kV to 4160 Volt, 5 MVA distribution style transformer (MTTR-7) on the South berm. Power from the transformer is routed to Unit Buses 3A and 3B using cable in an underground duct bank. The transformer is adequately sized to carry the electrical loads for Unit 3 post cessation of power operations.

3.11.4.1 Reliability Considerations

DC control power is used for the protective relaying schemes and the 230 kV breakers as follows:

a. Primary and backup relaying protection is provided for the 230 kV buses, transmission lines, and plant lines. Redundant, but not physically separate, current transformers are utilized for the redundant relay schemes. However, potential transformers used for the redundant schemes are not redundant.

   The primary and backup relay schemes are taken out of service for testing, repair or maintenance periodically. However, both schemes are never taken out of service simultaneously without the associated equipment being isolated from the 230 kV substation.

b. Each 230 kV breaker is provided with two electrically independent sets of tripping coils designated as primary and backup.

3.12 TESTS AND INSPECTIONS

The 230 kV circuit breakers can be inspected, maintained, and tested as follows:

Transmission line circuit breakers are tested on a routine basis. This can be accomplished on the breaker and a half scheme without removing the transmission line from service.

Transmission line protective relaying can be tested on a routine basis.

The ungrounded DC System has a detector to indicate when there is a ground existing on any leg of the system. A ground on one leg of the DC System will not cause any equipment to malfunction.

Grounds can be located by a logical isolation of individual circuits connected to the faulted system, while taking the necessary precautions to maintain the integrity of the vital bus supplies.

3.13 QUALITY CONTROL

Assurance that the electrical systems meet their design bases, insofar as the integrity of the systems is concerned, is obtained by analysis, inspection, and testing. The quality program for Crystal River 3 is described in Section 1.7. The offsite power circuit from the transmission system is maintained in accordance with normal utility maintenance procedures and practices.

3.14 OPERATING CONTROL STATIONS

Primary control for certain auxiliary systems are located in centralized areas, remote from the control room, when the system controlled does not involve power generation control or emergency functions.

3.14.1 NOT USED

3.14.2 INFORMATION DISPLAY AND CONTROL FUNCTION

Components on the control panels are arranged to facilitate communication between the controls and the operator.

Information displays are designed to provide the operator with sufficient information to make proper evaluations under the full range of conditions. The displays are arranged to facilitate evaluation and to avoid the possibility of confusing the operator.

3.14.3 SUMMARY OF ALARMS

Audible and visual alarms are initiated in appropriate areas by the related radiation monitoring channel and audible alarms are also sounded in the appropriate areas throughout Crystal River Unit 3 if high radiation conditions persist.

Audible alarms are also sounded if the automatic fire protection equipment is initiated. The audible alarm may be muted for a specified time period of one to ten minutes by the operators. The purpose of the delay is to allow the audible portion of the annunciator system to be inhibited during periods of numerous incoming alarms.

3.14.4 COMMUNICATION

A Telephone System with a battery back-up is provided utilizing handsets in the control room and in various areas in the plant. A paging system consisting of handsets and speakers has been provided for complete unit coverage. There is also an Evacuation and Fire Alarm System with complete unit coverage.

A Maintenance Communication System is provided that consists of portable headsets which are battery powered and self-contained and jack stations to interconnect the system.

Leased lines are utilized for out-of-plant communication. Cellular phones and a Satellite Phone are also available for out-of-plant communications.

Additional external circuits include the NRC Emergency Notification System (ENS) and the State of Florida Hot Ringdown Telephone System for contacting the State Watch Office (SWO).

A UHF radio system utilizing hand-held transceivers and six repeaters is also available for communications to any point in the plant or on the plant site. The UHF radio system is powered from two non-safety related 240VAC power sources. Additionally, a dedicated DC battery is provided which is capable of powering the equipment in the event of a loss of both AC power sources.

3.14.5 OCCUPANCY

Engineering Change 294154 has determined that the Control Complex no longer serves a safety function and has been reclassified as a Class III Structure.

3.14.5.1 Control Room Habitability

In regard to protection from toxic gas releases, all major sources of toxic gas have been removed from the site. The most significant sources remaining are ammonia at CR Units 4/5 and the potential to use chlorine and sulfur dioxide for short periods of time at the Helper Cooling Towers. Administrative limits have been established for the quantity of chlorine and sulfur dioxide that could be stored at the Helper Cooling Towers.

3.15 GAS TANKS AND CYLINDERS AS MISSILES

The tanks and cylinders containing compressed gases are installed in a manner which meets or exceeds the safety requirements of the Occupational Safety and Health Administration Standards given in 29 CFR 1910, Subpart H. Gas tanks and cylinders fall into two categories:

1. Tanks and cylinders that are mounted/secured and designed as system equipment, and,
2. Gas Tanks and cylinders that are stored.

All gas tanks and cylinders are evaluated for potential missile hazards from failure of components pressurized by gasses. The evaluation of gas tanks and cylinders becoming missiles is based on the following:

- Containers are designed, constructed and/or tested to rigid specifications;
- Relief valves are provided with setpoints below the design pressure of the tanks;
- Tanks are located in limited access areas, where possible;
- Anchoring and securing of tanks and cylinders is accomplished to minimize the potential of missiles in the event of failed piping and gas storage facilities.

Stored gas tanks and cylinders are in a remote location, and/or missile-proof walls exist between the storage area and essential equipment.

Portable gas cylinders are procedurally controlled during transportation, replacement and use. Plant walk-downs are performed on a regular basis to ensure that gas cylinders are properly secured.

By comparing the method of anchoring, weight, positioning, and location of the various tanks containing gas under pressure, the possibility of the tank becoming a missile would be extremely improbable.

3.16 COMPRESSED AIR

3.16.1 DESIGN BASES

The Instrument Air (IA) System has two electric driven air compressors located in the Turbine Building Elevation 95'. Each compressor will maintain a constant discharge pressure and meet the combined air demand of the Instrument and Station Air (SA) System during normal conditions. A single compressor will normally be operating

with the remaining compressor maintained in a standby "auto start" mode. A heatless desiccant dryer is installed in the discharge of each compressor. Each dryer is equipped with a duplex prefilter and duplex afterfilter. The outlets of the dryers are connected to a common header, supplying air to three receiver tanks. The outlets of the receiver tanks supply the air to the Instrument Air System distribution piping.

The Instrument Air System provides dry filtered air to the Station Air System through isolation valve IAV-30. The isolation valve closes on low IA System header pressure to isolate the IA System from the Station Air System.

The air systems equipment is normal industrial quality. The system is shown in Figure 3-24.

3.17 SYSTEM DESIGN AND OPERATION

3.17.1 Cycle Makeup and Water Treatment System

The demineralized water makeup requirements for the plant are supplied by the Cycle Makeup and Water Treatment System shown in Figure 3-33.

3.18 REFERENCES

1 Uniform Building Code, International Conference of Building Officials, Pasadena, California, 1964.

2 "Wind Forces on Structures," ASCE paper No. 3269, 1961.

3 ASME Boiler and Pressure Vessel Code.

4 Building Code Requirements for Reinforced Concrete - ACI 318-63.

5 "Specification for the Design, Fabrication, and Erection of Structural Steel for Buildings," AISC 1963.

6 Code for Pressure Piping, USAS (ANSI) B 31.1.

7 "Specification for Structural Concrete for Buildings," ACI 301-66.

8 "Standard Specification for Portland Cement," ASTM C-150-67.

9 "Specifications for Concrete Aggregates," ASTM C33-67.

10 "Specifications for Ready Mixed Concrete," ASTM C94-68.

11 - 15 Deleted

16 Welding Handbook, Section 3, 6th Edition, AWS.

17 "Safety Standard for Design, Fabrication and Maintenance of Steel Containment Structures for Stationary Nuclear Power Reactors," ASA N6.2, 1965.

18 - 41 Deleted

42 AISC, "Specification for the Design, Fabrication, and Erection of Structural Steel for Buildings," November 1983.

43 ACI, "Code Requirements for Nuclear Safety Related Concrete Structures" (ACI 349-85).

44 Regulatory Guide 1.142, "Safety-Related Concrete Structures for Nuclear Power Plants (Other Than Reactor Vessels and Containments)," Revision 1, October 1981.

45 Regulatory Guide 1.60, "Design Response Spectra for Seismic Design of Nuclear Power Plants," Revision 1, December 1973.

46 Regulatory Guide 1.61, "Damping Values for Seismic Design of Nuclear Power Plants," October 1973.

47 Regulatory Guide 1.76, "Design Basis Tornado for Nuclear Power Plants," April 1974.

48 Standard Review Plan 3.3.2, "Tornado Loadings," Revision 2, July 1981.

49 Standard Review Plan 3.5.1.4, "Missiles Generated by Natural Phenomena," Revision 2, July 1981.

50 Standard Review Plan, Section 3.8.4, "Other Seismic Category 1 Structures," Revision 1, July 1981.

51 ACI 301-84, "Specification for Structural Concrete for Buildings."

52 American Welding Society (AWS), "Structural Welding Code," AWS D1.1-84.

53 ASTM A36-81a, "Specification for Structural Steel."

54 ASTM A307-83a, "Specification for Carbon Steel Externally Threaded Standard Fasteners."

55 ASTM A354-83a, "Specification for Quenched and Tempered Alloy Steel Bolts, Studs, and Other Externally Threaded Fasteners."

56 ASTM A563-83a, "Specification for Carbon Steel and Alloy Steel Nuts."

57 ASTM F436-83b, "Specification for Hardened Steel Washers."

58 ASTM A615-82, "Specification for Deformed and Plain Billet-Steel Bars for Concrete Reinforcement."

59 ASTM C33-84, "Specification for Concrete Aggregates."

60 ASTM C94-83, "Specification for Ready-Mix Concrete."

61 ASTM C595-83a, "Specification for Blended Hydraulic Cements."

62 ASTM C150-84, "Specification for Portland Cement."

63 – 68 Deleted

69 ANSI B31, Case 70, "Design Criteria for Nuclear Power Piping Under Abnormal Conditions," January 1970.

70 Deleted

71 AISC Manual of Steel Construction, "Allowable Stress Design," 9th Edition, Including the Specification for Structural Steel Buildings, June, 1989.

72 ACI 349-97, "Code Requirements for Nuclear Safety Related Concrete Structures."

73 - 74 Deleted

75 Regulatory Guide 1.92, "Combining Modal Responses and Spatial Components in Seismic Response Analysis," Revision 1, February 1976.

76. Deleted

77 Calculation S10-0019, "Design Compressive Strength for RB External Cylinder Walls and Buttresses"

78 Specification CR3-C-0003, "Concrete Work for Restoration of the SGR Opening and Delamination Repair in the Containment Shell"

79 - 82 Deleted

83 ASME Section III, Division 2, Subsection CC-4331.2 (a) (2001 Edition with 2002 and 2003 Addenda)

84 Deleted

85 EC 275220, Revision 31, "Reactor Building Delamination Repair Phase 4 Concrete Placement"

86 EC 275221, Revision 15, "Reactor Building Delamination Repair Phase 5- Retension/Testing"

87 EC 282247, Revision 4, "Detension Vertical Tendons To 75% Of Design"

88 EC 282639, Revision 3, "Detension Tendons For Containment Repair (Not Fully Implemented)"

89 EC 290986, Revision 0, "Detensioning For Containment Stabilization"

90 Duke Energy Letter 3F0213-07, dated 2/20/2013, "Certification of Permanent Cessation of Power Operations

91 NRC letter dated 3-13-13, NRC's acknowledgement of Duke Energy letter 3F0213-07

92 Calculation S10-0006, Revision 1, "Containment Repair Project - Seismic, Wind, And Tornado Evaluation And Delamination Depth Evaluation For Detensioned State"

93 NCR 00395843, "Concrete Removal Exposed Horizontal & Vertical Cracks"

94 NCR 00453318, "Retensioning Work Stopped For AE In Bay 56"

95 NCR 00475123, "Bay 56 Crack Appears To Expanding"

96 NCR 00478650, "Spalled Concrete In IB"

97 EC 288572, Revision 0, "Containment Condition Assessment"

98 Calculation S13-0009, Revision 0, "Containment Repair Project- Analysis of Current Condition"

99 Calculation S13-0010, Revision 0, "Containment Repair Project- Design Code Checks for Current Condition"

100 - 144 Deleted

145 Engineering Change 293675, Control Complex Cooling Chiller CHHE-4C

146 - 147 Deleted

TABLE 3-3, LIST OF METHODS OF SEISMIC ANALYSIS FOR CLASS I STRUCTURES, SYSTEMS AND COMPONENTS

TABLE 3-3	Design Reference or Codes	Seismic Design
A. Ventilation Systems		
Ventilation System For Spent Fuel Cooling System Pump Area	SMACNA Stds	Dynamic
Reactor Building Recirculation System	SMACNA & AISC Stds	Dynamic
Ventilation System For Decay Heat Closed Cycle Cooling System Pump Area	SMACNA Stds	Dynamic
EFIC RM Air Handling Units (abandoned)	DSAR Section 3.8	Dynamic
Ventilation System For Control Complex	SMACNA Stds	Dynamic
B. Miscellaneous Systems and Components		
Emergency Feedwater System including:		
Reactor Building Polar Crane	AISC/EOCL	Static
Fuel Handling Area Crane	ASME NOG-1-2004 / NUREG-0554	Modal Analysis Response Spectrum
Spent Fuel Handling Bridge	*	
* To preclude dislodging the fuel handing bridge and associated bridge trolleys during a seismic disturbance, an anti-derailing device has been attached to the bridge frame and trolley frame. The device has been designed using a static analysis to a seismic loading equivalent to 0.5 g.		
Standpipe	NFPA-14s	Seismic I at Penetration (see DSAR Section 3.1.2.2)
C. Radioactive Waste Disposal System		
Reactor Coolant Bleed Tanks	DSAR Table 4-1, ASME III-c	Static
Miscellaneous Waste Storage Tank	DSAR Table 4-1, ASME III-c	Static
Spent Resin Storage Tank	DSAR Table 4-1, ASME III-c	Static
Concentrated Radioactive Waste Storage Tank	DSAR Table 4-1, ASME III-c	Static
Neutralizer Tank	DSAR Table 4-1, ASME III-c	Static
Reactor Building Sump Pumps	ASME III-c	Dynamic and Static
Decay Heat Pit Sump Pumps	ASME III-c	Static
Liquid Outlet Piping	DSAR Section 3.3.4	Modal Analysis Response Spectrum

TABLE 3-9, FIRE PROTECTION GOVERNING DOCUMENTS

Fire Hazard Analysis

The Fire Hazard Analysis (FHA) is a study of plant designs, potential fire hazards in the plant, potential threats of these hazards occurring and a fire loading for the various fire areas. The FHA identifies the location and designations of fire areas and zones within Crystal River Unit 3. The Fire Hazard Analysis is updated periodically to reflect plant modifications which affect the document. Changes to the FHA are performed under the 10 CFR 50.59 evaluation process per 10 CFR 50.48(f)(3).

Fire Protection Plan

The Fire Protection Plan has been developed to describe the operational elements of the Fire Protection Program for Crystal River Unit 3 in order to assure response to a fire emergency is timely and adequate. The Plan addresses the Fire Protection Program organization, responsibilities, authorities, administrative controls, automatic and manual fire detection and suppression equipment, procedures, training, and basic design change processes. The Fire Protection Plan is updated periodically, as necessary, in support of changes to the program. Changes to the Fire Protection Plan are performed under the 10 CFR 50.59 evaluation process per 10 CFR 50.48(f)(3).

CP-9300, Fire Protection - Minimum Requirements, Compensatory Measures, And Surveillance Requirements

CP-9300 has been developed to describe the minimum requirements for Fire Protections systems and components to maintain functionality. The procedure addresses the operating requirements, limitations, and surveillance requirements of the Fire Protection Program. Changes to CP-9300 are performed under the 10 CFR 50.59 evaluation process per 10 CFR 50.48(f)(3).

FIGURE 3-1, CLASS I STRUCTURES

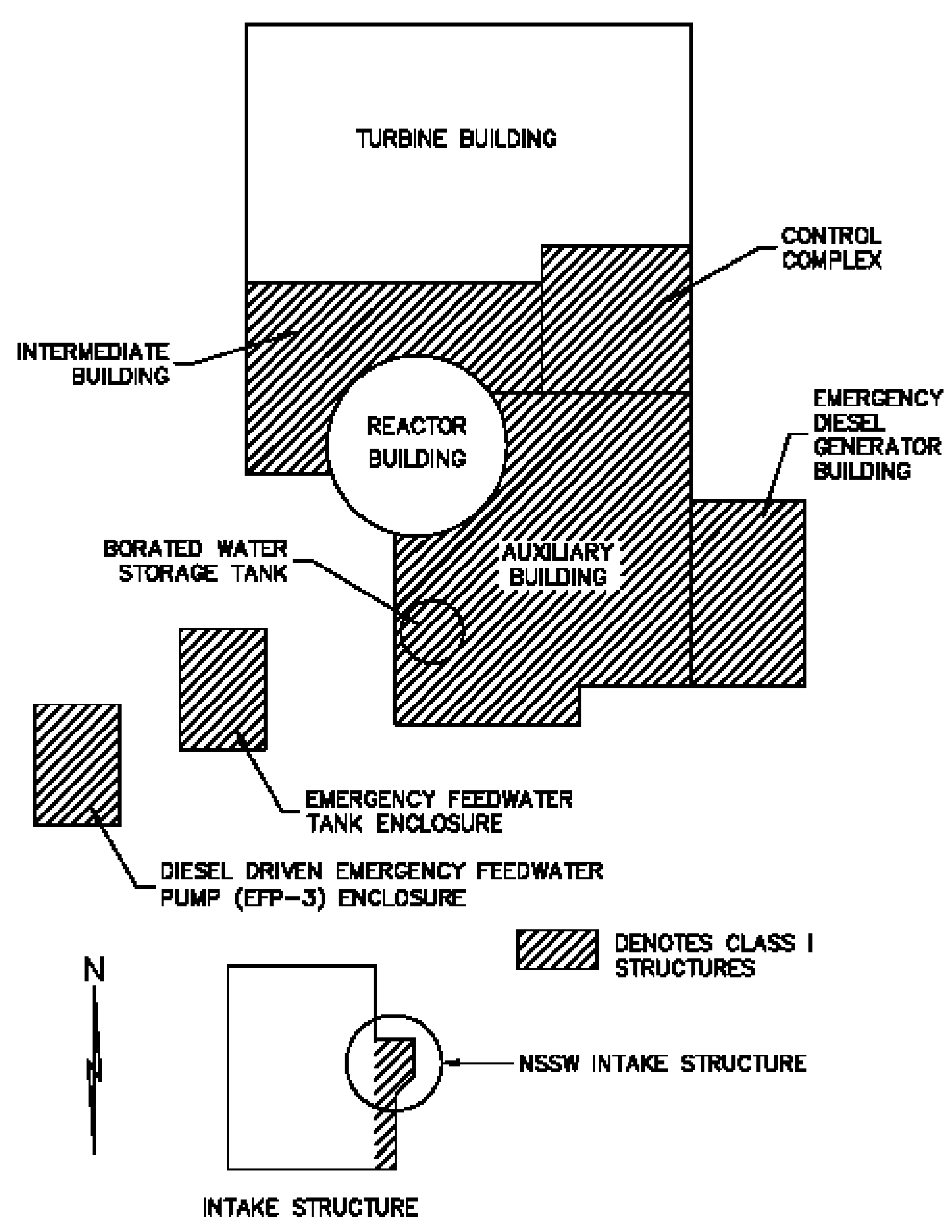


FIGURE 3-2, PENETRATION DETAILS

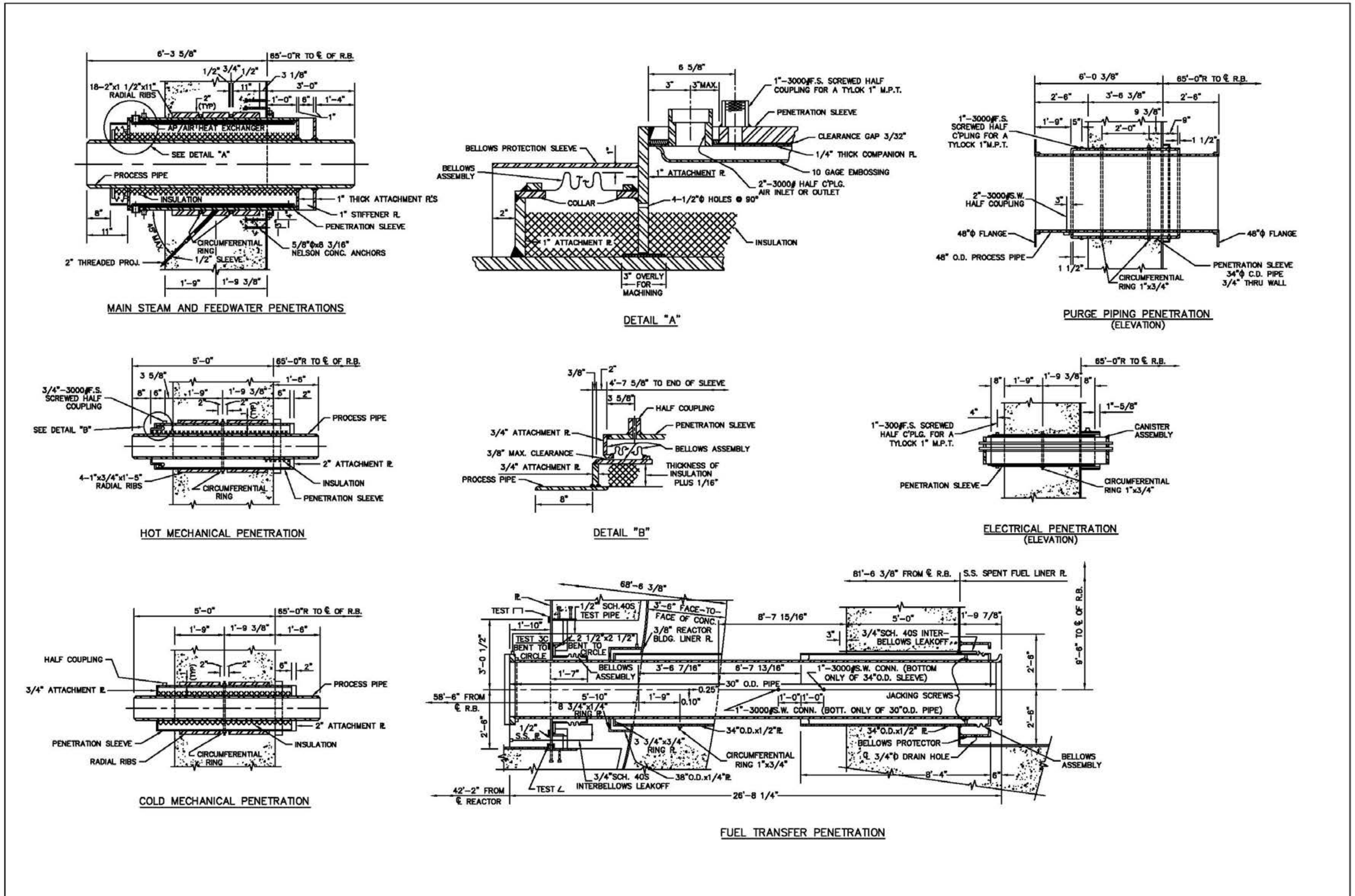


FIGURE 3-3, POLAR CRANE SEISMIC TIE-DOWN

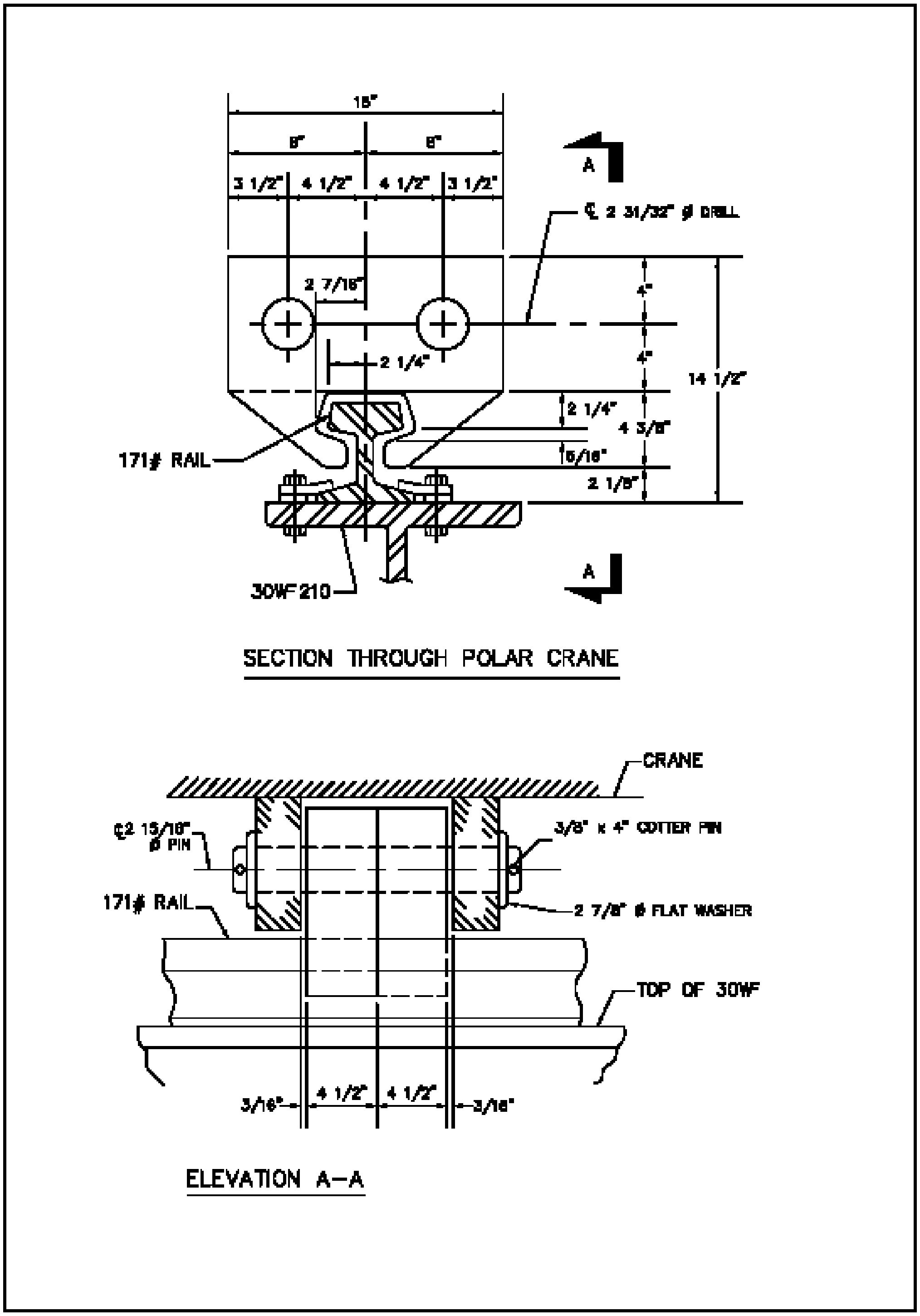


FIGURE 3-4, TYPICAL CRANE BRACKET DETAIL

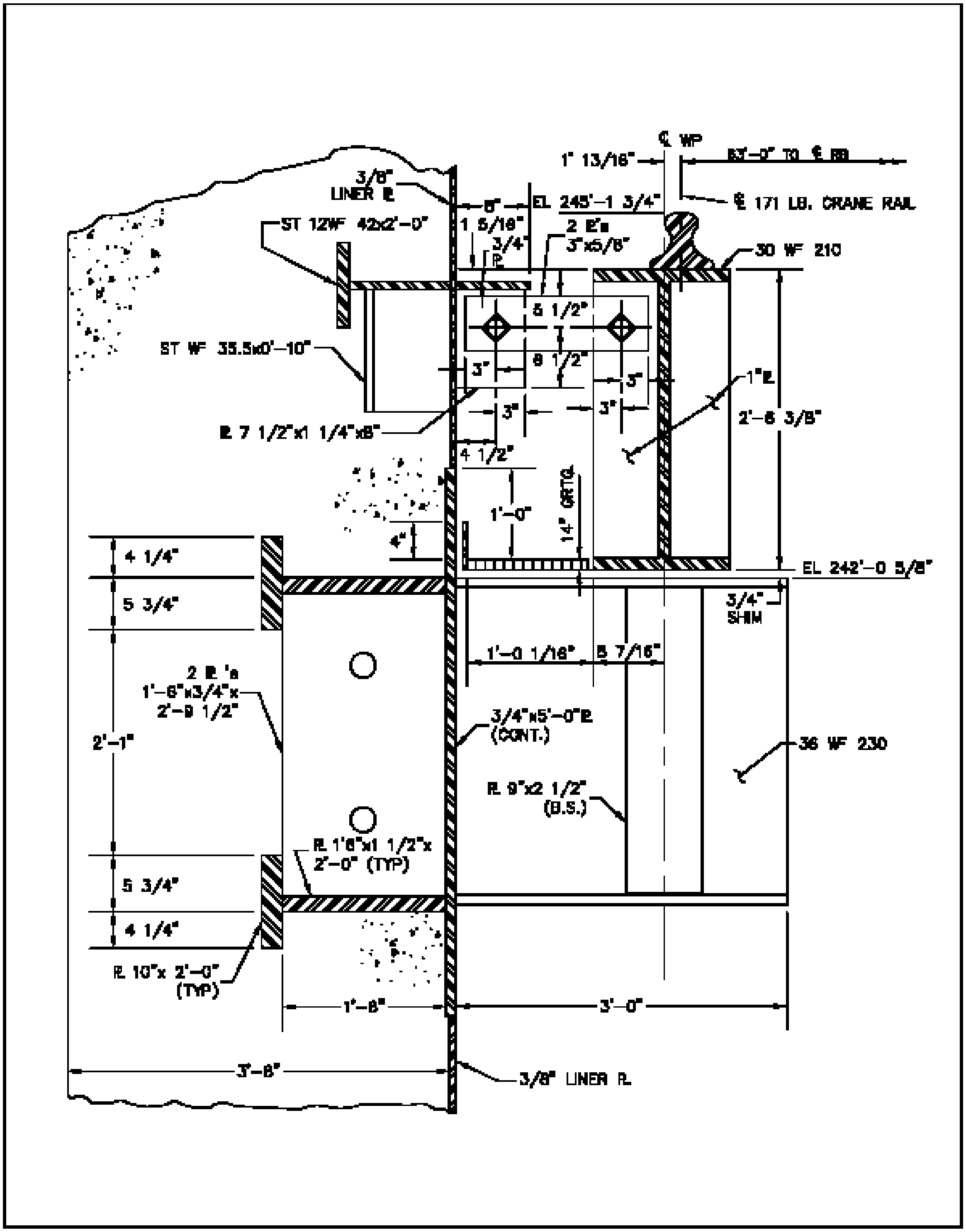


FIGURE 3-5, REACTOR BUILDING NORMAL VENTILATION SYSTEM, FD-302-751

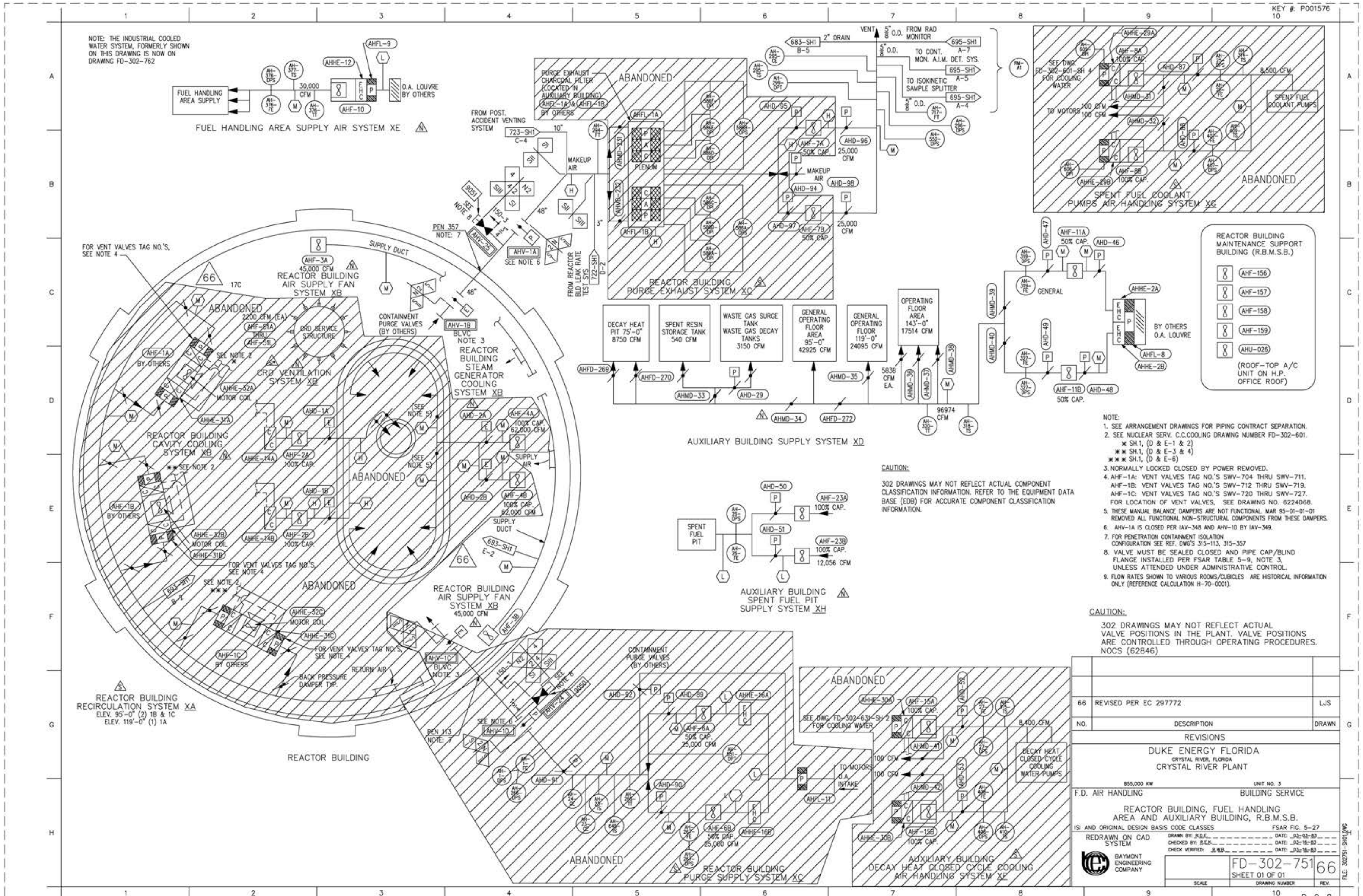


FIGURE 3-6, AUXILIARY BUILDING - FLOOR RESPONSE CURVES

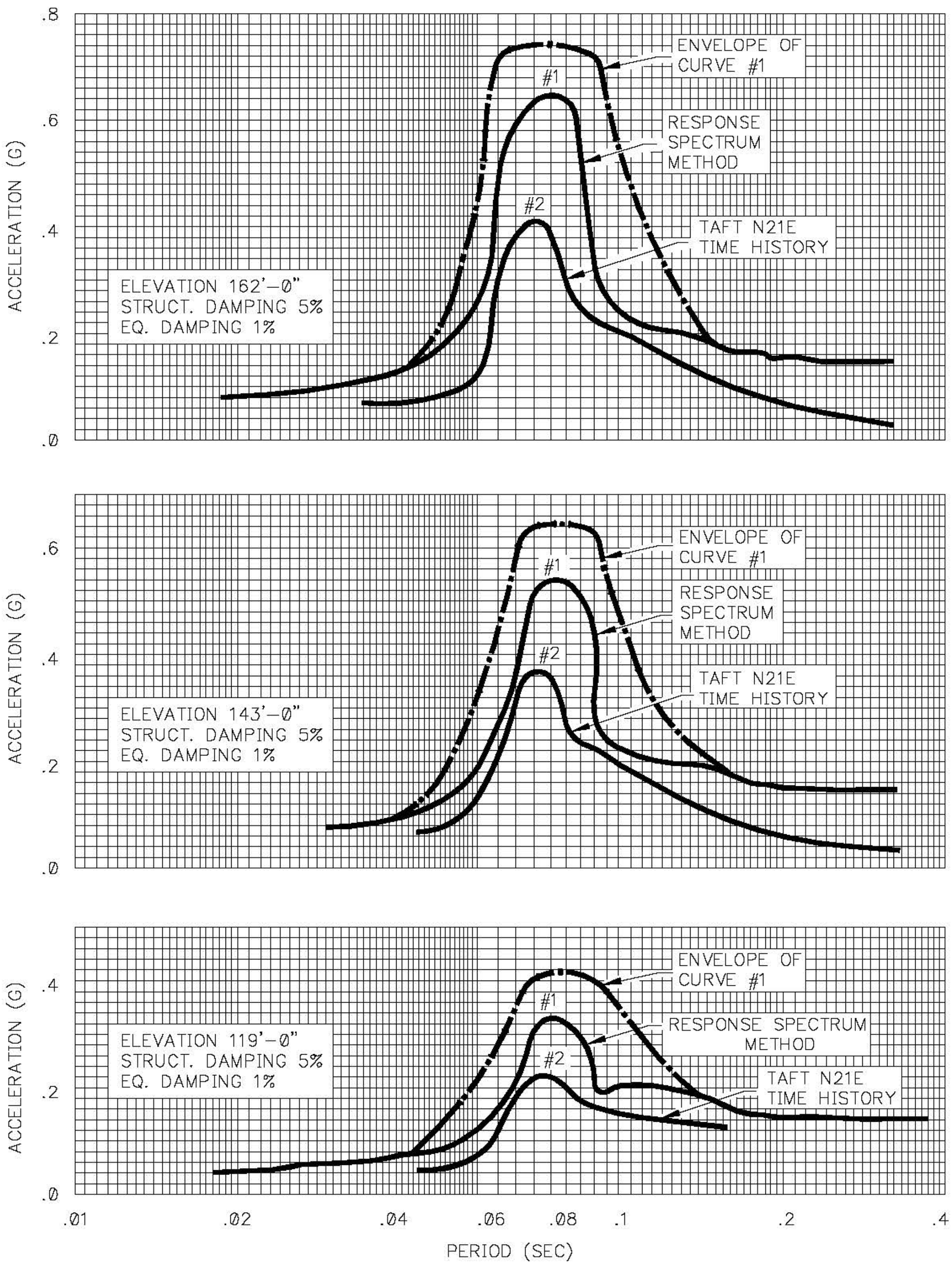


FIGURE 3-7, INTERIOR CONCRETE STRUCTURE FLOOR RESPONSE SPECTRA

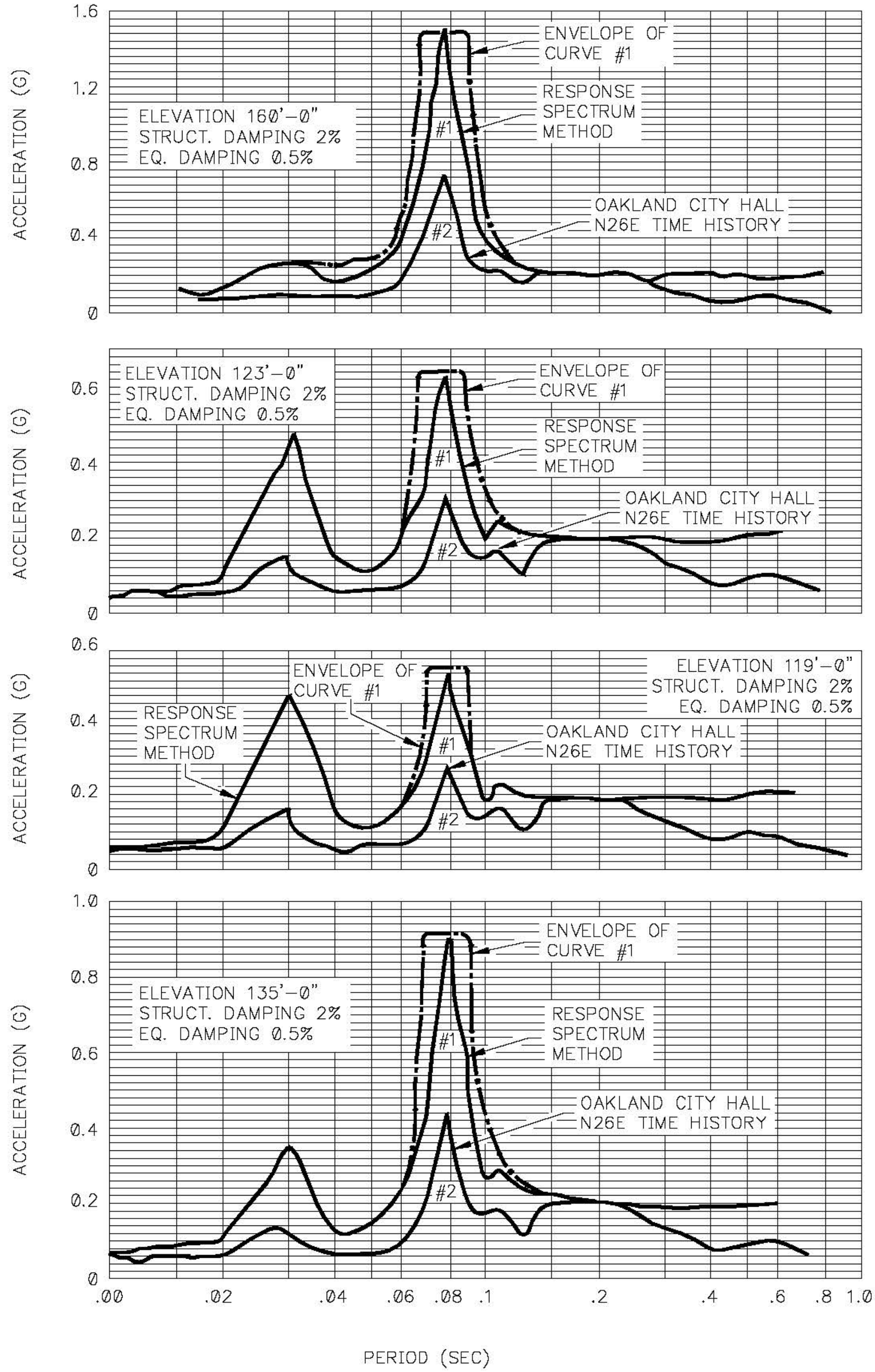


FIGURE 3-8, 230KV DISTRIBUTION

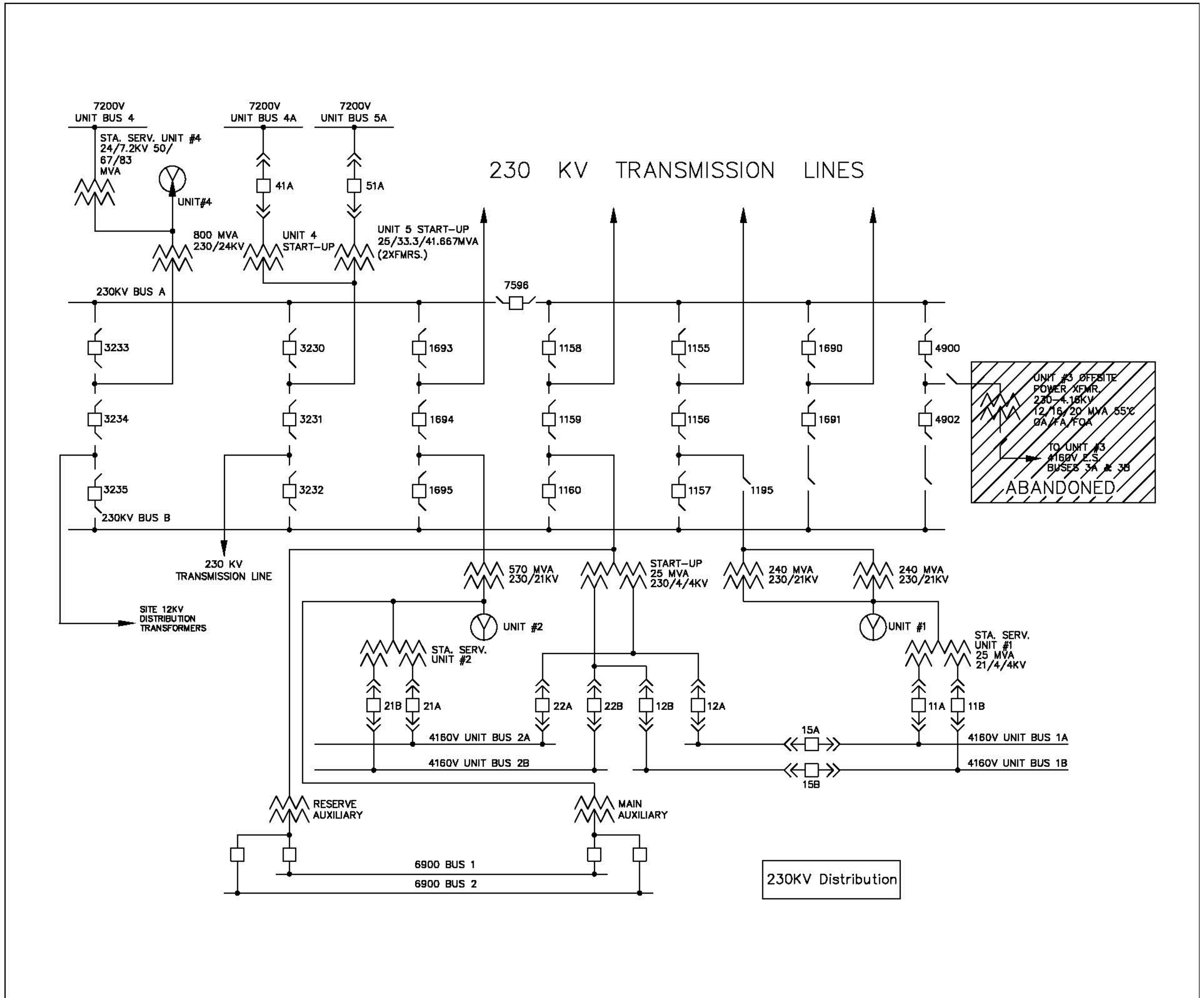


FIGURE 3-9, ELECTRICAL ONE LINE DIAGRAM: COMPOSITE, EC-206-011

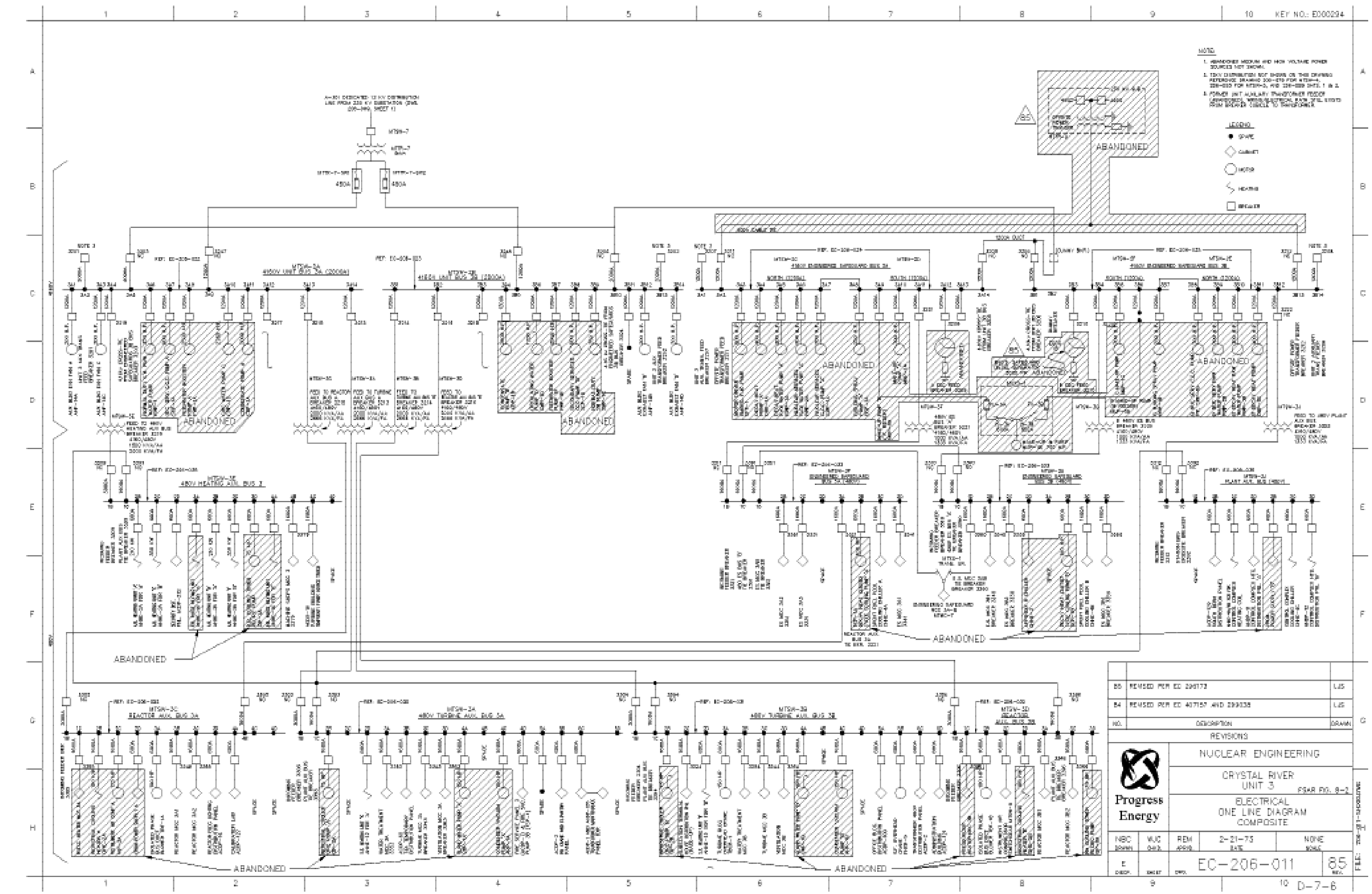


FIGURE 3-10, ELECTRICAL ONE LINE DIAGRAM: GENERATION & RELAYING 4160V BUS, EC-206-014

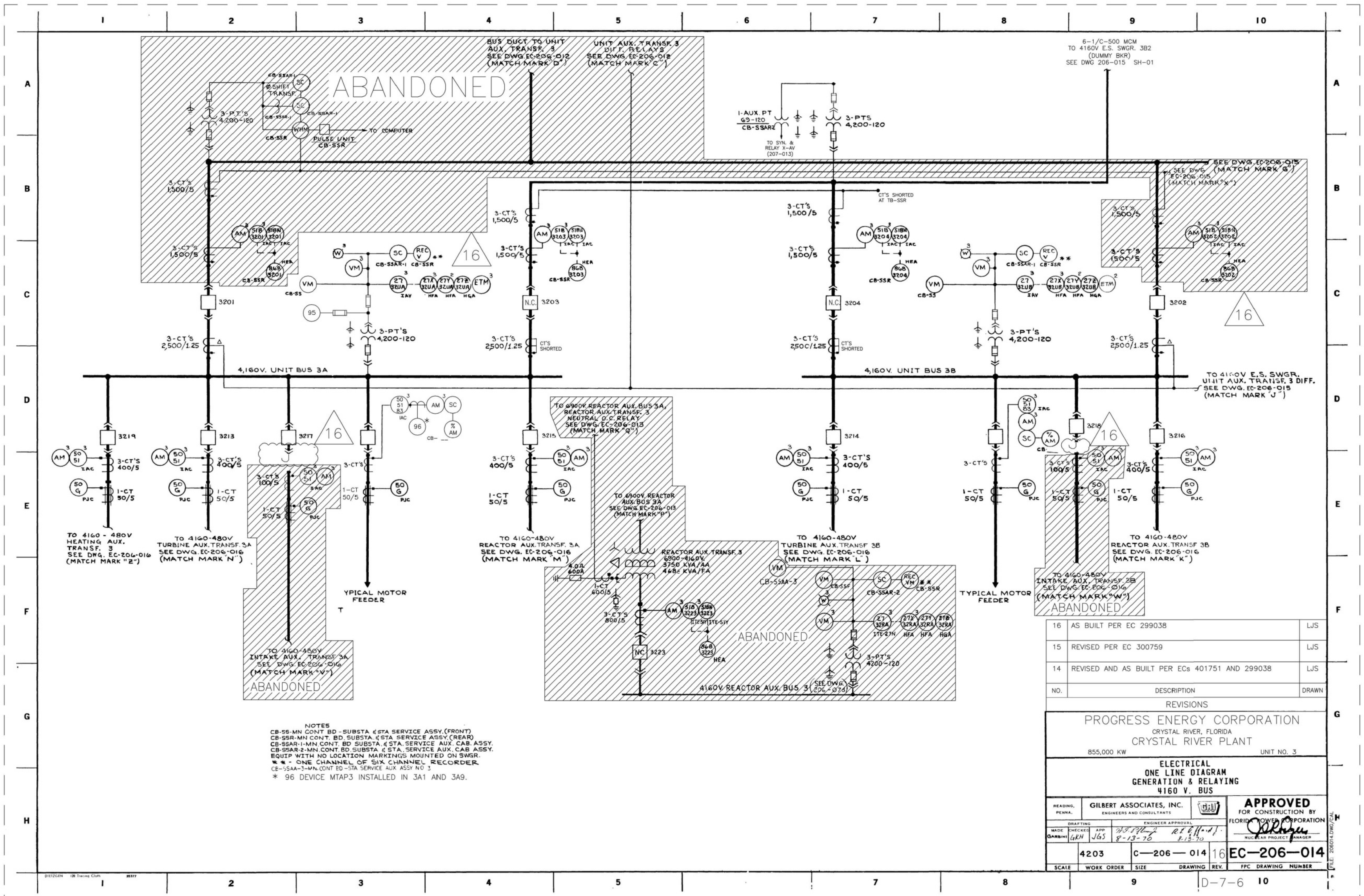


FIGURE 3-11, ELECTRICAL ONE LINE DIAGRM: GENERATION & RELAYING 4160V ENGINEERED SAFEGUARD BUS (1 SHEET), EC-206-015

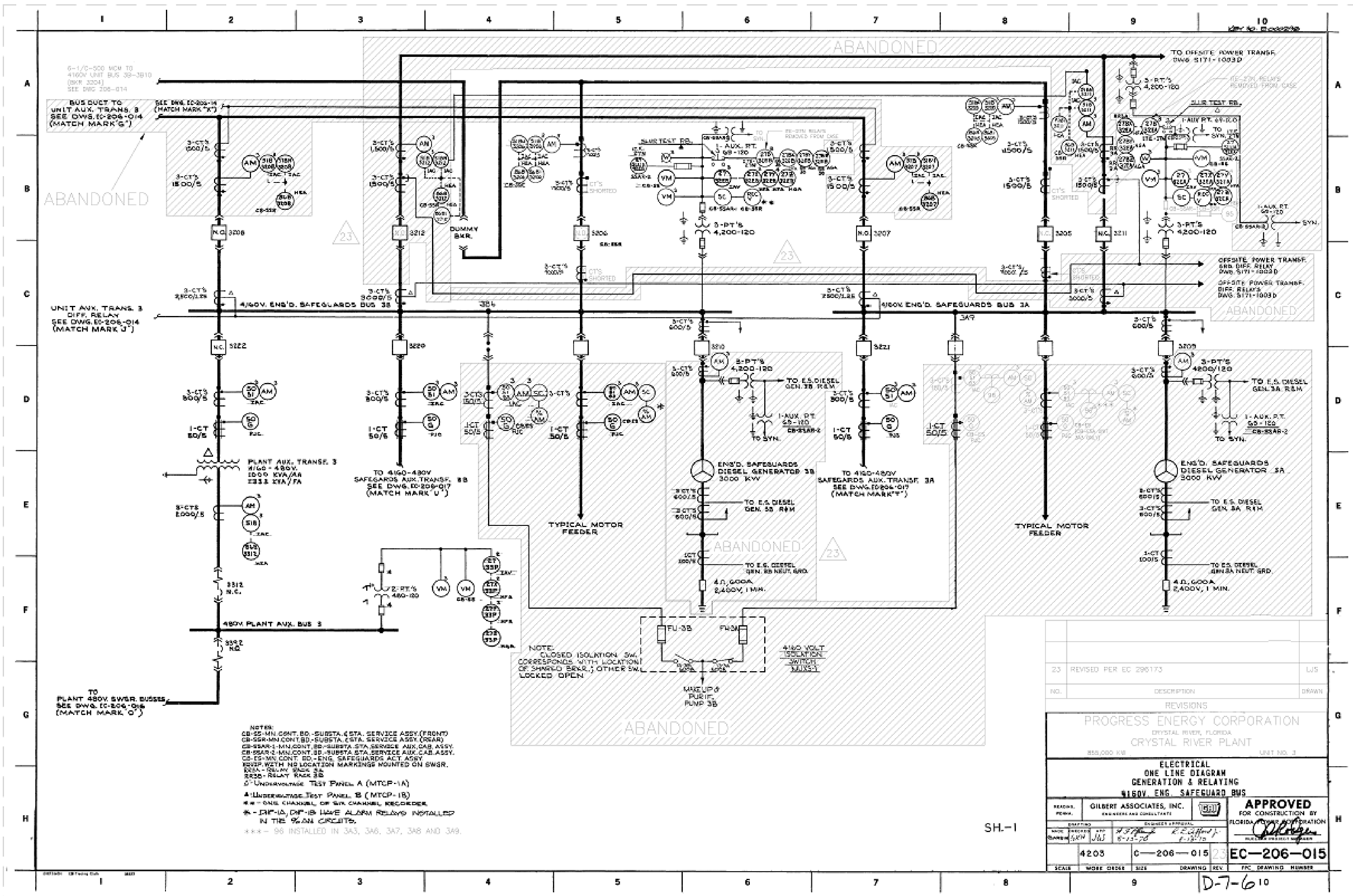


FIGURE 3-12, ELECTRICAL ONE LINE DIAGRAM: GENERATION & RELAYING 480V BUS, EC-206-016

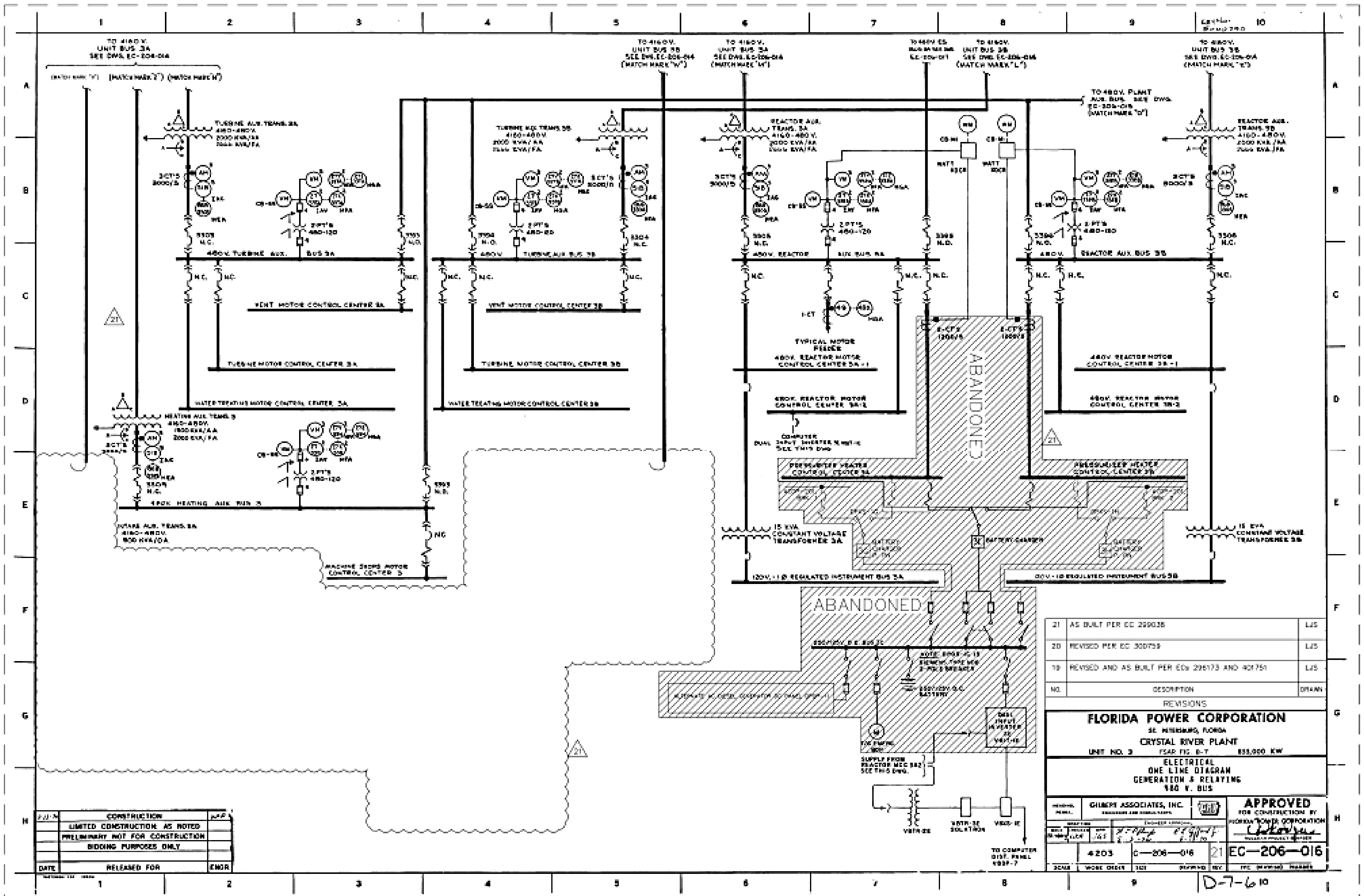


FIGURE 3-13, ELECTRICAL ONE LINE DIAGRAM: GENERATION & RELAYING 480V ENGINEERED SAFEGUARD BUS, EC-206-017

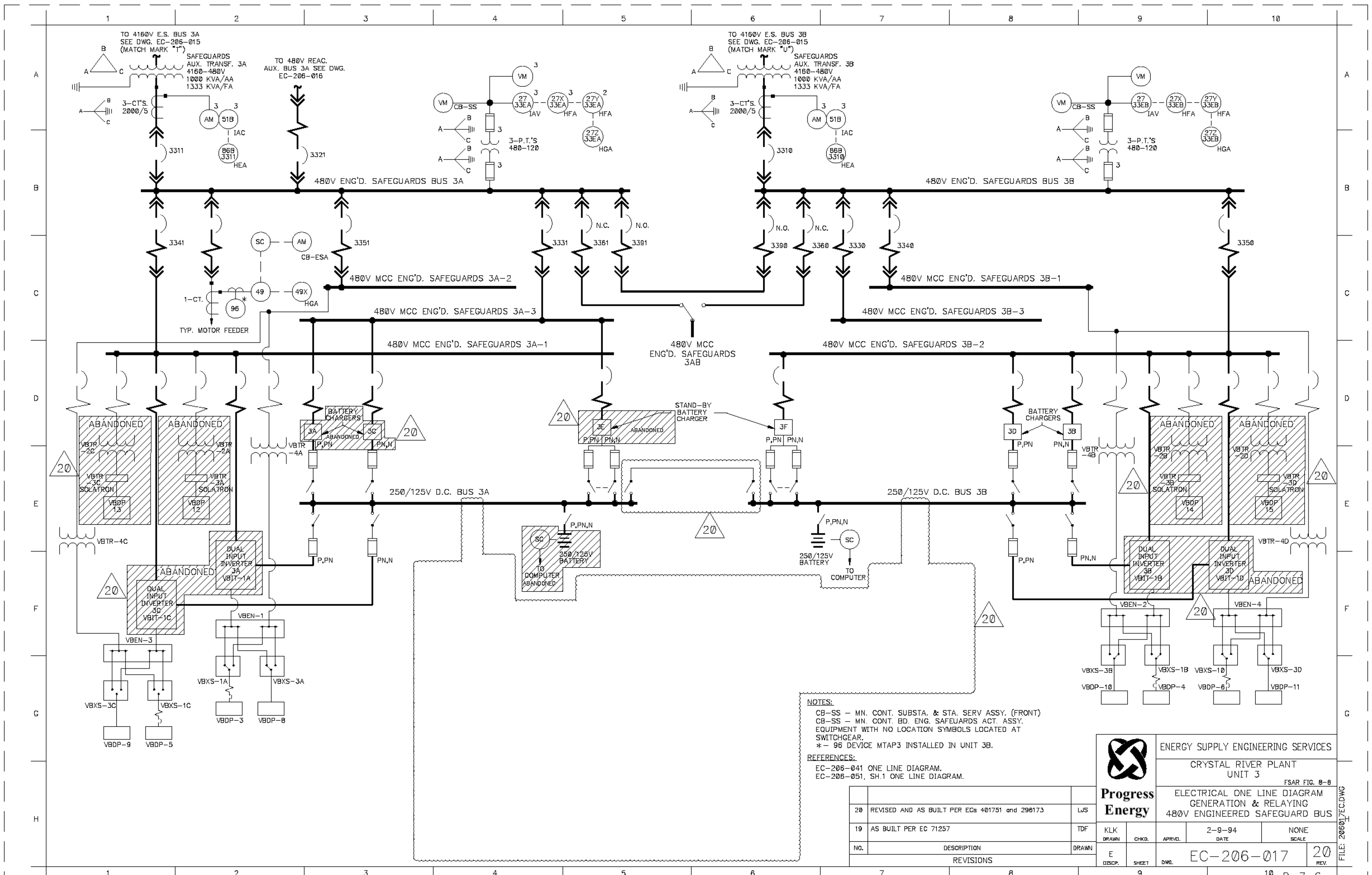


FIGURE 3-14, ELECTRICAL ONE LINE DIAGRAM: 250/125V DC SYSTEM, EC-206-051

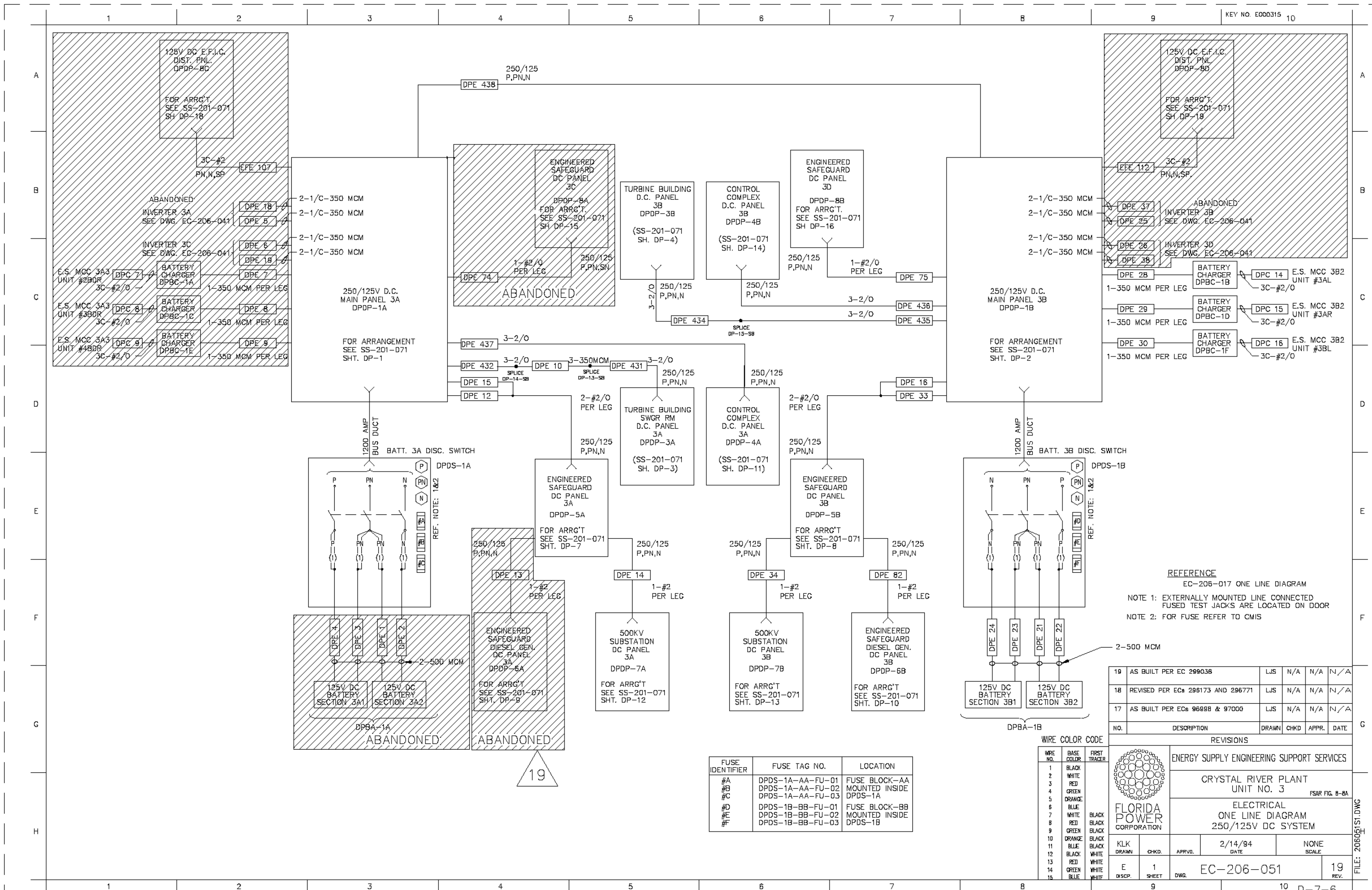


FIGURE 3-17A, FLOW DIAGRAM IDENTIFICATION, FD-302-001, SHEET 1

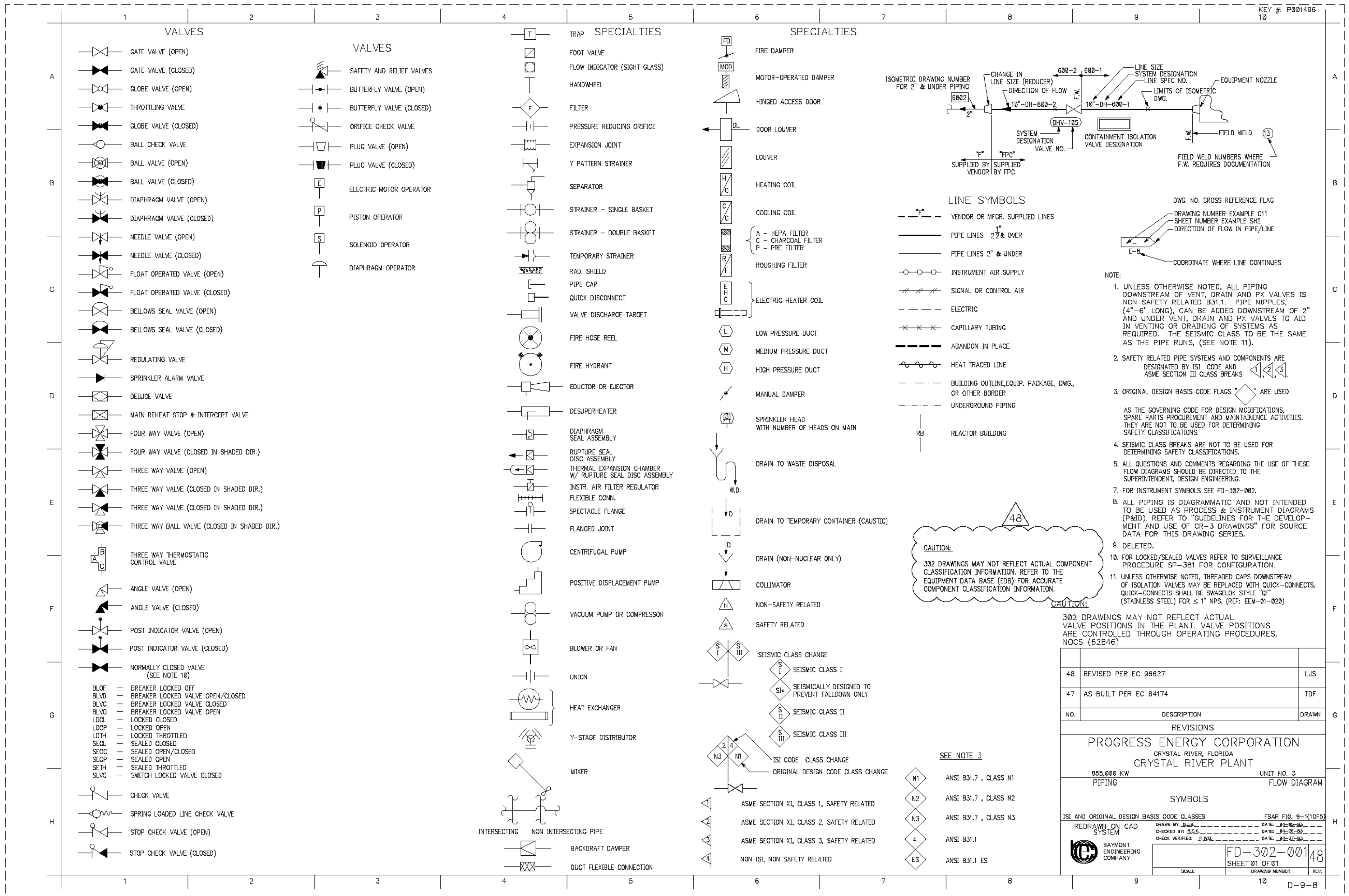
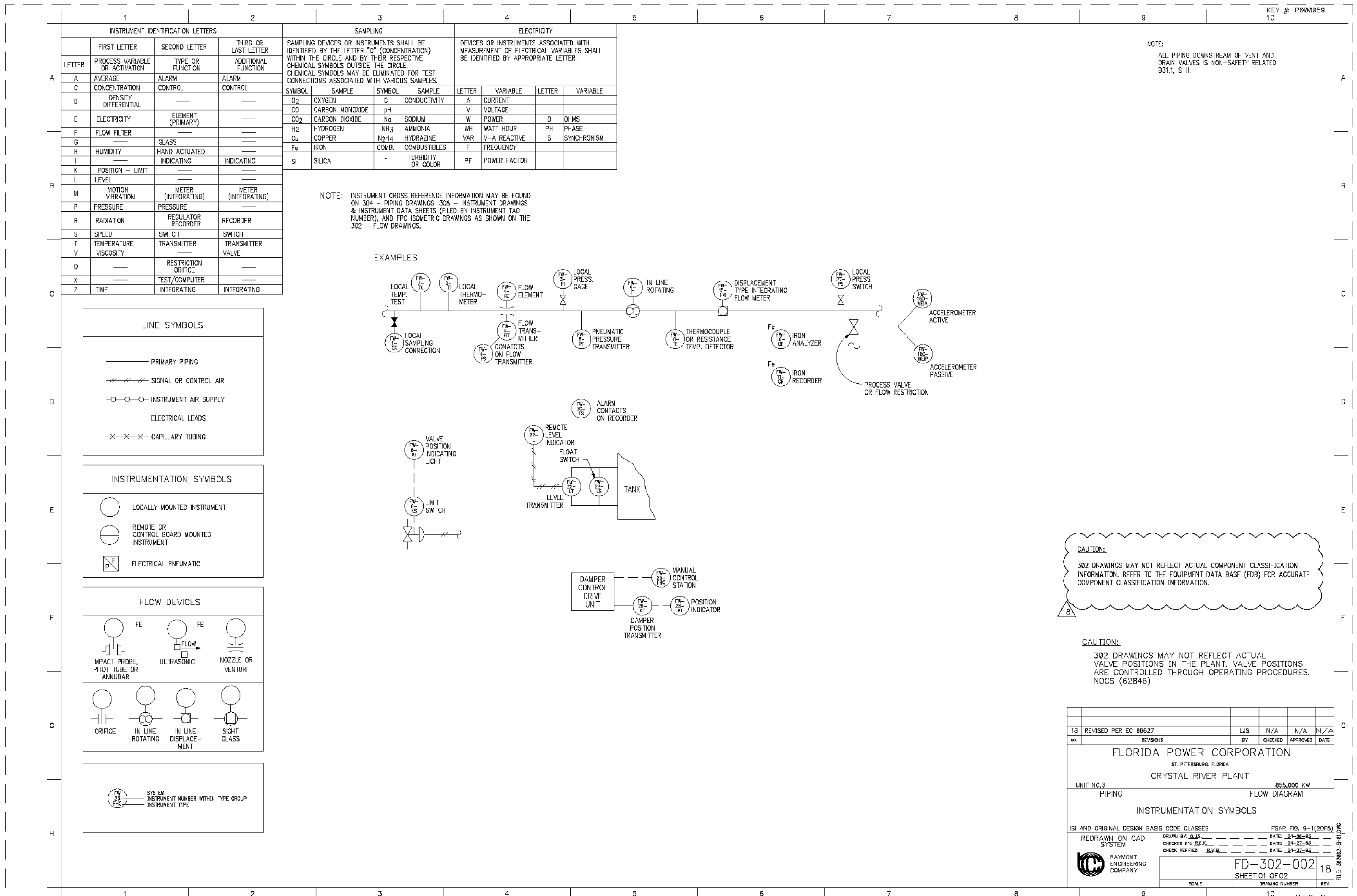


FIGURE 3-17B, FLOW DIAGRAM IDENTIFICATION, FD-302-002, SHEET 1



INSTRUMENT IDENTIFICATION LETTERS

LETTER	FIRST LETTER: PROCESS VARIABLE OR ACTIVATION	SECOND LETTER: TYPE OR FUNCTION	THIRD OR LAST LETTER: ADDITIONAL FUNCTION
A	AVERAGE	ALARM	ALARM
C	CONCENTRATION	CONTROL	CONTROL
D	DENSITY DIFFERENTIAL	—	—
E	ELECTRICITY	ELEMENT (PRIMARY)	—
F	FLOW FILTER	—	—
G	—	GLASS	—
H	HUMIDITY	HAND ACTUATED	—
I	—	INDICATING	INDICATING
K	POSITION – LIMIT	—	—
L	LEVEL	—	—
M	MOTION–VIBRATION	METER (INTEGRATING)	METER (INTEGRATING)
P	PRESSURE	PRESSURE	—
R	RADIATION	REGULATOR RECORDER	RECORDER
S	SPEED	SWITCH	SWITCH
T	TEMPERATURE	TRANSMITTER	TRANSMITTER
V	VISCOSITY	—	VALVE
O	—	RESTRICTION ORIFICE	—
X	—	TEST/COMPUTER	—
Z	TIME	INTEGRATING	INTEGRATING

SAMPLING

SAMPLING DEVICES OR INSTRUMENTS SHALL BE IDENTIFIED BY THE LETTER "C" (CONCENTRATION) WITHIN THE CIRCLE AND BY THEIR RESPECTIVE CHEMICAL SYMBOLS OUTSIDE THE CIRCLE. CHEMICAL SYMBOLS MAY BE ELIMINATED FOR TEST CONNECTIONS ASSOCIATED WITH VARIOUS SAMPLES.

SYMBOL	SAMPLE	SYMBOL	SAMPLE
O_2	OXYGEN	C	CONDUCTIVITY
CO	CARBON MONOXIDE	pH	
CO_2	CARBON DIOXIDE	Na	SODIUM
H_2	HYDROGEN	NH_3	AMMONIA
Cu	COPPER	N_2H_4	HYDRAZINE
Fe	IRON	COMB.	COMBUSTIBLES
Si	SILICA	T	TURBIDITY OR COLOR

ELECTRICITY

DEVICES OR INSTRUMENTS ASSOCIATED WITH MEASUREMENT OF ELECTRICAL VARIABLES SHALL BE IDENTIFIED BY APPROPRIATE LETTER.

LETTER	VARIABLE	LETTER	VARIABLE
A	CURRENT		
V	VOLTAGE		
W	POWER	O	OHMS
WH	WATT HOUR	PH	PHASE
VAR	V–A REACTIVE	S	SYNCHRONISM
F	FREQUENCY		
PF	POWER FACTOR		

FIGURE 3-17C, FLOW DIAGRAM IDENTIFICATION, FD-302-002, SHEET 2

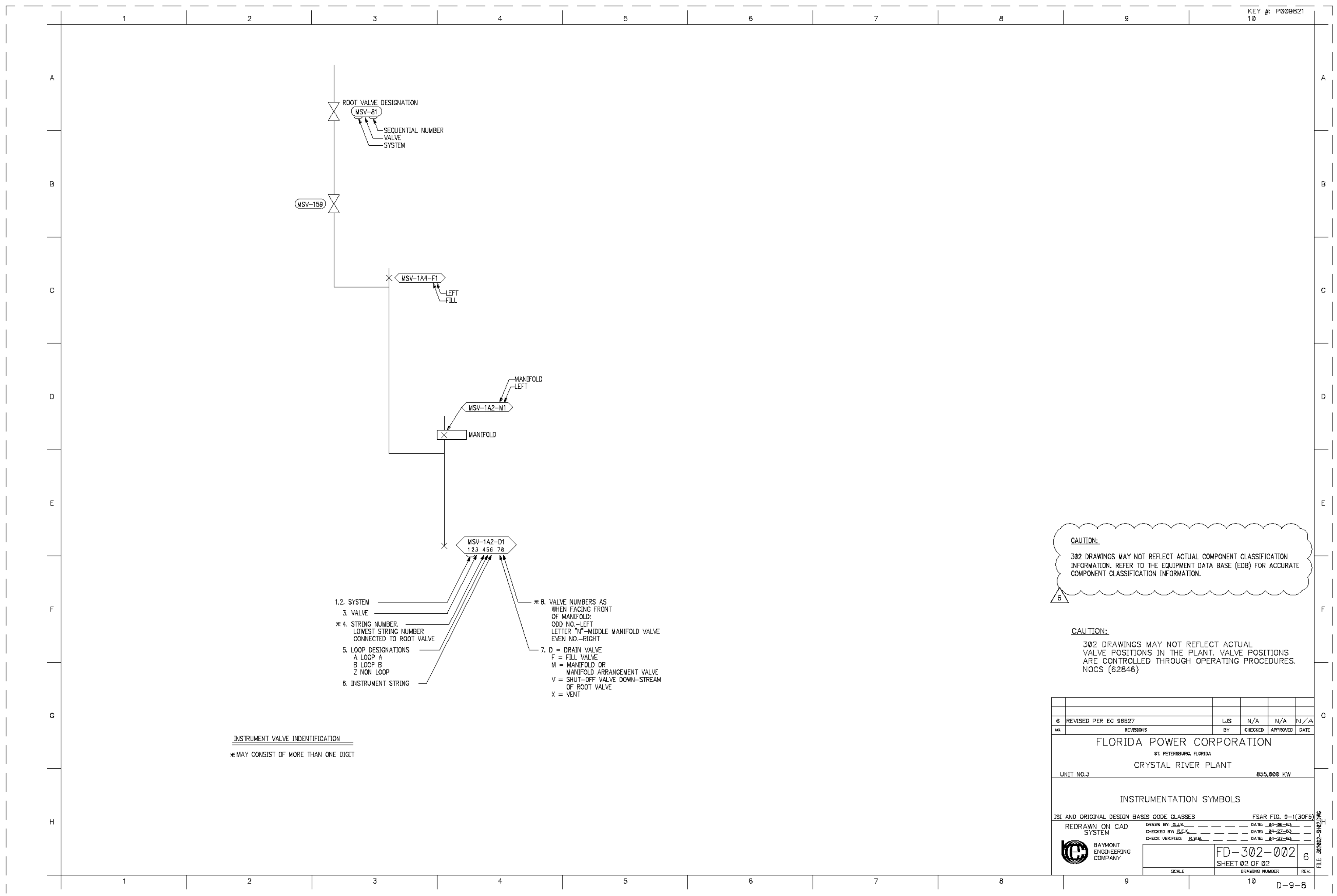


FIGURE 3-17D, FLOW DIAGRAM COMPONENT & SYSTEM IDENTIFICATION

COMPONENT IDENTIFICATION

COMPONENT PREFIX	COMPONENT
BA	BATTERY
BC	BATTERY CHARGER
BD	BUS DUCT
CC	CONTROL CENTER
CP	CONTROL PANEL
CR	CRANE OR BRIDGE
DG	DIESEL GENERATOR
DM	DEMINERALIZER
DP	DISTRIBUTION PANEL
DR	DRYER
DT	DRAIN TRAP
EJ	EXPANSION JOINT
EL	ELEVATOR
EV	EVAPORATOR
F	FAN OR EXHAUSTER
FL	FILTER
G	SLUICE GATE
GA	GAS ANALYZER
H	HANGER
HE	HEAT EXCHANGER, CONDENSOR, COOLING, DEARATOR
IN	INDUCTROL
M	MOTOR

COMPONENT PREFIX	COMPONENT
MS	MOISTURE SEPARATOR
MX	MIXER
P	PUMP, COMPRESSOR, VACUUM PUMP
PU	LUBE OIL PURIFIER
R	SEISMIC RESTRAINT
RD	CONTROL RODS AND DRIVE MECHANISMS
RE	REACTOR
S	STRAINER, RESIN TRAP
SB	SAMPLE BOMB
SG	STEAM GENERATOR
SN	SPRAY NOZZLE
SR	SAMPLE RACK
SW	SWITCHGEAR
T	TANK OR AIR RECEIVER
TB	TURBINE DRIVE
TG	TURBINE GENERATOR UNIT
TH	TRASH RAKE
TR	TRANSFORMER
TS	TRAVELING SCREEN
TU	TUBES
V	VALVES (ALL KINDS)
X	MISCELLANEOUS

SYSTEM IDENTIFICATION

SYSTEM PREFIX	SYSTEM
AC	MISCELLANEOUS A.C. DISTRIBUTION
AH	AIR HANDLING–VENTILATION AND COOLING
AR	CONDENSOR AIR REMOVAL
AS	AUXILIARY SYSTEM
BS	REACTOR BUILDING SPRAY
CA	CHEMICAL ADDITION AND SAMPLING–REACTOR
CC	CHEMICAL FEED–SECONDARY CYCLE
CD	CONDENSATE
CF	CORE FLOODING
CM	COMMUNICATIONS
CC	CO_2 GENERATOR PURGE
CP	COMPUTER
CW	CIRCULATING WATER
CX	CONDENSATION POLISHING DEMINERALIZER
DC	DECAY HEAT CLOSED CYCLE COOLING
DF	DIESEL FUEL
DH	DECAY HEAT REMOVAL
DO	DOMESTIC WATER
DP	D.C. SYSTEM
DR	CONTROL RODS
DW	DEMINERALIZER WATER–AUXILIARY BUILDING
EF	EMERGENCY FEEDWATER
EG	EMERGENCY DIESEL GENERATOR
ES	ENGINEERED SAFEGUARDS
EX	EXTRACTION STEAM
FH	NUCLEAR FUEL HANDLING
FS	FIRE SERVICE
FW	FEEDWATER
GS	GLAND STEAM
GW	GLAND SEAL WATER
HD	HEATER DRAINS
HT	HEAT TRACING
HV	HEATER VENTS
HY	HYDROGEN
IA	INSTRUMENT AIR
LO	LUBE OIL

SYSTEM PREFIX	SYSTEM
LR	REACTOR BUILDING LEAK RATE TESTING
MD	MISCELLANEOUS DRAINS
MS	MAIN STEAM
MT	AUXILIARY ELECTRICAL POWER
MU	MAKEUP AND PURIFICATION
MV	MISCELLANEOUS VENTS
NC	NUCLEAR INSTRUMENTATION IN–CORE AND OUT–OF–CORE
NI	NITROGEN
OX	OXYGEN
PC	PENETRATION COOLING
RC	REACTOR COOLANT
RD	ROOF DRAINS
RH	REHEAT STEAM
RM	RADIATION MONITORING
RV	RELIEF VALVE VENT
RW	NUCLEAR SERVICES & DECAY HEAT SEAWATER
SA	STATION AIR
SC	SECONDARY SERVICES CLOSED CYCLE COOLING
SD	STATION DRAINS
SF	SPENT FUEL COOLING
SP	SECONDARY PLANT
SS	SECONDARY CYCLE SAMPLING
SU	CYCLE START–UP
SW	NUCLEAR SERVICES CLOSED CYCLE COOLING
TB	TURBINE GENERATOR
TD	TURBINE DRAINS
VD	A.C. VITAL BUSES (120V DIST. PANELS)
WC	CIRCULATING WATER – CHLORINATION
WD	RADIOACTIVE WASTE DISPOSAL
WS	RADIOACTIVE WASTE SYSTEM SAMPLING
WT	CYCLE MAKEUP WATER TREATMENT

FIGURE 3-19, AUXILIARY BUILDING & FUEL HANDLING AREA: VENTILATION, FD-302-752

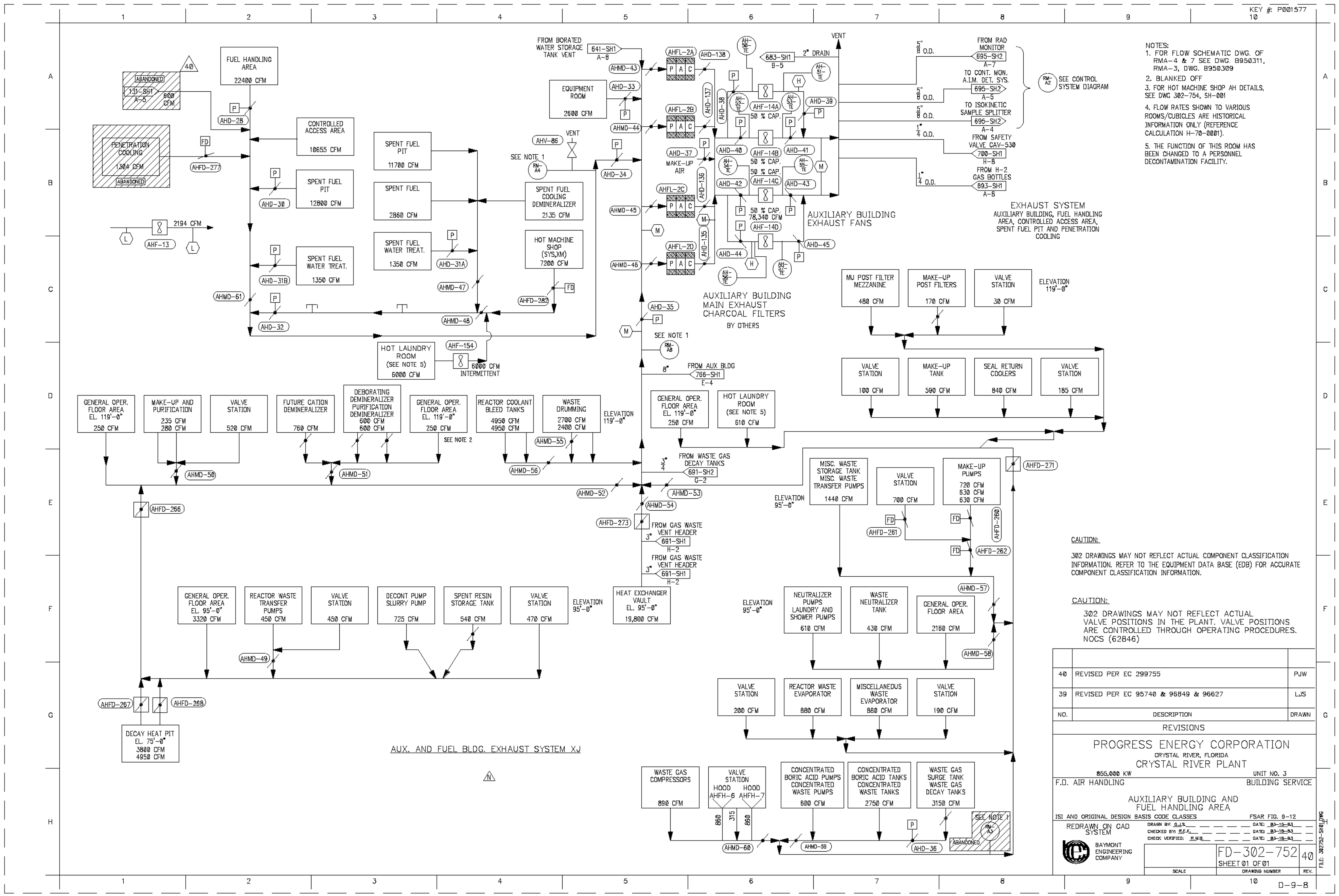


FIGURE 3-20A, CONTROL COMPLEX: VENTILATION, FD-302-753, SHEET 1

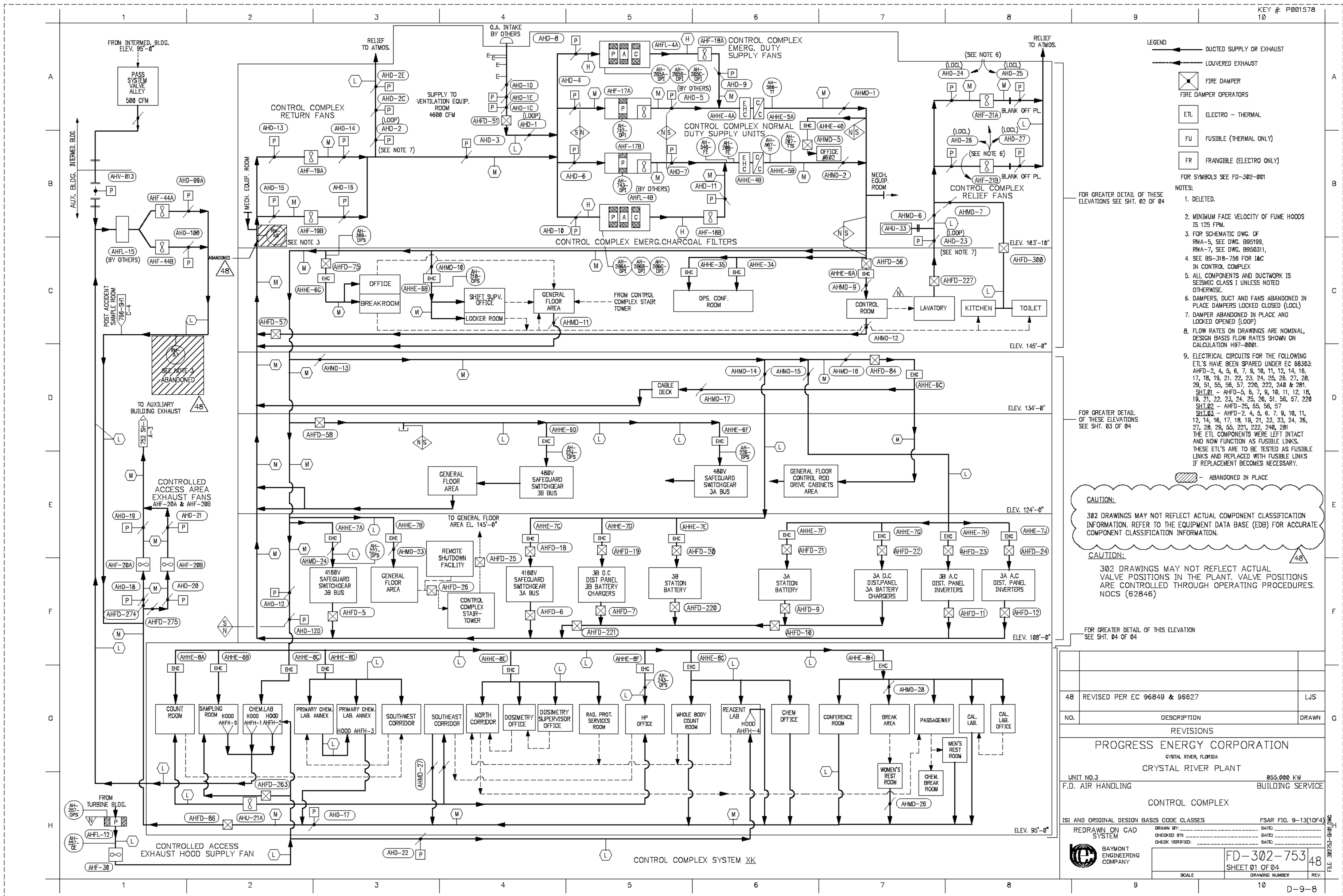


FIGURE 3-20B, CONTROL COMPLEX: VENTILATION, FD-302-753, SHEET 2

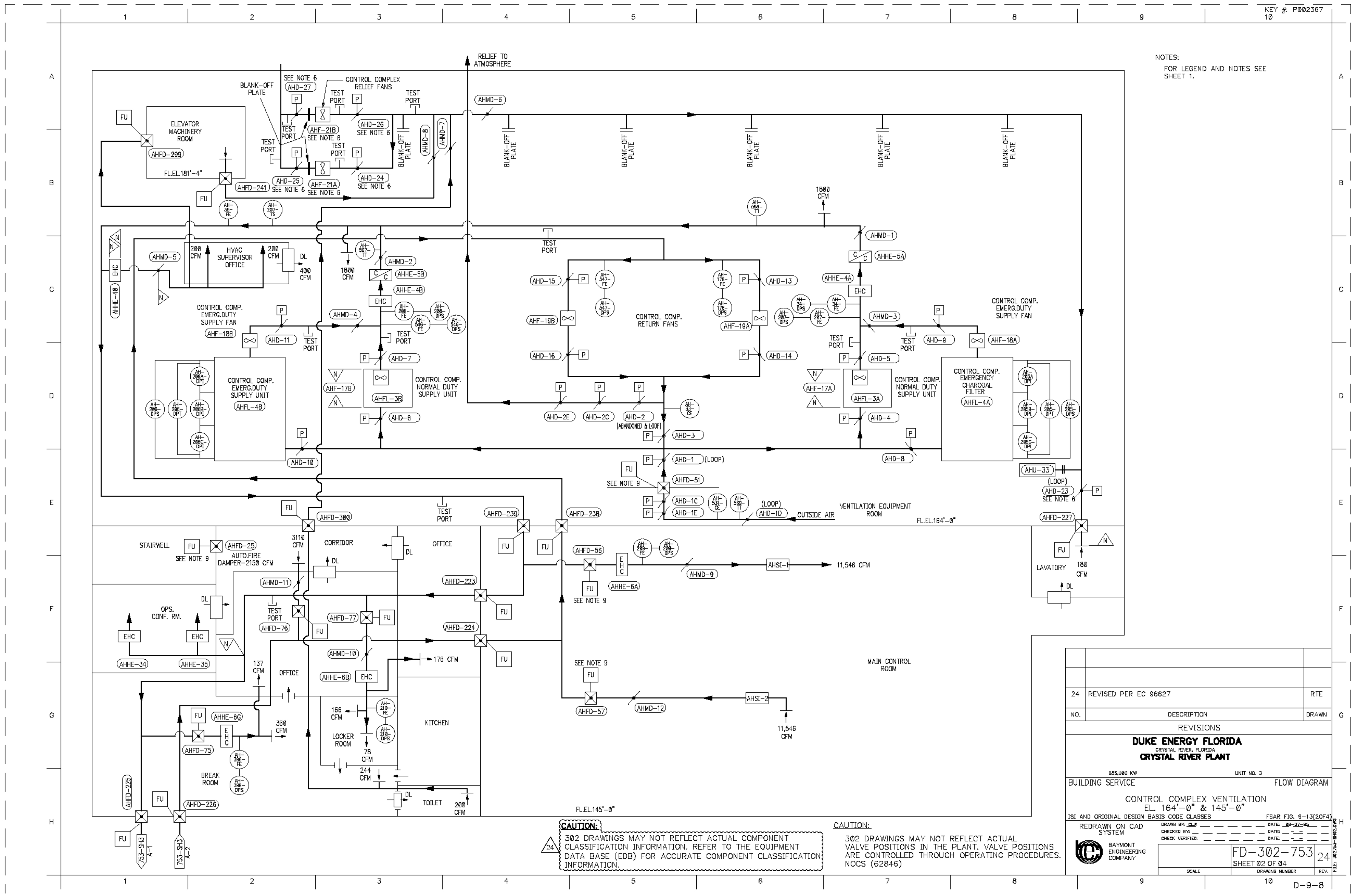


FIGURE 3-20C, CONTROL COMPLEX: VENTILATION, FD-302-753, SHEET 3

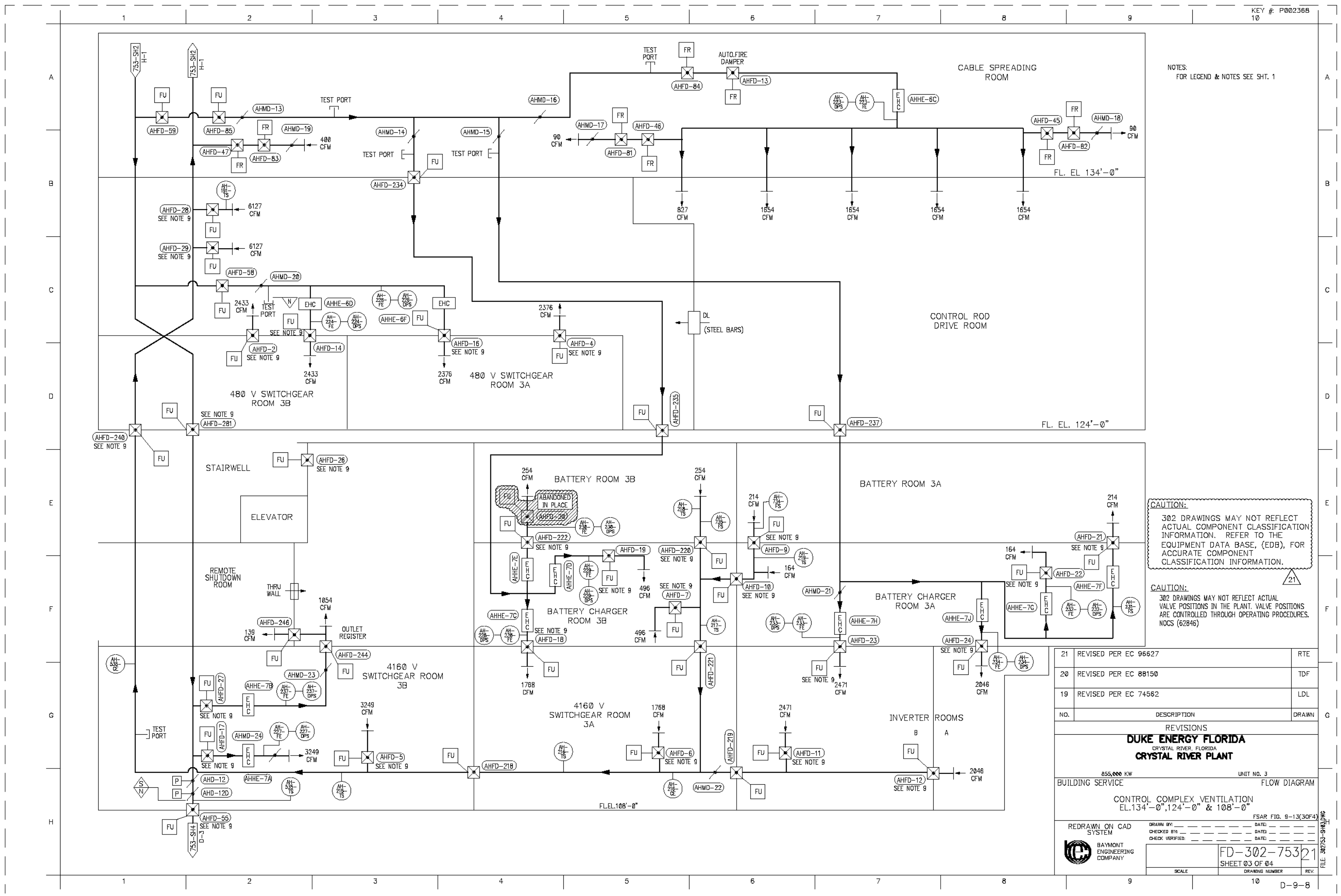
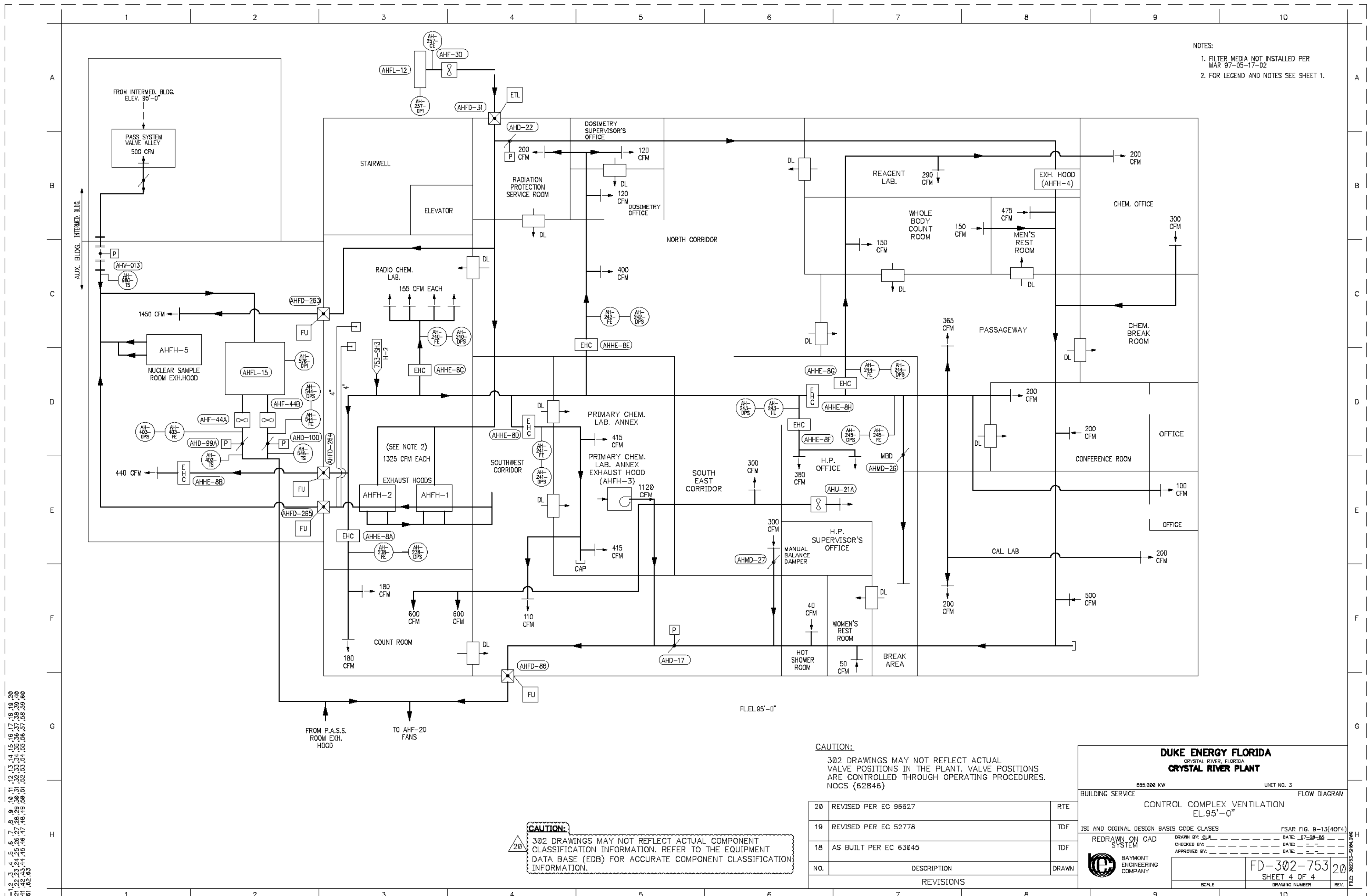


FIGURE 3-20D, CONTROL COMPLEX: VENTILATION, FD-302-753, SHEET 4



CAUTION:

302 DRAWINGS MAY NOT REFLECT ACTUAL VALVE POSITIONS IN THE PLANT. VALVE POSITIONS ARE CONTROLLED THROUGH OPERATING PROCEDURES. NOCS (62846)

NO.	DESCRIPTION	DRAWN
20	REVISED PER EC 96627	RTE
19	REVISED PER EC 52778	TDF
18	AS BUILT PER EC 63045	TDF

REVISIONS

DUKE ENERGY FLORIDA
CRYSTAL RIVER, FLORIDA
CRYSTAL RIVER PLANT
855,000 KW — UNIT NO. 3
BUILDING SERVICE — FLOW DIAGRAM
CONTROL COMPLEX VENTILATION EL.95'-0"
ISI AND OIGINAL DESIGN BASIS CODE CLASES — FSAR FIG. 9-13(4OF4)
REDRAWN ON CAD SYSTEM
DRAWN BY: CLW DATE: 07-28-85
CHECKED BY: DATE:
APPROVED BY: DATE:
BAYMONT ENGINEERING COMPANY
FD-302-753 SHEET 4 OF 4 — REV. 20
SCALE — DRAWING NUMBER — REV.
FILE: 302753-SH04.DWG

FIGURE 3-21A, TURBINE AREA & MISCELLANEOUS OUT BLDGS: VENTILATION, FD-302-754, SHEET 1

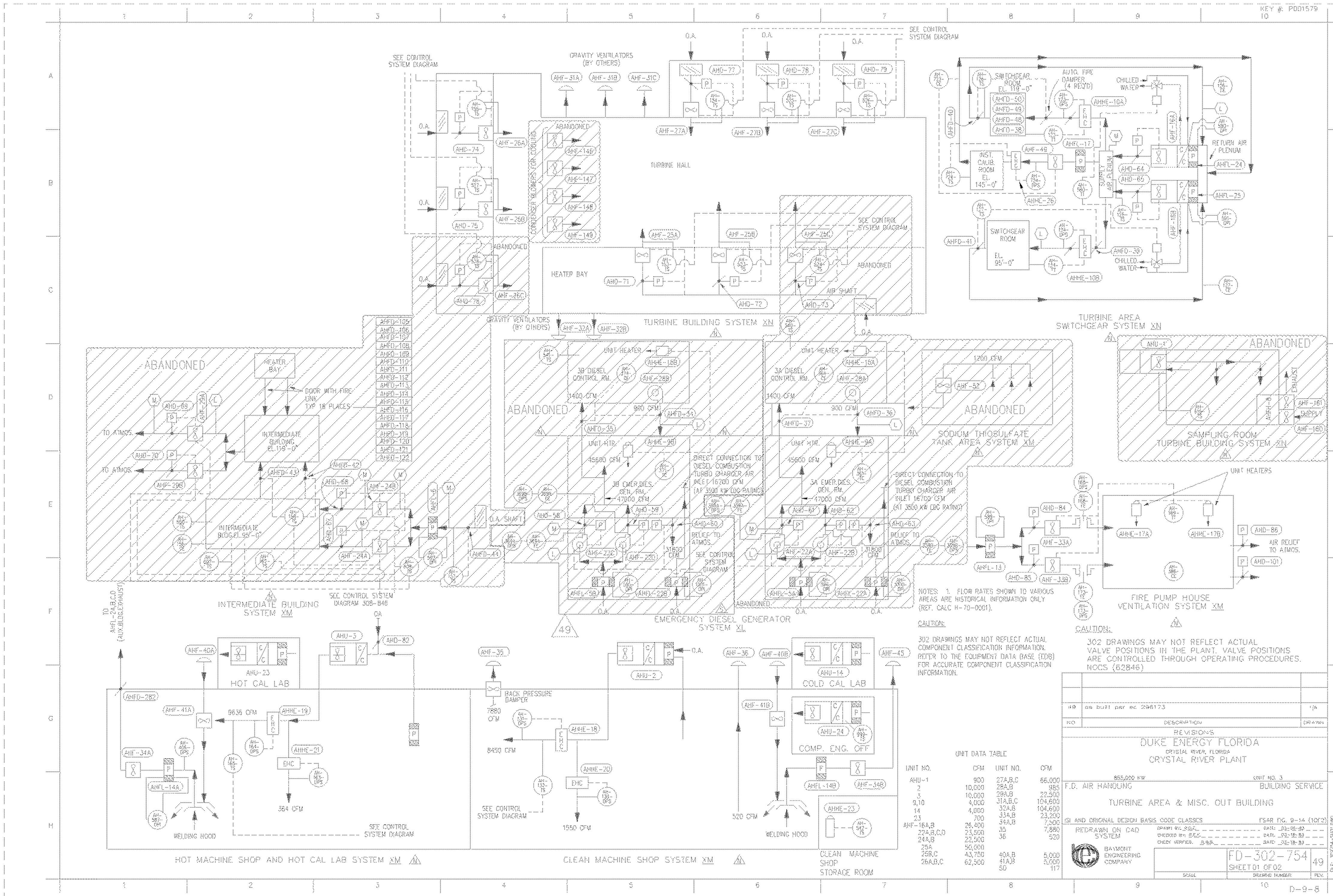


FIGURE 3-21B, TURBINE AREA & MISCELLANEOUS OUT BLDGS: VENTILATION, FD-302-754, SHEET 2

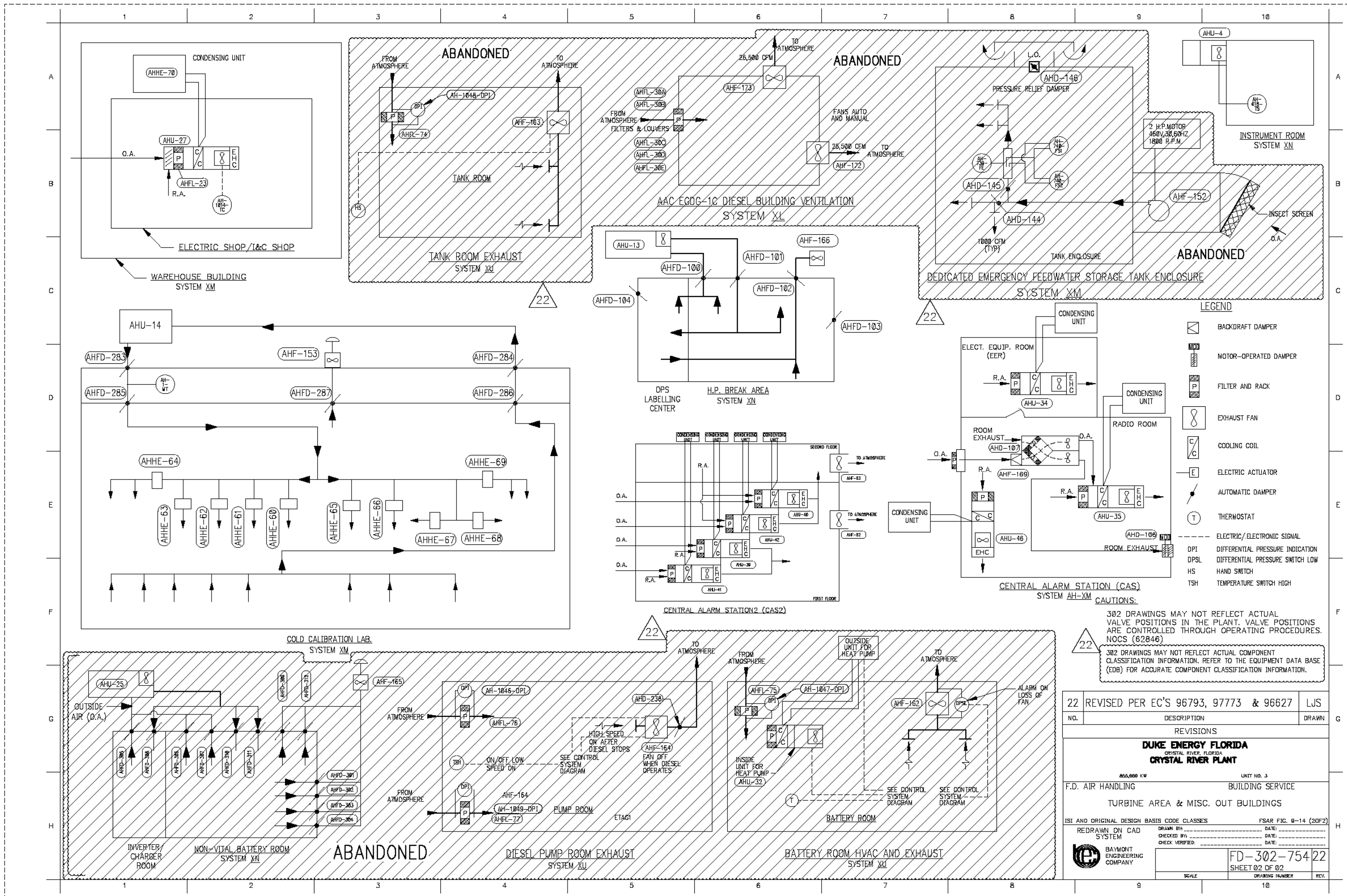


FIGURE 3-22, REACTOR BUILDING PENETRATION COOLING, FD-302-755

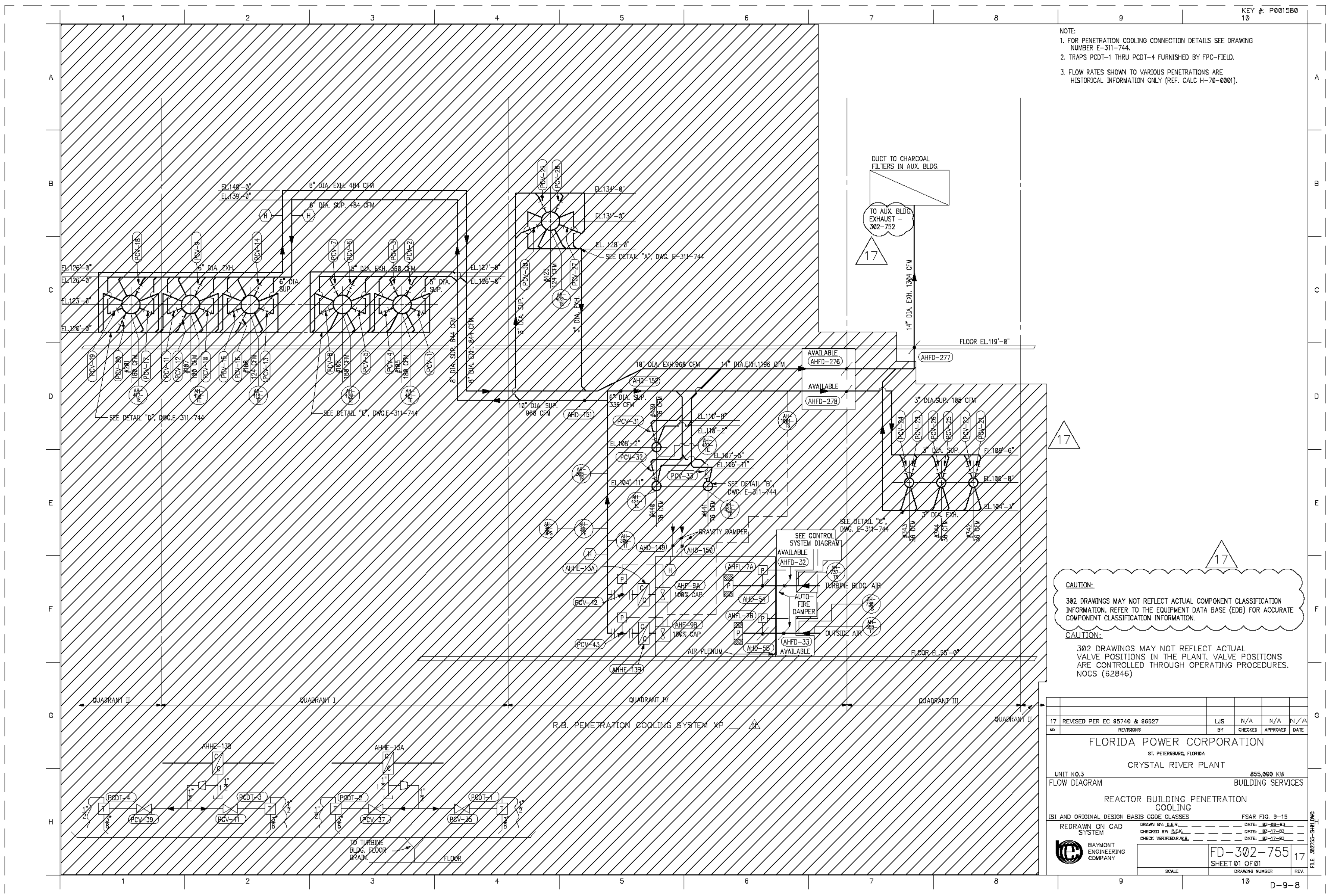


FIGURE 3-23, CHILLED WATER SYSTEM, FD-302-756

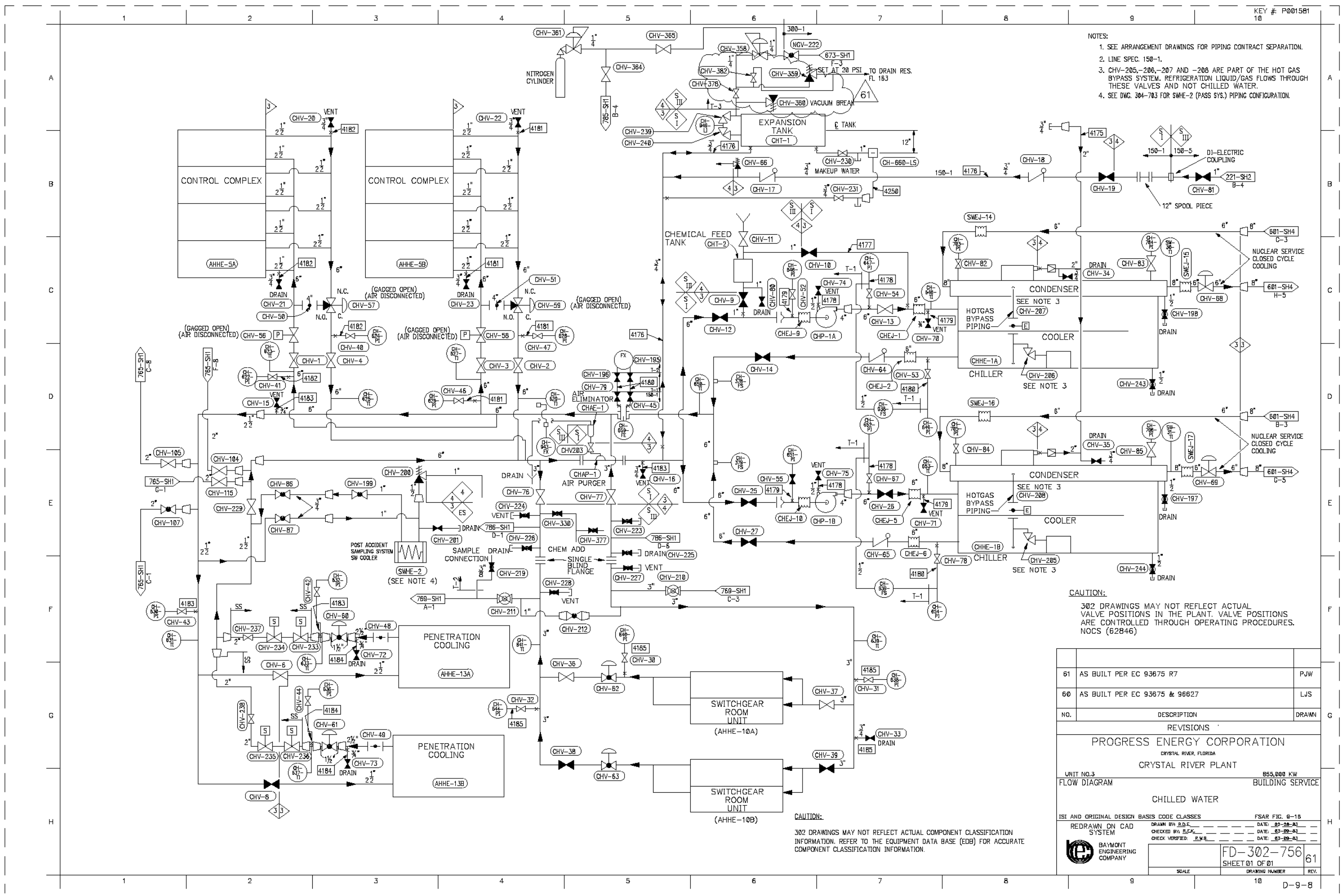


FIGURE 3-24A, INSTRUMENT AIR AND STATION SERVICE AIR, FD-302-271, SHEET 1

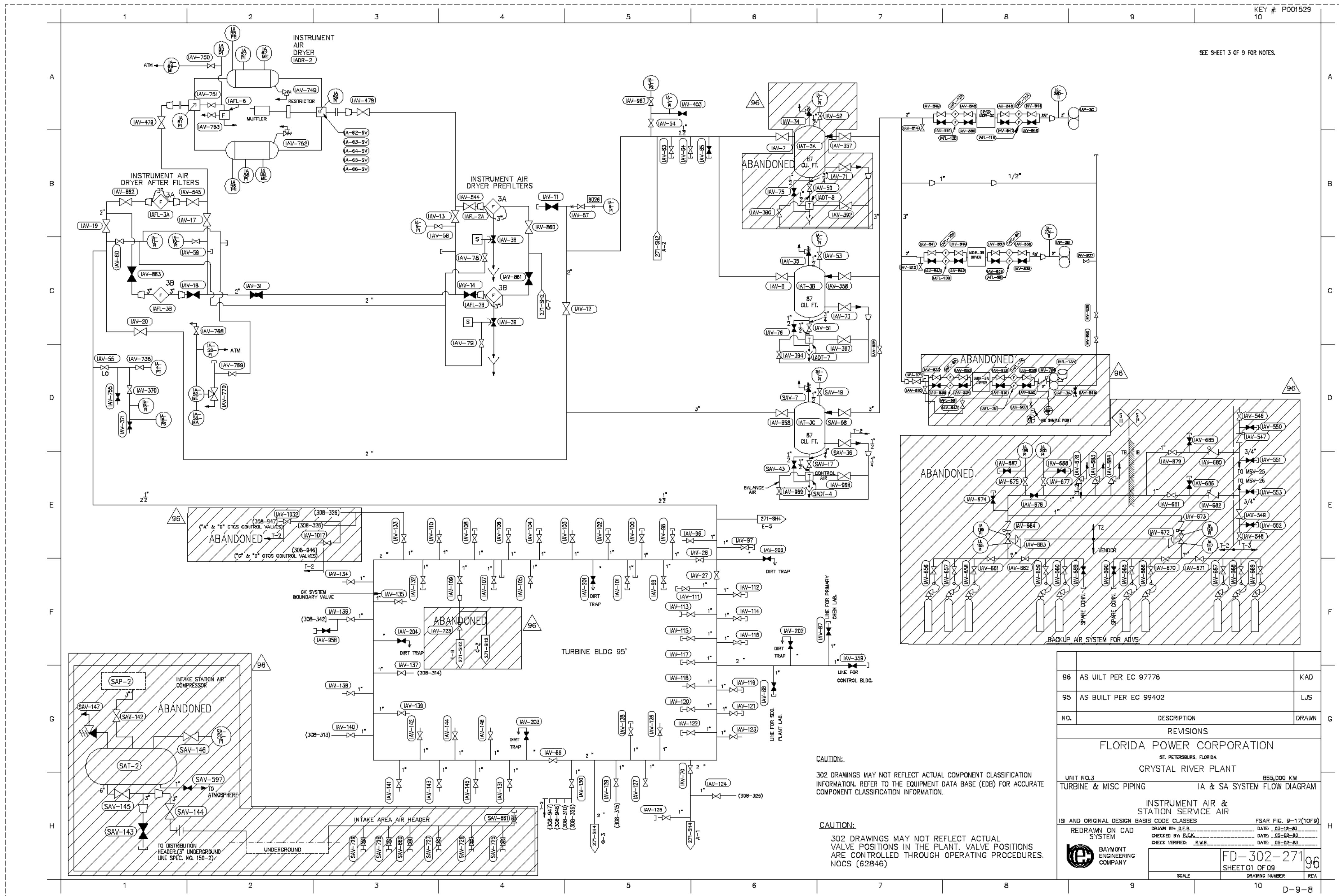


FIGURE 3-24B, INSTRUMENT AIR AND STATION SERVICE AIR, FD-302-271, SHEET 2

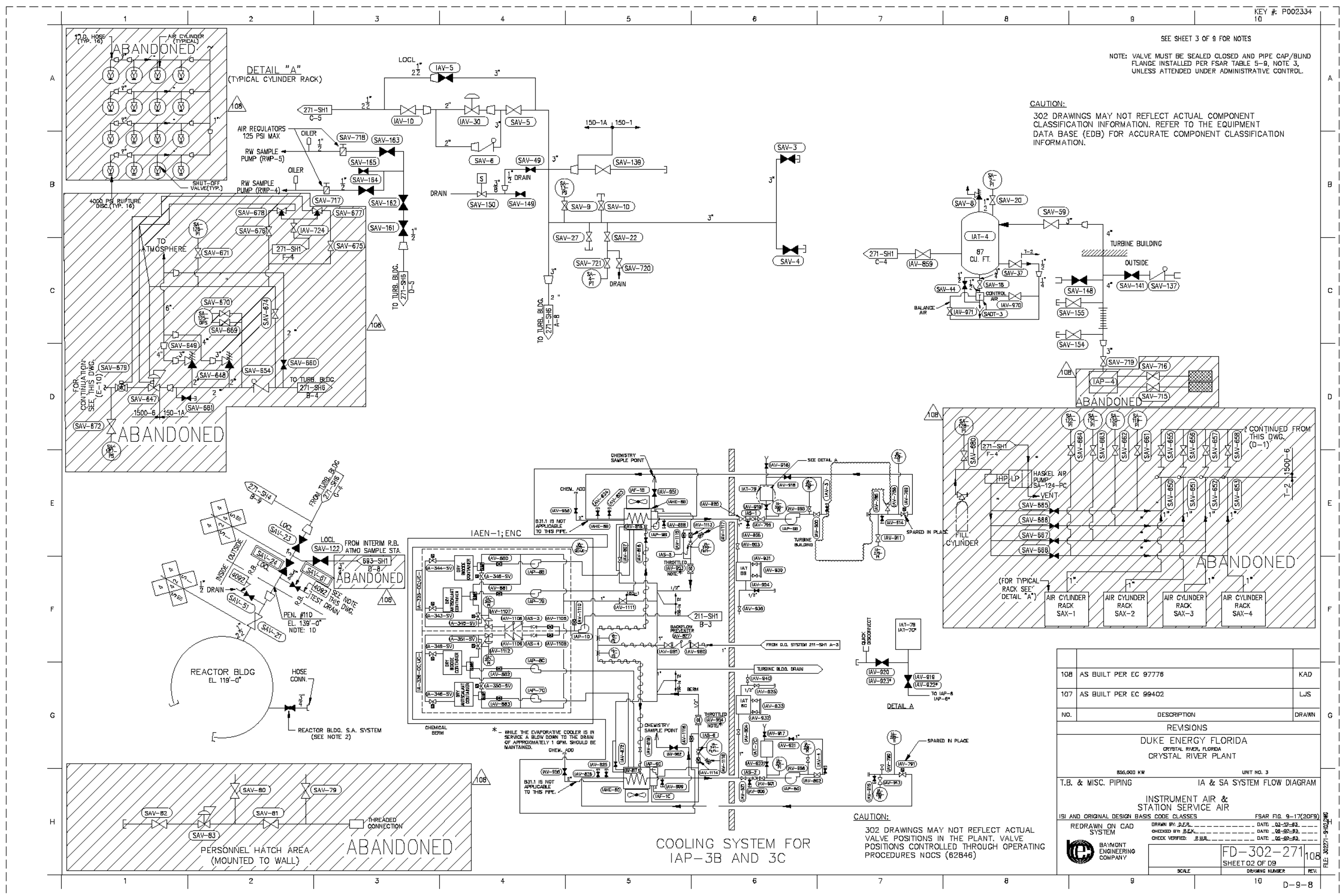


FIGURE 3-24C, INSTRUMENT AIR AND STATION SERVICE AIR, FD-302-271, SHEET 3

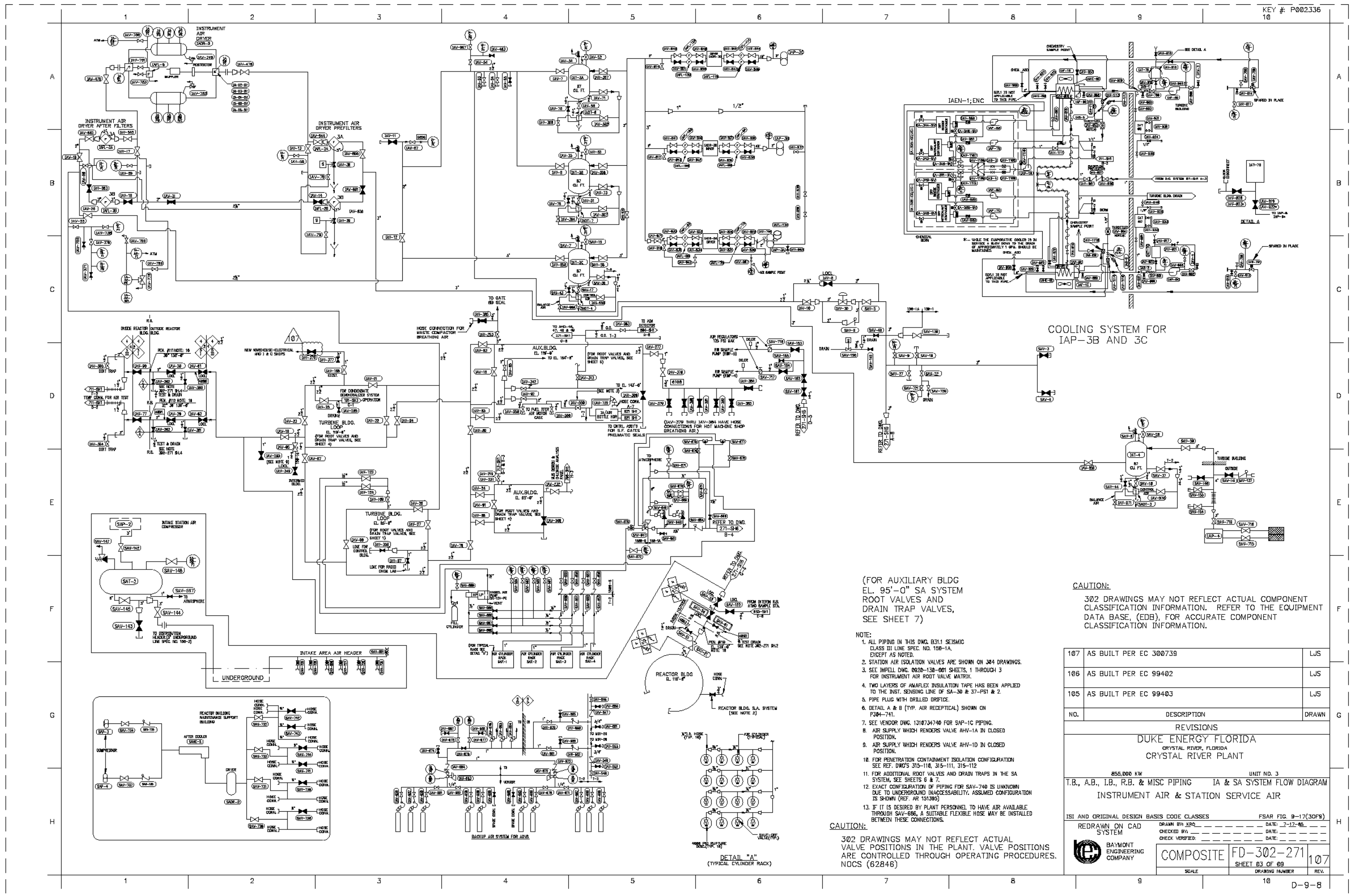


FIGURE 3-24D, INSTRUMENT AIR AND STATION SERVICE AIR, FD-302-271, SHEET 4

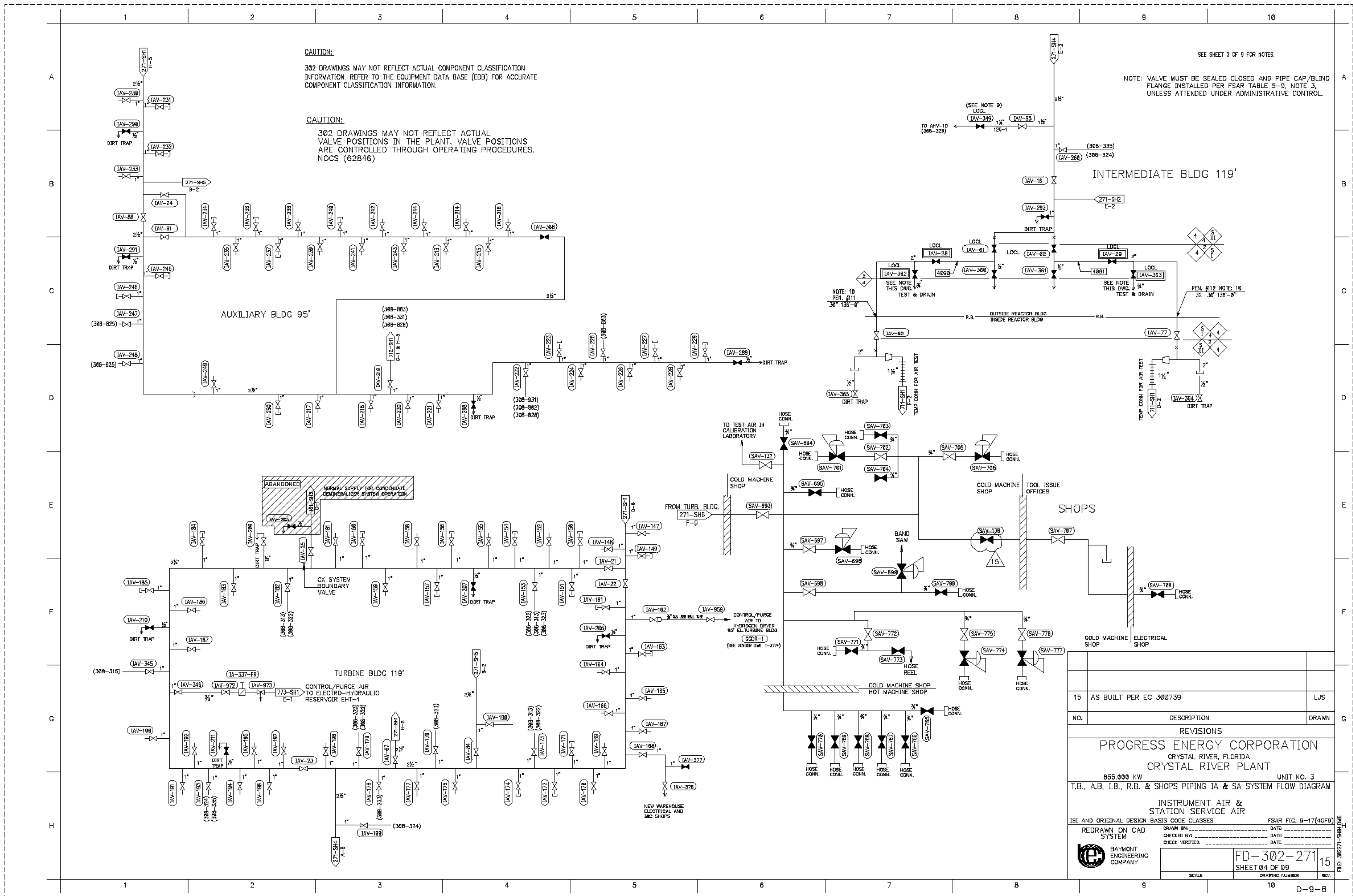


FIGURE 3-24E, INSTRUMENT AIR AND STATION SERVICE AIR, FD-302-271, SHEET 5

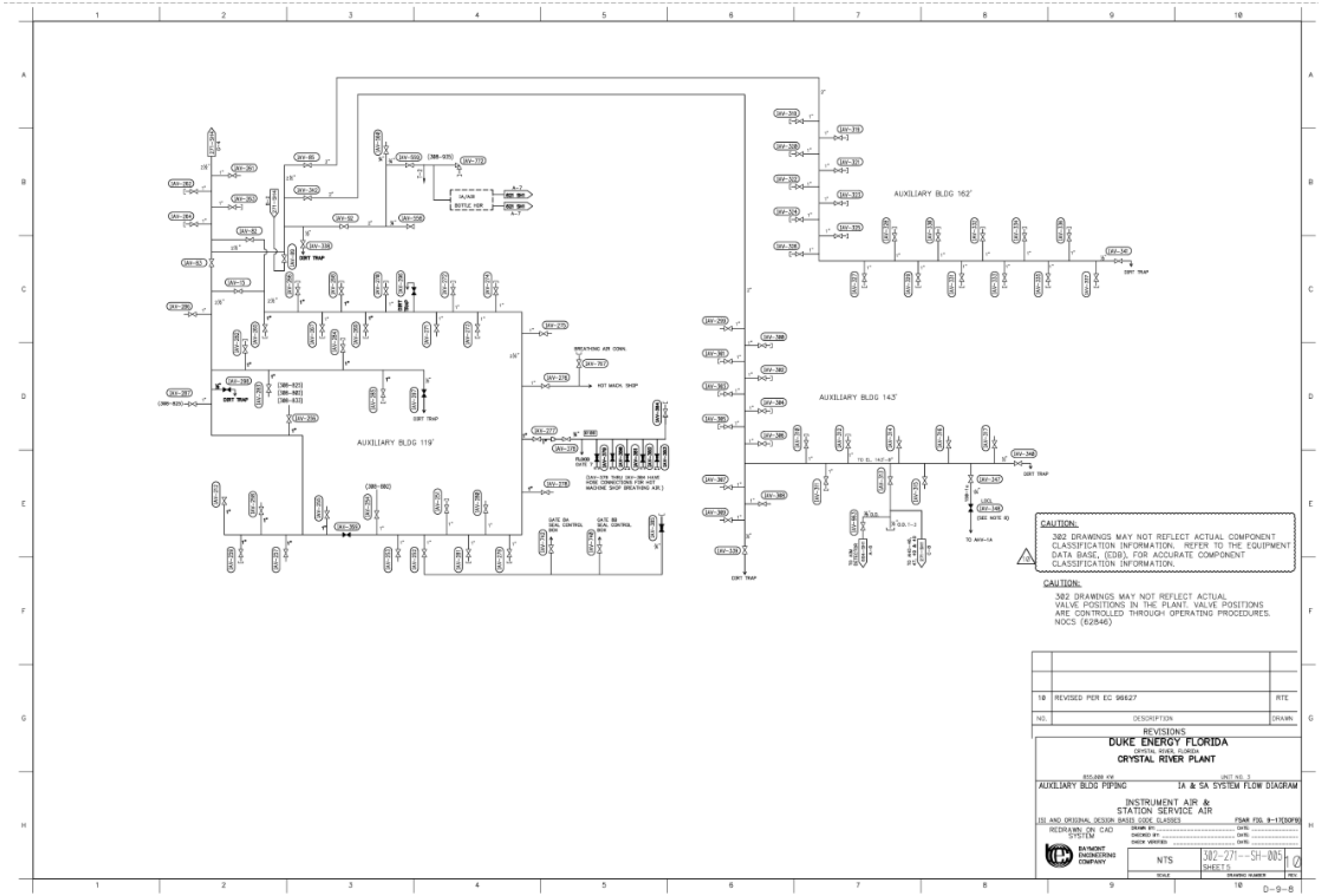


FIGURE 3-24F, INSTRUMENT AIR AND STATION SERVICE AIR, FD-302-271, SHEET 6

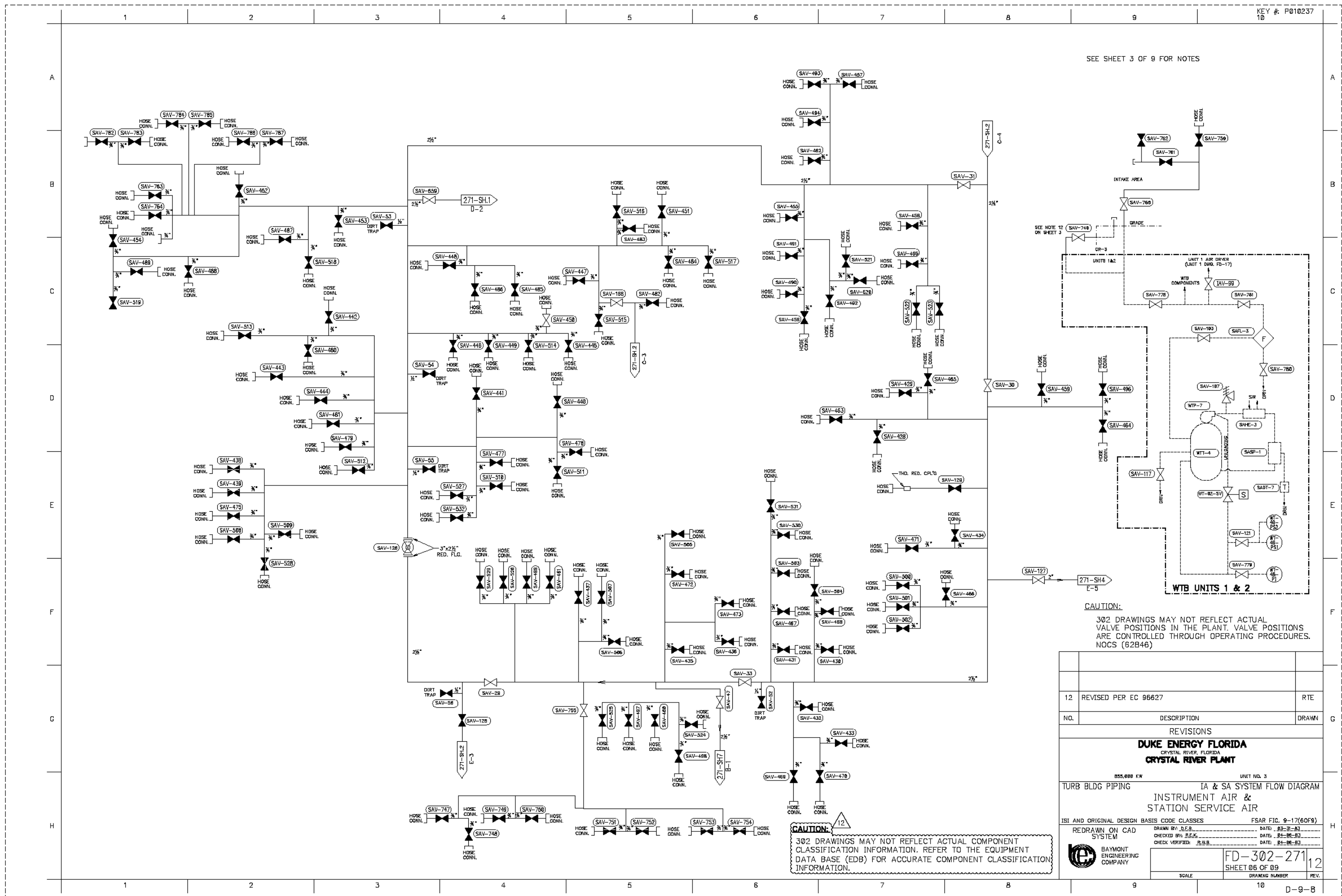


FIGURE 3-24G, INSTRUMENT AIR AND STATION SERVICE AIR, FD-302-271, SHEET 7

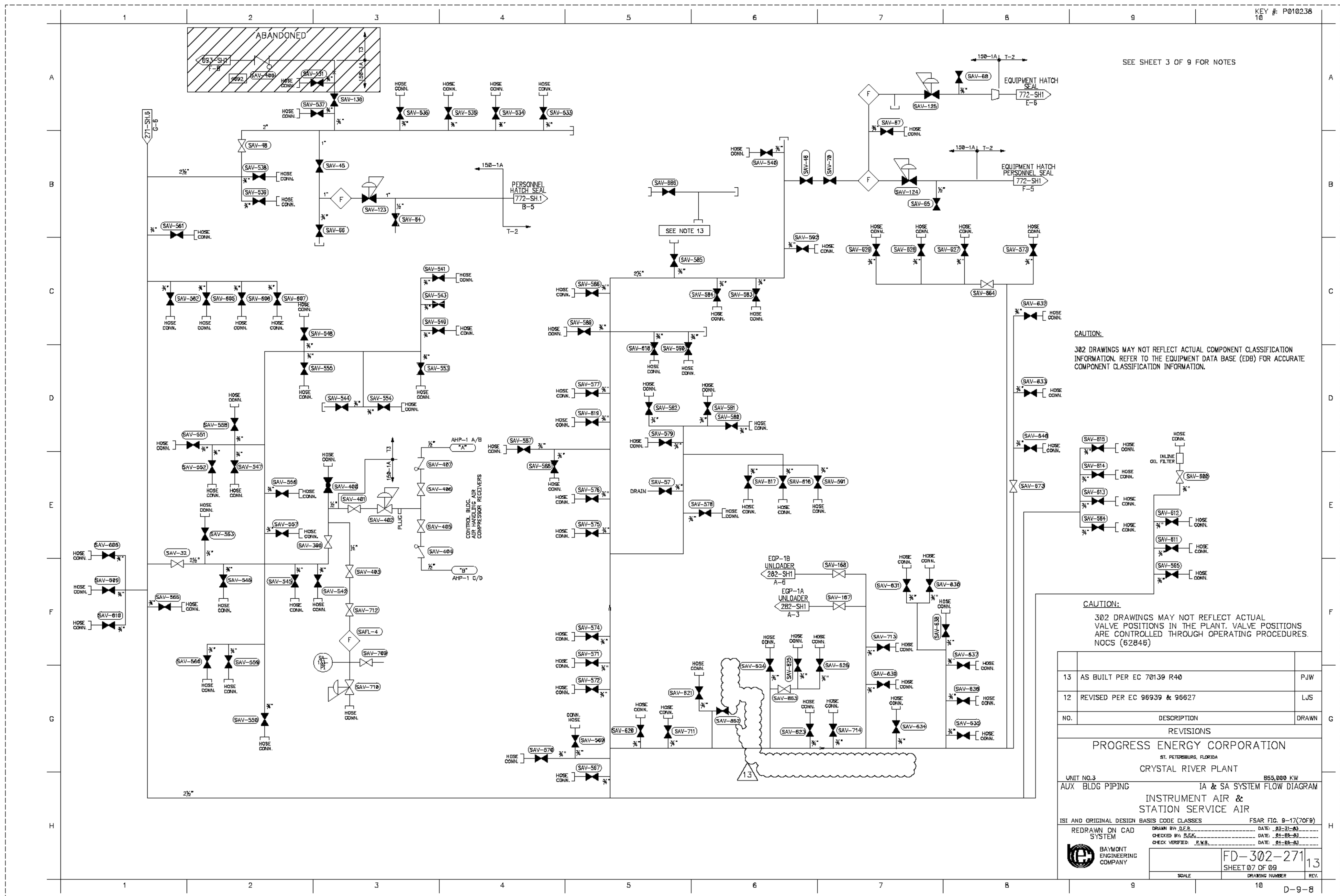


FIGURE 3-24H, INSTRUMENT AIR AND STATION SERVICE AIR, FD-302-271, SHEET 8

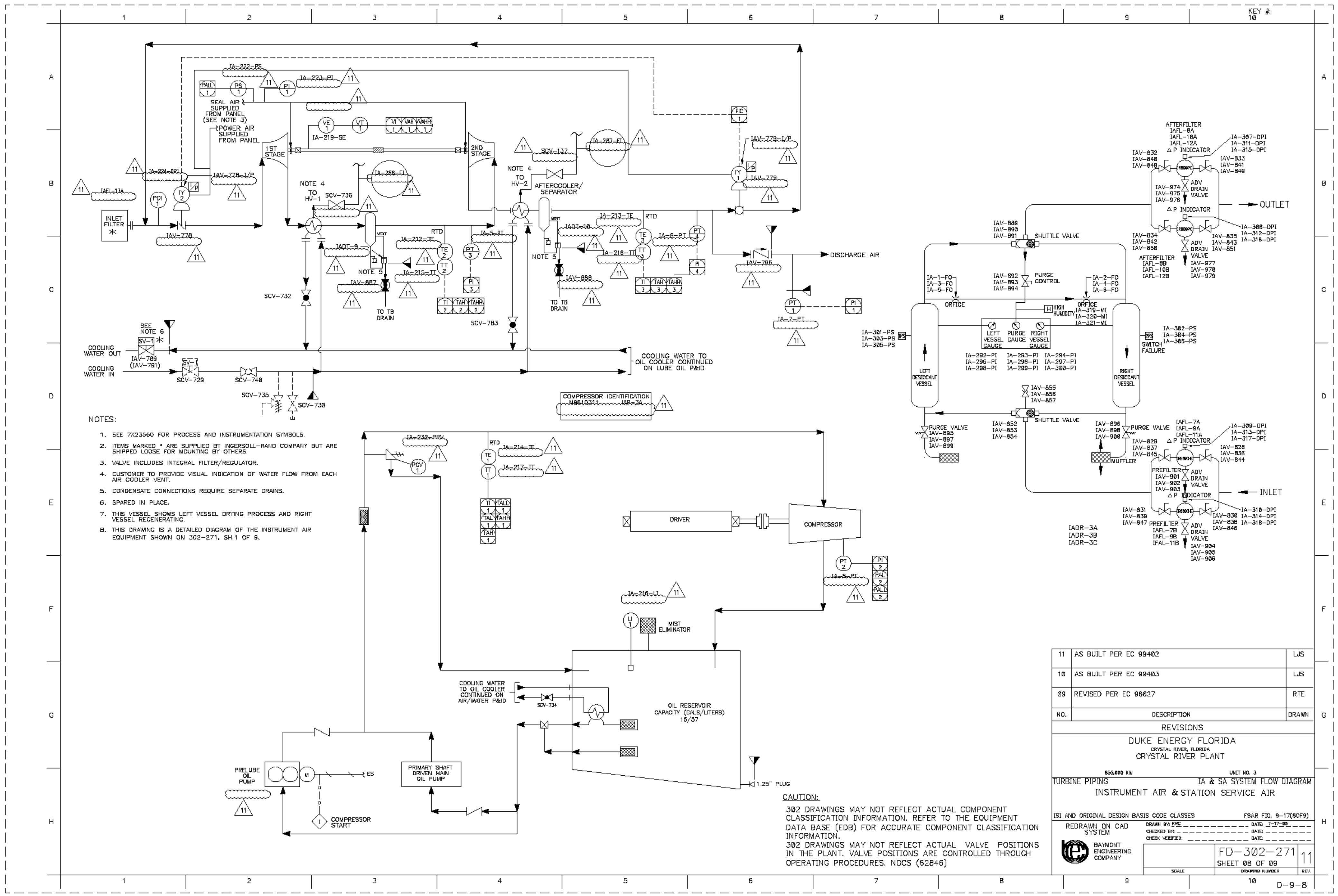


FIGURE 3-24I, INSTRUMENT AIR AND STATION SERVICE AIR, FD-302-271, SHEET 9

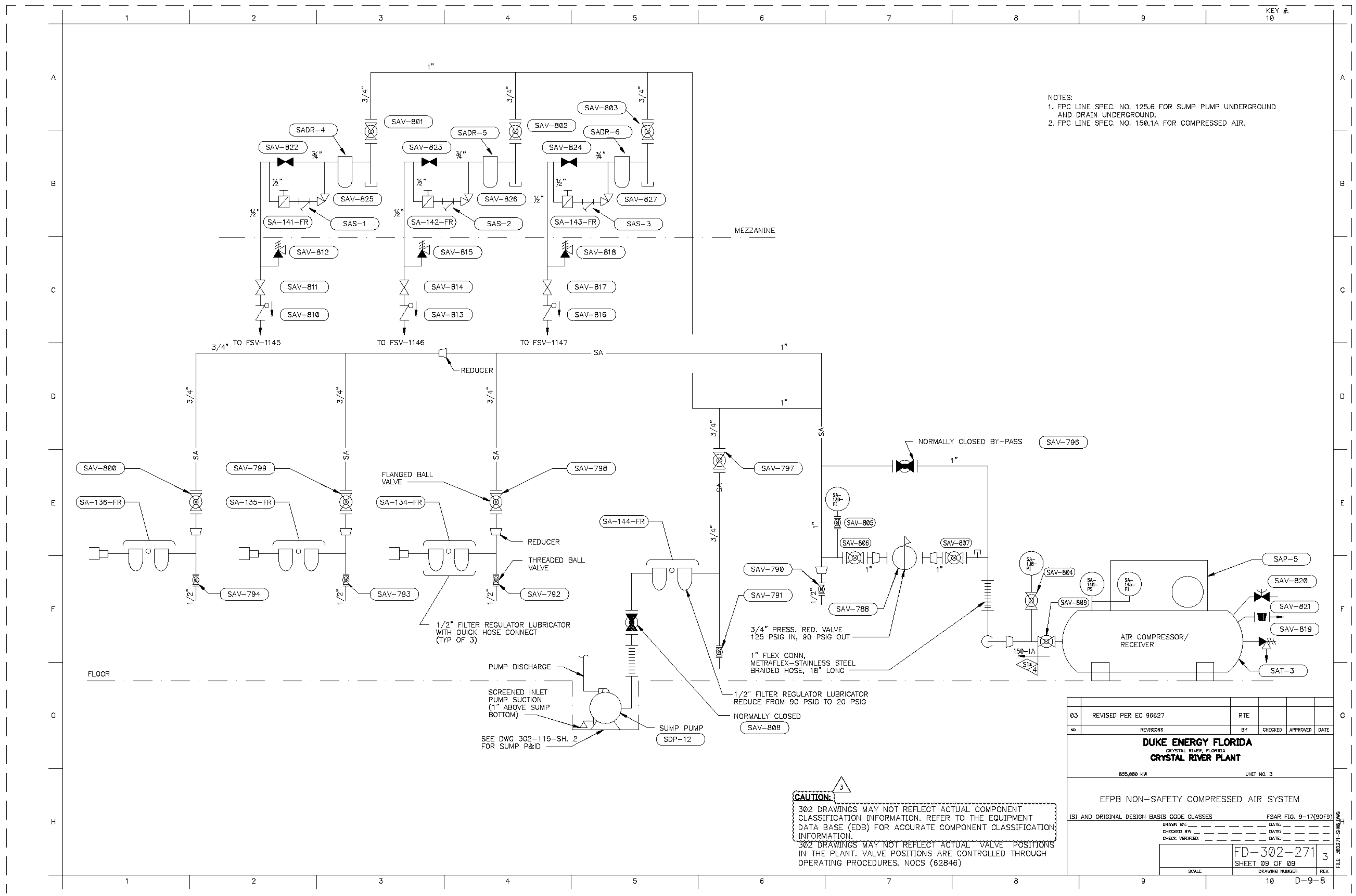


FIGURE 3-25A, FIRE SERVICE WATER, FD-302-231, SHEET 1

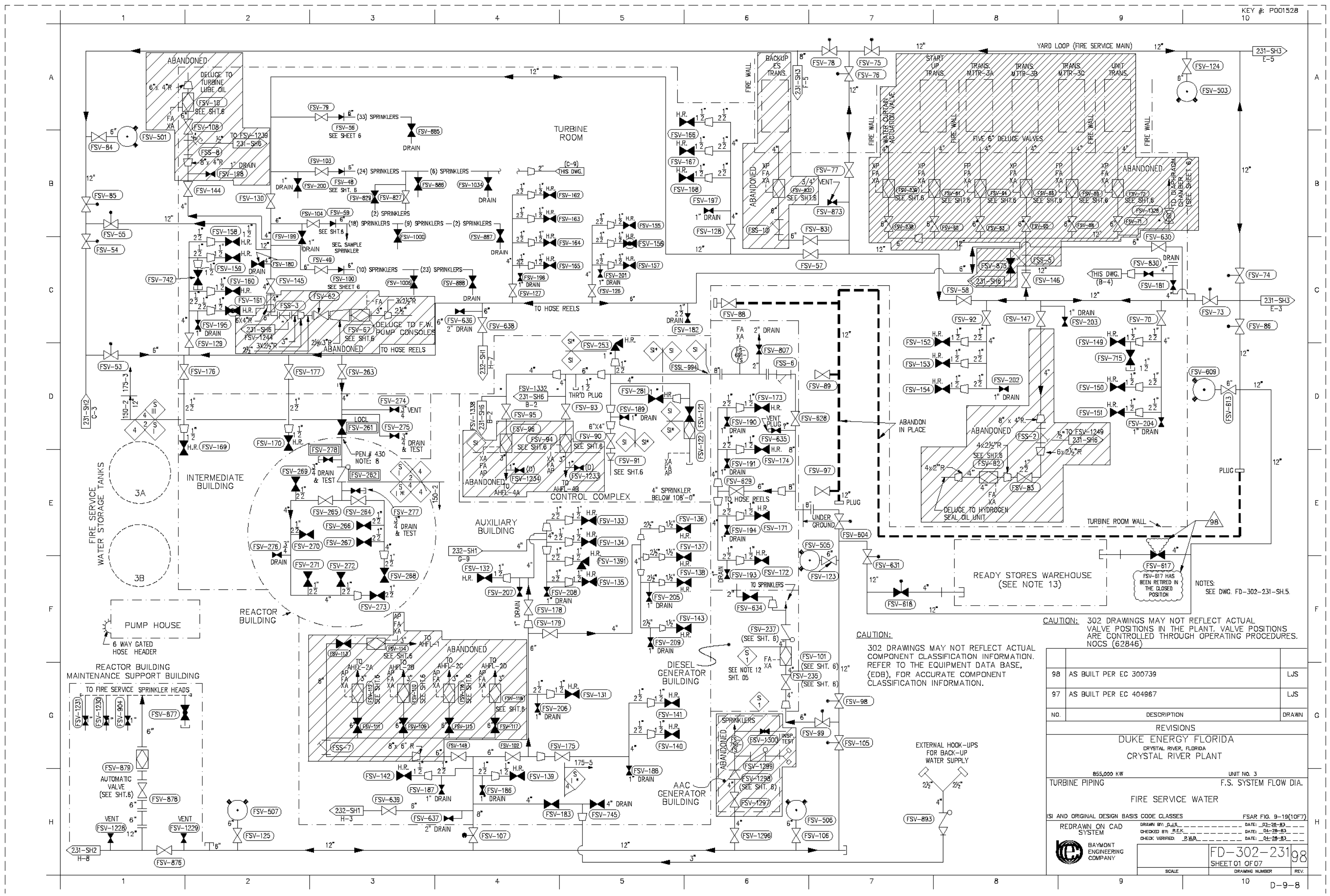


FIGURE 3-25B, FIRE SERVICE WATER, FD-302-231, SHEET 2

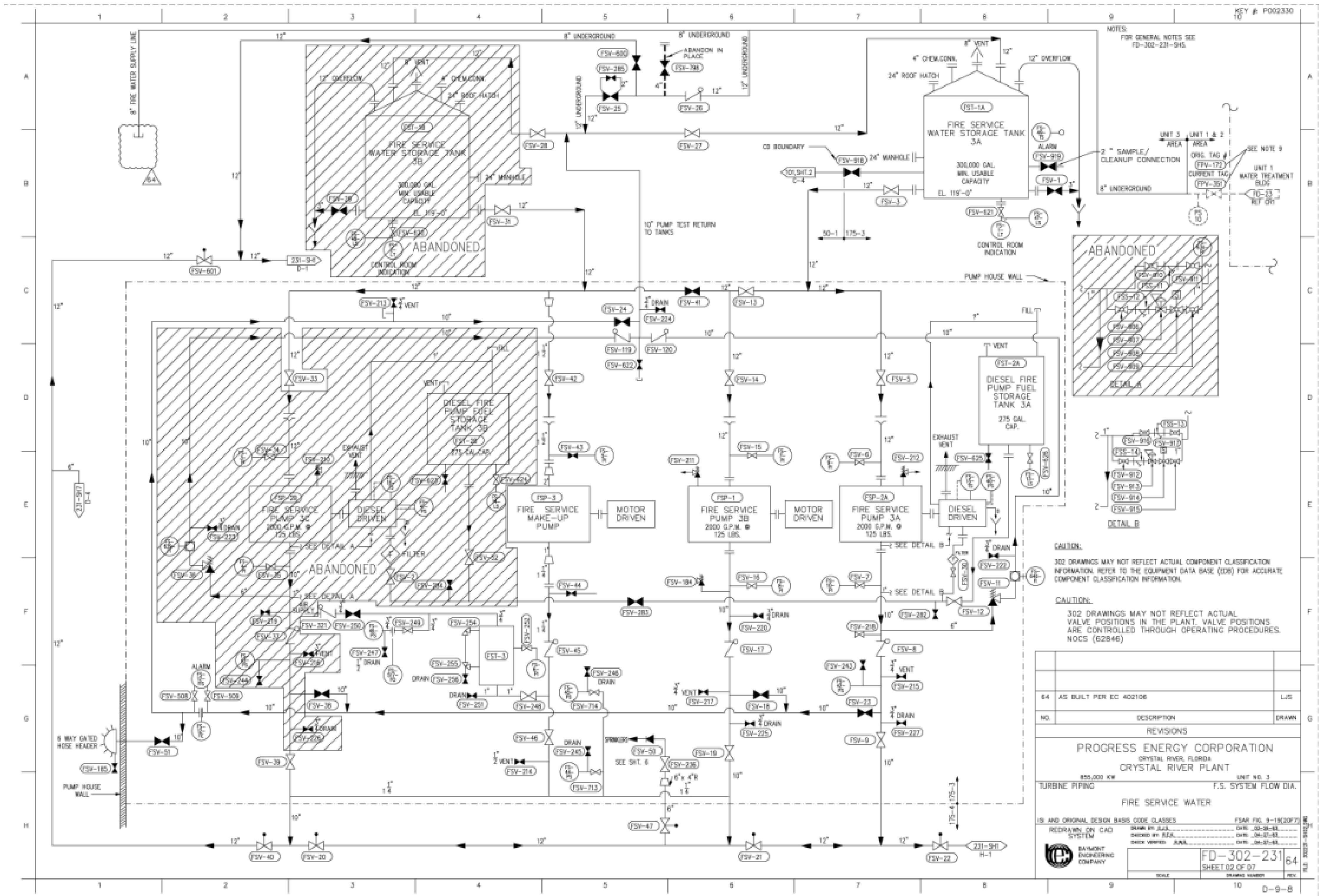


FIGURE 3-25C, FIRE SERVICE WATER, FD-302-231, SHEET 3

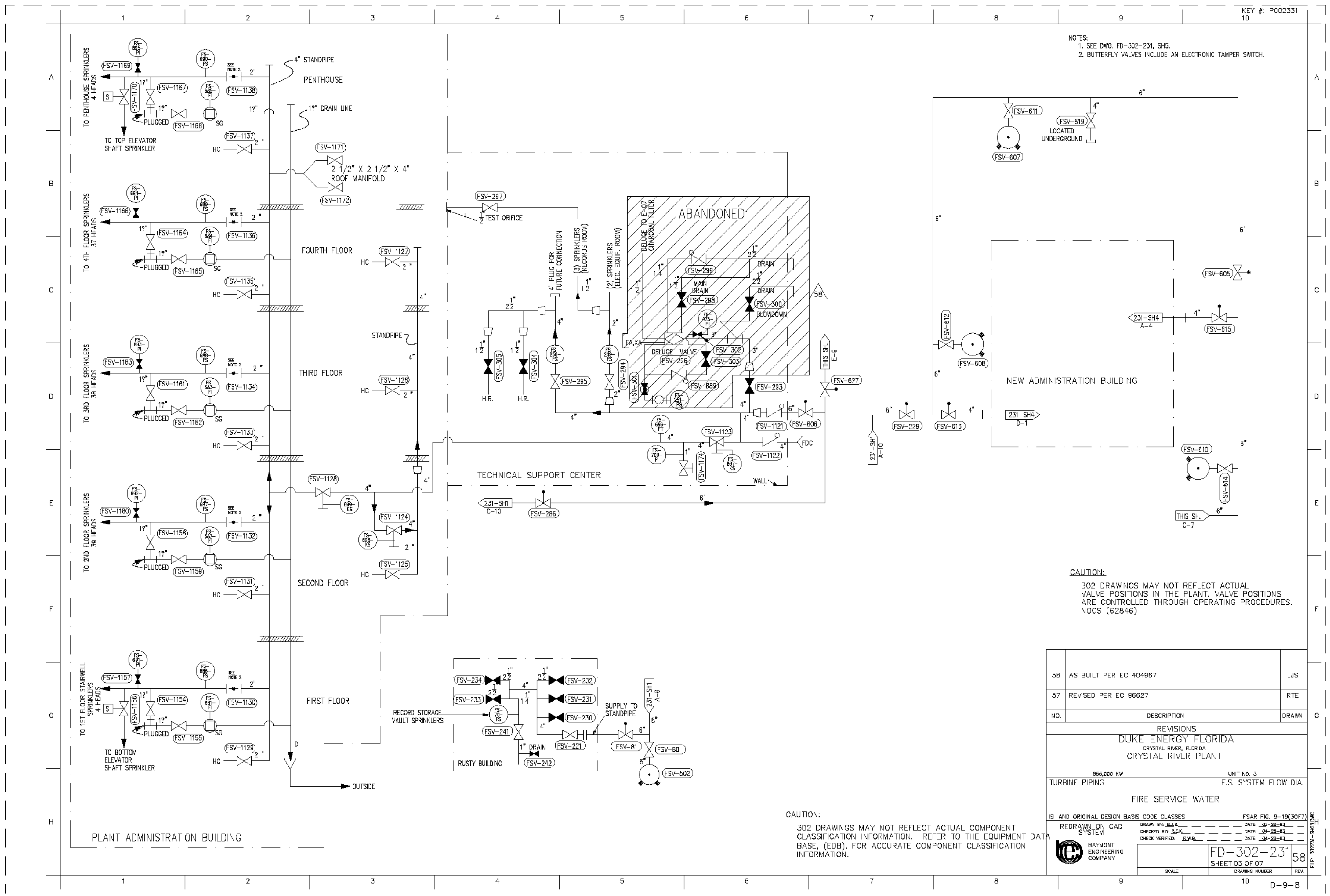


FIGURE 3-25D, FIRE SERVICE WATER, FD-302-231, SHEET 4

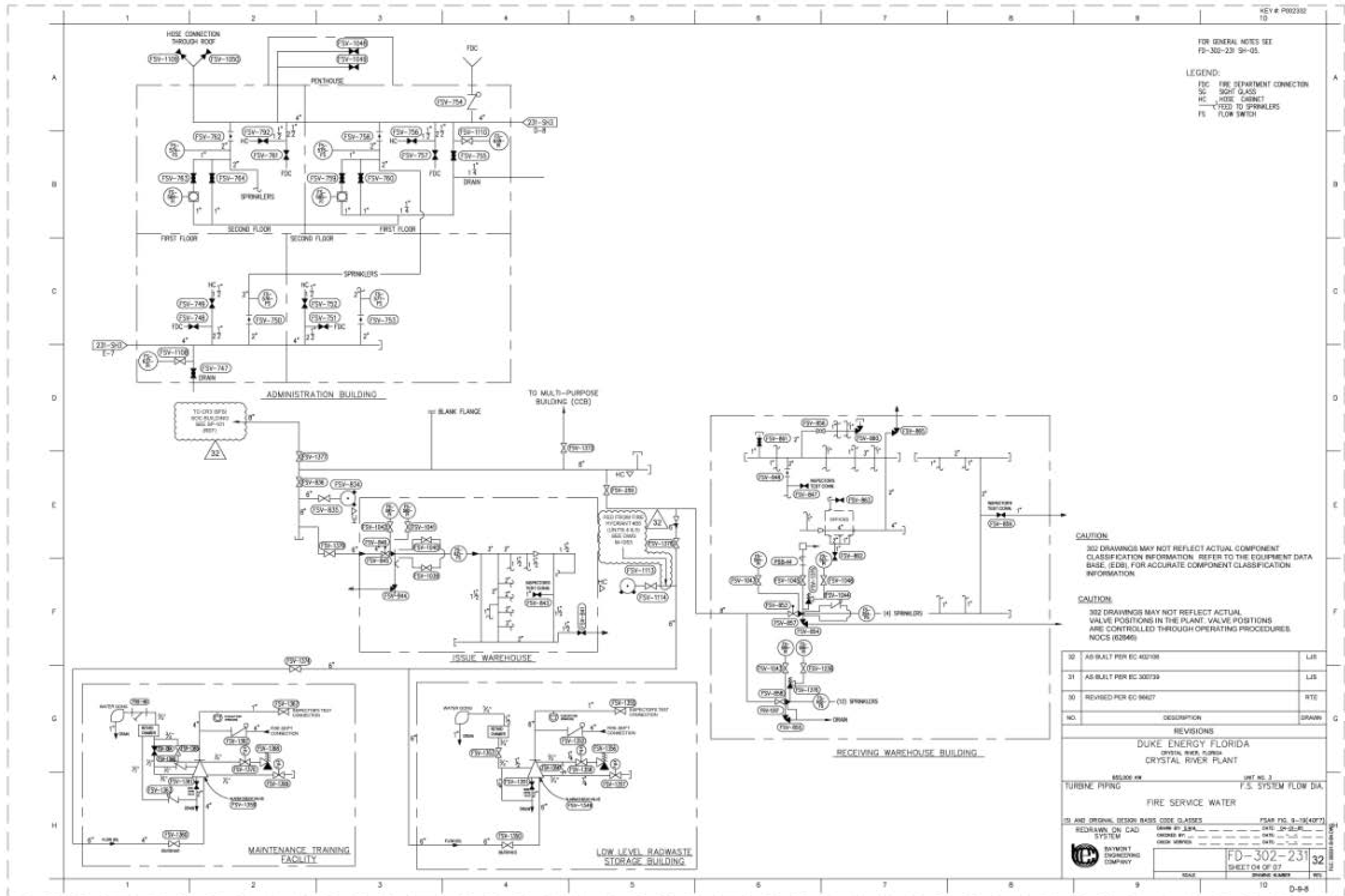


FIGURE 3-25E, FIRE SERVICE WATER, FD-302-231, SHEET 5

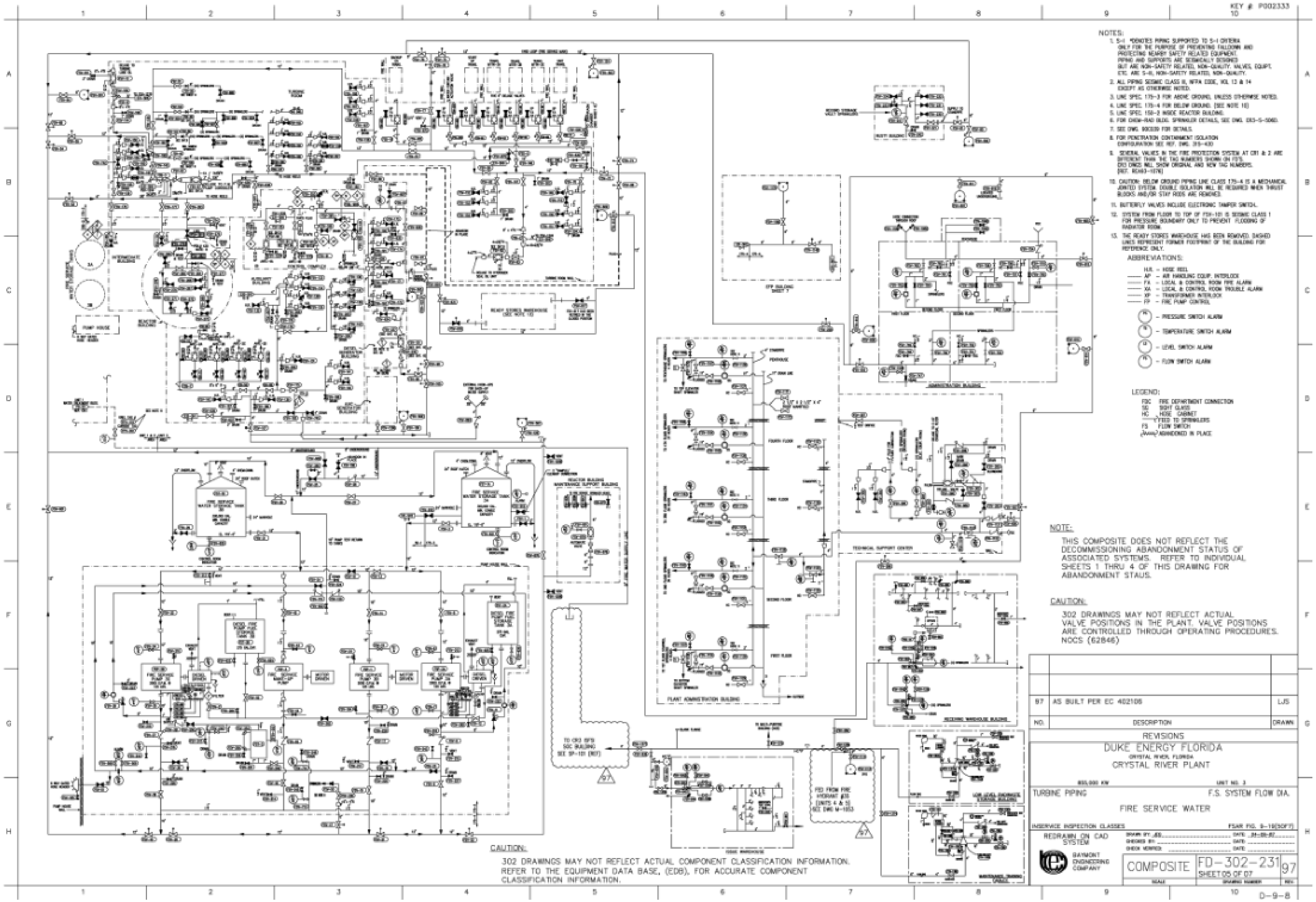


FIGURE 3-25F, FIRE SERVICE WATER, FD-302-231, SHEET 6

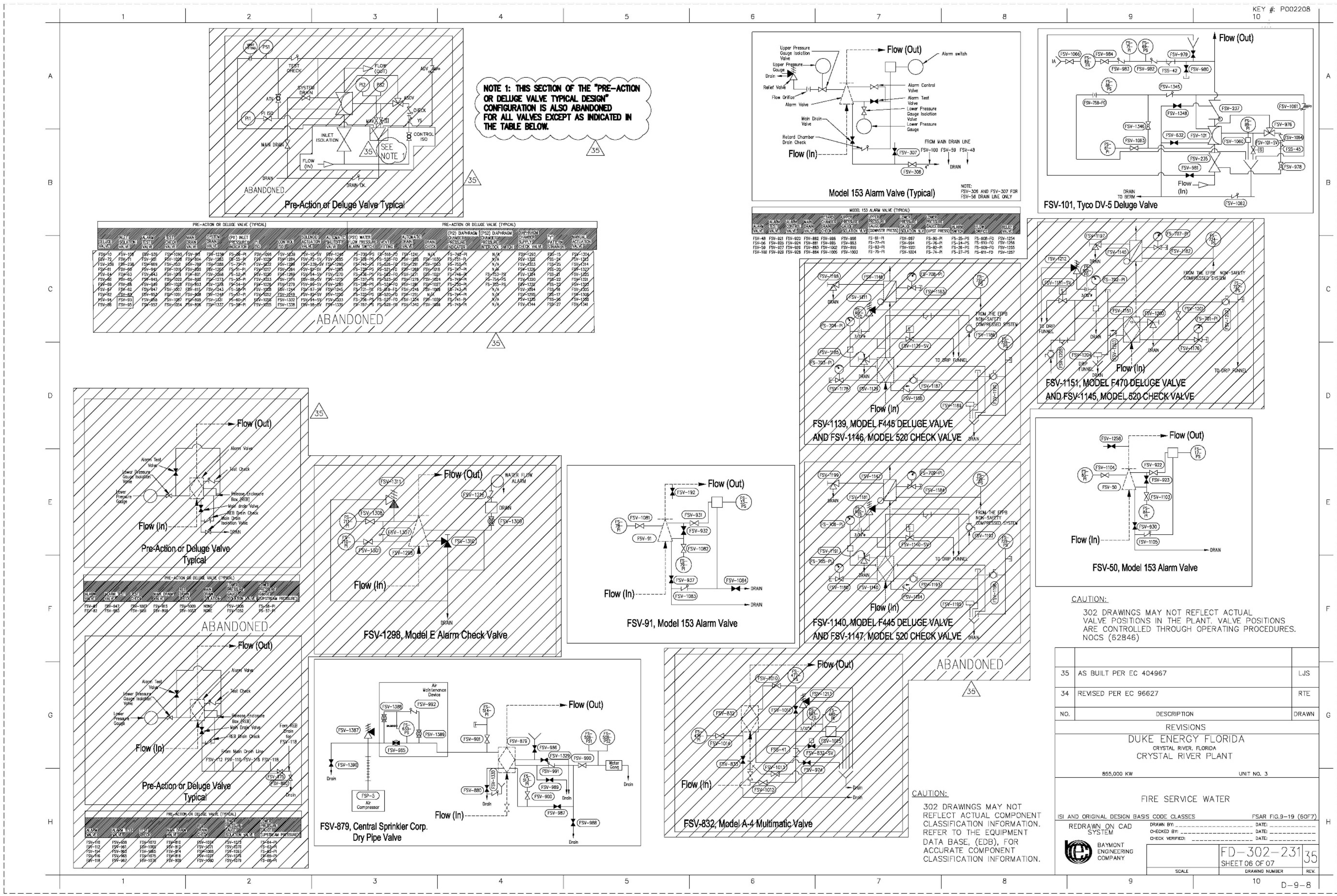


FIGURE 3-25G, FIRE SERVICE WATER, FD-302-231, SHEET 7

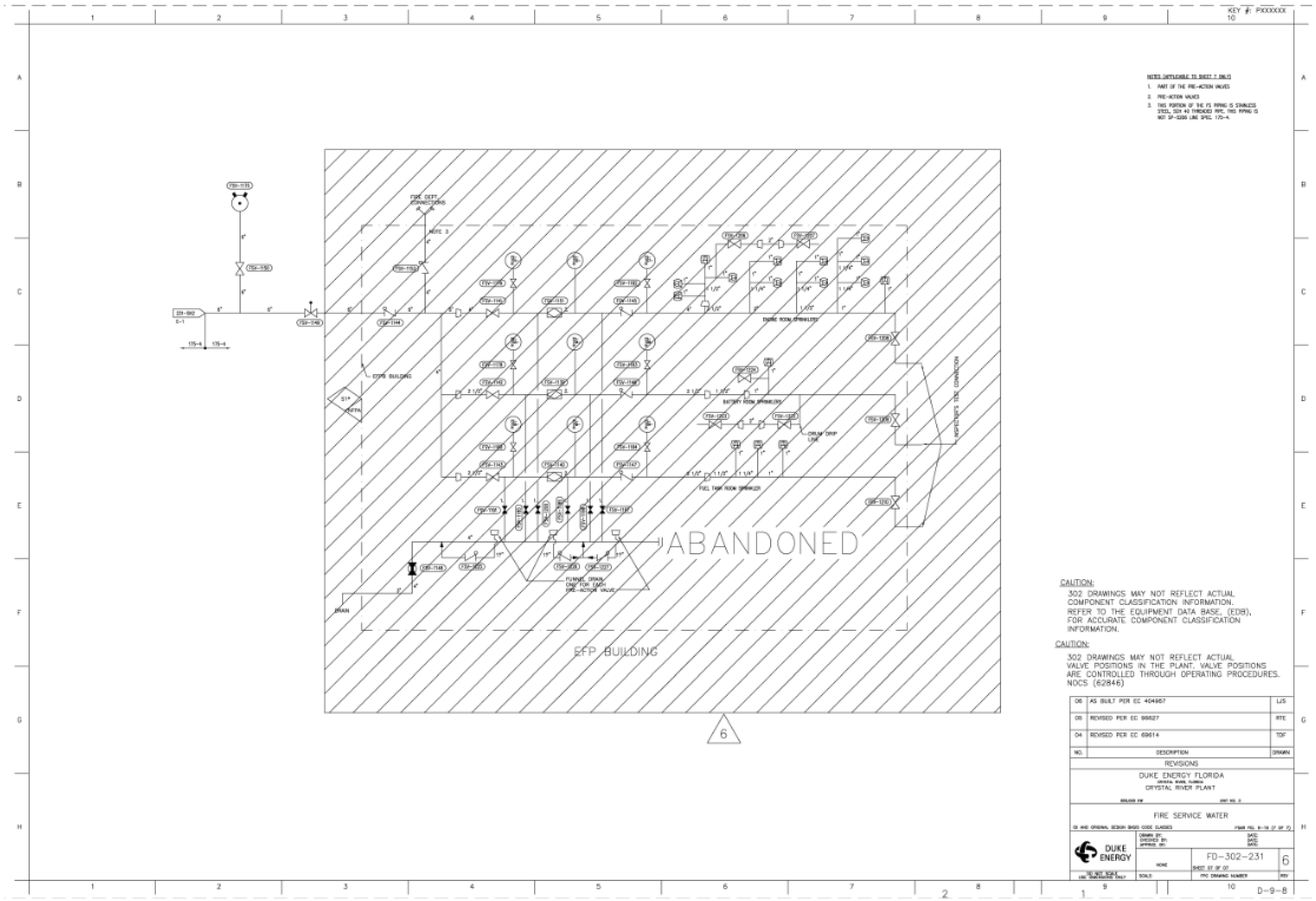


FIGURE 3-26, AUXILIARY BUILDING & CONTROL COMPLEX PLAN AT FLOOR ELEV 143′-0″ & 145′-0″ VENTILATION, BS-311-716

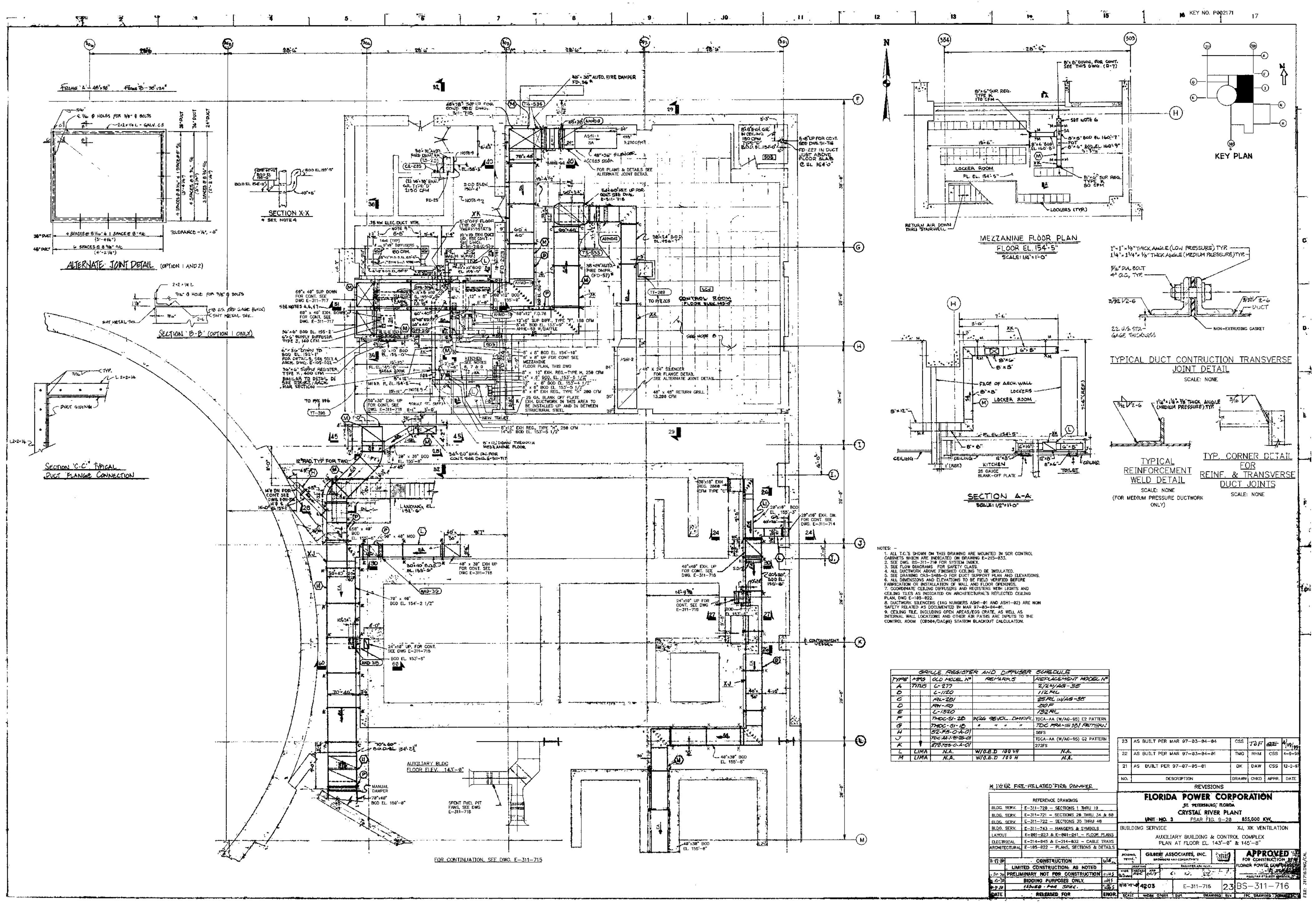


FIGURE 3-27, AUXILIARY BUILDING & CONTROL COMPLEX PLAN AT FLOOR ELEV 162′-0″ & 164′-0″ VENTILATION, BS-311-718

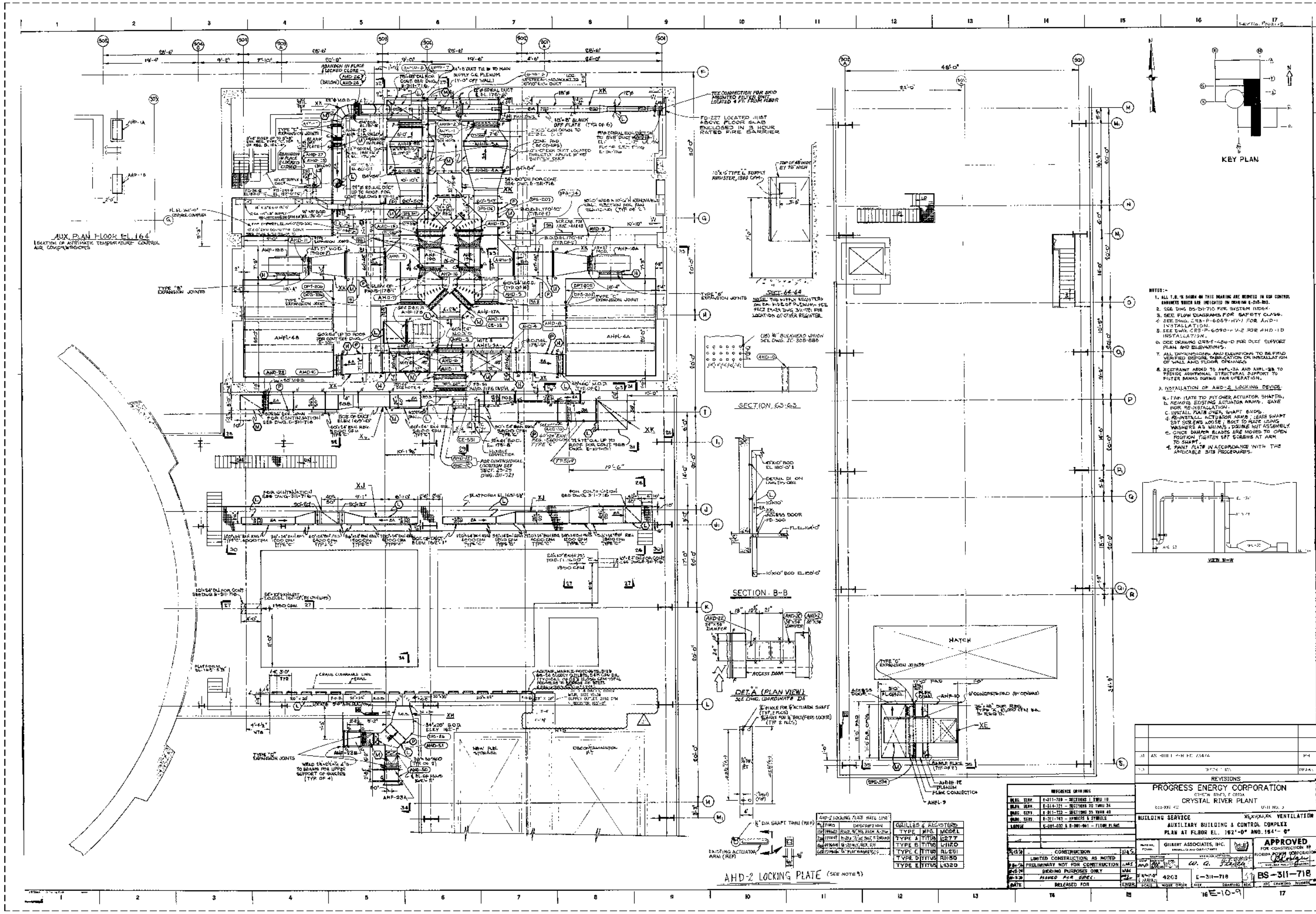


FIGURE 3-28, AUXILIARY BUILDING & CONTROL BUILDING SECTIONS VENTILATION, BS-311-721

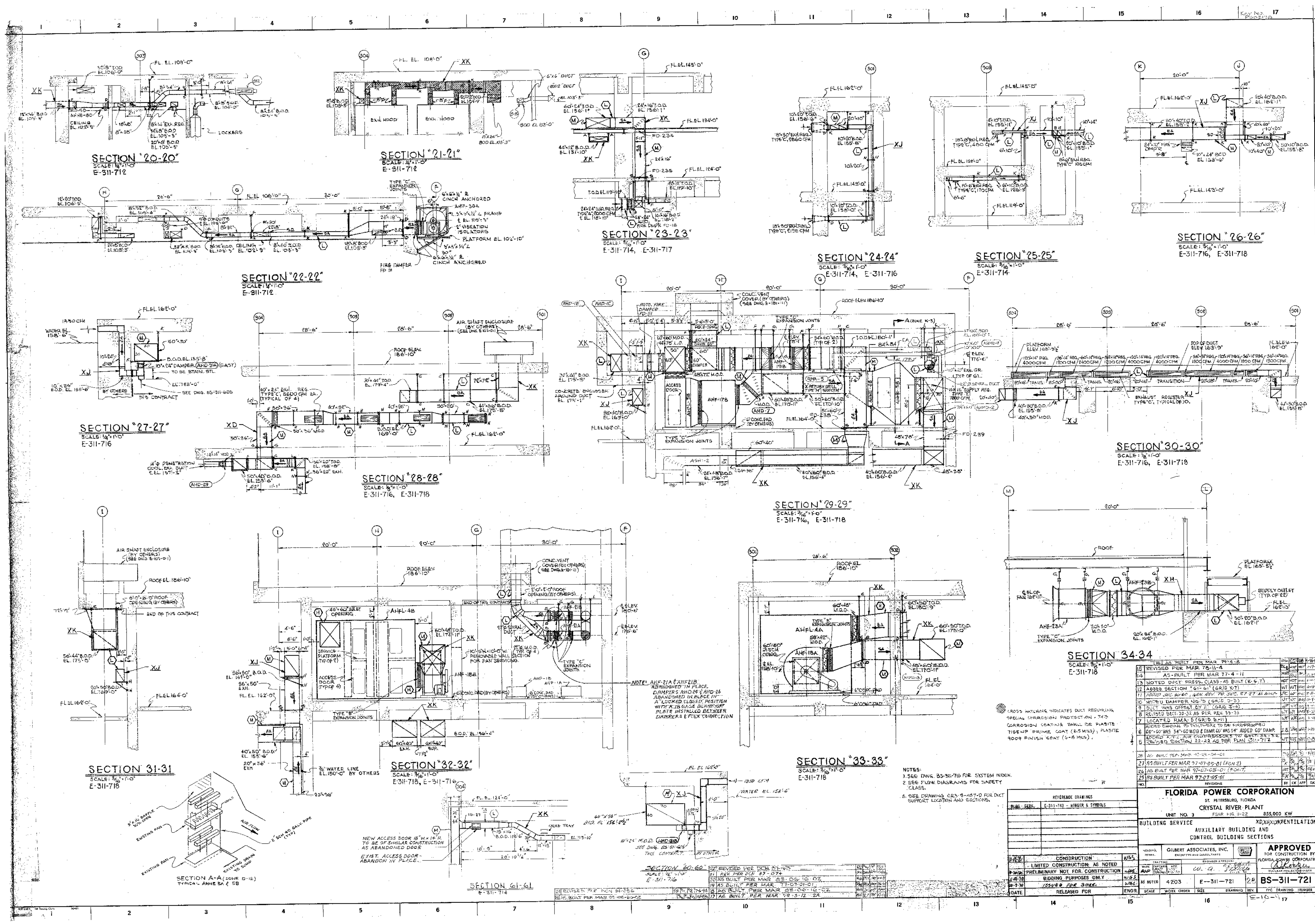


FIGURE 3-29, CONTROL COMPLEX - EFIC ROOMS EL. 124'-0" FLOW DIAGRAM, FD-302-765

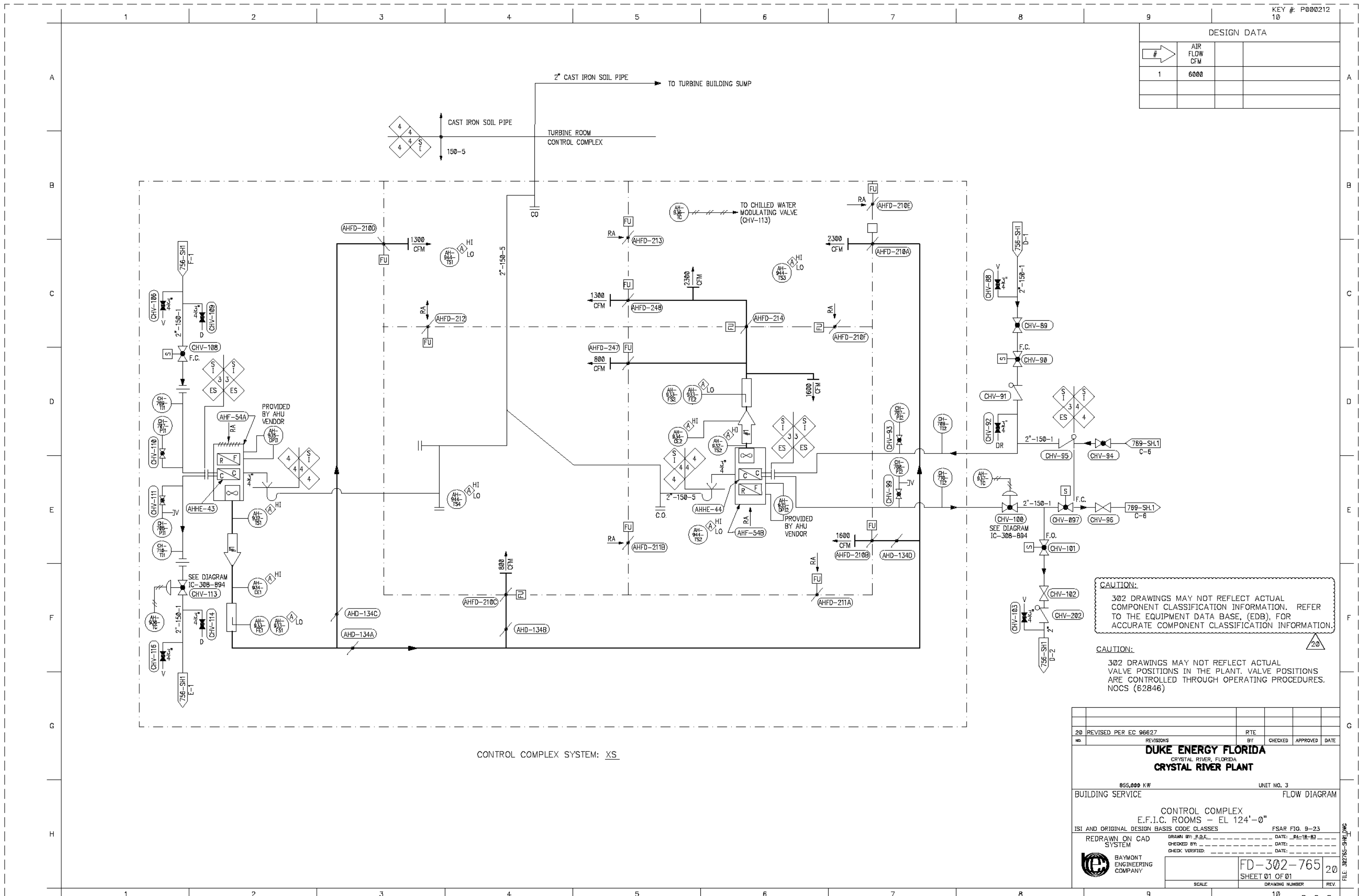


FIGURE 3-30, CONTROL COMPLEX - EFIC ROOMS EL. 124'-0" VENTILATION, BS-311-719

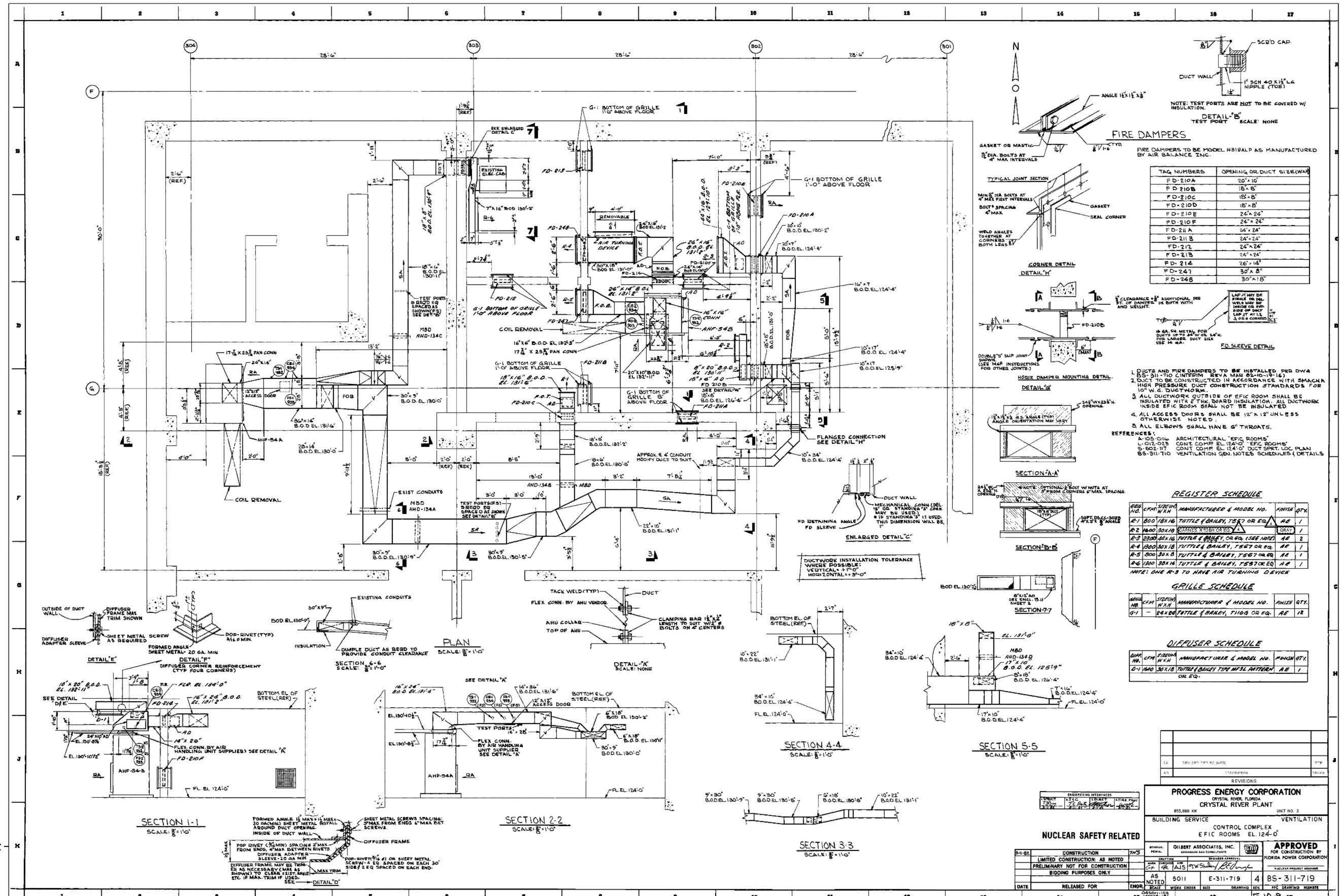


FIGURE 3-31, SPENT FUEL CHILLED WATER SYSTEM, FD-302-785

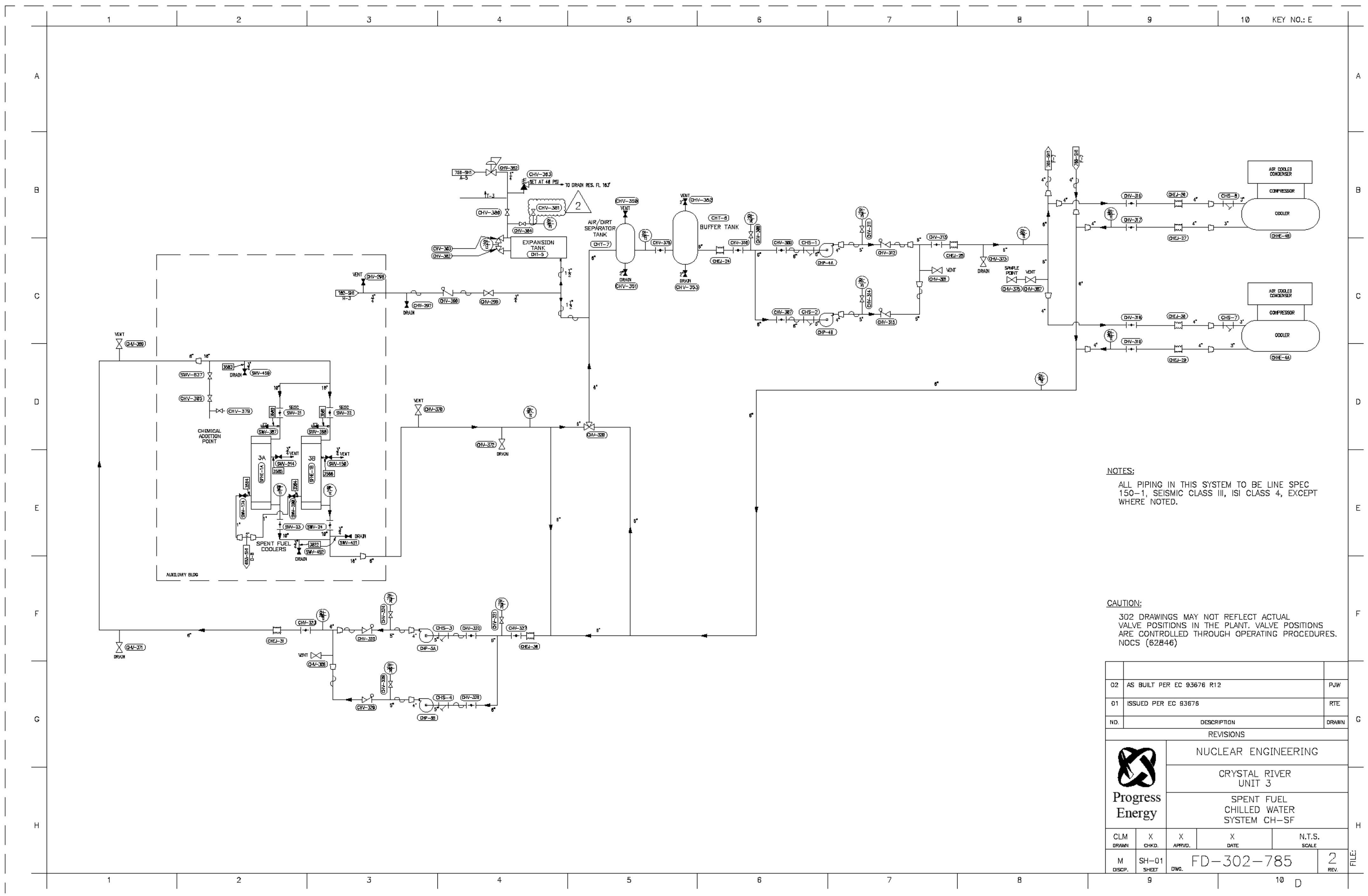


FIGURE 3-32, CONTROL COMPLEX CHILLED WATER, FD-302-786

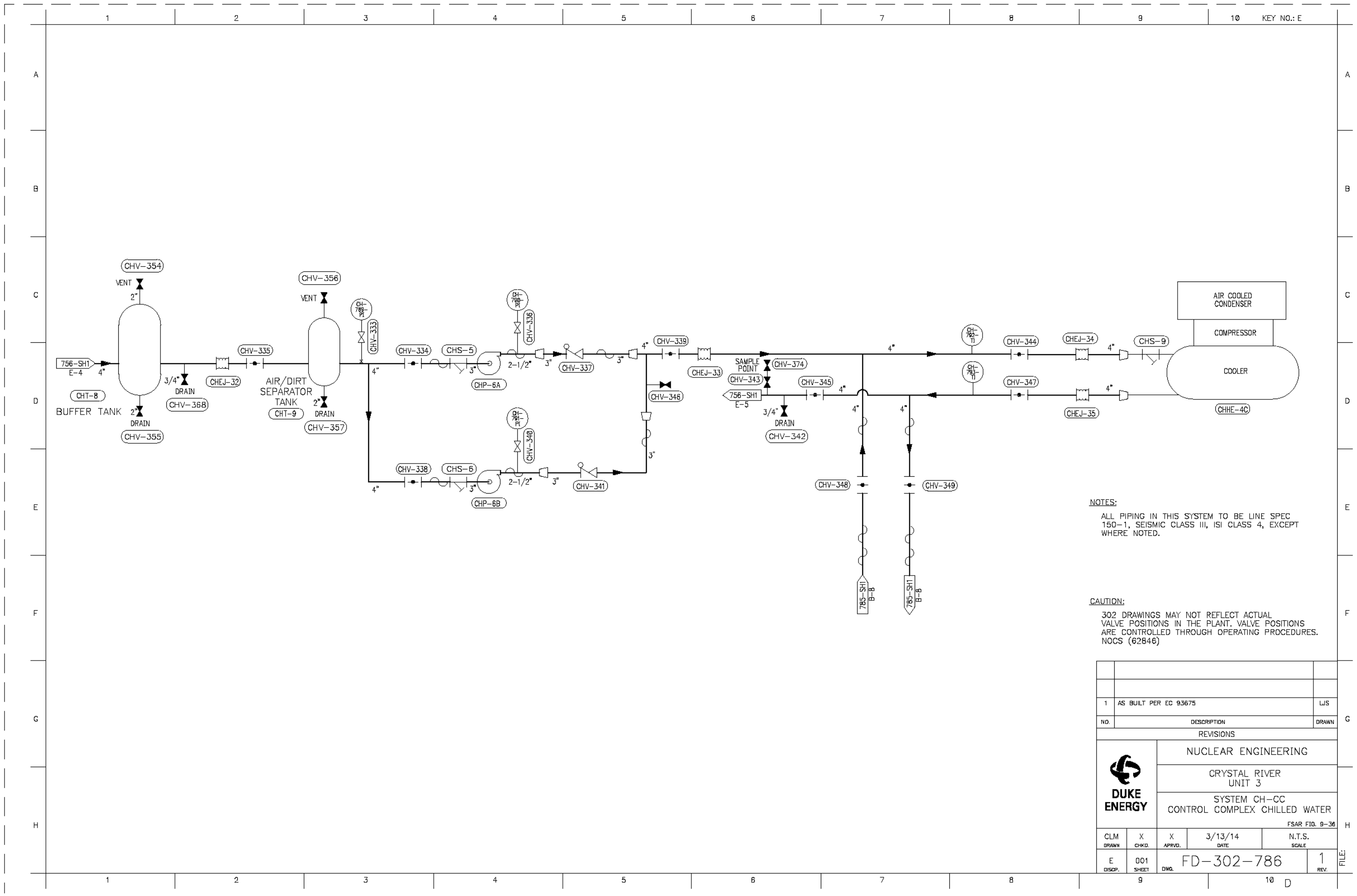
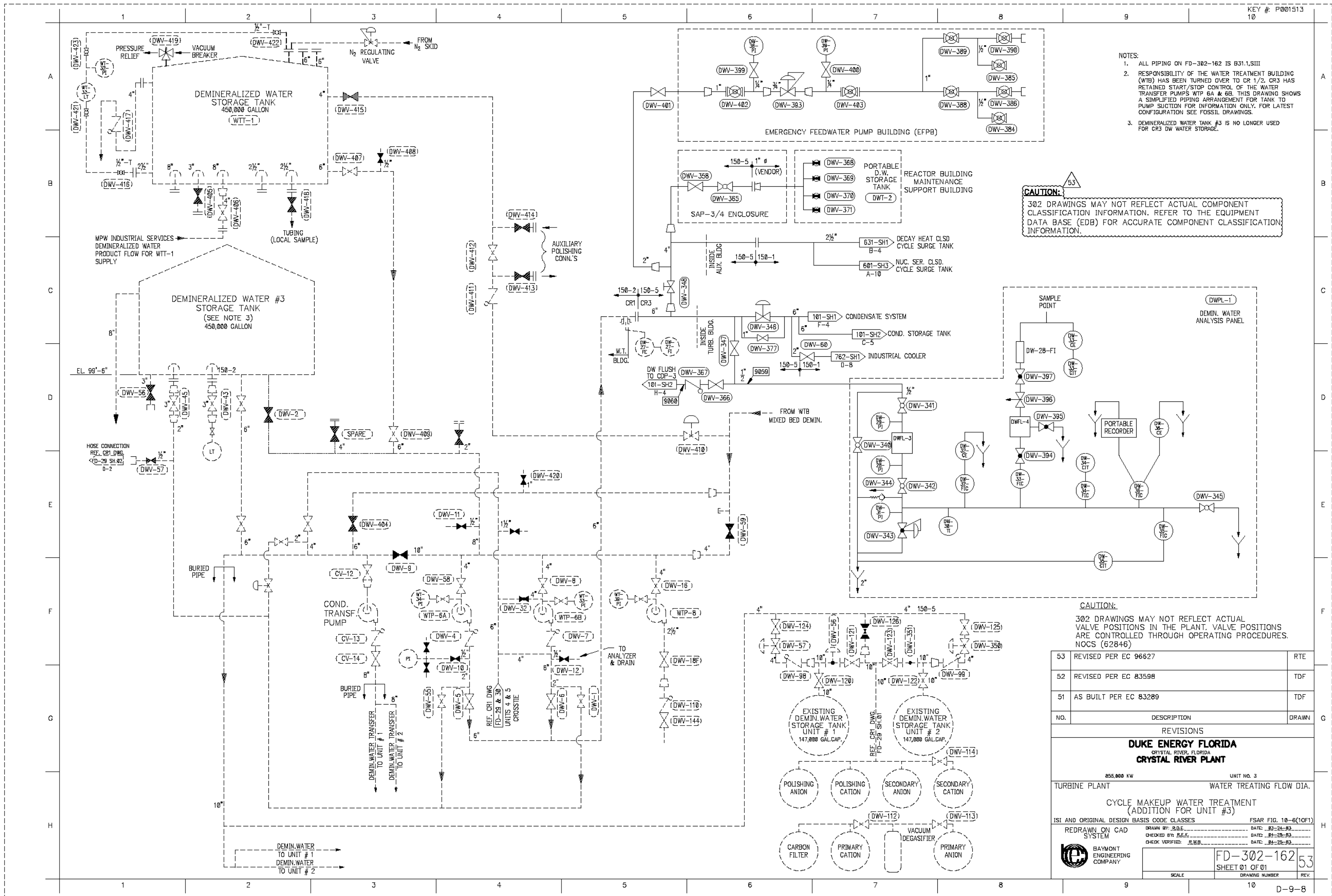


FIGURE 3-33, CYCLE MAKEUP WATER TREATMENT, FD-302-162



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4. RADIOACTIVE WASTE & RADIATION PROTECTION

4.1 DESIGN PERFORMANCE OBJECTIVES

The radioactive Waste Disposal Systems are designed to provide for controlled treatment of radioactive liquids and solids generated as a result of plant operation. These systems are designed and operated to assure that the quantities of radioactive materials released from the plant are as low as reasonably achievable.

Plant structures are designed and constructed, and equipment is located so as to maintain the radiation levels in the plant within applicable regulations for occupational exposure. The structures also provide shielding for the general public, such that applicable regulations for radiation exposure are not exceeded.

The Radiation Monitoring Systems (RMS) are designed to detect, measure, control or alarm, as required, radiation and releases of radioactive solids, liquids, and gases produced as a result of normal operation of the plant. These systems aid in the protection of the general public and of plant personnel from exposures to radiation or radioactive materials.

The radiation protection procedures and controls are designed to protect all personnel from unnecessary radiation exposure and to keep unavoidable radiation exposure to the absolute minimum consistent with work objectives, health physics philosophy, and federal and state regulations.

4.1.1 DESIGN BASIS

CR3 radioactive waste treatment systems were designed to assure that the limits of 10 CFR 20 were met. The adequacy of this design was demonstrated prior to plant operation by two evaluations, one of which was specific to the more limiting ALARA requirements of 10 CFR 50, Appendix I. Compliance with the methods and limits of the Offsite Dose Calculation Manual (ODCM), which implements the dose limits of 10CFR 50, Appendix I, assures that effluent concentration limits and public dose limits are met. ODCM limits have their bases in 10 CFR 20, 10 CFR 50 Appendix I, and 40 CFR 190.

When there are proposed changes to station design, the Safety Assessment required per the modification process requires an evaluation of the potential effects on normal radiological effluents. If the initial screening determines that there may be some effect, the more detailed radiological review considers the effects of the change on the actual measured effluents and the ability to meet the requirements of the ODCM.

4.2 RADIOACTIVE WASTE DISPOSAL SYSTEMS SUMMARY

The radioactive Waste Disposal System is provided to process radioactive liquid and solid wastes generated during plant operation. The functional operations controlling these waste products are:

a. Collection
b. Storage on-site
c. Processing and sampling for disposal
d. Packaging for shipment off-site

The Liquid Systems are designed to reduce the activity in liquid wastes, to restrict the concentration in routine releases from the plant to values significantly less than the regulatory limits. The liquid effluents are continuously monitored and the discharge is terminated if the effluent concentration exceeds the monitor setpoint. Sampling may also be performed in lieu of continuous monitoring. Solid wastes are packaged in DOT approved containers for offsite shipment or stored onsite until expiration of the CR3 license.

The NUS demineralizer system installed by MAR 87-10-21-01 decontaminates radioactive waste water by passing the contaminated water through pressure vessels containing filtration media and ion exchange resins. This vendor

supplied equipment supplements the plant liquid waste disposal system and substitutes for the Miscellaneous Waste Evaporator.

The radioactive waste holdup and processing equipment is located within reinforced concrete Seismic Class I structures, designed to withstand the Maximum Hypothetical Earthquake (MHE), tornadoes, and tornado driven missiles. The individual items of the waste holdup and processing equipment are designed, fabricated, and erected in conformance with all codes and standards applicable to equipment containing radioactive materials. The major components potentially presenting a radiation hazard to plant personnel are located within rooms separated from normally occupied areas and adjacent equipment by concrete walls. Each room is provided with its own floor and equipment drain to route any liquid leakage to stainless steel lined sumps. Area radiation monitors are strategically located to warn of excessive radiation levels within the area surveyed.

The building ventilation system continuously supplies a fresh air flow from normally occupied areas of the building, through the rooms containing the radioactive waste equipment to the ventilation exhaust discharge system. This system contains fans, roughing and HEPA particulate filters. The air flow is routed to the plant vent past a gas radiation monitor designed to automatically shutdown the ventilation system supply fans in the event its setpoint is exceeded or if sampling capability is lost. The capacity of the exhaust system is higher than the fresh air supply system to ensure the air pressure inside the building is slightly below the outside air pressure, to prevent an inadvertent or unmonitored release from the building atmosphere to the environment.

4.2.1 RADIOACTIVE LIQUID WASTE DISPOSAL SYSTEM

4.2.1.1 System Design and Operation

The radioactive Liquid Waste Disposal System is designed to collect, store, and process radioactive liquid wastes for disposal or reuse from various sources. The components of the Liquid Waste System are tanks, pumps, demineralizers, coolers, floor and equipment drains, sumps, valves, and piping. The liquid waste component data are given in Table 4-1.

The Liquid Waste Disposal System process flow diagram is shown in Figure 4-1.

The miscellaneous processing chain is used to process the miscellaneous wastes from (a) radioactive laboratory drains; (b) building and equipment drains and sumps; and (c) demineralizer backwash. The miscellaneous processing chain consists of the miscellaneous waste storage tank, and evaporator condensate storage tanks. The miscellaneous radwaste storage tank contents can be further processed through another flow path consisting of a disposable activated carbon filter and a high capacity anion and cation resin bed prior to entering the evaporator condensate storage tank. The contents of the evaporator condensate tanks may be transferred to the Nuclear Service and Decay Heat Sea Water Systems (RW) for release to the discharge canal.

The floor and equipment drainage system provides for the safe collection, measurement, sampling, and segregation of equipment and floor drainage solutions. The major collection vessels are:

a. Reactor building sump
b. Auxiliary building sump
c. Decay heat pit sump, A
d. Decay heat pit sump, B
e. Laundry/hot shower sump
f. Miscellaneous waste storage tank

The specific sources, or potential sources, of liquid routed to each of these sumps or tanks are shown on Figure 4-1 and Figure 4-2.

The tanks and sumps, previously listed as major collection vessels, each contain remote liquid level indicators and level alarms.

Some of the major collection vessels can be sampled to determine the gross radioactivity, the specific radioactivity of components, or the chemical compositions of the solutions. Periodic radiation surveys of these sumps and tanks are made with portable radiation instruments.

Tanks with ample storage capacity are provided to permit segregation of waste and to ensure processing, packaging, and disposal of waste can be accomplished in a convenient and timely manner. Spent resins produced as a result of radioactive system operation are stored in the spent resin storage tank for holdup and decay prior to off-site shipment. Spent resins are processed through the Solid Waste Packaging System.

Most radioactive liquid wastes released from the primary side of the plant are from one of the two 10,000 gallon evaporator condensate storage tanks. Only one tank is released at a time with the other on standby, and isolated from the release path. Small amounts of liquid waste are also released from the laundry tanks. During plant dormancy the ODCM may provide for the direct release to RW of liquid wastes contained in other tanks.

The release of liquid wastes to the effluent of the RW System is automatically terminated when the setpoint of the activity monitor (RM-L2/RM-L7) is exceeded. The release will be either automatically or manually terminated upon loss of dilution flow from the RW system. When RW System recirculation is being employed, liquid waste releases are made using an existing cross-tie to the RW System at a nominal flow of 800 gpm in accordance with plant operating procedures. Additional dilution of the activity released in the RW System is accomplished in the discharge canal where the liquid wastes are mixed with the circulating water discharge from Units 1 and 2 when operating.

4.2.1.2 Radioactive Releases

All releases of liquid effluent to the environment are under strict administrative control. Liquid effluent releases are in the batch mode, and a numbered discharge permit is issued for each batch release. Details of the sampling and analysis criteria are given in the Off-site Dose Calculation Manual (ODCM).

4.2.1.3 System Design Evaluation

The Liquid Waste Disposal System is designed to process radioactive fluids so that the quantity of radioactivity released to the environment are as low as reasonably achievable and regulatory limits are met.

4.2.1.4 Miscellaneous Activity Releases from the Auxiliary Building

This activity is discharged via the Auxiliary Building Ventilation System and is filtered (roughing, HEPA) prior to release. The monitors in the plant vent measure and record the activity released and alarm if the setpoint is exceeded.

4.2.1.5 Radioactive Releases from the Turbine Building

For the Turbine Building sump and SDT-1 releases, the activity associated with these releases is insignificant. A radiation monitor, RM-L7, is installed in the discharge line to the RW system from SDT-1 to provide continuous monitoring capability during a release. An interlock is provided in the line, designed to terminate the release upon indication of high activity in the effluent.

4.2.2 RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

A program is required to monitor the radiation and radionuclides in the environs of the plant. The program provides (1) representative measurements of radioactivity in the highest potential exposure pathways, and (2) verification of the accuracy of the effluent monitoring program and modeling of environmental pathways. The program is (1) contained in the Offsite Dose Calculation Manual (ODCM), (2) conforms to the guidance of 10 CFR 50, Appendix I, and (3) includes the following:

a. Monitoring, sampling, analysis, and reporting of radiation and radionuclides in the environment in accordance with the methodology and parameters in the ODCM.

b. A Land Use Census to ensure that changes in the use of areas at and beyond the site boundary are identified and that modifications to the monitoring are made if required by the results of this census, and

c. Participation in an Interlaboratory Comparison Program to ensure that independent checks on the precision and accuracy of the measurements of radioactive materials in environmental sample matrices are performed as part of the quality assurance program for environmental monitoring. [TSA149]

4.2.2.1 Summary of Estimated Doses

Estimates of the maximum whole body dose to an individual and the maximum organ dose to an individual that would be received by the general public at the site boundary as a result of the release of liquid and gaseous effluents and can be found in the Annual Radioactive Effluent Release Reports submitted to the NRC.

4.2.3 RADIOACTIVE SOLID WASTE PACKAGING SYSTEM

The radioactive solid waste management process is designed to safely package, store and transport radioactive waste, while minimizing radiation exposure to personnel. Wastes are packaged for storage, shipment to offsite waste processors, or shipment to burial facilities. Waste can also be returned to CR3 from offsite waste processors for long-term storage. Solid waste packaging and transportation is performed in accordance with both Department of Transportation (DOT) and NRC regulations.

4.2.3.1 Process Evaluation

The types of radioactive waste generated at CR3 include:

1. Dry Active Waste (DAW);
2. Spent resins;
3. Tank and sump sludge; and
4. Spent filters

DAW is contaminated waste that is collected onsite and packaged in appropriate containers to meet processor and/or burial site acceptance criteria. DAW consists of contaminated paper, plastic, cloth, rubber, glass and metals. DAW can be placed into a strong, tight container for shipment to an offsite processor, or compacted into 55 gallon drums by a radioactive waste compactor.

Liquid waste processing resins are sluiced directly from the demineralizer into HICs. Resins are de-watered prior to shipment for offsite processing or direct disposal.

TSA149 *This paragraph was added when Technical Specification Amendment 149 was issued. Obtain Licensing & Regulatory Affairs concurrence before changing.*

Tank and sump sludge is generated during the cleaning of various tanks and sumps located in the Auxiliary and Reactor Buildings. The sludge is transferred into suitable containers and de-watered. Sludge can be processed into a form suitable for disposal by offsite waste processors utilizing their Process Control Program (PCP) and applicable procedures. The waste processor's procedures and PCP are approved by CR3 prior to solidification of waste. Solidification can be performed either offsite at the waste processor facilities or onsite at CR3.

Spent filters are removed from service and stored to allow radioactive decay. Filters are loaded for shipment into appropriate containers (e.g., HICs or 55 gallon drums).

Throughout the packaging and shipping operations, radiation exposure to personnel is minimized by the use of various ALARA techniques, as appropriate, including:

a. Administrative controls
b. A shielded cask in the truck loading area.
c. A shielded drum storage area.
d. Use of shielded carts for transporting plant filters

Waste containers are surveyed for radiological conditions and stored in designated storage areas. Storage locations include outside storage areas.

4.2.3.2 Radioactive Waste Handling Event

A radioactive waste handling event has been postulated to be the limiting event for decommissioning activities at CR3. The event is postulated to be the airborne dispersal of radioactive waste resin upon dropping of a high integrity container (HIC) outside of the power block. Although an airborne release is not expected to occur with a drop, or while in storage awaiting shipment due to the low flammability and reactivity of the spent resin, a release is nevertheless postulated.

The event is based on a release of radioactive material with activity and isotopic mix taken from the resin shipments which occurred during a 5-1/2 year period. Resin shipments made from January 2008 through June 2013 were reviewed and the isotopic distribution (except for cobalt-60) was obtained from the shipment with the highest overall activity. Cobalt-60 activity was taken from a different shipment to assure that the highest activity was used and dose was maximized. This created a composite maximum shipment having a total activity of approximately 116 curies, which is approximately twice the activity of the average shipment made during this period. A release fraction of 10% is assumed. No deposition is assumed along the plume path out to the exclusion area boundary (EAB) as a conservatism which maximizes dose at the EAB.

Table 4-3 contains the design inputs for calculation of the event consequences. The results of the event analysis determined that a receptor at the closest point on the EAB would receive a dose of 40 millirem TEDE.

4.2.4 HAZARDOUS/MIXED WASTE ACCUMULATION FACILITY

The Hazardous/Mixed Waste Accumulation Facility is designed to facilitate the interim storage of those plant wastes which are both radioactive and hazardous, as regulated by the Environmental Protection Agency. There are limited disposal options for these mixed wastes. Regulatory changes with regard to hazardous waste have necessitated the implementation of this Facility which was not envisioned in the original plant design.

4.2.4.1 System Design Evaluation

The Hazardous/Mixed Waste Accumulation Facility consists of specially designed modular hazardous material storage buildings. The Facility is located within the Protected Area on the southeast corner of the berm and access is controlled by the Nuclear Facility Services and the Health Physics Department for the purposes of radiation protection. The buildings provide the waste containers protection from personnel intrusion, fire, high wind, rain, heat, chemical and mechanical damage. Each building offers spill protection in the form of a chemical resistant

20% capacity sump. The facility provides sufficient aisle space for weekly inspection and for the isolation of incompatible wastes.

4.3 RADIATION SHIELDING

Radiation shielding is provided principally by concrete walls, floors, ceilings, and roofs that are also structural members of buildings, which are designed to meet the requirements of the controlling criteria (i.e., structural strength or radiation shielding). Therefore, where structural strength is the controlling criteria, the radiation shielding provided is in excess of requirements.

4.3.1 DESIGN CRITERIA

The design criteria for radiation shielding are based on:

- Ensuring that during normal plant operation the radiation dose to operating personnel and to the general public is within the limits set forth in 10 CFR 20.

To comply with radiation exposure limits specified in 10 CFR 20, the shielding is designed to attenuate radiation levels throughout the plant, from direct and scattered radiation to the dose rate levels specified for each of the areas indicated below:

Zone		Dose Rate (mr/hr)	Typical Locations
0	Unlimited Occupancy	≤ 0.5	Office, Control Room and Turbine Building
I	Normal Continuous Occupancy	≤ 2.5	Auxiliary Building / Accessible Areas
II	Controlled, Limited Access	≤ 15.0	Valve Galleries
III	Controlled, Limited Access	≤ 25.0	Reactor Building / Accessible Areas
IV	Restricted Access	≥ 100.0	Restricted Access / Inside Secondary Shield

All zones that are not designed as restricted access (i.e., ≥ 100 mrem/Hr) were intended to be ≤ 25 mrem/hr. Layout drawings of the plant showing these radiation zones are provided in Figures 4-6 through 4-9. These zone drawings represent the historical design criteria for the original plant shielding. They should not be used to determine current plant radiological conditions.

4.3.2 GENERAL DESCRIPTION

Radiation shielding is divided into the following categories: primary, secondary, reactor building, spent fuel (including fuel transfer), and auxiliary shielding. Each of these categories are described below.

4.3.2.1 Primary Shield

The primary shield serves to protect plant personnel, surrounding structures, and primary system components from the neutron and gamma radiation escaping the reactor vessel. It is an annular cylinder of reinforced concrete which surrounds the reactor vessel and extends upward from the reactor building floor to the bottom of the fuel transfer canal. The shield thickness is 5 feet up to the bottom of the reactor vessel flange, where the thickness is reduced to 4.5 feet. It is designed to meet the following specific radiation attenuation objectives:

a. To sufficiently reduce the radiation escaping the reactor vessel after shutdown to permit limited access to the Reactor Coolant System (RCS) equipment.

b. To sufficiently attenuate the neutron flux escaping the reactor vessel to prevent excessive activation of plant components and structures over the life of the plant.

4.3.2.2 Secondary Shield

The secondary shield is a 4 foot thick reinforced concrete structure surrounding the primary shield and the bulk of the primary system equipment, including piping, pumps, and steam generators. The secondary shield supplements the primary shield by attenuating neutron and gamma radiation escaping from the primary shield.

4.3.2.3 Reactor Building Shield

The reactor building shield is a reinforced, prestressed containment structure with 3.5 foot thick cylindrical walls and a 3.0 foot thick dome that encloses the Nuclear Steam Supply System (NSSS).

4.3.2.4 Auxiliary Shield

Auxiliary shielding includes concrete walls, covers, doors, and removable blocks which provide radiation protection of personnel from the numerous sources of radiation occurring from equipment housed within the auxiliary building.

4.3.2.5 Spent Fuel Shielding

Two radiation criteria were considered in the design of the spent fuel pool to ensure shielding adequacy, i.e., first, the concrete shielding surrounding the pool and second, the minimum water depth above the fuel elements. The spent fuel pool is surrounded by 5 foot thick concrete walls which will limit the maximum continuous radiation levels in working areas to less than 2 mrem/hour. Operating experience shows dose rates of 0.5 to 2.0 mrem/hour either at the edge or above the center of the Spent Fuel Pools regardless of the quantity of fuel stored. The shielding is adequate to maintain below tolerance dose levels for normal contamination of the pool water by particulates. Similarly, gaseous activity coming out of solution from the pool water is picked up by a push-pull type ventilation system over the pool. The spent fuel pool filters and demineralizers are located in a shielded cubicle not in a readily accessible area, and thus, will not present any radiation hazard should they become contaminated.

A minimum cover of water of 8.0 feet has been specified during the transfer of the fuel element (assuming only one element is transferred at a time) to maintain the radiation level below 2 mrem/hr.

4.3.3 MATERIALS

The material used for the primary, secondary, reactor building, control room, auxiliary, and spent fuel shields is ordinary concrete with a density of approximately 150 lb/ft^3.

4.3.4 DESIGN FEATURES

The following features are utilized in the design of the plant shielding to minimize personnel exposure during operation and maintenance of equipment:

a. Radioactive process equipment is located in shielded cubicles designed to attenuate radiation levels to acceptable values consistent with the radiation zone designations shown in Figures 4-6 through 4-9. Access labyrinths are provided on these rooms to preclude a direct radiation path from the equipment to the accessible area.

b. Penetrations, ducts, and voids in radiation shield walls are located to prevent the possibility of streaming from a high-to-low radiation area or otherwise are adequately shielded.

c. Shielding discontinuities caused by shield plugs, hatch covers, and shield doors are provided with off-sets to reduce radiation streaming.

d. Radioactive piping is routed through high radiation areas where practicable or in shielded pipe chases in low radiation areas.

e. The reactor vessel functions as a storage vessel for the activated core support internals and the incore assemblies. See Figure 4-2.

4.4 RADIATION MONITORING SYSTEM

4.4.1 DESIGN BASIS

The Radiation Monitoring System (RMS) detects, indicates, alarms, and records the radiation levels and concentrations of radioactivity at selected locations inside of the plant to verify compliance with 10 CFR 20.

4.4.2 SYSTEM DESIGN

4.4.2.1 System Description

The RMS, as shown in Figure 4-3A and Figure 4-3B, comprises the following systems:

a. Area Gamma Monitoring System
b. Atmospheric Monitoring System
c. Liquid Monitoring System

The RMS receives 120 volt, single-phase, 60-hertz power from essential service power supply buses. The pump assemblies receive power from the 480 volt Engineered Safeguards (ES) Power System.

RMS equipment is labeled to identify the range and sensitivity of each instrument and its location.

Each channel has an adjustable, high radiation alarm, alert low level alarm, and power supply failure alarm. Contacting type meters are not used.

Solid state circuitry is used except for primary detectors.

Each radiation channel is capable of being calibrated periodically with radiation sources.

The systems are designed so that an alarm is initiated upon power failure or a signal loss occurring in the channel.

All channels display their respective radiation levels and alarm conditions on radiation monitoring panels located in the control room. A radiation monitoring recorder panel is located in the control room. It contains the recorders used to obtain data of the radiation level and concentrations at selected locations in the plant.

As shown in Figure 4-4, the RMS provides signals for control or interlocking functions for other systems.

4.4.2.1.1 Area Gamma Monitoring System

The Area Gamma Monitoring System comprises multiple channels, each of which is provided with a gamma sensitive detector. Detectors are installed at selected locations, as indicated in Figure 4-3A, to aid in determining the radiation levels throughout the plant.

All channels have a range from 0.1 mr/hour to 10^4 mr/hour.

RM-G14 is the Fuel Storage Pool Area monitor which is required when spent fuel is in the pool or the building. The alarm setpoint for RM-G14 is 15 mr/hr. Procedures exist to qualitatively assess, functionally test, and calibrate this instrument. [TSA149]

Channels RM-G1, RM-G14, and RM-G15 use G-M detectors with their energy response modified by a thin lead shield. Remote actuated check sources mounted in the detector housing provide in-service functional testing and a live zero for positive indication of operation. The detector is a high range ion chamber designed to operate in an ambient temperature up to 350°F and a maximum pressure of 70 psig with uniform energy response (± 20%) from 0.01 to 3 MeV and a sensitivity down to 60 KeV gammas.

TSA149 *This paragraph was added when Technical Specification Amendment 149 was issued. Obtain Licensing & Regulatory Affairs concurrence before changing.*

The initial alarm setpoints for high radiation alarms were based on the shielding design criteria at their locations as indicated in Section 4.3.1, and may be adjusted up or down based on radiation surveys made during actual operation. The warning or low level alarm setpoints for the area monitors are established above the normal operating plant background and below the high alarm setpoint to provide an alert in the event of a change of the normal plant operating conditions.

4.4.2.1.2 Atmospheric Monitoring System

The Atmospheric Monitoring System, as shown in Figure 4-3A, comprises monitors of which one is the movable cart type. The following monitor is equipped with a particulate prefilter ahead of the gaseous channel. The particulate prefilter is removed periodically and taken to the laboratory for analysis.

- RM-A4, Fuel handling and spent fuel area exhaust duct

The final gas discharge from the plant to the atmosphere is monitored by RM-A2, auxiliary and fuel handling building exhaust duct monitor, which has particulate sample filters in the sample line upstream of the noble gas detector. The sample filters are removable for offline analysis.

An interlock is provided from RM-A2 to terminate any gaseous release upon failure of radiation monitoring sampling systems.
The locations, functions, and ranges of monitors located inside of the plant are as follows:

a. Auxiliary Building and Fuel Handling Area Exhaust Duct Monitor RM-A2

   This monitor detects the activity of the air released by the auxiliary building and fuel handling area exhaust duct. The monitor is located close to the combined exhaust duct for the auxiliary building and fuel handling area to minimize length of the sample piping from the isokinetic nozzle to the particulate filter. The isokinetic nozzle takes an air sample from the center of the exhaust duct and is located at a point selected for reduced turbulence.

   The high alarm gas channel setpoint is established to assure that the site boundary dose rate does not exceed 500 mrem/yr. If gases are released such that the site boundary dose rate is exceeded, the monitor will alarm, shutdown fans supplying air to the Auxiliary Building (AHF-9A, AHF-9B, AHF-10, AHF-11A, AHF-11B, AHF-30 and AHF-34A, including AHU-3).

b. Fuel Handling and Spent Fuel Handling Area Exhaust Duct Monitor RM-A4

   The high radiation alarm setpoint is set at a level based on plant operating conditions. If the alarm actuates, the Fuel Handling Area Supply Fan (AHF-10) is shut down.

   This monitor is provided with particulate prefilter and a continuous gaseous detection channel with a G-M detector. The function of this monitor is to provide an early detection of radioactivity release. The range of the gaseous channel is $2x10^{-6}$ to $1x\ 10^{-2}$ µCi/cc Kr-85.

4.4.2.1.3 Liquid Monitoring System

The Liquid Monitoring System, as shown in Figure 4-3B.

a. Plant Discharge Line (prior to dilution) RM-L2

   This channel monitors the radioactivity of the effluent from the liquid waste processing system. A lead shielded sampler equipped with scintillation detector is located prior to the discharge dilution point and has a minimum range of 1×10^{-6} to 1×10^{-2} µCi/ml Cs-137.

   The rate meter provides indication 1×10^{1} to 1×10^{6} CPM, compatible with the detector utilized.

All liquids to be discharged from the Liquid Waste Disposal System are also first sampled and analyzed in the laboratory before dilution and disposal.

The monitor provides a backup to administrative procedures to prevent release of activity in excess of the planned discharge based on analysis. The high radioactivity alarm will automatically initiate the closure of the waste processing discharge valve. The activity level at which this interlock is actuated can be adjusted as a function of the radioactivity content of the waste batch and/or available dilution.

b. Station Drain Discharge Monitor RM-L7

A radiation monitor, RM-L7, is provided in the discharge path from the condensate demineralizer regeneration neutralization tank (SDT-1) to the discharge canal via the Nuclear Service and Decay Heat Sea Water System (RW). The minimum range of this monitor is 1 x 10^{-6} to 1 x 10^{-2} μCi/ml Cs-137. The high radioactivity alarm will automatically initiate the closure of the waste processing discharge valve (SDV-90). The activity level at which this interlock is actuated can be adjusted as a function of the radioactivity content of the waste and/or available dilution.

4.4.3 SYSTEM EVALUATION

The RMS continuously monitors the normal station effluent discharge paths during steady state, transient or accident conditions, and after an accident will provide information in determining the magnitude of the accident. Interlock functions are also provided to control effluent releases.

The monitors located in the normal station effluent discharge paths and manual sampling will provide information on the rate and location of releases.

The RMS is powered from reliable and diversified sources to assure continuity of operation in the event of a loss of off-site power.

Testing and maintenance features such as remote-operated check sources and calibration and flushing connections (decontamination type for maintenance) are provided for periodic system check and/or calibration.

The calibration of the monitors located in the normal station effluent discharge paths will be compared to the results of an initial calibration utilizing radioactive sources traceable to the National Institute of Standards and Technology (NIST). Calibration curves and solid point sources will be used to periodically calibrate the detector.

The area monitor channels RM-G1, RM-G14 and RM-G15 are calibrated by exposure to a point source. Exposure of the detector to a known gamma flux from the calibration source placed at different fixed positions will verify the dose response for a minimum of two data points.

The gaseous channel monitors are calibrated by comparison of the response of the detector to a calibrated point source. Also, the electronic characteristics and adjustments of the signal processing and display components are verified. The point source calibration data is referenced to initial calibration, established by measurement of sensitivity to a gas sample with a known concentration of a gaseous radioisotope.

The liquid monitors RM-L2 and RM-L7 are calibrated by measurement of the scintillation detector response to a calibrated point source. Also, the electronic characteristics and adjustments of the rate meter are verified. The point source calibration data is referenced to prior calibration of sensitivity to a liquid sample with a known radioisotope concentration.

Auxiliary equipment, such as air flow indicators and system recorders, are routinely tested and/or verified according to vendor instruction manuals.

Some interlock alarm setpoints are verified by substitution of the detector signal with a measured electrical input signal. Where it is practical to do so, alarms are verified by exposure of the detector to a radionuclide source and an actuation of associated equipment interlocks, rather than an artificial electrical input signal.

The calibration data is recorded and maintained for all radiation monitoring channels.

The radiation monitors located in the gaseous release paths sample noble gases and particulate radioactivity with sufficient sensitivity to demonstrate compliance with 10 CFR 20 limits. They provide samples which, when analyzed with laboratory equipment, permit demonstration of compliance to restrictive regulations.

4.5 HEALTH PHYSICS

4.5.1 PROGRAM DESCRIPTION

Crystal River Unit 3's Radiation Protection Program resides within the Health Physics and Radiation Safety Procedures. These procedures describe the programmatic content and operating philosophy of the Radiation Protection Program. The Health Physics Section of the Radiation Protection Department is responsible for the administration of these procedures in accordance with the requirements of the Plant Operating Manual (POM).

The Radiation Protection Program is based upon a Risk versus Benefit ALARA methodology and is designed to prevent the occurrence of non-stochastic health effects and to minimize the probability of occurrence of stochastic health effects within three distinct populations: individual radiation workers, the workforce, and members of the general public. [TSA149]

4.5.2 TRAINING (TRAINING AND HEALTH PHYSICS)

The Radiation Protection Department is responsible for orientation and training of personnel in radiation protection principles and procedures to maintain exposures As Low As Reasonably Achievable (ALARA).

4.5.3 TOTAL RISK ASSESSMENT

The health and safety of employees is of paramount importance to Duke Energy Florida (DEF). Therefore, before prescribing the use of any personal protective measures, all of the risk factors associated with the task to be performed will be evaluated and the protective measures chosen will be those that offer the best protection against the greatest risk present. To the extent practical, industrial, environmental and radiological risks will be eliminated in the planning phase of the work control process. Personnel monitoring devises, protective clothing, portable shielding and respiratory protection equipment are available for use when conditions warrant.

4.5.4 ACCESS CONTROL AND THE RWP (ALL ORGANIZATIONS)

Only trained/qualified individuals are granted unescorted access into a Radiation Controlled Area (RCA). The Radiation Work Permit (RWP) is the mechanism used to authorize and document the requirements for entry into an RCA. Personnel entering an RCA are required to read and understand the information presented by the RWP authorizing their entry.

A prospective assessment is performed for each category of radiation worker and is reviewed on an annual basis during the yearly Radiation Protection Program self-assessment. This prospective assessment and the requirements specified on the RWP will determine the need for and level of personnel monitoring required for the various work activities to be performed inside of the Restricted Area.

An alternative to access control to High Radiation Areas may be provided as follows:

> In lieu of the "control device" or "alarm signal" required by paragraph 20.1601(a) of 10 CFR 20, a High Radiation Area in which the intensity of radiation is greater than 100 mrem/hr but less than 1000 mrem/hr shall be barricaded and conspicuously posted as a High Radiation Area and entrance thereto shall be controlled by issuance of a Radiation Work Permit and any individual or group of individuals permitted to enter such areas shall be provided with one or more of the following:

TSA149 *This Section was added when Technical Specification Amendment 149 was issued. Obtain Licensing & Regulatory Affairs concurrence before changing.*

a. A radiation monitoring device which continuously indicates the radiation dose rate in the area, or

b. An integrating alarming dosimeter which alarms when a preset integrated dose or dose rate is received. Entry into such areas with this alarming dosimeter may be made after the dose rate levels in the area have been established and personnel have been made knowledgeable of them, or

c. An individual qualified in Health Physics Procedures with a radiation dose rate monitoring device, who is responsible for providing positive control over activities in the area and who performs periodic radiation surveillance at the frequency specified by the Radiation Work Permit.

This provision is addressed in Permanently Defueled Technical Specification 5.8 "High Radiation Area" and must be maintained as stated unless prior NRC approval is granted in accordance with 10 CFR 50.90 and 10 CFR 20.1601(c).

4.5.5 HEALTH PHYSICS RESOURCES

The necessary manpower, instrumentation and related equipment needed to support the Radiation Protection Program as described in the appropriate procedures are provided to support the safe operation of the unit. Instrumentation and equipment are available to sample for the various forms of radioactive materials (i.e., gaseous, liquid and solid) and provide quantitative and qualitative data as necessary. The instrumentation and equipment is periodically checked and calibrated to assure quality performance.

Health Physics instrumentation and equipment is situated in various locations throughout the facility. The majority of the equipment is located on the first floor of the Control Complex which also functions to separate the Turbine and other secondary buildings from the Primary RCA consisting of the Auxiliary and Reactor Buildings.

4.5.6 RELATED MEDICAL PROGRAMS

Medical qualifications and health physics bioassay requirements must be met to enter Crystal River Unit 3's Respiratory Protection Program. Bioassay services are provided on a routine basis and are available based upon recommendations received by either Health Services or Health Physics personnel. Any employee may request bioassay services and their results at any time. personnel in the respiratory protection program must pass initial and annual physical requirements. Bioassay services (i.e., Invivo counting, urine analysis, etc.) are provided on a routine basis and available on recommendation by either Health Services, Health Physics, or the employee.

TABLE 4-1, DISPOSAL SYSTEM COMPONENT DATA

Item Number	Name	Type[2]	Capacity, Ft^3 (each tank)		Design		Material		Vented To	Design Code	Seismic Design	Comments
			Total	Liquid	Temp (°F)	Press. psig	Body	Lining				
WDT-3A WDT-3B WDT-3C	R.C. Bleed Tanks (Abandoned)	V/S	11,050	10,150	250	25	SS	None	V.H.	ASME III-C	Class I	Maximum operating temp/press is 150F/±3 psig. Contains nearly one primary system volume.
WDT-4	Misc. Waste Storage Tank	H/S	3,150	2,750	250	25	SS	None	V.H.	ASME III-C	Class I	Maximum operating temp/press is 150F/±3 psig.
WDT-5	Reactor Coolant Drain Tank (Abandoned)	V/L	843.88	562	300	100	SS	None	V.H.	ASME III-C	Class I	Rupture disk provides overpressure relief. Internal plate coils provide cooling.
WDT-6	Spent Resin Storage Tank	V/L	920	860	150	15	SS	None	V.H.	ASME III-C	Class I	Nominal resin capacity 800 ft^3 or two year's retention as design basis.
WDT-7A WDT-7B	Concentrated Waste Storage Tanks (Abandoned)	V/L	920	728	200	15	SS	None	V.H.	ASME III-C	Class I	Nominal one year's retention of evaporator concentrate. (Evaporator was eliminated via MAR 99-08-02-01.)

[2]Legend: V/S = vertical skirt V/L = vertical legs
H/S = horizontal saddle V.H. = vent header

(Table continues on the following page)

Table 4-1, Disposal System Component Data (continued)

Item Number	Name	Type[2]	Capacity, Ft ^{3} (each tank)		Design		Material		Vented To	Design Code	Seismic Design	Comments
			Total	Liquid	Temp (°F)	Press. psig	Body	Lining				
WDT-8A WDT-8B	Concentrated Boric Acid Tanks (Abandoned)	V/L	920	728	200	15	SS	None	V.H.	ASME III-C	Class I	
WDT-9	Neutralizer Tank (Abandoned)	V/L	530	470	150	15	CS	Rubber	V.H.	ASME III-C	Class I	
WDT-10A WDT-10B	Evaporator Condensate Storage Tanks	H/S	1,340	1,100	150	15	CS	Rubber	Atm.	ASME III-C	Class I	The inventories in these 2 tanks principally govern releases of radioactive liquids to the environment.
WDT-11A WDT-11B	Laundry/Hot Shower Monitoring Tanks	V/S	127	120	Amb.	Atm.	SS	None	Atm.	ASME VIII	None	

[2]Legend: V/S = vertical skirt; V/L = vertical legs; H/S = horizontal saddle; V.H. = vent header

(Table continues on the following page, with different Headings …)

Table 4-1, Disposal System Component Data (continued)

Item Number	Name	Resin Volume Ft^3 (each)	Design Flow Rate, gpm (each)	Type Of Resin	Tank Volume, Ft^3 (each)	Design Temp (°F)	Design Press psig	Material	Vented To	Design Code	Seismic Design	Comments
WDDM-1A WDDM-1B	Deborating Demineralizers (Abandoned)	40	70	Anion	80	120	150	SS	V.H.	ASME III-C	Class I	Resin is generated in place. Spent resin dumped to spent resin tank.
WDDM-2A WDDM-2B	Cation Demineralizers (Abandoned)	36	70	Cation*	80	120	150	SS	V.H.	ASME III-C	Class I	Spent resin dumped to spent resin tank.
WDDM-3A WDDM-3B	Evaporator Condensate Demineralizers (Abandoned)	12	15	Mixed Bed	16	120	150	SS	V.H.	ASME III-C	Class I	Spent resin dumped to spent resin tank.

* Note: One of the cation demineralizers is normally loaded with mixed resin (anion and cation). If high coolant activity is detected by isotopic analysis, resin loadings will be adjusted as necessary to maintain the limits required by the ODCM.

(Table continues on the following page, with different Headings …)

Table 4-1, Disposal System Component Data (continued)

Item Number	Name	Type	Design Flow Rate, gpm	Decontamination Factor	Feed Tank Volume, Ft^3	Material	Vented To	Design Code	Seismic Design
WDEV-1 WDEV-2 WDEV-3 WDEV-4	Reactor Coolant Evaporator Package/Misc. Waste Evaporator Package (NOTE: These evaporator packages were eliminated via MAR 99-08-02-01.)	Vacuum Distillation	12.5	$> 10^4$	133	SS	V.H.	ASME III-C	Class I
Total Liquid Capacity In Liquid Waste System Process Equipment = 673 ft^3									

Item Number	Name/Function	Type	Flow gpm	Head Ft H_2O	Design		Materials *Wetted Parts	Seismic Design
					Temp (°F)	Press (psig)		
WDP-5A WDP-5B WDP-5C	Waste Transfer Pumps inject feed to primary system via Makeup & Purification System & process reactor coolant (Abandoned)	Horizontal Canned Centrifugal	140	325	150	150	Stainless Steel *	Class III
WDP-6A WDP-6B	Miscellaneous Radwaste Transfer Pumps transfer miscellaneous wastes to processing	Horizontal Centrifugal CSO	35	225	150	150	Stainless Steel *	Class III
WDP-7	Reactor Coolant Drain Pump Drains primary system (Abandoned)	Horizontal Canned Centrifugal	200	100	150	150	Stainless Steel *	Class III
WDP-8	R.C. Drain Tank Pump Circulate drain tank contents for cooling or transfer to waste processing (Abandoned)	Horizontal Canned Centrifugal	30	100	150	150	Stainless Steel *	Class III
WDP-9A WDP-9B	Neutralizer Pumps Transfer neutralized wastes (Abandoned)	Horizontal Canned Centrifugal	30	25	150	150	Stainless Steel *	Class III

(Table continues on the following page)

Table 4-1, Disposal System Component Data (continued)

Item Number	Name/Function	Type	Flow gpm	Head Ft H_2O	Design Temp (°F)	Design Press (psig)	Materials *Wetted Parts	Seismic Design
WDP-10	Decant Pump Transfer demineralized water	Horizontal Canned Centrifugal	30	29	150	150	Stainless Steel *	Class III
WDP-11	Slurry Pump Transfer slurry	Positive Displacement (Progressing) Cavity	120	115.5	150	150	Stainless Steel *	Class III
WDP-12A WDP-12B	Concentrated Liquid Waste Pumps Transfer concentrated wastes (Abandoned)	Horizontal Canned Centrifugal	30	53	180	150	Stainless Steel*	Class III
WDP-13A WDP-13B	Boric Acid Recycle Pumps Transfer reclaimed boric acid (Abandoned)	Horizontal Canned Centrifugal	30	69	180	150	Stainless Steel*	Class III
WDP-14A WDP-14B	Evaporator Condensate Pumps/transfer radioactive liquid	Horizontal Centrifugal	100	40	120	150	Stainless Steel*	None
WDP-23A WDP-23B	Laundry/Hot Shower Monitoring Pumps/transfer and recirculate wastes	Vertical Centrifugal	40	140	100	200	Stainless Steel*	None

(Table continues on the following page, with different Headings …)

Table 4-1, Disposal System Component Data (continued)

Item Number	Name/Function	Type	Capacity, Ft^3 (gal)	Sump Depth	Flow gpm	Head Ft H_2O	Comments
WDP-2A WDP-2B WDSU-1	Reactor Building sump and Pumps collect water from floor and miscellaneous drains within Reactor Building	SS Lined Concrete Pit	1000 (7500)	11'-6"	100	45	Pumps to Miscellaneous Waste Storage Tank
WDP-4A WDP-4B WDSU-2	Auxiliary Building sump and Pumps collect water from floor and miscellaneous drains within auxiliary and fuel handling buildings	SS Lined Concrete Pit	560 (4190)	8 '	125	50	Pumps to Miscellaneous Waste Storage Tank
WDP-3A WDP-3B WDSU-4 WDSU-5	Decay Heat Pit Sumps A & B and Pumps collect leakage from equipment in Decay Heat Pits A & B, respectively	SS Lined Concrete Pit	64 (480) each	4 '	30	54	Pumps to Miscellaneous Waste Storage Tank
WDP-22A WDP-22B WDSU-3	Laundry and Shower Sump and Pumps	SS Lined Concrete Pit	420 (3142)	6 '	40	100	Pumps to laundry/hot shower monitoring tanks

END OF TABLE

TABLE 4-2, PRINCIPAL SHIELDING

Component	Concrete Thickness (ft)
Reactor Building	
Primary Shield (below flange)	5
Primary Shield (above flange)	4 to 4.5
Secondary Shield	4
Reactor Building Vertical Walls	3.5
Reactor Building Dome	3
Side Walls of Fuel Transfer Canal	4
End Walls of Fuel Transfer Canal	3 to 4
Floor of Fuel Transfer Canal	4
Reactor Coolant Drain Tank	3.0
Minimum Water Over Active Fuel During Transfer	8.0
Auxiliary Building	
Misc. Waste Storage Tank	3
Reactor Coolant Bleed Tanks	2 to 3
Waste Transfer Pumps	2 to 4
Waste Gas Compressors	2 to 4
Waste Gas Decay Tank	2 to 6
Boric Acid Recycle Pumps	2 to 4
Concentrated Radioactive Waste Pumps	2 to 4
Concentrated Boric Acid Tanks	2 to 5
Concentrated Waste Storage Tanks	2 to 5
Spent Resin Storage Tank	3 to 5
Makeup Tank	2 to 4
Makeup and Purification Pumps	2 to 3
Waste Gas Decay Tank	2 to 5
Evaporator Condensate Demineralizer	3 to 5
Makeup and Purification Demineralizers	3 to 5
Deborating Demineralizers	3 to 5
Waste Drumming Area	2 to 3
Fuel Handling Building	
Side Walls of Spent Fuel Pools	5
End Walls of Spent Fuel Pools	5
Bottom of Spent Fuel Pools	5
Pools Watergate Wall	4
Spent Fuel Shipping Cask Pit Walls	3

TABLE 4-3, ASSUMPTIONS FOR RADIOACTIVE WASTE HANDLING EVENT

Input Assumption	Basis
1. Source activity, except for Cobalt-60: 99.7 Ci of mixed fission and activation products	1. Primary Resin Shipment 08-030 Highest total activity of all resin shipments since 2008
2. Source activity for Cobalt-60: 16.1 Ci	2. Primary Resin Shipment 12-006 Highest quantity of Cobalt-60 of all resin shipments since 2008.
3. EAB χ/Q: 1.78E-04 sec/m^3	3. Calculation N12-0002, DSAR Section 2.3.4
4. Breathing Rate: 3.5E-04 m^3/s	4. Regulatory Guide 1.183
5. Internal dose conversion factors	5. Federal Guidance Report 11
6. External dose conversion factors	6. Federal Guidance Report 12
7. Release Fraction: 10%	7. 10 times greater than 10 CFR 30.72, Schedule C release fraction for mixed fission and activation products

FIGURE 4-1A, LIQUID WASTE DISPOSAL, FD-302-681, SHEET 1 OF 6

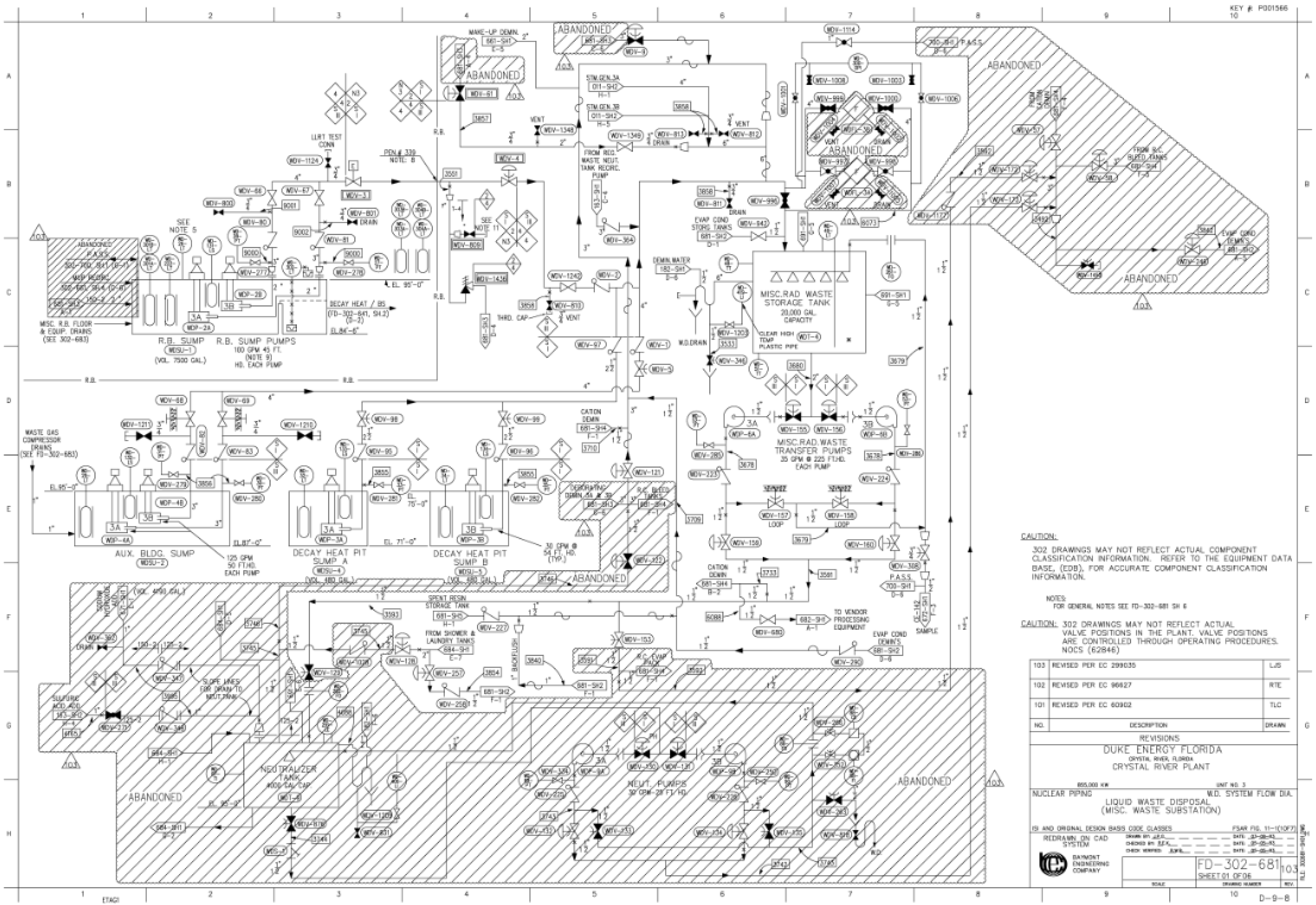


FIGURE 4-1B, LIQUID WASTE DISPOSAL, FD-302-681, SHEET 2 OF 6

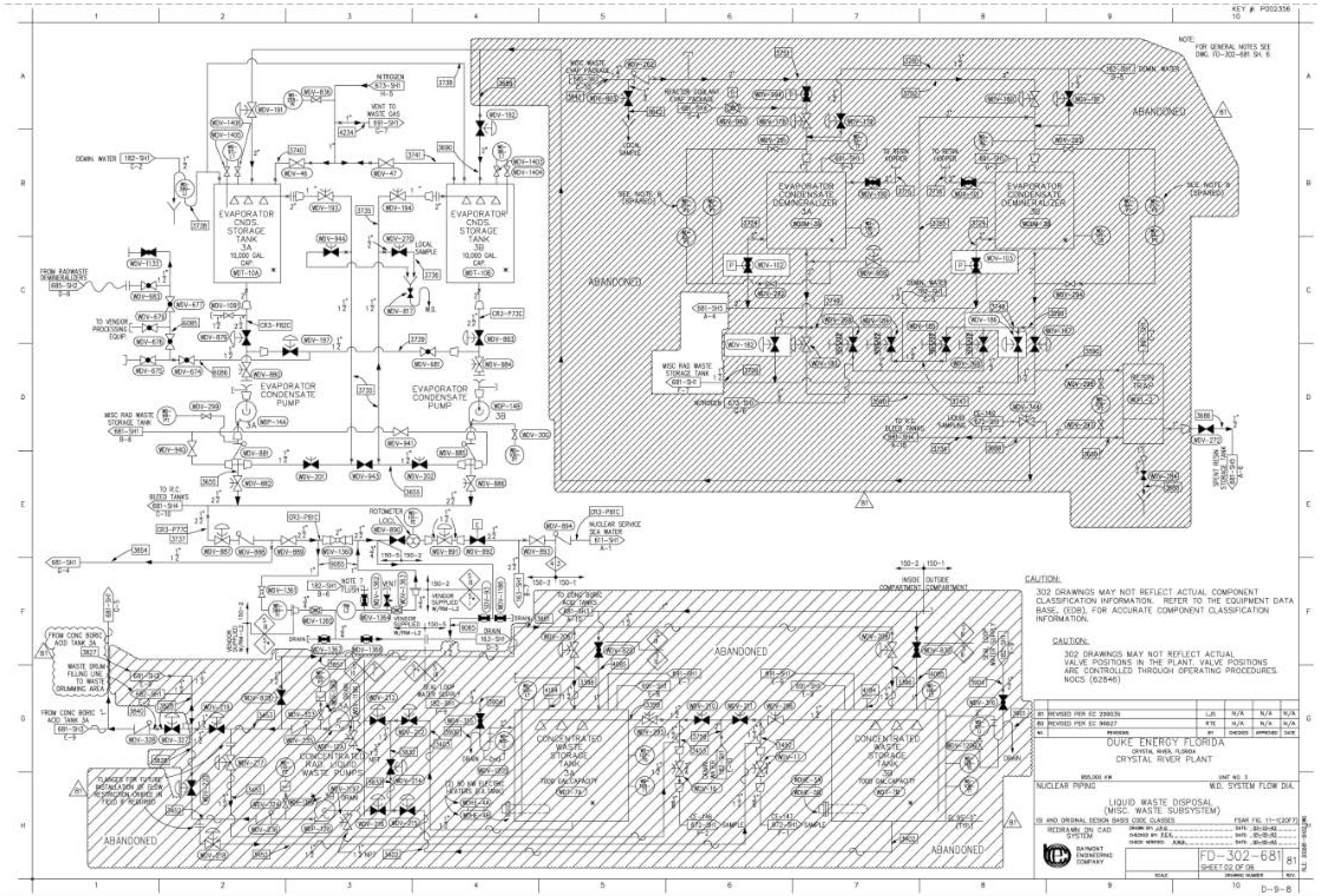


FIGURE 4-1C, LIQUID WASTE DISPOSAL, FD-302-681, SHEET 3 OF 6

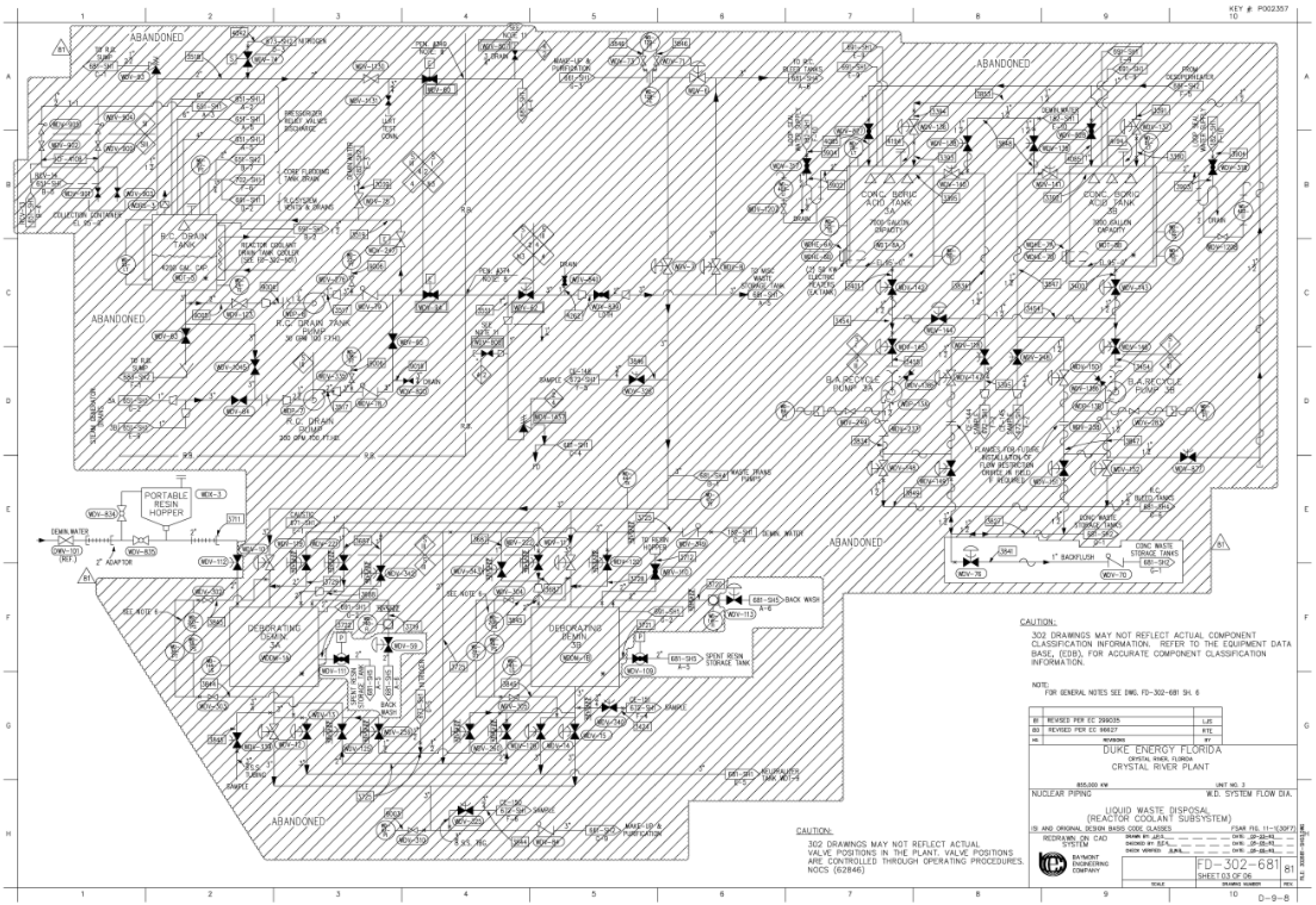


FIGURE 4-1D, LIQUID WASTE DISPOSAL, FD-302-681, SHEET 4 OF 6

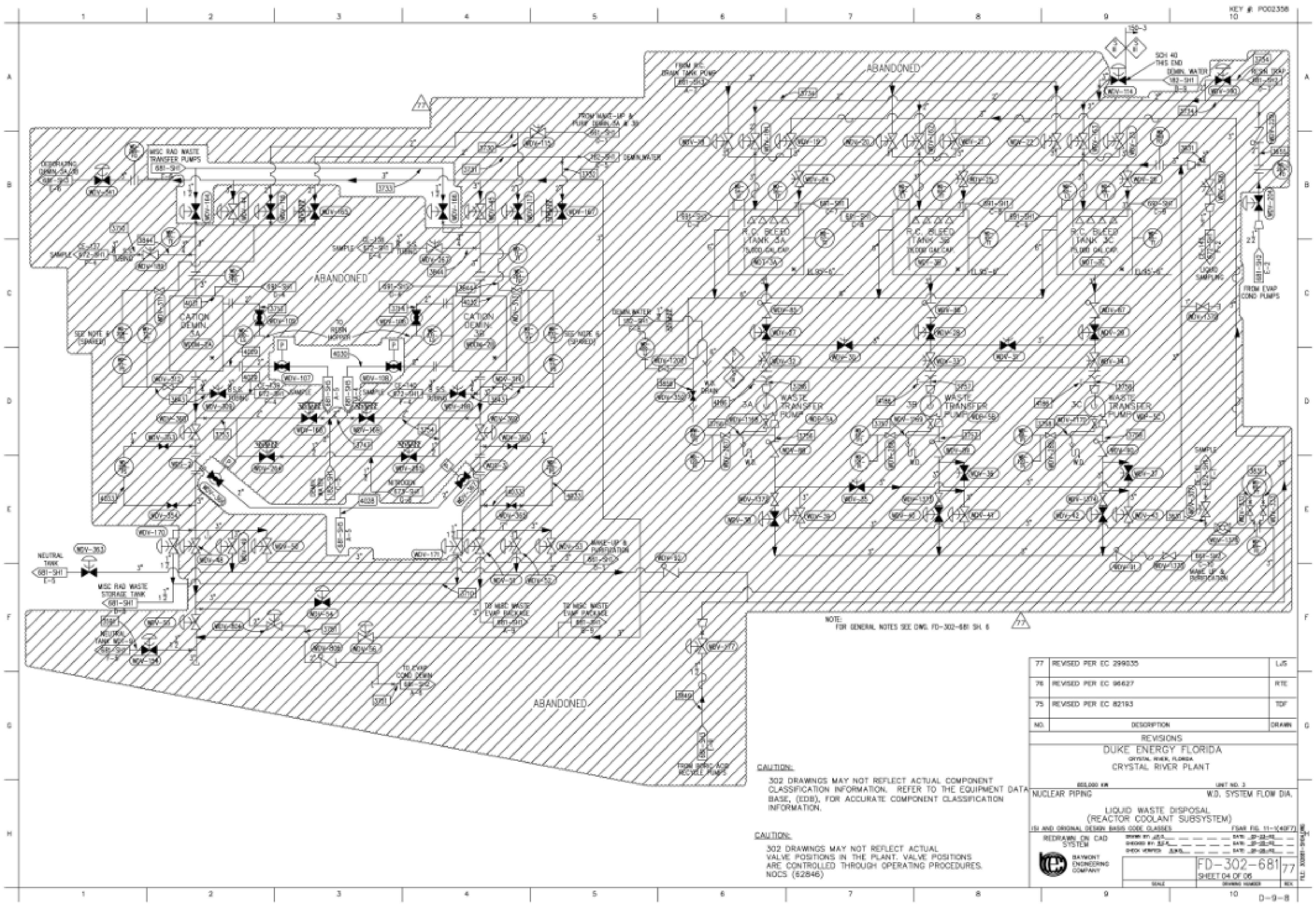


FIGURE 4-1E, LIQUID WASTE DISPOSAL, FD-302-681, SHEET 5 OF 6

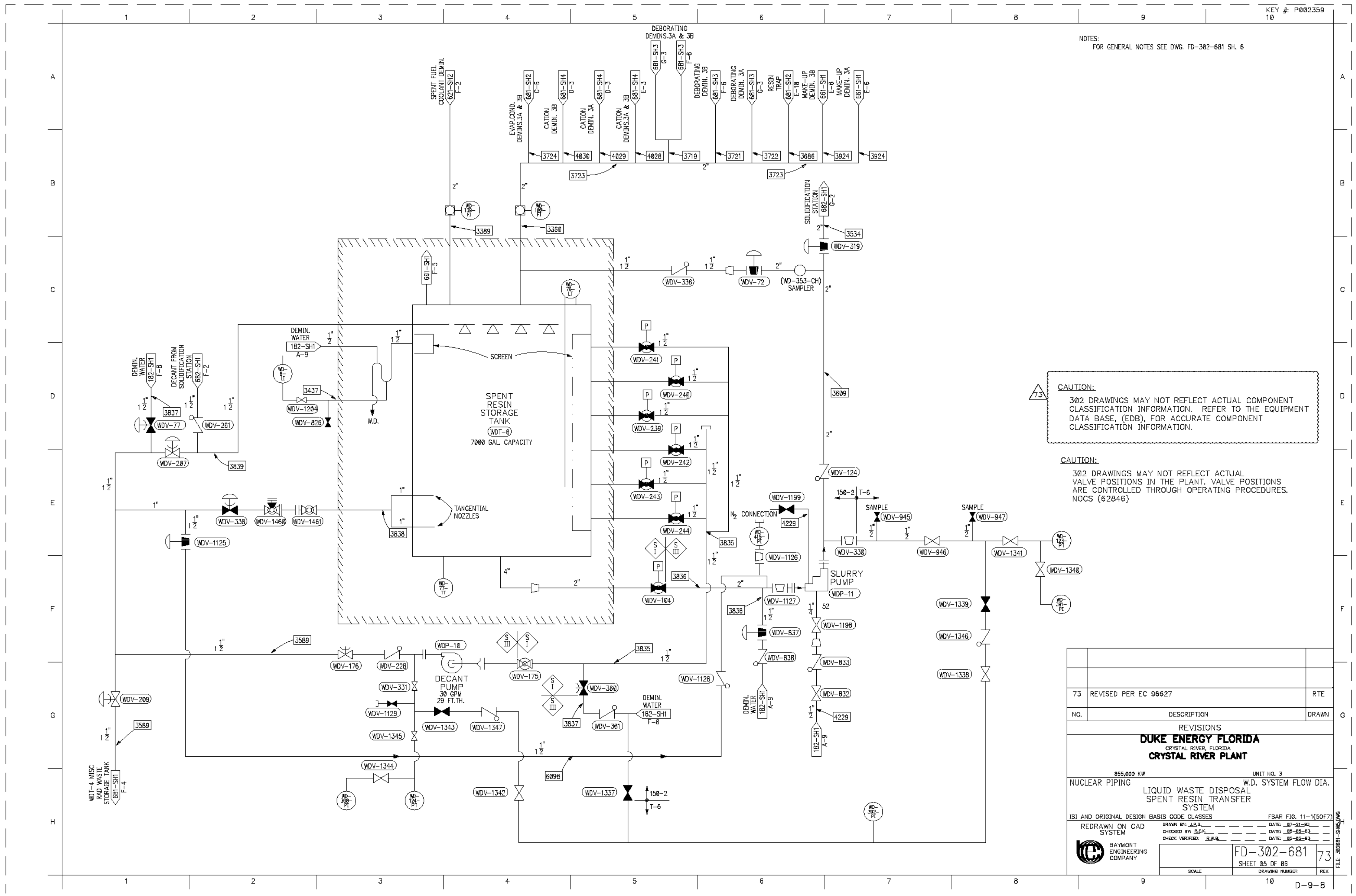


FIGURE 4-1F, LIQUID WASTE DISPOSAL, FD-302-681, SHEET 6 OF 6

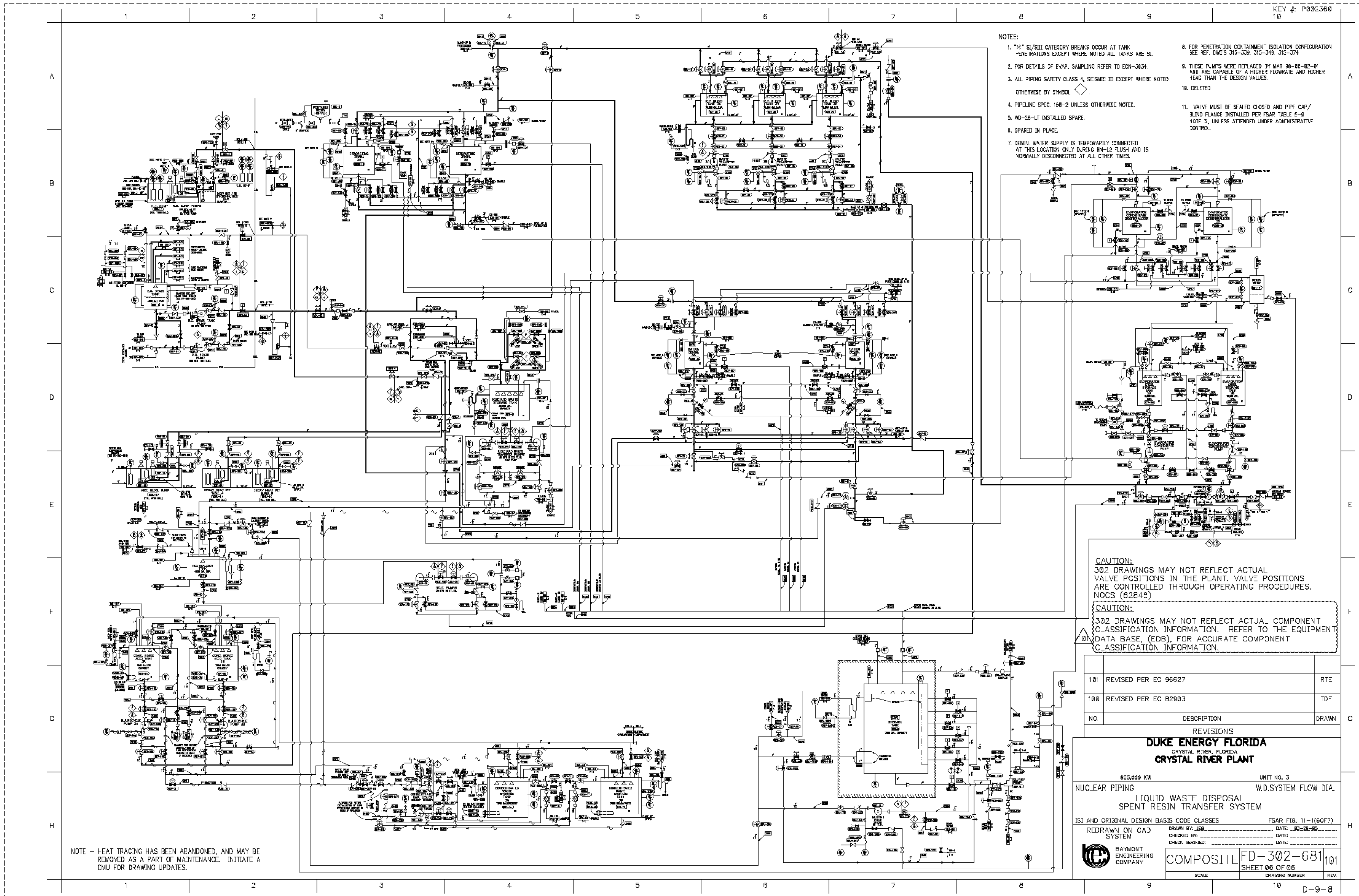


FIGURE 4-1G, LIQUID WASTE DISPOSAL, FD-302-684, SHEET 1

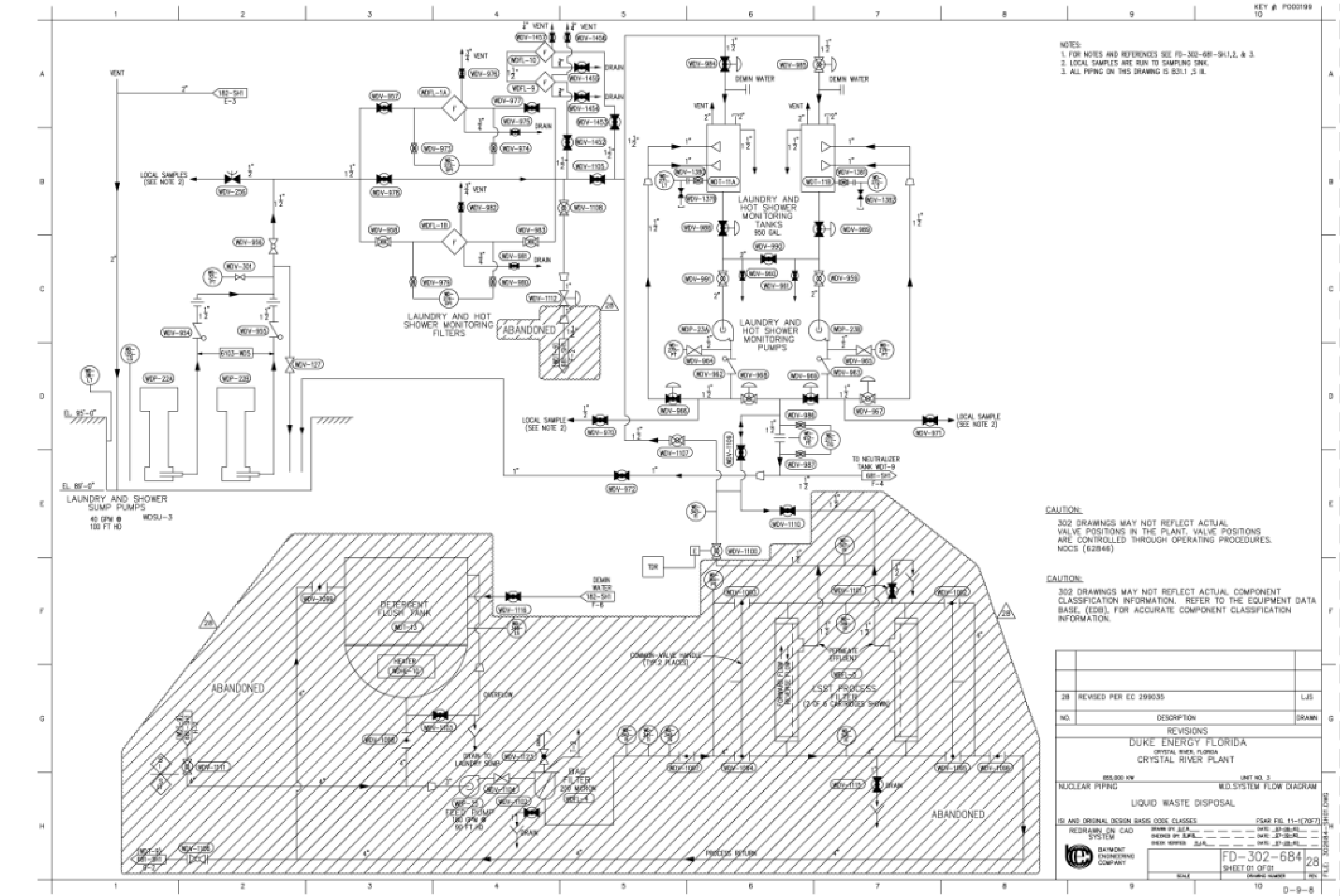


FIGURE 4-2A, FLOOR AND EQUIPMENT DRAINS, AUXILIARY BUILDING AND REACTOR BUILDING, FD-302-683, SHEET 1 OF 3

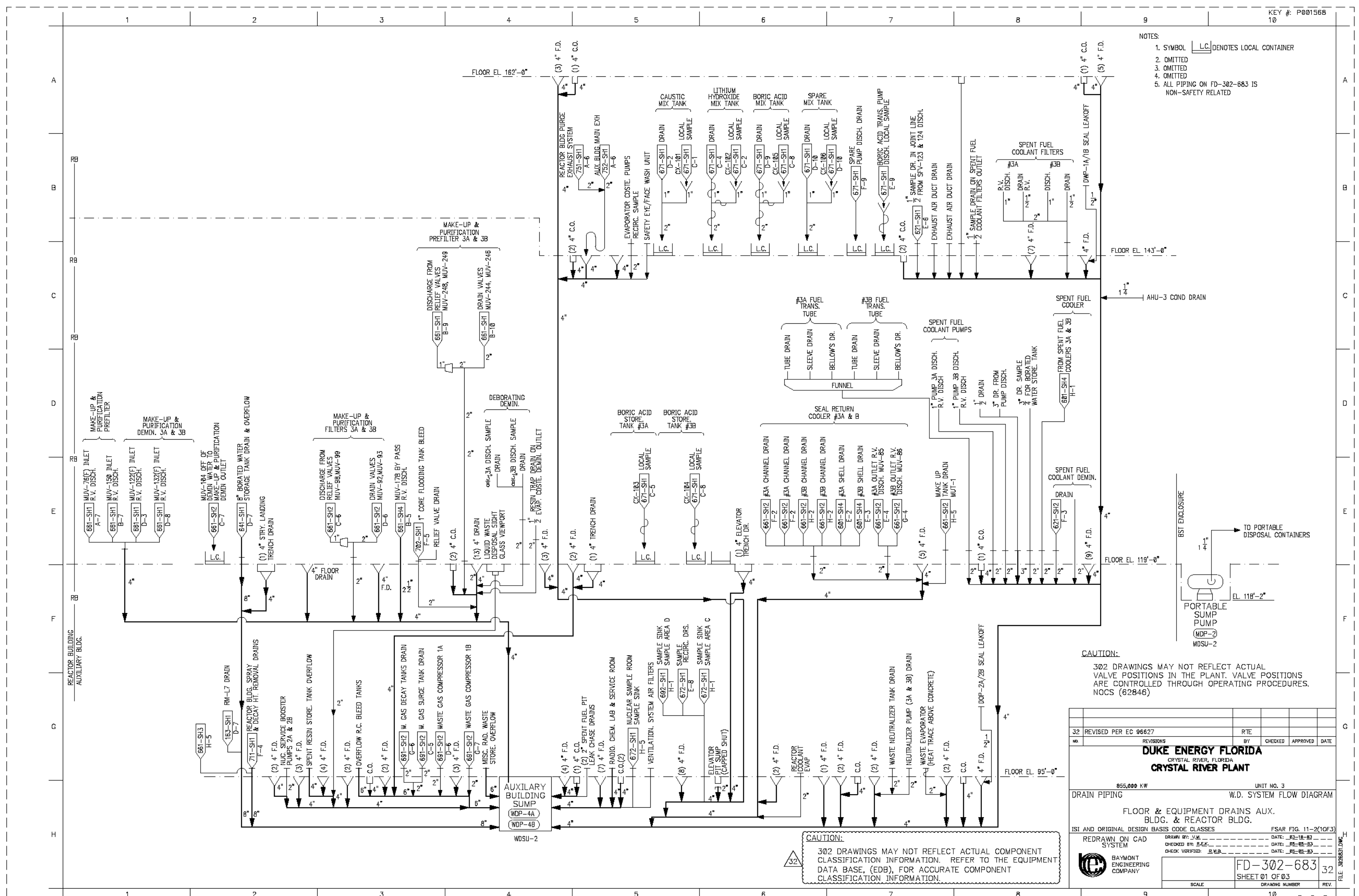


FIGURE 4-2B, FLOOR AND EQUIPMENT DRAINS, AUXILIARY BUILDING AND REACTOR BUILDING, FD-302-683, SHEET 2 OF 3

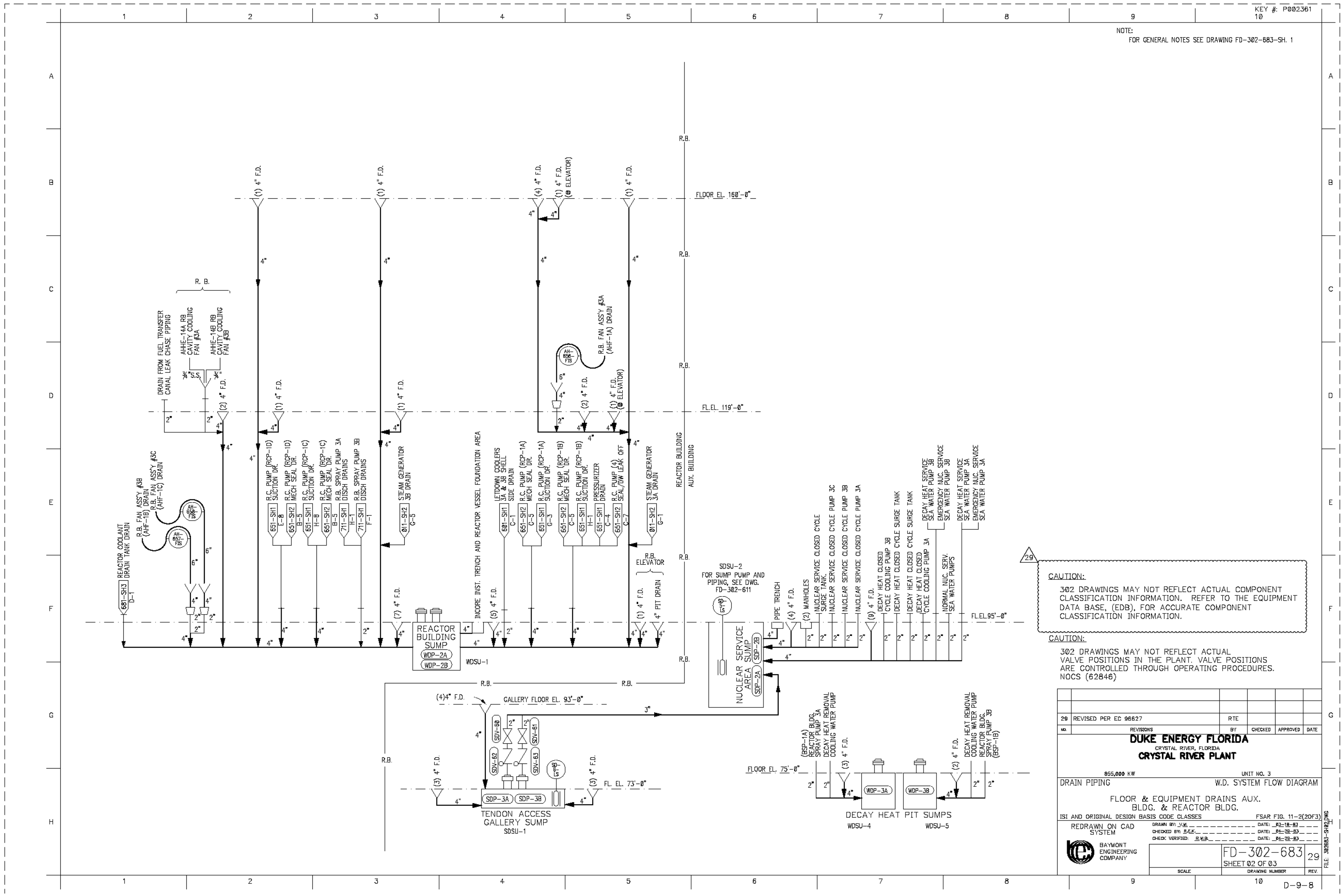


FIGURE 4-2C, FLOOR AND EQUIPMENT DRAINS, AUXILIARY BUILDING AND REACTOR BUILDING, FD-302-683, SHEET 3 OF 3

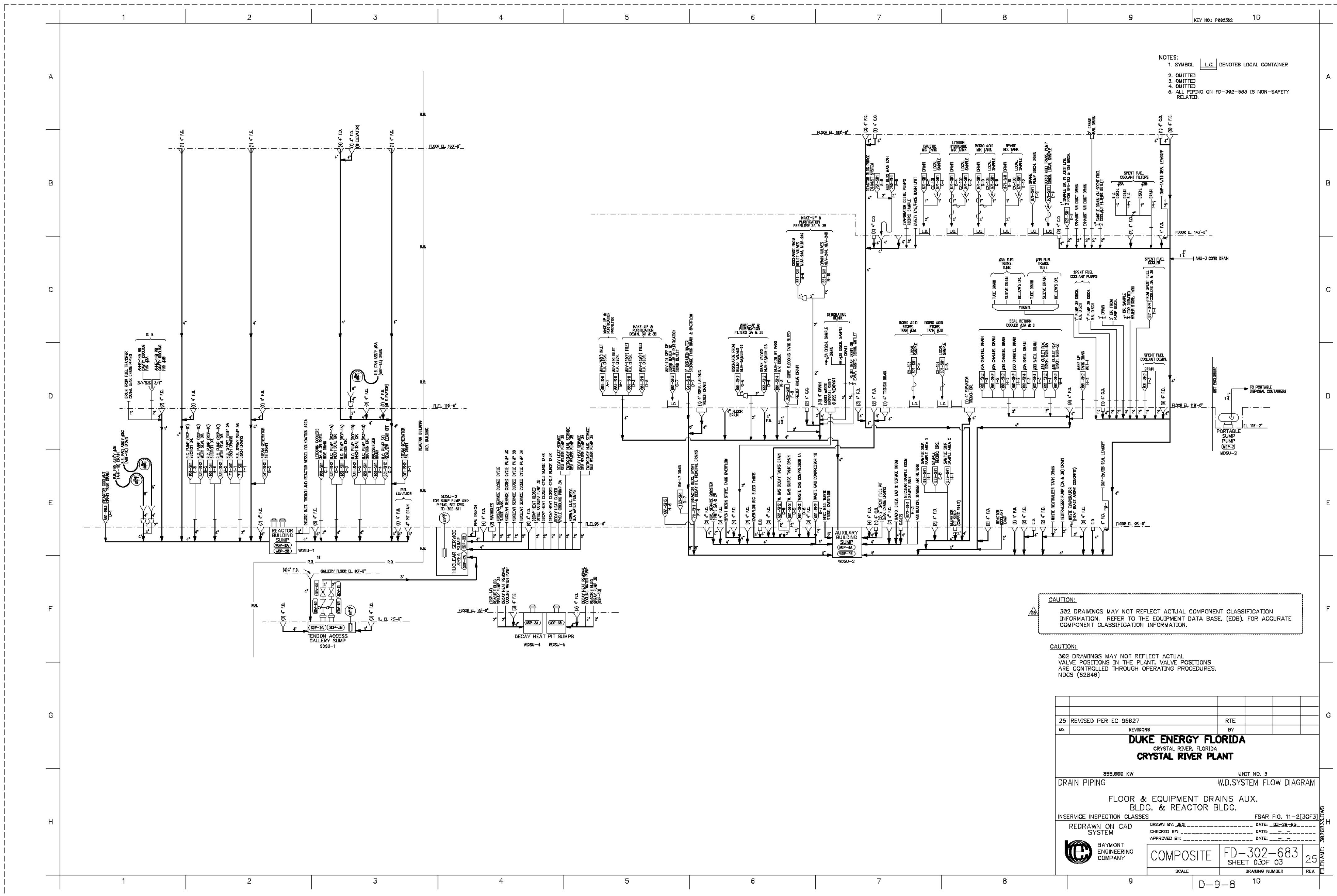


FIGURE 4-3A, RADIATION MONITORING SYSTEM BLOCK DIAGRAM (SHEET 1)

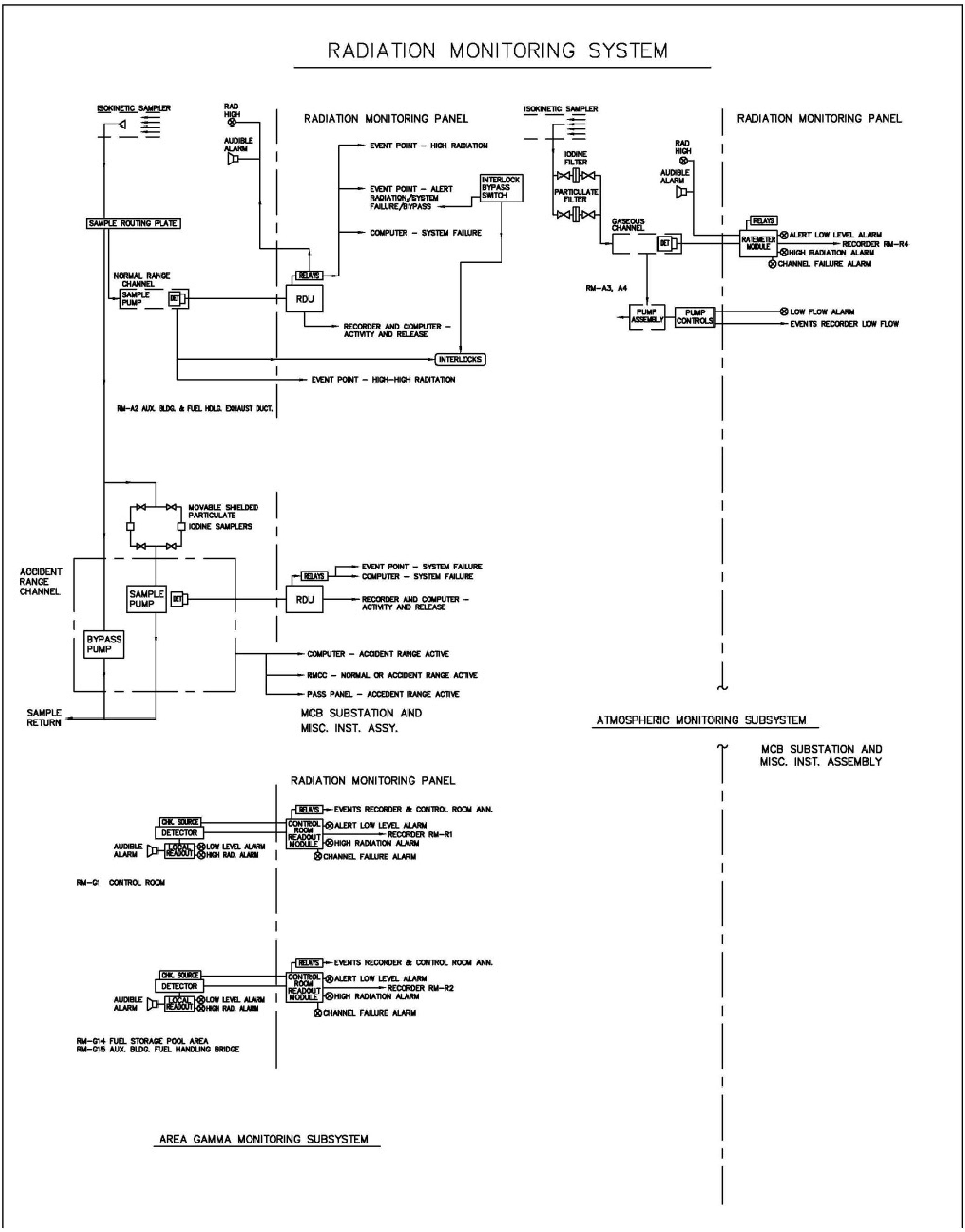


FIGURE 4-3B, RADIATION MONITORING SYSTEM BLOCK DIAGRAM (SHEET 2)

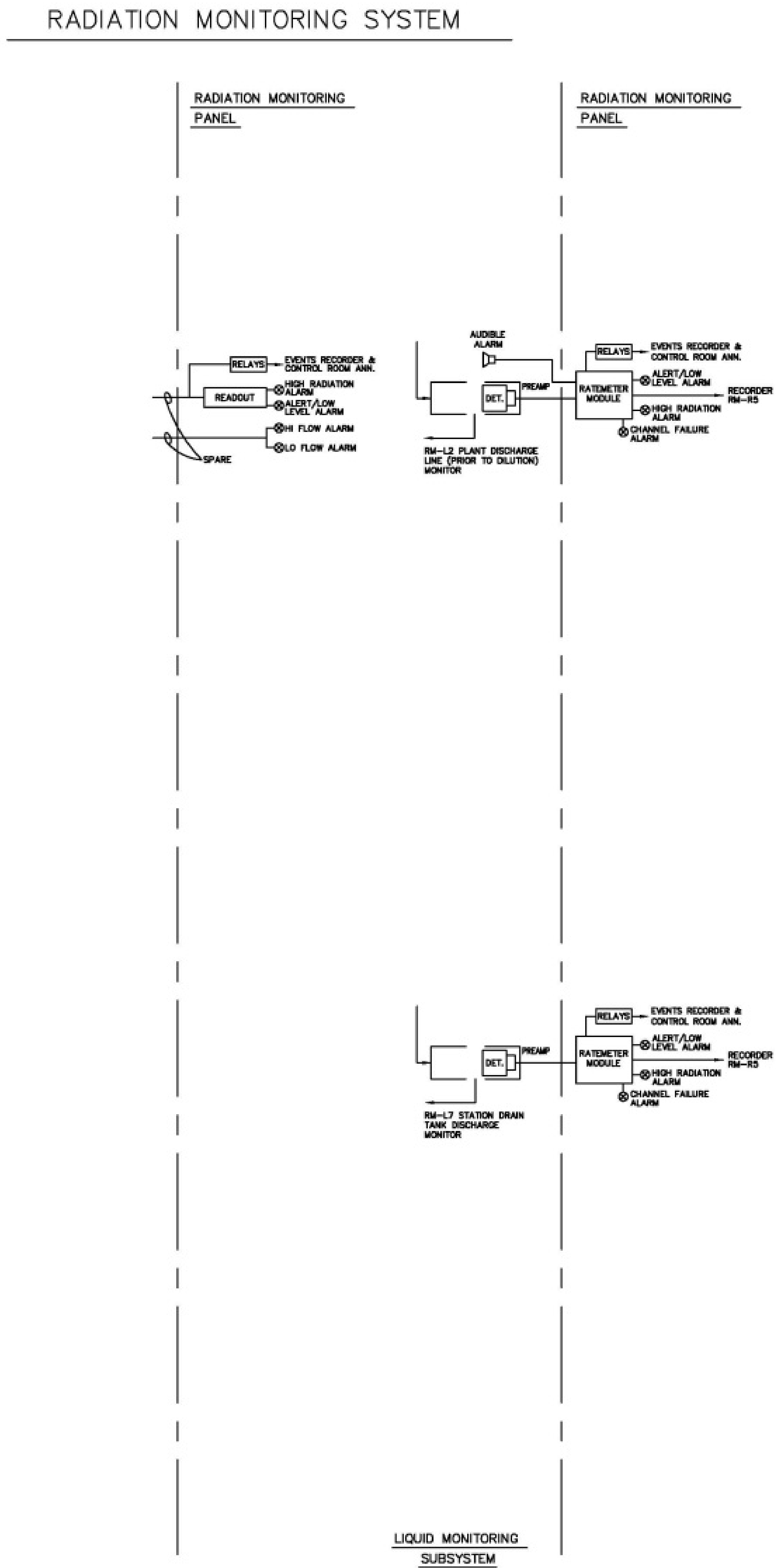


FIGURE 4-4, RADIATION MONITORING SYSTEM INTERLOCKS

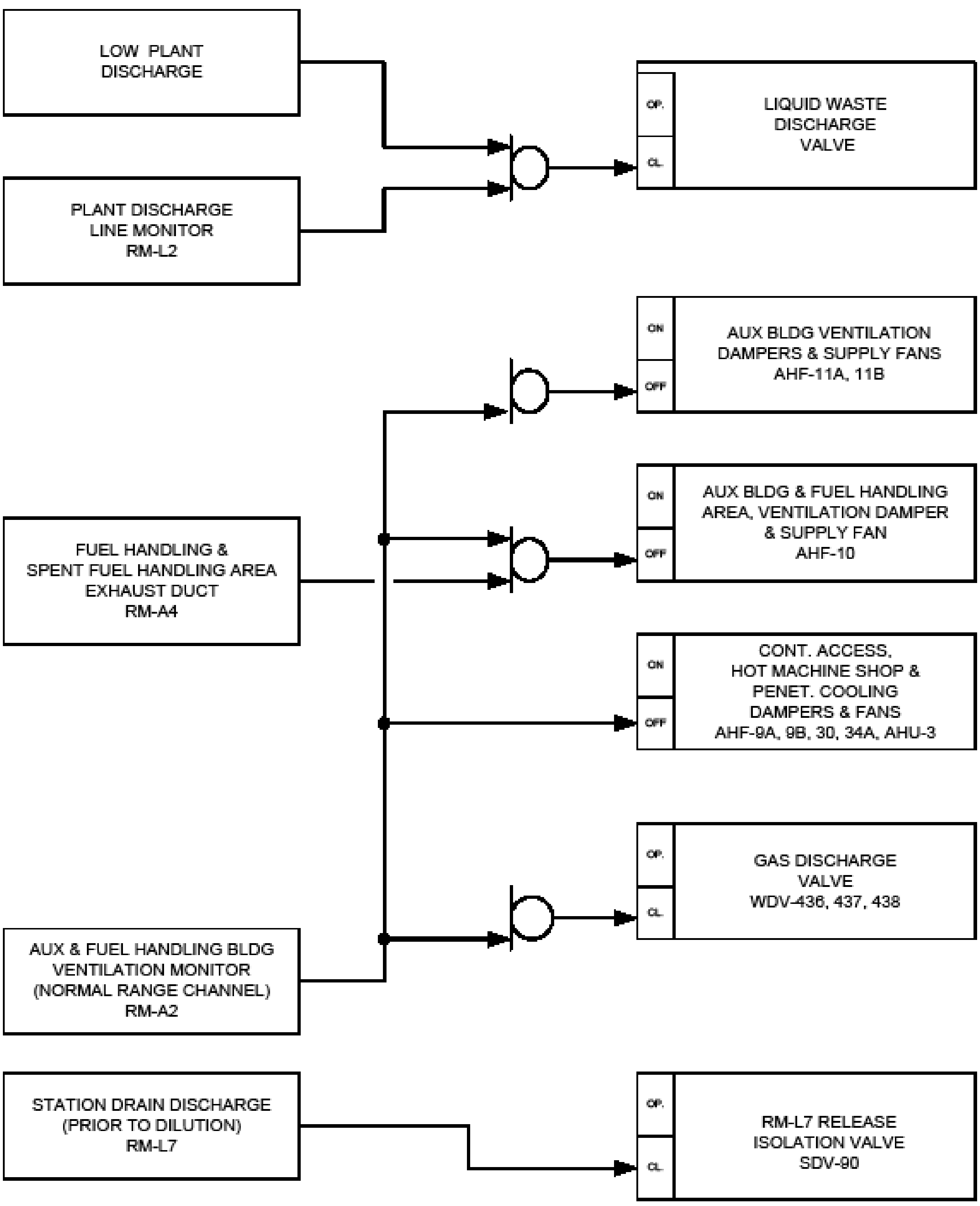


FIGURE 4-5, CROSS SECTION - REACTOR BUILDING AND AUXILIARY BUILDING RADIATION ZONES

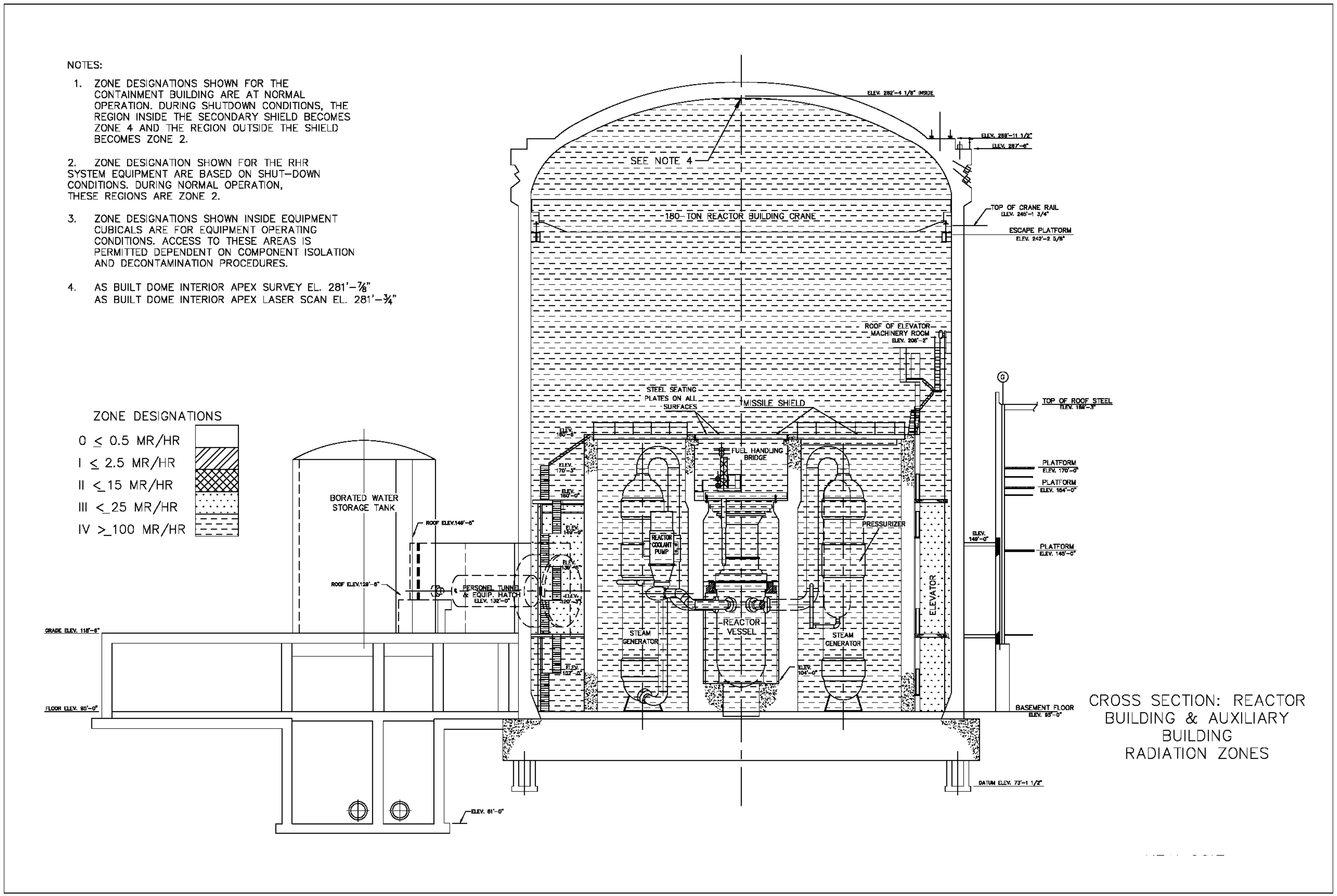


FIGURE 4-6, LONGITUDINAL SECTION - REACTOR BUILDING AND SPENT FUEL BUILDING RADIATION ZONES

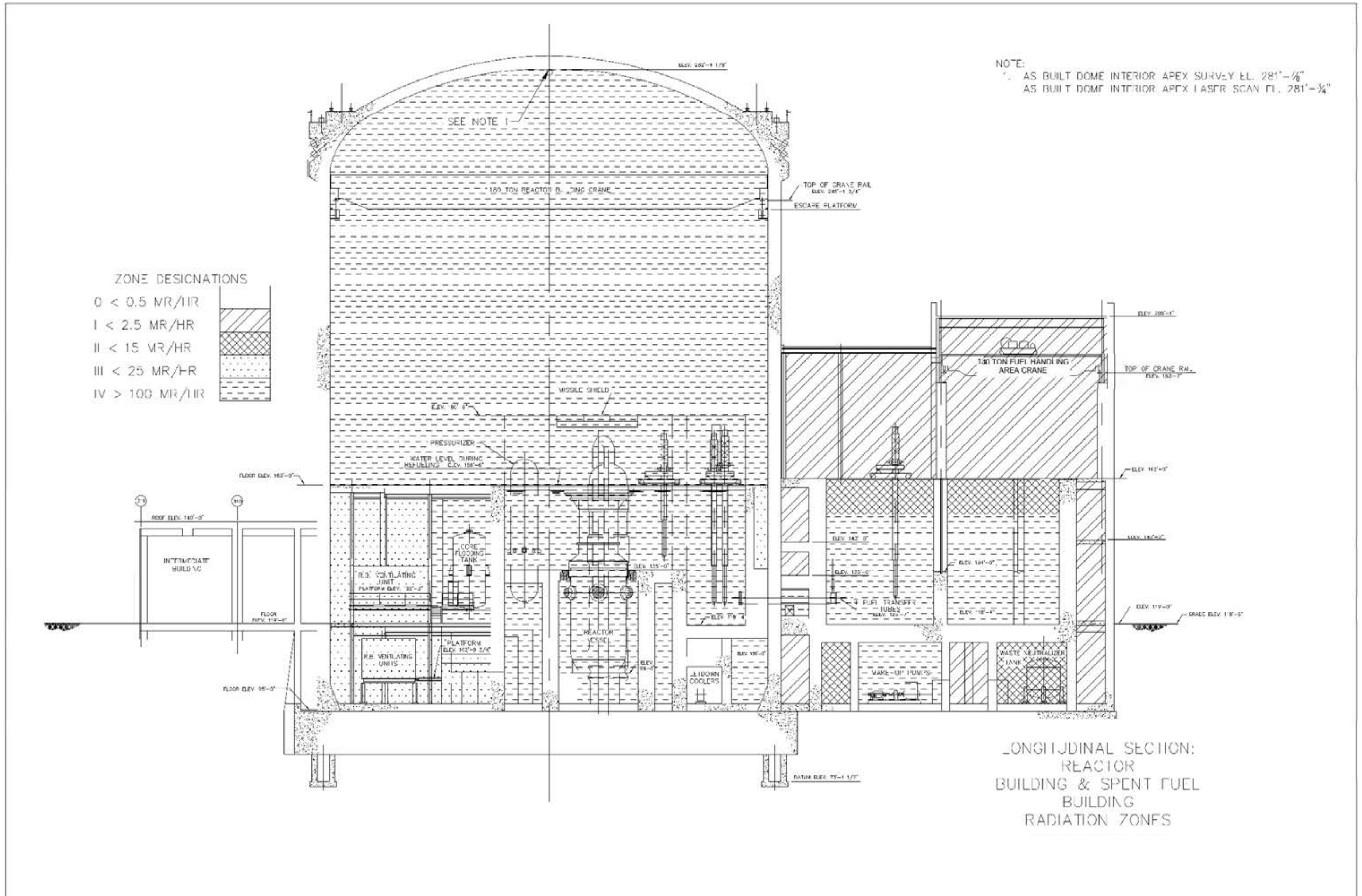


FIGURE 4-7, BASEMENT FLOOR ELEVATION 75'-0" RADIATION ZONES

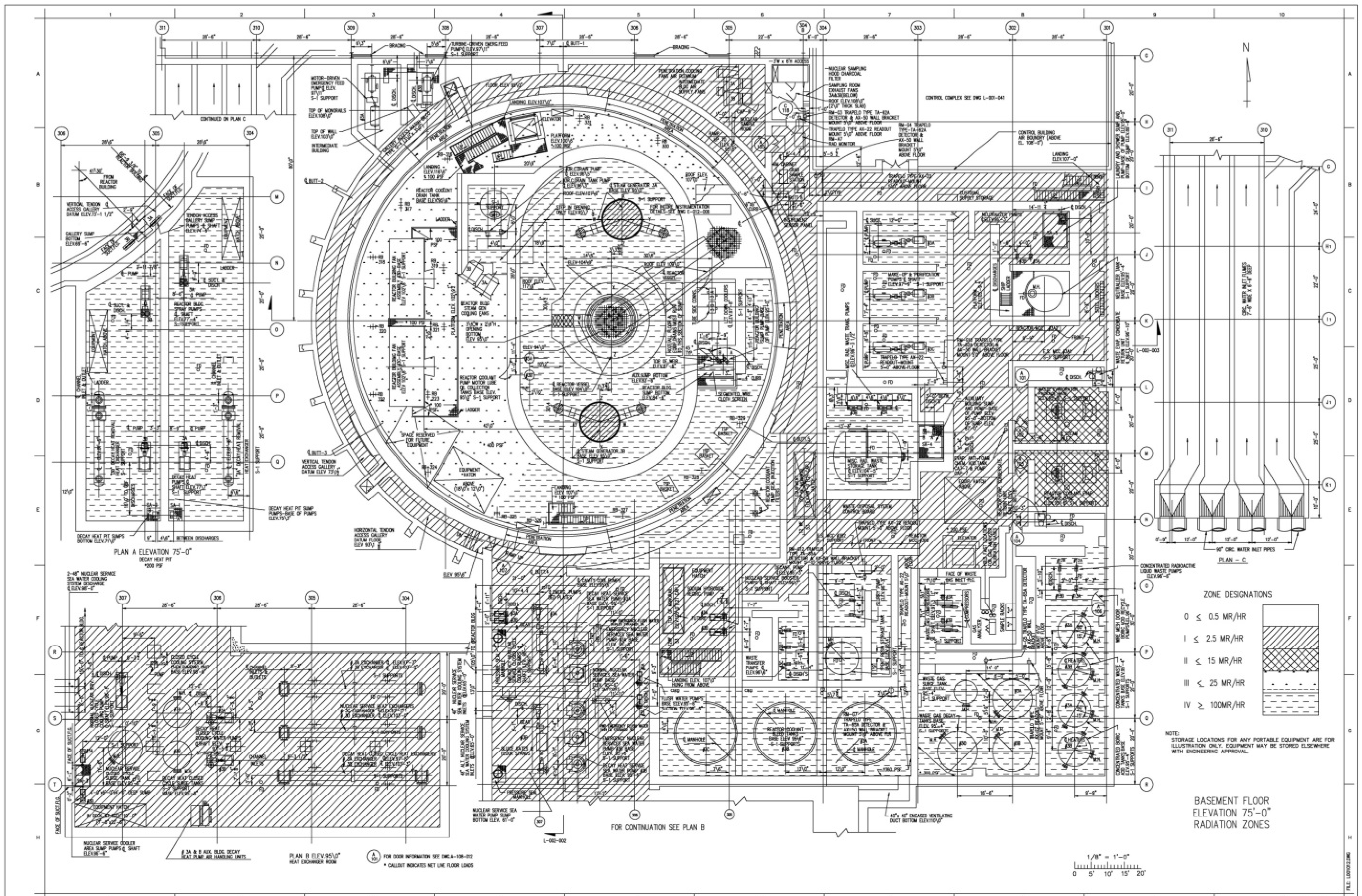


FIGURE 4-8, MEZZANINE FLOOR ELEVATION 119'-0" RADIATION ZONES

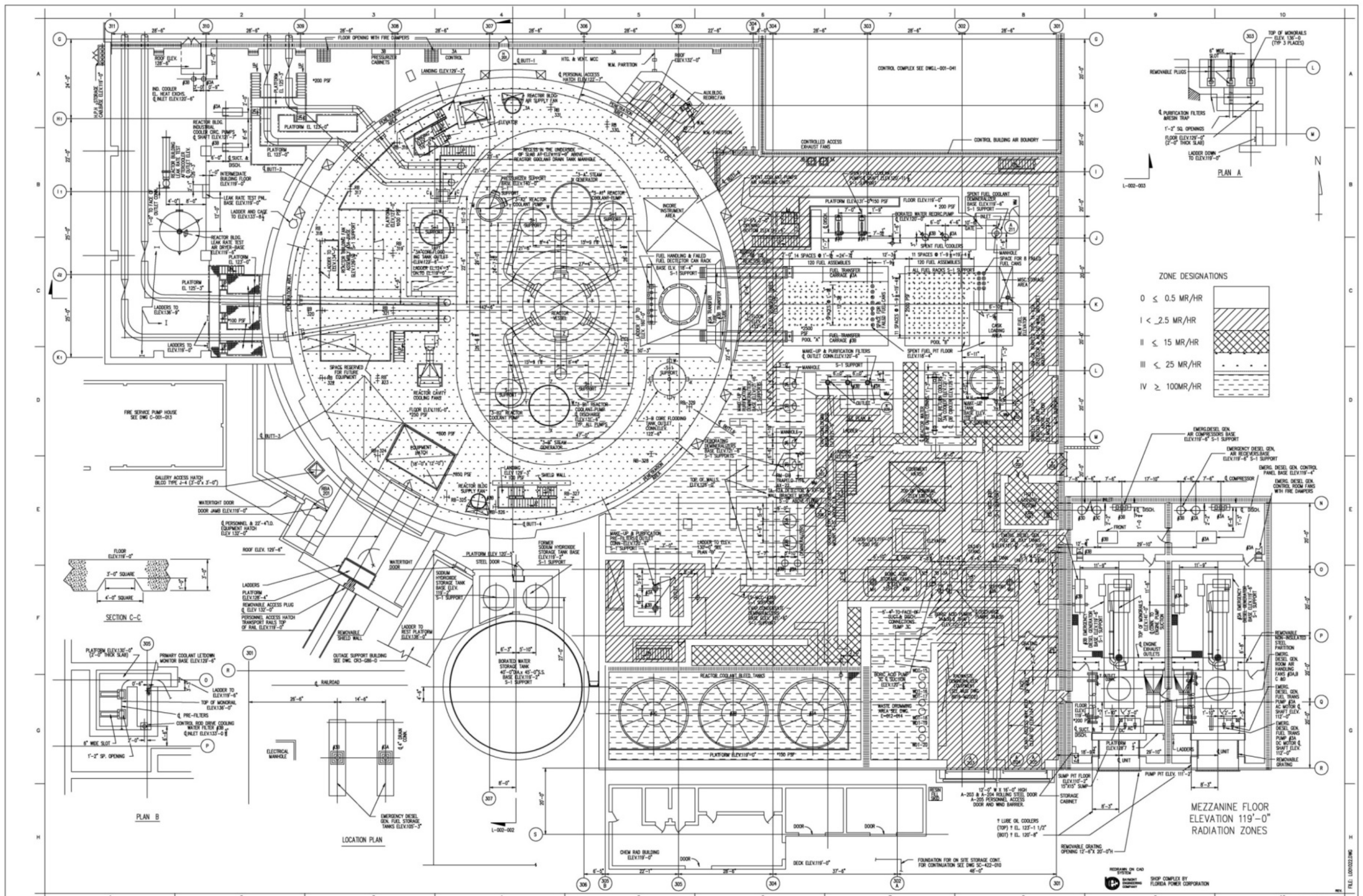
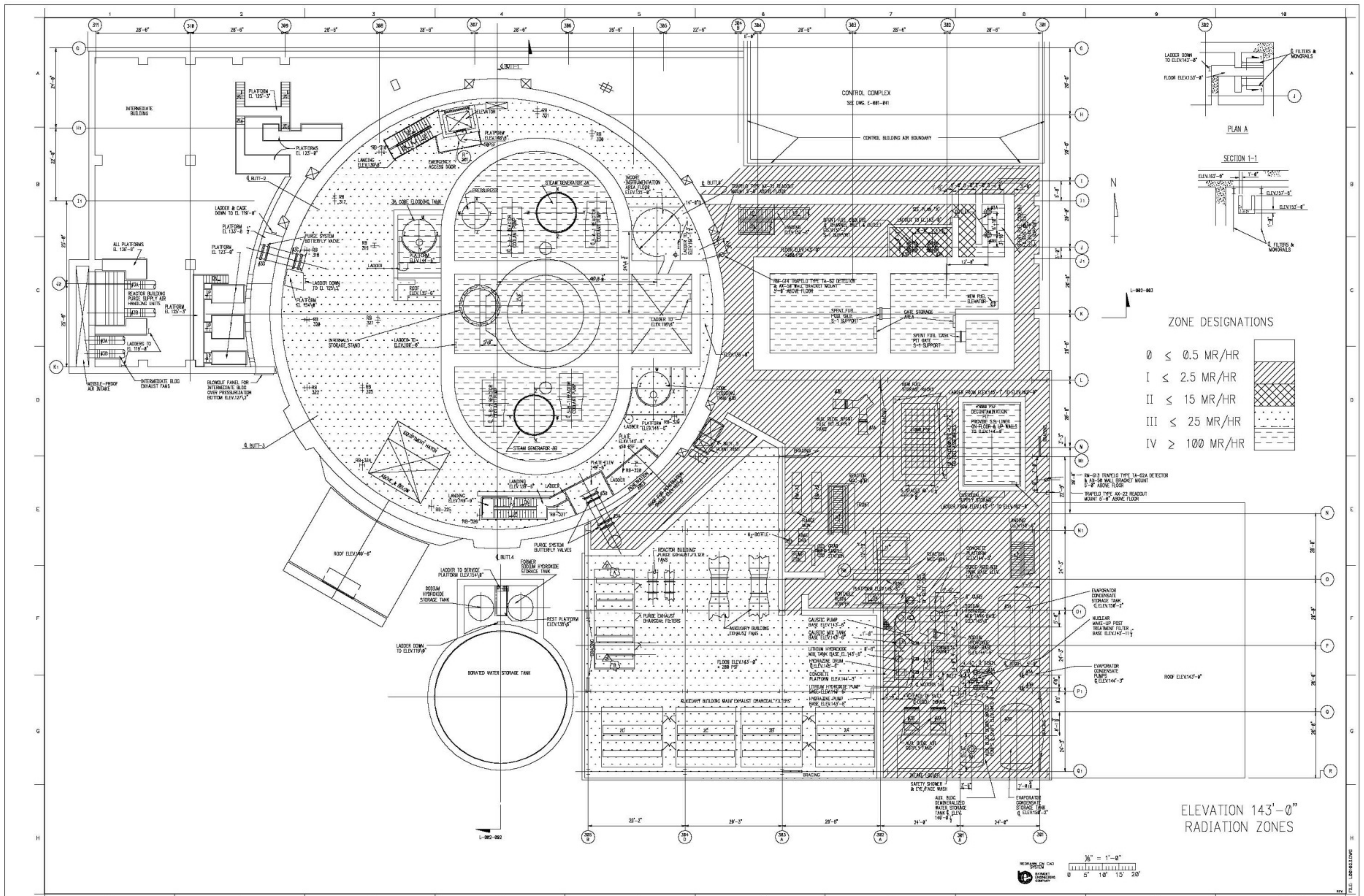


FIGURE 4-9, ELEVATION 143'-0" RADIATION ZONES



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5. SAFETY ANALYSIS

5.0 PLANT CHARACTERISTICS CONSIDERED IN SAFETY EVALUATION

5.0.1 GENERAL SAFETY ANALYSIS

Since CR3 has permanently ceased operation and all fuel is stored in the ISFSI, the safety analyses that assume operation of the reactor no longer apply.

5.1 STANDBY SAFEGUARDS ANALYSIS

5.1.1 SITUATION ANALYSES AND CAUSES

This Section presents an analysis of accidents in which one or more of the protective barriers are not effective and standby safeguards are required.

5.1.2 ACCIDENT ANALYSES - DELETED

END OF CHAPTER

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6. CONDUCT OF OPERATIONS

6.1 ORGANIZATION AND RESPONSIBILITY

6.1.1 FUNCTIONAL ORGANIZATION

The organization responsible for the Crystal River Unit 3 (CR3) plant is headed by the Senior Vice President, Operations Support. Additional description of the plant and corporate organization is provided in Section 1.7

6.1.2 OPERATING ORGANIZATION

The CR3 organization is discussed in Section 1.7.

6.1.3 QUALIFICATIONS

Crystal River Unit 3 personnel have the combination of education, skill, health, and experience commensurate with their level of responsibility. These qualifications provide reasonable assurance that decisions and actions during normal and abnormal conditions are such that the plant is maintained in a safe and efficient manner. The qualifications of plant managerial, supervisory, operating, technical support and technician personnel meet or exceed the requirements set forth in ANSI N18.1-1971 as endorsed by Regulatory Guide 1.8, Rev. 1 (9/75) and clarified by DSAR Chapter 1, Section 1.7, Table 1-3, "CR3 Quality Program Commitments" and the Permanently Defueled Technical Specification Section 5.3.1.

6.2 TRAINING

6.2.1 CONCEPT

Duke Energy Florida has implemented a comprehensive training program designed to indoctrinate personnel in the administrative and technical aspects of plant operations. A retraining and replacement training program for the facility staff will be maintained under the direction of the Senior Vice President, Operations Support and will meet or exceed the requirements and recommendations of Section 5.5 of ANSI N18.1-1971 and 10 CFR 50.120. The program includes both in-plant and out-of-plant personnel, and consists of academic and "hands-on" training developed to supplement an employee's general knowledge and experience. The overall conduct and administration of the training program of plant personnel is the responsibility of the General Manager Decommissioning. [TSA149]

6.2.2 PLANT TRAINING PROGRAMS

6.2.2.1 General Employee Training

All new employees will receive General Employee Indoctrination Training prior to being granted unescorted access to Crystal River Unit 3. The program is designed to familiarize new employees with Crystal River Unit 3 policies and commitments in the areas of Security, Administrative Instructions, Emergency Procedures, Industrial Safety, and Compliance.

Additional training will be provided to those employees whose jobs require work in the Radiation Controlled Area (RCA) of the Crystal River Nuclear Plant. This training will familiarize personnel with radiological procedures.

[TSA149] *This paragraph was added when Technical Specification Amendment 149 was issued. Obtain Licensing & Regulatory Affairs concurrence before changing.*

6.2.2.2 General Respiratory Training

All employees whose job at the Crystal River Nuclear Plant requires the use of respiratory equipment will receive Respiratory Protection Training.

6.2.2.3 Certified Fuel Handler Training - Deleted

6.2.2.4 Certified Fuel Handler Retraining - Deleted

6.2.2.5 Non-Certified Operator Training

This training program provides background training to Crystal River Unit 3 personnel who are actively and extensively engaged in monitoring and ensuring the safe storage of nuclear fuel.

6.2.2.6 Technical Training

All craftsmen will participate in a systems training program. The program will provide training to craft personnel in areas pertinent to their job functions.

6.2.2.7 Special Training

Special training will be provided as necessary to upgrade personnel in areas of deficiencies or areas of general interest appropriate for the day-to-day operation of the plant.

From time to time, special "one-time" courses may be required. One example of this type of training is the "Independent Spent Fuel Storage Installation Overview" training Crystal River Unit 3 presented to plant employees.

6.2.2.8 Team Training

This program provides initial and requalification training to Crystal River Unit 3 emergency teams as required to maintain a high level of proficiency in their job functions. Teams included as a part of this program are: Radiation Emergency Team, Environmental Survey Team, Medical Team, and Fire Brigade.

6.2.2.9 Contractor and Off-Site Employee Training

All contractor and off-site Duke Energy Florida personnel requiring unescorted access to CR3 will be required to participate in the General Employee Training Program.

General Respiratory Training will be provided under the guidance of Section 6.2.2.2, "General Respiratory Training."

6.2.3 RETRAINING

6.2.3.1 Chemistry and Health Physics Technicians Retraining

Retraining for Chemistry and Radiation Protection personnel will be conducted under the Technical Training Program (see Section 6.2.2.6). The program combines academic, hands-on training, and self-study.

6.2.3.2 Maintenance Section Retraining

Retraining for Mechanical Maintenance Personnel, Electrical Maintenance Personnel, and Instrumentation and Control Technicians will be conducted under the Technical Training Program (see Section 6.2.2.6). The program combines academic, hands-on training, and self-study.

6.2.3.3 Contractor Retraining

Retraining for contractor personnel will be conducted under the guidance of Section 6.2.2.9, "Contractor and Off-Site Employee Training."

6.2.4 EXAMINATIONS

Various types of examinations will be administered during the training of Crystal River Unit 3 personnel. A minimum score of 70% will be used as a cut-off for passing unless otherwise stated.

6.2.5 DOCUMENTATION

All training will be documented. Training documentation will be maintained by the Technical Support Manager.

6.3 INDUSTRIAL SECURITY

Duke Energy Florida has developed and submitted to the NRC a Physical Security Plan and a Safeguards Contingency Plan. Both documents, pursuant to 10 CFR 2.790(d), have been determined to contain proprietary and/or safeguards information and shall be withheld from public disclosure. For the latest revisions of the Plans, contact Crystal River Unit 3 Security or Licensing.

6.4 EMERGENCY PLAN

Duke Energy Florida has developed the ISFSI Only Emergency Plan to describe the elements of an integrated preparedness program to respond to potential emergencies at CR3. In the event of an emergency at CR3, actions are required to identify and assess the nature of the emergency and bring it under control in a manner that protects the health and safety of plant personnel.

The ISFSI Only Emergency Plan describes the organization and responsibilities of Duke Energy Florida and its CR3 and Corporate staffs for implementing emergency measures. It describes interfaces with Federal, State of Florida, and Citrus County organizations, to be notified in the event of an emergency and may provide, or be requested to provide, assistance.

CR3 is licensed under the requirements of 10 CFR 50, "Domestic Licensing of Production and Utilization Facilities," and 10 CFR 72, “Licensing Requirements For The Independent Storage of Spent Nuclear Fuel, High Level Radioactive Waste, and Reactor-Related Greater Than Class C Waste.” Consistent with the requirements of 10 CFR 50, this Plan is based upon the requirements of 10 CFR 50, Section 50.47(b) and Appendix E, "Emergency Planning and Preparedness for Production and Utilization Facilities," as applicable to CR3 in its permanently shutdown and defueled status. Sections 5.0 through 20.0 of this Plan address the standards delineated in 10 CFR 50.47(b)(1) through (16). In addition, the Plan is also intended to meet appropriate State of Florida and U.S. Nuclear Regulatory Commission (NRC) regulations in accordance with Duke Energy Florida’s Operating License (No. DPR-72).

Because the analyses of the credible design basis events and consequences indicates there are no postulated accidents that would result in off-site dose consequences that require off-site emergency planning, emergencies are divided into two classifications: 1) Notification of an Unusual Event (Unusual Event) and 2) Alert. This classification scheme has been discussed and agreed upon with responsible off-site organizations and is compatible with the State Plan.

Duke Energy Florida is responsible for planning and implementing emergency measures within the CR3 OWNER CONTROLLED AREA. This Plan is provided to meet that responsibility. To carry out specific emergency measures discussed in the Plan, detailed implementing procedures are established and maintained. The Plan provides a listing of the implementing procedures.

In addition to the description of activities and steps that can be implemented during a potential emergency, this Plan provides a general description of the steps taken to recover from an emergency situation. It also describes the training, exercises, planning, and coordination appropriate to maintain an adequate level of emergency preparedness.

6.5 FIRE PROTECTION PROGRAM

Refer to Section 3.9.

6.6 PLANT PROCEDURES

Plant procedures provide detailed written instructions designed to govern the normal and emergency conditions under which the plant is operated or might have to operate. These procedures are developed to meet the intent of Appendix A of Regulatory Guide 1.33, "Quality Assurance Requirements (Operation)," Revision 2, February 1978, and are in accordance with the operational Quality Assurance Program described in Section 1.7.

Plant procedures are written, reviewed, and implemented by the plant staff in accordance with the Quality Assurance Program described in Section 1.7. These procedures provide instructions for the safe operation, control, and maintenance of plant systems and equipment and furnish documentation of actions taken.

The following procedure categories are to be included in the Plant Operating Manual (POM):

a. Administrative Instructions

   Administrative Instructions cover the following topics: administrative policies, plant organization and responsibilities, conduct of plant operations, conduct of maintenance, conduct of training, and housekeeping.

b. Operating Procedures

   Operating Procedures are designed to provide instructions for normal operations and include: electrical system operation, instrumentation and control system operation, spent fuel cooling, and waste disposal.

c. Emergency Plans and Implementing Procedures

   These procedures provide instructions and outline responsibilities of on-site and off-site personnel in the event of a site emergency. The Emergency Plan, fire protection, emergency classification, evacuation and notification, medical emergencies, and violent weather documents are included in this category.

d. Abnormal Procedures

   Abnormal Procedures are operating procedures designed to control plant transients and abnormal conditions before they develop further. Events considered as abnormal conditions are: fire, earthquake, loss of spent fuel cooling, and spent fuel pool level lowering.

e. Maintenance Procedures

Maintenance Procedures provide written instructions on the maintenance, repair, and replacement of equipment and components.

f. Fueling Procedures

Fueling Procedures cover all phases of fuel handling operations including the following: spent fuel shipment, failed fuel identification and handling, neutron source handling, and fuel handling bridge operations.

g. Waste Handling, Chemistry, and Radiation Protection Procedures

These procedures provide written instructions governing the surveillance, scheduling, and control of waste handling, chemistry, and radiological protection activities.

h. Compliance Procedures

Compliance Procedures provide a means to ensure plant commitments and requirements are met. Activities controlled by these procedures include: processing of Licensing correspondence, commitment management, administration of some regulatory required programs, and reporting requirements.

i. Surveillance Procedures

Surveillance Procedures include those tests, checks, calculations, calibrations, inspections, and reports required to be performed on a periodic basis. These include all phases of plant operation and plant safety to ensure all systems/components are functional and/or available.

j. Special Nuclear Materials Handling and Accountability Manual

This document describes the procedures to be followed at Crystal River Unit 3 to meet the requirements of 10 CFR 70, governing the handling and accountability of special nuclear material.

k. Security Procedures

Security Procedures encompass the entire spectrum of actions, planned or implemented, to maintain site security. Included as a part of these procedures are the following areas: guard force organization, operation and responsibilities, alarm systems, personnel and materials access, communications, security equipment, contingency actions, emergency actions, loss or breach of security, and security training.

l. Preventive Maintenance Procedures

Preventive Maintenance Procedures are written to ensure proper maintenance is provided for equipment/components to prevent the unscheduled outage of such equipment due to sudden failure.

m. Quality Control Procedures

Quality Control Procedures include those procedures designed to control materials, equipment, services, drawings, documents, instructions, and special processes to support plant operations. These procedures also include the training requirements of Quality Assurance and Quality Control inspection and compliance personnel.

n. Annunciator Response Procedures

Annunciator Response Procedures provide the main control room operators with a reference document for each annunciator window on a particular lampbox. The procedures establish operator actions for valid annunciator alarms on the lampbox and provide a reference to other procedures to address operator actions for valid annunciator alarms.

o. Nuclear Fleet or equivalent Standard Procedures

These procedures are typically applicable to more than one department and are developed when a single consistent process is advantageous to the Nuclear Fleet. Nuclear Fleet or equivalent Standard Procedures detail administrative requirements, responsibilities, activities and actions required to implement or comply with regulatory requirements, commitments and Nuclear Fleet Directives.

p. Other procedures may be used which provide detailed written instructions for the performance of processes not governed by any of the aforementioned procedures. These procedures provide a preplanned method of conducting activities in order to reduce errors. These procedures may include instructions for temporary procedures, work controls, information technology, and other functions, and are used by the plant staff to ensure that normal operations are conducted in a safe manner.

6.7 RECORDS

6.7.1 OPERATING RECORDS

The following records of operation will be kept. These will be preserved as required.

6.7.1.1 Main Control Room Log

The Main Control Room log is maintained on a shift basis to record the plant status and events in chronological order. It contains information relative to plant operation. At the end of each shift the Non-Certified Operator signs the Main Control Room log, signifying the entries are a complete and accurate record of plant operations.

6.7.1.2 Shift Supervisor's Log

The Shift Supervisor summarizes plant conditions and events during his shift in the Shift Supervisor's log. The Shift Supervisor's log shall begin with plant status information and should include any changes in status of the availability of systems, unusual occurrences, etc. At the end of each shift, the Shift Supervisor signs the log, signifying the report is a comprehensive, accurate summary of plant events and activities.

6.7.2 ADMINISTRATIVE RECORDS

The following is the responsibility of the General Manager Decommissioning:

Investigation and reporting of abnormalities and corrective action taken including, but not limited to, the following:

a. Personnel overexposure
b. Loss or theft of licensed radioactive material

Corrective action will be taken immediately.

6.7.3 MAINTENANCE RECORDS

The Maintenance Section Shop Supervisors are responsible for the maintenance records for plant equipment and instrumentation.

6.7.4 HEALTH PHYSICS RECORDS

The Manager Radiation Protection and Chemistry maintains the following Health Physics records:

6.7.4.1 Personnel Exposure

The Supervisor, Nuclear Radiation Protection maintains the following:

a. Dosimetry records (monthly or more frequently)
b. Radio bioassay records (as deemed necessary)
c. Records of radiation exposure history and current exposure status (as required by 10 CFR 20)

6.7.4.2 Radiological Surveys

The Supervisor, Nuclear Radiation Protection is responsible for routine radiological surveys and job-specific radiological surveys.

6.7.4.3 Survey Instrument Calibration

The Supervisor, Nuclear Radiation Protection maintains all HP survey meters in accordance with 10 CFR 20.2103.

6.7.4.4 Radiological Monitoring and Waste Disposal

The Manager Radiation Protection and Chemistry is responsible for the following:

a. Liquid waste discharged
b. Gaseous activity released
c. Monitoring reports

6.7.4.5 Water Quality

The Manager Radiation Protection and Chemistry is responsible for Water Quality records.

6.7.4.6 Instrumentation Calibration

The Manager Radiation Protection and Chemistry maintains all environmental monitors.

6.7.4.7 Shipping, Receiving and Inventory of Radioactive Material

The Manager Radiation Protection and Chemistry is responsible for the following:

a. Records in sufficient detail to satisfy the appropriate sections of 10 CFR 20, 10 CFR 70, 10 CFR 71 and 49 CFR 173.
b. Solid radioactive waste shipped.

6.7.5 OTHER RECORDS

6.7.5.1 Special Nuclear Material Inventory

The Site SNM Custodian is responsible for records in sufficient detail to satisfy the requirements of 10 CFR 70.

6.8 ADMINISTRATIVE CONTROL

Administrative controls are established to ensure plant operations, maintenance tests, and emergency responses are performed in accordance with reviewed and approved procedures. The General Manager Decommissioning has the responsibility and authority to operate the plant within the limits of the administrative controls.

A review of the operating logs, charts, and other data is performed by Operations, Engineering, and other personnel.

Non-routine operations are closely reviewed. In addition to these reviews, periodic plant staff meetings are held to keep operating personnel advised of current plant conditions.

6.8.1 PLANT NUCLEAR SAFETY COMMITTEE (PNSC)

A Plant Nuclear Safety Review Committee (PNSC) is established to review activities in accordance with Section 6.8.1.5. The PNSC recommends to the General Manager Decommissioning approval or disapproval of the activities reviewed in Section 6.8.1.5.

6.8.1.1 Composition

The PNSC is composed of a minimum of four members and one chairman from CR3 personnel with cumulative expertise in the following areas: operations, health physics, chemistry, maintenance, modifications, engineering, nuclear safety and licensing. These positions will be designated by the General Manager Decommissioning in administrative procedures or policies. The PNSC members and alternate members shall, as a minimum, meet equivalent qualification criteria as specified in Section 4.3 of ANSI N18.1 – 1971.

6.8.1.2 Alternates

Alternate members are appointed in writing by the PNSC Chairman.

6.8.1.3 Meeting Frequency

The PNSC meets on an as-needed basis as convened by the PNSC Chairman or his designated alternate.

6.8.1.4 Quorum

A quorum of the PNSC necessary for the performance of the PNSC responsibility and authority provisions of this DSAR shall consist of the Chairman or his designated alternate and three members, including alternates.

6.8.1.5 Responsibilities

a. Proposed changes to the facility as described in the DSAR. This review is to confirm that the change does not adversely affect safety and if a Permanently Defueled Technical Specification change or NRC review is required.

b. Proposed changes to procedures as described in the DSAR and tests or experiments not described in the DSAR. This review is to confirm that the change does not adversely affect safety and if a Permanently Defueled Technical Specification change or NRC review is required.

c. Proposed Permanently Defueled Technical Specification changes and other license amendments, except in those cases where the change is identical to a previously reviewed change.

d. Review of all proposed changes to the Technical Specification Bases that have been initially determined to require a license amendment.

e. License Event Reports that are required to be submitted to the NRC. This review includes results of any investigations made and recommendations resulting from such investigations to prevent or reduce the probability of recurrence of the event.

f. Investigation of all violations of the Technical Specifications, including the preparation and forwarding of reports covering evaluation and recommendations to prevent recurrence, to the General Manager Decommissioning.

g. Review of unit operations to detect potential hazards to spent nuclear fuel safety.

h. Review proposed changes to the Security Plan that have been initially determined to decrease the effectiveness of the plan.

i. Review proposed changes to the Permanently Defueled Emergency Plan that have been initially determined to decrease the effectiveness of the plan.

j. Review of any accidental, unplanned or uncontrolled radioactive release, including the preparation of reports covering evaluation, recommendations and disposition of the corrective action to prevent recurrence, and the forwarding of these reports to the General Manager Decommissioning.

k. Review, prior to implementation, changes to the Offsite Dose Calculation Manual (ODCM).

6.8.1.6 Requirements

Render determinations in writing with regard to whether or not each item considered under Section 6.8.1.5 complies with the facility license and governing regulations.

6.8.1.7 Records

The PNSC shall maintain written minutes of each PNSC meeting which, at a minimum, document the results of all PNSC activities performed under the responsibility provisions of this DSAR. Copies shall be provided to the General Manager Decommissioning.

END-OF-CHAPTER