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3.1 CONFORMANCE WITH NRC GENERAL DESIGN CRITERIA

3.1.1 SUMMARY DESCRIPTION

This section contains an evaluation of the design bases of the Energy Northwest Nuclear Project No. 2 (Columbia Generating Station (CGS)) nuclear generating station as compared to the NRC General Design Criteria (GDC) for Nuclear Power Plants, Appendix A of 10 CFR Part 50, effective May 21, 1971, and subsequently amended on July 7, 1971. The GDC, which are divided into six groups and total 55 in number, are intended to establish minimum requirements for the design of nuclear power plants.

The GDC were not written specifically for the boiling water reactor (BWR); rather, they were intended to guide the design of all water-cooled nuclear power plants. As a result, the criteria are generic in nature and subject to interpretation. For this reason, there are some cases where conformance to a particular criterion is not directly comparable. In these cases, the conformance of plant design to the interpretation of the criterion is discussed. For each of the 55 criteria, a specific assessment of the plant design is made, and a complete list of section references are included to identify where detailed design information pertinent to each criterion is discussed.

Based on the content herein, Energy Northwest concludes that CGS is in compliance with the GDC.

3.1.2 CRITERION CONFORMANCE

3.1.2.1 Group I - Overall Requirements

3.1.2.1.1 Criterion 1 - Quality Standards and Records

> Structures, systems, and components important to safety shall be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed. Where generally recognized codes and standards are used, they shall be identified and evaluated to determine their applicability, adequacy, and sufficiency and shall be supplemented or modified as necessary to assure a quality product in keeping with the required safety function. A quality assurance program shall be established and implemented in order to provide adequate assurance that these structures, systems, and components will satisfactorily perform their safety

functions. Appropriate records of the design, fabrication, erection, and testing of structures, systems, and components important to safety shall be maintained by, or under the control of, the nuclear power unit licensee throughout the life of the unit (GDC 1).

Evaluation Against Criterion 1

Structures, systems, and components important to safety are identified in Section 3.2. The quality assurance program used during the operations phase is described in EN-QA-004, Energy Northwest Operational Quality Assurance Program Description. The quality assurance program used during the design and construction of the plant was provided at the Preliminary Safety Analysis Report (PSAR) stage and has been applied to the items contained in Table 3.2-1. The intent of the quality assurance program is to ensure sound engineering in all phases of design and construction through conformity to regulatory requirements and design bases described in the license application. In addition, the program ensures adherence to specified standards of workmanship and implementation of recognized codes and standards in fabrication and construction. It also includes the observance of proper preoperational and operational testing and maintenance procedures as well as keeping appropriate records. The total quality assurance program of Energy Northwest and its principal contractors is responsive to and satisfies the quality-related requirements of 10 CFR Part 50, including Appendix B.

Structures, systems, and components are first classified in Section 3.2 with respect to their location and service and their relationship to the safety function to be performed. Recognized codes and standards are applied to the equipment to ensure a quality product in keeping with the required safety function. In cases where codes are not available or the existing code must be modified, an explanation is provided.

Documents are maintained which demonstrate that the requirements of the quality assurance program are being satisfied. This documentation shows that appropriate codes, standards, and regulatory requirements are observed, specified materials are used, correct procedures are utilized, qualified personnel are provided, and that the finished parts and components meet the applicable specifications for safe and reliable operation. These records are available so that any desired item of information is retrievable for reference. These records will be maintained in accordance with the Operational Quality Assurance Program Description.

The detailed quality assurance program developed by the applicant and its contractors satisfies the requirements of Criterion 1.

For further discussion see the following sections:

a. Principal design criteria 1.2

b. Plant description 1.2

c. Classification of structures, components, and systems 3.2

d. Quality assurance 17

3.1.2.1.2 Criterion 2 - Design Bases for Protection Against Natural Phenomena

Structures, systems, and components important to safety shall be designed to withstand the effects of natural phenomena such as earthquakes, tornadoes, hurricanes, floods, tsunami, and seiches without loss of capability to perform their safety functions. The design bases for these structures, systems, and components shall reflect: (1) appropriate consideration of the most severe of natural phenomena that have been historically reported for the site and surrounding area, with sufficient margin for the limited accuracy, quantity, and period of time in which the historical data have been accumulated, (2) appropriate combinations of the effects of normal and accident conditions with the effects of the natural phenomena, and (3) the importance of the safety functions to be performed (GDC 2).

Evaluation Against Criterion 2

The design criteria adopted for structures, systems, and components considers the magnitude and the probability of occurrence of natural phenomena at the specific site. The designs are based on the most severe natural phenomena recorded for the site with an appropriate margin to account for uncertainties in the historical data. Detailed discussion of the various phenomena considered and the design criteria developed are presented in the sections listed below.

The design criteria developed meet the requirements of Criterion 2.

For further discussion, see the following sections:

a. Meteorology 2.3

b. Hydrologic engineering 2.4

c. Geology and seismology 2.5

d. Classification of structures, components, and systems 3.2

e. Wind and tornado design loadings 3.3

f. Water level (flood) design 3.4

g.	Missile protection	3.5
h.	Seismic design	3.7
i.	Design of Seismic Category I structures	3.8
j.	Mechanical systems and components	3.9
k.	Seismic qualification of Category I instrumentation and electrical equipment	3.10
l.	Environmental design of mechanical and electrical equipment	3.11

3.1.2.1.3 Criterion 3 - Fire Protection

> Structures, systems, and components important to safety shall be designed and located to minimize, consistent with other safety requirements, the probability and effect of fires and explosions. Noncombustible and heat resistant materials shall be used wherever practical throughout the unit, particularly in locations such as the containment and control room. Fire detection and fighting systems of appropriate capacity and capability shall be provided and designed to minimize the adverse effect of fires on structures, systems, and components important to safety. Firefighting systems shall be designed to assure that their rupture or inadvertent operation does not significantly impair the safety capability of these structures, systems, and components (GDC 3).

Evaluation Against Criterion 3

Insurer and National Fire Protection Association guidelines were used for the design of the plant.

Noncombustible and fire resistant materials are used wherever practical throughout the facility, particularly in areas containing critical portions of the plant such as containment structure, control room, and components of systems important to safety. These systems are designed and located to minimize the effects of fires or explosions on their redundant components. Facilities for the storage of combustible materials such as fuel oil are located, designed, and protected to minimize both the probability and the effects of a fire.

Equipment and facilities for detecting, annunciating, and extinguishing fires are provided to protect both plant and personnel from fire or explosion.

Administrative controls are utilized where applicable throughout the facility to minimize the probability and consequences of fires and explosions.

The fire protection system is designed such that a failure of any component of the system will not

a. Generate an accident resulting in significant release of radioactivity to the environment, or

b. Impair the ability of redundant equipment to safely shut down and isolate the reactor or limit the release of radioactivity to the environment in the event of a loss-of-coolant accident (LOCA).

For further discussion, see the following sections:

a.	Design of Seismic Category I structures	3.8
b.	Fire protection system	9.5.1, Appendix F

3.1.2.1.4 Criterion 4 - Environmental and Missile Design Bases

> Structures, systems, and components important to safety shall be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, including loss-of-coolant accidents. These structures, systems, and components shall be appropriately protected against dynamic effects, including the effects of missiles, pipe whipping, and discharging fluids, that may result from equipment failures and from events and conditions outside the nuclear power unit (GDC 4).

Evaluation Against Criterion 4

Structures, systems, and components important to safety are designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, including LOCAs. (See Section 3.11.)

These structures, systems, and components are appropriately protected against dynamic effects including the effects of missiles, pipe whipping, and discharging fluids that may result from equipment failures.

The electrical equipment, instrumentation, and associated cables of protection and engineered safety features (ESF) system which are located inside the containment are discussed in the sections listed below. The design requirements in terms of the time that each must survive the extreme environmental conditions following a LOCA are indicated.

The design of these structures, systems, and components meets the requirements of Criterion 4.

For further discussion, see the following sections:

a.	Classification of structures, components, and systems	3.2
b.	Missile protection	3.5
c.	Protection against dynamic effects associated with the postulated rupture of piping	3.6
d.	Design of Seismic Category I structures	3.8
e.	Mechanical system and components	3.9
f.	Environmental design of mechanical and electrical equipment	3.11
g.	Integrity of reactor coolant pressure boundary (RCPB)	5.2
h.	ESF	6
i.	Instrumentation and controls	7
j.	Electric power	8

3.1.2.1.5 Criterion 5 - Sharing of Structures, Systems, and Components

> Structures, systems, and components important to safety shall not be shared among nuclear power units unless it can be shown that such sharing will not significantly impair their ability to perform their safety functions, including, in the event of an accident in one unit, an orderly shutdown and cooldown of the remaining units (GDC 5).

Evaluation Against Criterion 5

CGS is a single unit plant, and therefore Criterion 5 does not apply.

3.1.2.2 Group II - Protection by Multiple Fission Product Barriers

3.1.2.2.1 Criterion 10 - Reactor Design

> The reactor core and associated coolant, control, and protection systems shall be designed with appropriate margin to assure that specified acceptable fuel design limits are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences (GDC 10).

Evaluation Against Criterion 10

The reactor core components consist of fuel assemblies, control rods, in-core ion chambers, neutron sources, and related items. The mechanical design is based on conservative application of stress limits, operating experience, and experimental test results. The fuel is designed to maintain integrity over a complete range of power levels including transient conditions. The core is sized with sufficient heat transfer area and coolant flow to ensure that fuel design limits are not exceeded under normal conditions or anticipated operational occurrences.

The reactor protection system (RPS) is designed to monitor certain reactor parameters, sense abnormalities, and to scram the reactor thereby preventing fuel design limits from being exceeded. Scram trip setpoints are selected on operating experience and by the safety design basis. There is no case in which the scram trip setpoints allow the core to exceed the thermal hydraulic safety limits. Power for the RPS is supplied by two independent ride-through ac power supplies. An alternate power source is available for each bus.

An analysis and evaluation has been made of the effects on core fuel following adverse plant operating conditions. The results of abnormal operational transients are presented in Chapter 15 and show that the minimum critical power ratio (MCPR) does not fall below the transient MCPR limit, thereby satisfying the transient design basis.

The reactor core and associated coolant, control, and protection systems are designed to ensure that the specified fuel design limits are not exceeded during the conditions of normal or abnormal operation and therefore meet the requirements of Criterion 10.

For further discussion, see the following sections:

a. Principal design criteria 1.2.1

b. Plant description 1.2.2

c. Fuel system design 4.2

d.	Nuclear design	4.3
e.	Thermal and hydraulic design	4.4
f.	Reactor materials	4.5
g.	Functional design of reactivity control systems	4.6
h.	Reactor coolant system and connected systems	5
i.	Reactor trip system	7.2
j.	Accident analyses	15

3.1.2.2.2 Criterion 11 - Reactor Inherent Protection

The reactor core and associated coolant systems shall be designed so that in the power operating range the net effect of the prompt inherent nuclear feedback characteristics tends to compensate for a rapid increase in reactivity (GDC 11).

Evaluation Against Criterion 11

The reactor core is designed to have a reactivity response that regulates or damps changes in power level and spatial distributions of power production to a level consistent with safe and efficient operation.

The inherent dynamic behavior of the core is characterized in terms of the fuel temperature or doppler coefficient, the moderator void coefficient, and the moderator temperature coefficient. The combined effect of these coefficients in the power range is termed the power coefficient.

Doppler reactivity feedback occurs simultaneously with a change in fuel temperature and opposes the power change that caused it; this contributes to system stability. Since the doppler reactivity opposes load changes, it is desirable to maintain a large ratio of moderator void coefficient to doppler coefficient for optimum load-following capability. The BWR has an inherently large moderator-to-doppler coefficient ratio which permits the use of the coolant flow rate for load following.

In a BWR, the moderator void coefficient is of importance during operation at power. Nuclear design requires the void coefficient inside the fuel channel to be negative. The negative void reactivity coefficient provides an inherent negative feedback during power transients. Because of the large negative moderator coefficient of reactivity, the BWR has a number of inherent advantages, such as the use of coolant flow as opposed to control rods for load following, inherent self-flattening of the radial power distribution, ease of control, and spatial xenon stability.

The reactor is designed so that the moderator temperature coefficient is small and positive in the cold condition; however, the overall power reactivity coefficient is negative. Typically, the power coefficient at full power is about -0.04 $\Delta k/k/\Delta P/P$ at the beginning of life and about -0.3 $\Delta k/k/\Delta P/P$ at 10,000 MWd/t. These values are well within the range required for adequate damping of power and spatial xenon disturbances.

The reactor core and associated coolant system are designed so that in the power operating range prompt inherent dynamic behavior tends to compensate for any rapid increase in reactivity in accordance with Criterion 11.

For further discussion, see the following sections:

a. Principal design criteria 1.2

b. Nuclear design 4.3

c. Thermal and hydraulic design 4.4

3.1.2.2.3 Criterion 12 - Suppression of Reactor Power Oscillations

The reactor core and associated coolant, control, and protection systems shall be designed to assure that power oscillations which can result in conditions exceeding specified acceptable fuel design limits are not possible or can be reliably and readily detected and suppressed (GDC 12).

Evaluation Against Criterion 12

The reactor core is designed to ensure that no power oscillation will cause fuel design limits to be exceeded. The power reactivity coefficient is the composite simultaneous effect of the fuel temperature or doppler coefficient, moderator void coefficient, and moderator temperature coefficient to a change in power level. It is negative and well within the range required for adequate damping of power and spatial xenon disturbances. Analytical studies indicate that for large BWRs, under-damped, unacceptable power disturbance behavior could only be expected to occur with power coefficients more positive than about -0.01 $\Delta k/k/\Delta P/P$. Operating experience has shown large BWRs to be inherently stable against xenon-induced power instability. The large negative operating coefficients provide

a. Good load following with well-damped behavior and little undershoot or overshoot in the heat transfer response,

b. Load following with recirculation flow control, and

c. Strong damping of spatial power disturbances.

The RPS design provides protection from excessive fuel cladding temperatures and protects the RCPB from excessive pressures that threaten the integrity of the system. Local abnormalities are sensed, and if protection system limits are reached, corrective action is initiated through an automatic scram. Integrity of the protection system is achieved through the combination of logic arrangement, trip channel redundancy, power supply redundancy, and physical separation.

The reactor core and associated coolant, control, and protection systems are designed to suppress any power oscillations which could result in exceeding fuel design limits. These systems ensure that Criterion 12 is met.

For further discussion, see the following sections:

a. Principal design criteria 1.2

b. Reactivity control systems 4.1

c. Nuclear design 4.3

d. Thermal and hydraulic design 4.4

e. Functional design of reactivity control systems 4.6

f. Overpressurization protection 5.2.2

g. Reactor trip system 7.2

h. Reactor manual control system instrumentation and control 7.7

i. Accident analyses 15

3.1.2.2.4 Criterion 13 - Instrumentation and Control

Instrumentation shall be provided to monitor variables and systems over their anticipated ranges for normal operation, for anticipated operational occurrences, and for accident conditions as appropriate to assure adequate safety, including those variables and systems that can affect the fission process, the integrity of the reactor core, the reactor coolant pressure boundary, and the containment and its associated systems. Appropriate controls shall be provided to maintain these variables and systems within prescribed operating ranges (GDC 13).

Evaluation Against Criterion 13

The fission process is monitored and controlled for all conditions from source range through power operating range. The intermediate and power ranges of the neutron monitoring system detect core conditions that threaten the overall integrity of the fuel barrier due to excess power generation and provide a signal to the RPS. Fission detectors, located in the core, are used for neutron detection. The detectors are located to provide optimum monitoring in the intermediate and power ranges.

The intermediate range monitors (IRM) measure neutron flux from the upper portion of the source range monitors (SRM) to the lower portion of the local power range monitor (LPRM) subsystem. The IRM is capable of generating a trip signal to scram the reactor.

The LPRM subsystem consists of fission chambers located throughout the core, signal conditioning equipment, and trip functions. Local power range monitor signals are also used to block rod withdrawal and to generate the necessary trip signal for reactor scram [average power range monitor (APRM)].

The RPS protects the fuel barriers and the nuclear process barrier by monitoring plant parameters and causing a reactor scram when predetermined setpoints are exceeded. Separation of the scram and normal rod control function prevents failures in the reactor manual control circuitry from affecting the scram circuitry.

To provide protection against the consequences of accidents involving the release of radioactive materials from the fuel and RCPB, the containment and reactor vessel isolation control system initiates automatic isolation of appropriate pipelines whenever monitored variables exceed preselected operational limits.

Nuclear system leakage limits are established so that appropriate action can be taken to ensure the integrity of the RCPB. Nuclear system leakage rates are classified as identified and unidentified, which corresponds respectively to the flow to the equipment drain and floor drain sumps. The permissible total leakage rate limit to these sumps is based on the makeup capabilities of various reactor component systems. Flow integrator and recorders are used to determine the leakage flow pumped from the drain sumps. The unidentified leakage rate as established in Section 5.2.5 is less than the value that has been conservatively calculated to be a minimum leakage from a crack large enough to propagate rapidly, but which still allows time for identification and corrective action before the integrity of the process barrier is threatened.

The process radiation monitoring system monitors radiation levels of various processes and provides trip signals to the containment and reactor vessel isolation control system whenever preestablished limits are exceeded.

As noted above, adequate instrumentation is provided to monitor system variables in the reactor core, RCPB, and reactor containment. Appropriate controls are provided to maintain the variables in the operating range and to initiate the necessary corrective action in the event of abnormal operational occurrence or accident. These instrumentation and controls meet the requirements of Criterion 13.

For further discussion, see the following sections:

a.	Principal design criteria	1.2
b.	Functional design of reactivity control systems	4.6
c.	Detection of leakage through RCPB	5.2.5
d.	Main steam line isolation system	5.4.5
e.	Containment systems	6.2
f.	Primary containment and reactor vessel isolation control system instrumentation and control	7.1
g.	Reactor trip system	7.2
h.	ESF systems	7.3
i.	Systems required for safe shutdown	7.4
j.	Safety-related display instrumentation	7.5
k.	All other instrumentation systems required for safety	7.6
l.	Control systems not required for safety	7.7

3.1.2.2.5 Criterion 14 - Reactor Coolant Pressure Boundary

> The reactor coolant pressure boundary shall be designed, fabricated, erected, and tested so as to have an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture (GDC 14).

Evaluation Against Criterion 14

The piping and equipment pressure parts within the RCPB through the outer isolation valves are designed, fabricated, erected, and tested to provide a high degree of integrity throughout the plant lifetime. Section 3.2 classifies systems and components within the RCPB to Quality Group A Requirements. The design requirements and codes and standards applied to this quality group ensure a quality product in keeping with the safety functions to be performed.

To minimize the possibility of brittle fracture within the RCPB, the fracture toughness properties and the operating temperature of ferritic materials are controlled to ensure adequate toughness. Section 5.2.3 describes the methods used to control toughness properties. Materials are impact tested in accordance with ASME Code Section III, where applicable. Where RCPB piping penetrates the containment, the fracture toughness temperature requirements of the RCPB materials apply.

Piping and equipment pressure parts of the RCPB are assembled and erected by welding unless applicable codes permit flanged or screwed joints. Welding procedures are employed which produce welds of complete fusion and free of unacceptable defects. All welding procedures, welders, and welding machine operators used in producing pressure-containing welds are qualified in accordance with the requirements of ASME Code Section IX, for the materials to be welded. Qualification records including the results of procedure and performance qualification tests and identification symbols assigned to each welder are maintained.

Section 5.2 contains the detailed materials and examination requirements for the piping and equipment of the RCPB prior to and after its assembly and erection. Leakage testing and surveillance is accomplished as described in the evaluation against Criterion 30 of the GDC.

The design, fabrication, erection, and testing of the RCPB ensure an extremely low probability of failure or abnormal leakage, thus satisfying the requirements of Criterion 14.

For further discussion, see the following sections:

a. Principal design criteria — 1.2

b. Design of structures, components, equipment, and systems — 3

c. Integrity of RCPB — 5.2

d. Overpressurization protection — 5.2.2

e. Reactor vessel — 5.3

f. Reactor coolant piping 5.4.3

g. Reactor trip system 7.2

h. Accident analyses 15

3.1.2.2.6 Criterion 15 - Reactor Coolant System Design

The reactor coolant system and associated auxiliary, control, and protection systems shall be designed with sufficient margin to assure that the design conditions of the reactor coolant pressure boundary are not exceeded during any condition of normal operation, including anticipated operational occurrences (GDC 15).

Evaluation Against Criterion 15

The reactor coolant system consists of the reactor vessel and appurtenances, the reactor recirculation system, the nuclear system pressure relief system, the main steam lines, the reactor core isolation cooling (RCIC) system, and the residual heat removal (RHR) system.

These systems are designed, fabricated, erected, and tested to stringent quality requirements and appropriate codes and standards that ensure the integrity of the RCPB throughout the plant lifetime. The reactor coolant system was designed and fabricated to meet the requirements of the ASME Code Section III, as indicated in Section 5.2.1.

The auxiliary, control, and protection systems associated with the reactor coolant system act to provide sufficient margin to ensure that the design conditions of the RCPB are not exceeded during any condition of normal operation, including anticipated operational occurrences. As described in the evaluation of Criterion 13, instrumentation is provided to monitor essential variables to ensure that they are within prescribed operating limits. If the monitored variables exceed their predetermined settings, the auxiliary, control, and protection systems automatically respond to maintain the variables and systems within allowable design limits.

An example of the integrated protective action scheme, which provides sufficient margin to ensure that the design conditions of the RCPB are not exceeded, is the automatic initiation of the nuclear system pressure relief system on receipt of an overpressure signal. To accomplish overpressure protection, a number of pressure-operated relief valves are provided that can discharge steam from the nuclear system to the suppression pool. The nuclear system pressure relief system also provides for automatic depressurization of the nuclear system in the event of a LOCA in which the vessel is not depressurized by the accident. The depressurization of the nuclear system in this situation allows operation of the low-pressure emergency core cooling system (ECCS) to supply enough cooling water to adequately cool the core. In a similar manner, other auxiliary, control, and protection systems provide assurance that the design conditions of the RCPB are not exceeded during any conditions of normal operation, including anticipated operational occurrences.

The application of appropriate codes and standards and high quality requirements to the reactor coolant system and the design features of its associated auxiliary, control, and protection systems ensure that the requirements of Criterion 15 are satisfied.

For further discussion, see the following sections:

a.	Principal design criteria	1.2
b.	Design of structures, components, equipment, and systems	3
c.	Integrity of RCPB	5.2
d.	Overpressurization protection	5.2.2
e.	Detection of leakage through the reactor coolant pressure boundary	5.2.5
f.	Reactor vessel	5.3
g.	Reactor coolant piping	5.4
h.	Reactor trip system	7.2
i.	Accident analyses	15

3.1.2.2.7 Criterion 16 - Containment Design

> Reactor containment and associated systems shall be provided to establish an essentially leak-tight barrier against the uncontrolled release of radioactivity to the environment and to assure that the containment design conditions important to safety are not exceeded for as long as postulated accident conditions require (GDC 16).

Evaluation Against Criterion 16

An essentially leaktight barrier is provided by a steel primary containment consisting of a drywell and suppression chamber and a containment isolation system. Release of radioactivity from the primary containment to the environment is further controlled by the reactor building, which provides secondary containment, and the standby gas treatment system (SGTS). These systems are designed to protect the public from the consequences of a LOCA, based on a postulated break of the reactor coolant piping up to and including a double-ended break of the largest reactor coolant pipe.

The primary containment, reactor building, and the associated ESF systems are designed to safely sustain all internal and external environmental conditions that may be postulated to occur during the life of the plant, including both short- and long-term effects following a LOCA.

Containment temperature and pressure following an accident are limited by the suppression pool and the RHR system.

For further discussion, see the following sections:

a.	General plant description	1.2
b.	Steel containment	3.8
c.	Containment systems	6.2
d.	Accident analyses	15

3.1.2.2.8 Criterion 17 - Electric Power Systems

An onsite electric power system and an offsite electric power system shall be provided to permit functioning of structures, systems, and components important to safety. The safety function for each system (assuming the other system is not functioning) shall be to provide sufficient capacity and capability to assure that (1) specified acceptable fuel design limits and design conditions of the reactor coolant pressure boundary are not exceeded as a result of anticipated operational occurrences and (2) the core is cooled and containment integrity and other vital functions are maintained in the event of postulated accidents.

The onsite electric power supplies, including the batteries, and the onsite electric distribution system, shall have sufficient independence, redundancy, and testability to perform their safety functions assuming a single failure.

Electric power from the transmission network to the onsite electric distribution system shall be supplied by two physically independent circuits (not necessarily on separate rights of way) designed and located so as to minimize to the extent practical the likelihood of their simultaneous failure under operating and postulated accident and environmental conditions. A switchyard common to both circuits is acceptable. Each of these circuits shall be designed to be available in sufficient time following a loss of all onsite alternating current power supplies and the other offsite electric power circuit, to assure that specified acceptable fuel design limits and design conditions of the reactor coolant pressure boundary are not exceeded. One of these circuits shall be designed to be available to assure that core coolant, primary containment integrity, and other vital safety functions are maintained.

Provisions shall be included to minimize the probability of losing electric power from any of the remaining supplies as a result of, or coincident with, the loss of power generated by the nuclear power unit, the loss of power from the transmission network, or the loss of power from the onsite electric power supplies (GDC 17).

Evaluation Against Criterion 17

Offsite power is transmitted to the plant transformer yard by two physically independent transmission lines which approach the plant along separate rights-of-way. Offsite power is available for plant startup or shutdown utilizing the 230-kV startup transformer. This transformer has sufficient capacity to carry all plant normal loads, in addition to those ESF loads required for plant shutdown or mitigation of the consequences of an accident.

A 115-kV transmission line, located on a separate right-of-way, provides power to a backup transformer which is capable of providing power for the plant safety-related systems to mitigate the effects of an accident. This power source is automatically transferred to the plant safety-related system on loss of all other offsite power sources.

The three independent onsite ac load groups provide independence and redundancy of equipment function. They meet the safety requirements assuming a single failure since the load groups are connected to offsite power by circuits having independent routes from the Class 1E switchgear.

For each of the three ac load groups there are independent batteries that furnish dc load and control power for the corresponding divisions.

The reactor protection instrumentation is powered from two independent ride-through ac power sources.

During normal plant operation, the station auxiliary power is supplied from the main generator through the plant auxiliary transformers. On loss of power from the normal auxiliary transformers, there will be an automatic transfer to a source of offsite power from the 230-kV startup or 115-kV backup transformers. In the event of a loss of the 230-kV and 115-kV offsite power sources, three onsite diesel generator sets and redundant sets of station batteries provide the necessary ac and dc power for safe shutdown or, in the event of an accident, to restrict the consequences to within acceptable limits. The onsite emergency ac and dc power systems contain sufficient redundancy and independence such that a single failure does not prevent the systems from performing their safety function.

The power systems as designed meet the requirements of Criterion 17.

For further discussion, see the following sections:

a.	General plant description	1.2
b.	Seismic qualification of Seismic Category I instrumentation and electrical equipment	3.10
c.	Environmental design of mechanical and electrical equipment	3.11
d.	Electric power	8

3.1.2.2.9 Criterion 18 - Inspection and Testing of Electric Power Systems

> Electric power systems important to safety shall be designed to permit appropriate periodic inspection and testing of important areas and features, such as wiring, insulation, connections, and switchboards, to assess the continuity of the systems and the condition of their components. The systems shall be designed with a capability to test periodically (1) the operability and functional performance of the components of the systems, such as onsite power sources, relays, switches, and buses, and (2) the operability of the systems as a whole and, under conditions as close to design as practical, the full operation sequence that brings the systems into operation, including operation of applicable portions of the protection system, and the transfer of power among the nuclear power unit, the offsite power system, and the onsite power system (GDC 18).

Evaluation Against Criterion 18

The engineered safety systems power supply buses and associated diesel generators are arranged for periodic inspection and testing of each diesel generator independently.

At periodic intervals when the reactor is not at power operation, tests simulating loss of power and accident signals will be initiated to demonstrate the capability of the power systems to meet the starting and loading requirements.

Periodically during plant operation each diesel generator will be manually started and loaded. The redundant (Division 1 and 2) units will be synchronized to the startup power source and loaded. The high-pressure core spray (HPCS) (Division 3) unit will be loaded by synchronizing to the startup power source.

These tests are designed to prove the operability of the onsite power system under conditions as close to design as possible, to assess the continuity of the systems and condition of the components.

The electrical power systems are configured to provide access for appropriate periodic inspection. For further discussion, see the following sections:

a. Onsite power systems 8.3

b. Initial test program 14

3.1.2.2.10 Criterion 19 - Control Room

A control room shall be provided from which actions can be taken to operate the nuclear power unit safely under normal conditions and to maintain it in a safe condition under accident conditions, including loss-of-coolant accidents. Adequate radiation protection shall be provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 5 rem whole body, or its equivalent to any part of the body, for the duration of the accident.

Equipment at appropriate locations outside the control room shall be provided (1) with a design capability for prompt hot shutdown of the reactor, including necessary instrumentation and controls to maintain the unit in a safe condition during hot shutdown, and (2) with a potential capability for subsequent cold shutdown of the reactor through the use of suitable procedures (GDC 19).

Evaluation Against Criterion 19

A control room is provided from which the nuclear power plant can be safely operated under normal conditions and can be maintained in a safe condition following postulated accidents.

The control room contains the following equipment: controls and necessary surveillance equipment for operation of the plant functions, such as the reactor and its auxiliary systems, ESF, turbine generator, steam and power conversion systems, and station electrical distribution boards.

The control room is located in a Safety Class 3, Seismic Category I structure. Safe occupancy of the control room during abnormal conditions is ensured by the design. Adequate shielding is provided to maintain tolerable radiation levels in the control room in the event of a design basis accident for the duration of the accident.

The control room ventilation system has redundant and spatially separated fresh air intakes and redundant equipment, radiation detectors, and smoke detectors with appropriate alarms and interlocks. Redundant systems are provided for the control room air to be recirculated through high-efficiency particulate air (HEPA) and charcoal filters during isolation of the control room from outside air.

The control room can be continuously occupied under all operating and accident conditions. In the unlikely event that the control room must be vacated and access is restricted, instrumentation and controls are provided outside the control room from which safe shutdown of the reactor can be effected.

The above demonstrates that the control room design meets the requirements of Criterion 19.

For further discussion, see the following sections:

a.	General plant description	1.2
b.	Radwaste and control building	3.8
c.	Classification of structures, components, and systems	3.2
d.	Control room habitability	6.4
e.	Instrumentation and control	7
f.	Shutdown outside the control room	7.4
g.	Control room area ventilation system	9.4
h.	Fire protection	Appendix F
i.	Shielding	12.3
j.	Ventilation	12.3

3.1.2.3 Group III - Protection and Reactivity Control System

3.1.2.3.1 Criterion 20 - Protection System Functions

The protection system shall be designed (1) to initiate automatically the operation of appropriate systems including the reactivity control systems, to assure that specified acceptable fuel design limits are not exceeded as a result of anticipated operational occurrences, and (2) to sense accident conditions and to initiate the operation of systems and components important to safety (GDC 20).

Evaluation Against Criterion 20

The RPS is designed to provide timely protection against the onset and consequences of conditions that threaten the integrity of the fuel barrier and the RCPB barrier. Fuel damage is prevented by the initiation of an automatic reactor shutdown if monitored system variables exceed limits of anticipated operational occurrences. Scram trip settings are selected and verified to be far enough above or below operating levels to provide proper protection but not be subject to spurious scrams. The RPS includes the high-inertia motor generator power system, sensors, relays, bypass circuitry, and switches that signal the control rod system to scram and shut down the reactor. The scrams initiated by neutron monitoring system variables, nuclear system high pressure, turbine stop valve closure, turbine control valve fast closure, MSIV closure, and reactor vessel low water level will prevent fuel damage following abnormal operational transients. Specifically, the process parameters initiate a scram in time to prevent the core from exceeding thermal hydraulic safety limits during abnormal operational transients. In addition, protection systems are provided to sense accident conditions and initiate automatically the operation of other systems and components important to safety.

The design of the protection system satisfies the functional requirements as specified in Criterion 20.

For further discussion, see the following sections:

a.	Principal design criteria	1.2
b.	Fuel system design	4.2
c.	Functional design of reactivity control systems	4.6
d.	Main steam line isolation system	5.4.5
e.	Reactor trip system	7.2
f.	Accident analyses	15

3.1.2.3.2 Criterion 21 - Protection System Reliability and Testability

> The protection system shall be designed for high functional reliability and inservice testability commensurate with the safety functions to be performed. Redundancy and independence designed into the protection system shall be sufficient to assure that (1) no single failure results in loss of the protection function, and (2) removal from service of any component or channel does not result in loss of the required minimum redundancy unless the acceptable reliability of operation of the protection system can be otherwise demonstrated. The protection system shall be designed to permit periodic testing of its functioning when the reactor is in operation, including a capability to test channels independently to determine failures and losses of redundancy that may have occurred (GDC 21).

Evaluation Against Criterion 21

The RPS design provides assurance that, through redundancy, each channel has sufficient reliability to fulfill the single-failure criterion. No single component failure, intentional bypass maintenance operation, calibration operation, or test to verify operational availability will impair the ability of the system to perform its intended safety function. Additionally, the system design ensures that when a scram trip point is exceeded there is a high scram probability. However, should a scram not occur, other monitored components will scram the reactor if their trip points are exceeded. There is sufficient electrical and physical separation between channels and between trip logics monitoring the same variable to prevent environmental factors, electrical transients, and physical events from impairing the ability of the system to respond correctly.

The RPS includes design features that permit inservice testing. This ensures the functional reliability of the system should the monitored reactor parameter exceed the corrective action setpoint.

The RPS initiates an automatic reactor shutdown if the monitored plant variables exceed preestablished limits. This system is arranged as two separately powered trip systems. Each trip system has two trip channels. An automatic or manual trip in either or both trip channels constitutes a trip system trip. A scram results when both trip systems have tripped. The logic scheme is a one-out-of-two twice arrangement. The RPS can be tested during reactor operation. Manual scram testing is performed by operating one of the four manual scram controls. Two manual scram controls are associated with each trip system, one in each trip channel. Operating one manual scram control tests one trip channel and one trip system. The total test verifies the ability to deenergize the scram pilot valve solenoids. Indicating lights verify that the actuator contacts have opened. This capability for a thorough testing program significantly increases reliability.

Control rod drive (CRD) operability can be tested during normal reactor operation. Drive position indicator and in-core neutron detectors are used to verify control rod movement. Each control rod can be inserted one notch and then withdrawn to the original position without significantly perturbing the nuclear system. One control rod is tested at a time. Control rod drive mechanism over-travel tests demonstrate rod-to-drive coupling integrity. Hydraulic supply subsystem pressures can be observed on control room instrumentation. More importantly, the hydraulic control unit scram accumulator and the scram discharge volume level are continuously monitored.

The MSIVs may be tested during full reactor operation. Individually, they can be closed to 90% of full open position without affecting the reactor operation. If reactor power is reduced sufficiently, the isolation valves on a single main steam supply line may be fully closed. Provisions are provided to evaluate valve stem leakage during reactor shutdown. During refueling operation, valve leakage rates can be determined.

The high functional reliability, redundancy, and inservice testability of the protection system satisfy the requirements specified in Criterion 21.

For further discussion, see the following sections:

a.	Principal design criteria	1.2
b.	Functional design of reactivity control systems	4.6
c.	Main steam line isolation system	5.4.5
d.	Reactor trip system	7.2

3.1.2.3.3 Criterion 22 - Protection System Independence

> The protection system shall be designed to assure that the effects of natural phenomena, and of normal operating, maintenance, testing, and postulated accident conditions on redundant channels do not result in loss of the protection function, or shall be demonstrated to be acceptable on some other defined basis. Design techniques, such as functional diversity or diversity in component design and principles of operation, shall be used to the extent practical to prevent loss of the protection function (GDC 22).

Evaluation Against Criterion 22

The components of protection systems are designed so that the mechanical and thermal environmental conditions resulting from any emergency situation in which the components are required to function will not interfere with the operation of that function. The safety-related wiring and cabling necessary to support the operation of the RPS, outside of enclosures, are routed within totally enclosed and grounded raceway systems. The RPS has four independent initiation channels providing input to two trip systems for redundancy. Cabling for the four channels is routed in separate raceway systems to preserve channel independence. The wires from duplicate sensors on a common process tap are run in separate wireways. The system sensors are electrically and physically separated. Only one trip actuator logic circuit from each trip system may be run in the same wireway.

The RPS is designed to permit maintenance, calibration, and testing activities while the reactor is operating without restricting plant operation or affecting system safety function completion. This is accomplished by limiting these activities to only one of the independent initiation sensors or associated logic channels at a time leaving at least two channels per monitored parameter available to respond to an accident or transient. Only a single channel per trip system is required to initiate a scram. The protection system meets the design requirements for functional and physical independence as specified in Criterion 22.

For further discussion, see the following sections:

a. Principal design criteria 1.2

b. Functional design of reactivity control systems 4.6

c. Main steam line isolation system 5.4

d. Reactor trip system 7.2

3.1.2.3.4 Criterion 23 - Protection System Failure Modes

> The protection system shall be designed to fail into a safe state or into a state demonstrated to be acceptable on some other defined basis if conditions such as disconnection of the system, loss of energy (e.g., electric power, instrument air), or postulated adverse environments (e.g., extreme heat or cold, fire, pressure, steam, water, and radiation) are experienced (GDC 23).

Evaluation Against Criterion 23

The RPS components/logic are designed to fail into a safe state (system actuation) when conditions such as loss of electrical power source or fire occur. This is accomplished by maintaining the input/output logic components and final actuated devices in a normally energized state with deenergization resulting in the completion of their intended safety functions. Additionally, RPS cables that are located outside of panels are enclosed within totally enclosed and grounded raceways providing additional fail safe design.

Reactor protection system components that perform safety-related functions to mitigate a design basis event are designed to complete their intended safety functions even when exposed to adverse environmental conditions (e.g., temperature, pressure, steam, water, and radiation) postulated to occur during those events.

The failure modes of the protection system are such that it will fall into a safe state as required by Criterion 23.

For further discussion, see the following sections:

a.	Principal design criteria	1.2
b.	Environmental design of mechanical and electrical equipment	3.11
c.	Functional design of reactivity control systems	4.6
d.	Reactor trip system	7.2
e.	Electric power	8.3

3.1.2.3.5 Criterion 24 - Separation of Protection and Control Systems

> The protection system shall be separated from control systems to the extent that failure of any single control system component or channel, or failure or removal from service of any single protection system component or channel which is common to the control and protection systems leaves intact a system satisfying all reliability redundancy, and independence requirements of the protection system. Interconnection of the protection and control systems shall be limited so as to assure that safety is not significantly impaired (GDC 24).

Evaluation Against Criterion 24

There is separation between the RPS and the process control systems. Sensors, trip channels, and trip logics of the RPS are not used directly for automatic control of process systems. Therefore, failure in the controls and instrumentation of process systems cannot induce failure of any portion of the protection system. High scram reliability is designed into the RPS and hydraulic control unit for the CRD. The scram signal and mode of operation overrides all other signals.

The protection system is separated from control systems as required in Criterion 24.

For further discussion, see the following sections:

a.	Principal design criteria	1.2
b.	Functional design of reactivity control systems	4.6
c.	Reactor trip system	7.2
d.	Reactor manual control system instrumentation and controls	7.7

3.1.2.3.6 Criterion 25 - Protection System Requirements for Reactivity Control Malfunctions

The protection system shall be designed to assure that specified acceptable fuel design limits are not exceeded for any single malfunction of the reactivity control systems, such as accidental withdrawal (not ejection or dropout) of control rods (GDC 25).

Evaluation Against Criterion 25

The RPS provides protection against the onset and consequences of conditions that threaten the integrity of the fuel barrier and the RCPB. Any monitored variable exceeding the scram setpoint will initiate an automatic scram and not impair the remaining variables from being monitored. If one channel fails, the remaining portions of the RPS shall function.

The reactor manual control system is designed so that no single failure can negate the effectiveness of a reactor scram. The circuitry for the reactor manual control system is completely independent of the circuitry controlling the scram valves. This separation of the scram and normal rod control functions prevents failures in the reactor manual control circuitry from affecting the scram circuitry. Because each control rod is controlled as an individual unit, a failure that results in energizing any of the insert or withdraw solenoid valves can affect only one control rod. The effectiveness of a reactor scram is not impaired by the malfunctioning of any one control rod.

The most serious rod withdrawal errors are considered to be when the reactor is just subcritical and an out-of-sequence rod is continuously withdrawn. The rod worth minimizer would normally prevent the withdrawal of out-of-sequence rods. If such a continuous rod withdrawal were to occur, the increase in fuel temperature subsequent to scram would not be sufficient to exceed acceptable fuel design limits.

The design of the protection system ensures that specified acceptable fuel limits are not exceeded for any single malfunction of the reactivity control systems as specified in Criterion 25.

For further discussion, see the following sections:

a.	Principal design criteria	1.2
b.	Nuclear design	4.3
c.	Thermal and hydraulic design	4.4
d.	Functional design of reactivity control systems	4.6
e.	Reactor trip system	7.2
f.	Reactor manual control system instrumentation and controls	7.7
g.	Accident analyses	15

3.1.2.3.7 Criterion 26 - Reactivity Control System Redundancy and Capability

> Two independent reactivity control systems of different design principles shall be provided. One of the systems shall use control rods, preferably including a positive means for inserting the rods, and shall be capable of reliably controlling reactivity changes to assure that under conditions of normal operation, including anticipated operational occurrences, and with appropriate margin for malfunctions such as stuck rods, specified acceptable fuel design limits are not exceeded. The second reactivity control system shall be capable of reliably controlling the rate of reactivity changes resulting from planned, normal power changes (including xenon burnout) to assure acceptable fuel design limits are not exceeded. One of the systems shall be capable of holding the reactor core subcritical under cold conditions (GDC 26).

Evaluation Against Criterion 26

Two independent reactivity control systems using different design principles are provided. The normal method of reactivity control employs control rod assemblies. Positive insertion of these control rods is provided by means of the CRD hydraulic system. The control rods are capable of reliably controlling reactivity changes during normal operation (e.g., power changes, power shaping, xenon burnout, normal startup and shutdown) via operator-controlled insertions and withdrawals. The control rods are also capable of maintaining the core within acceptable fuel design limits during anticipated operational occurrences via the automatic scram function. The unlikely occurrence of a limited number of stuck rods during a scram will not adversely affect the capability to maintain the core within fuel design limits.

The circuitry for manual insertion or withdrawal of control rods is completely independent of the circuitry for reactor scram. This separation of the scram and normal rod control functions prevents failures in the reactor manual control circuitry from affecting the scram circuitry. Two sources of scram energy (accumulator pressure and reactor vessel pressure) provide needed scram performance over the entire range of reactor pressure, i.e., from operating conditions to cold shutdown. The design of the control rod system includes appropriate margin for malfunctions such as stuck rods. Control rod withdrawal sequences and patterns are selected prior to operation to achieve optimum core performance, and simultaneously, low individual rod worths. The operating procedures to accomplish such patterns are supplemented by the rod worth minimizer, which prevents rod withdrawals yielding a rod worth greater than permitted by the preselected rod withdrawal pattern. Because of the carefully planned and regulated rod withdrawal sequence, prompt shutdown of the reactor can be achieved with the insertion of a small number of the many independent control rods. In the event that a reactor scram is necessary, the unlikely occurrence of a limited number of stuck rods will not hinder the capability of the control rod system to render the core subcritical.

The second independent reactivity control system is provided by the reactor coolant recirculation system. By varying reactor flow, it is possible to affect the type of reactivity changes necessary for planned, normal power changes (including xenon burnout). In the event that reactor flow is suddenly increased to its maximum value (pump runout), the core will not exceed fuel design limits because the power flow map defines the allowable initial operating states such that the pump runout will not violate these limits.

The control rod system is capable of holding the reactor core subcritical under cold conditions, even when the control rod of highest worth is assumed to be stuck in the fully withdrawn position. This shutdown capability of the control rod system is made possible by designing the fuel with burnable poison (Gd_2O_3) to control the high reactivity of fresh fuel. In addition, the standby liquid control system is available to add soluble boron to the core and render it subcritical, as discussed in Section 3.1.2.3.8.

The redundancy and capabilities of the reactivity control systems for the BWR satisfy the requirements of Criterion 26.

For further discussion, see the following sections:

a.	Principal design criteria	1.2
b.	Reactivity control system	4.3
c.	Functional design reactivity control system	4.6

d.	Reactor trip system	7.2
e.	Process computer system instrumentation and controls (rod worth minimizer)	7.7
f.	Reactor manual control system instrumentation and controls	7.7
g.	Reactor recirculation system	5.4

3.1.2.3.8 Criterion 27 - Combined Reactivity Control Systems Capability

> The reactivity control systems shall be designed to have a combined capability, in conjunction with poison addition by the emergency core cooling system, of reliably controlling reactivity changes to assure that under postulated accident conditions and with appropriate margin for stuck rods, the capability to cool the core is maintained (GDC 27).

Evaluation Against Criterion 27

There is no credible event applicable to the BWR which requires combined capability of the control rod system and poison additions by the emergency core cooling network. The BWR design is capable of maintaining the reactor core subcritical, including allowance for a stuck rod, without addition of any poison to the reactor coolant. The primary reactivity control system for the BWR during postulated accident conditions is the control rod system. Abnormalities are sensed and, if protection system limits are reached, corrective action is initiated through automatic insertion of control rods. High integrity of the protection system is achieved through the combination of logic arrangement, actuator redundancy, power supply redundancy, and physical separation. High reliability of reactor scram is further achieved by separation of scram and manual control circuitry, individual control units for each control rod, and fail-safe design features built into the rod drive system. Response by the RPS is prompt and the total scram time is short.

In the event that more than one control rod fails to insert, and the core cannot be maintained in a subcritical condition by control rods alone as the reactor is cooled down subsequent to initial shutdown, the standby liquid control (SLC) system will be actuated to insert soluble boron into the reactor core. The SLC system has sufficient capacity to ensure that the reactor can always be maintained subcritical; and hence, only decay heat will be generated by the core which can be removed by the RHR system, thereby ensuring that the core will always be coolable.

The design of the reactivity control system ensures reliable control of reactivity under postulated accident conditions with appropriate margin for stuck rods. The capability to cool is maintained under all postulated accident conditions; thus, Criterion 27 is satisfied.

For further discussion, see the following sections:

a.	Principal design criteria	1.2
b.	Reactivity control system	4.3
c.	Nuclear design	4.3
d.	Thermal and hydraulic design	4.4
e.	Functional design of reactivity control system	4.6
f.	Reactor trip system	7.2
g.	Reactor manual control system	7.7
h.	Process computer system instrumentation and controls (rod worth minimizer)	7.7
i.	Standby liquid control system	9.3.5
j.	Accident analyses	15

3.1.2.3.9 Criterion 28 - Reactivity Limits

> The reactivity control systems shall be designed with appropriate limits on the potential amount and rate of reactivity increase to assure that the effects of postulated reactivity accidents can neither (1) result in damage to the reactor coolant pressure boundary greater than limited local yielding nor (2) sufficiently disturb the core, its support structures, or other reactor pressure vessel internals to impair significantly the capability to cool the core. These postulated reactivity accidents shall include consideration of rod ejection (unless prevented by positive means), rod dropout, steam line rupture, changes in reactor coolant temperature and pressure, and cold water addition (GDC 28).

Evaluation Against Criterion 28

The control rod system design incorporates appropriate limits on the amount and rate of reactivity increase. Control rod withdrawal sequences and patterns are selected to achieve optimum core performance and low individual rod worths. The rod worth minimizer prevents withdrawal other than by the preselected rod withdrawal pattern. The rod worth minimizer function assists the operator with an effective backup control rod monitoring routine that enforces adherence to established startup, shutdown, and low power level operations control rod procedures.

The control rod mechanical design incorporates a hydraulic velocity limiter in the control rod which prevents rapid rod ejection. This protects against a high reactivity insertion rate by limiting the control rod velocity to less than 5 ft/sec. Normal rod movement is limited to 6 in. increments and the rod withdrawal rate is limited through the hydraulic valve to 3 in./sec.

The accident analyses (Chapter 15) evaluates postulated reactivity accidents as well as abnormal operational transients in detail. Analyses are included for rod dropout, steam line rupture, changes in reactor coolant temperature and pressure, and cold water addition. The initial conditions, assumptions, calculational models, sequences of events, and anticipated results of each postulated occurrence are covered in detail. The results of these analyses indicate that none of the postulated reactivity transients or accidents would result in damage to the RCPB. In addition, the integrity of the core, its support structures or other reactor pressure vessel (RPV) internals, are maintained so that the capability to cool the core is not impaired for any of the postulated reactivity accidents described in the accident analyses.

The design features of the reactivity control system, which limit the potential amount and rate of reactivity increase, ensure that Criterion 28 is satisfied for all postulated reactivity accidents.

For further discussion, see the following sections:

a.	Principal design criteria	1.2
b.	CRD systems	3.9
c.	Reactor core support structures and internals mechanical design	4.2
d.	Reactivity control system	4.3
e.	Nuclear design	4.3
f.	CRD housing supports	4.5.3
g.	Overpressurization protection	5.2.2
h.	Reactor vessel and appurtenances	5.3

i.	Main steam line flow restrictor	5.4.4
j.	MSIV system	5.4.5
k.	Process computer system	7.7.1.9
l.	Accident analyses	15

3.1.2.3.10 Criterion 29 - Protection Against Anticipated Operational Occurrences

> The protection and reactivity control system shall be designed to assure an extremely high probability of accomplishing their safety functions in the event of anticipated operational occurrences (GDC 29).

<u>Evaluation Against Criterion 29</u>

The high functional reliability of the protection and reactivity control system is achieved through the combination of logic arrangements, redundancy, physical and electrical independence, functional separation, fail-safe design, and inservice testability. These design features are discussed in detail in Criteria 21, 22, 23, 24, and 26.

An extremely high reliability of timely response to anticipated operational occurrences is maintained by a thorough program of inservice testing and surveillance. Components important to safety such as CRDs and MSIVs are tested during normal operation. The capability for inservice testing ensures the high functional reliability of protection and reactivity control systems should a reactor variable exceed the corrective action setpoint.

The capabilities of the protection and reactivity control systems to perform their safety functions in the event of anticipated operational occurrences are satisfied in agreement with the requirements of Criterion 29.

For further discussion, see the following sections:

a.	Principal design criteria	1.2
b.	Reactivity control system	4.6
c.	MSIV system	5.4.5
d.	Reactor trip system	7.2

3.1.2.4 Group IV - Fluid Systems

3.1.2.4.1 Criterion 30 - Quality of Reactor Coolant Pressure Boundary

> Components which are part of the reactor coolant pressure boundary shall be designed, fabricated, erected, and tested to the highest quality standards practical. Means shall be provided for detecting and, to the extent practical, identifying the location of the source of reactor coolant leakage (GDC 30).

Evaluation Against Criterion 30

By utilizing conservative design practices and detailed quality control procedures, the pressure retaining components of the RCPB are designed and fabricated to retain their integrity during normal and postulated accident conditions. Accordingly, components that compose the RCPB are designed, fabricated, erected, and tested in accordance with recognized industry codes and standards listed in Table 3.2-1 and Chapter 5. Further, product and process quality planning is provided to ensure conformance with the applicable codes and standards and to retain appropriate documented evidence verifying compliance. Because the subject matter of this criterion deals with aspects of the RCPB, further discussion on this subject is treated in the response to Criterion 14, Reactor Coolant Pressure Boundary.

Means are provided for detecting reactor coolant leakage. The leak detection system consists of sensors and instruments to detect, annunciate, and in some cases, isolate the RCPB from potential hazardous leaks before predetermined limits are exceeded. Small leaks are detected by temperature and pressure changes, increased frequency of sump pump operation, and by measuring fission product concentration. In addition to these means of detection, large leaks are detected by changes in flow rates in process lines and changes in reactor water level. The allowable leakage rates have been based on the predicted and experimentally determined behavior of cracks in pipes, the ability to make up coolant system leakage, the normally expected background leakage due to equipment design, and the detection capability of the various sensors and instruments. The total leakage rate limit is established so that in the absence of normal ac power with loss of feedwater supply, makeup capabilities are provided by the RCIC system. While the leak detection system provides protection from small leaks, the ECCS network provides protection for the complete range of discharges from ruptured pipes. Thus, protection is provided for the full spectrum of possible discharges.

The RCPB and leak detection system are designed to meet the requirements of Criterion 30.

For further discussion, see the following sections:

a. Principal design criteria 1.2

b.	Design of structures, components, equipment, and systems	3
c.	Overpressurization protection	5.2.2
d.	Integrity of RCPB	5.2
e.	Detection of leakage through the RCPB	5.2.5
f.	Reactor vessel	5.3
g.	Component and subsystem design	5.4
h.	Reactor recirculation system	5.4
i.	Instrumentation and control systems	7
j.	Primary containment and reactor vessel isolation control system	7.3
k.	Leak detection system - instrumentation and control	7.6
l.	Quality assurance	17
m.	Inservice inspection and testing of the RCPB	5.2.4

3.1.2.4.2 Criterion 31 - Fracture Prevention of Reactor Coolant Pressure Boundary

The reactor coolant pressure boundary shall be designed with sufficient margin to assure that when stressed under operating, maintenance, testing, and postulated accident conditions (1) the boundary behaves in a nonbrittle manner; and (2) the probability of rapidly propagating fracture is minimized. The design shall reflect consideration of service temperatures and other conditions of the boundary material under operating, maintenance, testing, and postulated accident conditions and the uncertainties in determining (1) material properties, (2) the effects of irradiation on material properties, (3) residual, steady-state and transient stresses, and (4) size of flaws (GDC 31).

<u>Evaluation Against Criterion 31</u>

Brittle fracture control of pressure-retaining ferritic materials is provided to ensure protection against nonductile fracture. To minimize the possibility of brittle fracture failure of the RPV, the RPV was designed to meet the requirements of ASME Code Section III.

The nil-ductility transition (NDT) temperature is defined as the temperature below which ferritic steel breaks in a brittle rather than ductile manner. The NDT temperature increases as a function of neutron exposure at integrated neutron exposures greater than about 1 x 10^{17} nvt with neutron of energies in excess of 1 MeV.

The reactor assembly design provides an annular space from the outermost fuel assemblies to the inner surface of the reactor vessel that serves to attenuate the fast neutron flux incident up on the reactor vessel wall. This annular volume contains the core shroud, jet pump assemblies, and reactor coolant. Assuming plant operation at rated power and availability of 100% for the plant life time, the neutron fluence at the inner surface of the vessel causes a slight shift in the transition temperature. Expected shifts in transition temperature during design life as a result of environmental conditions, such as neutron flux, are considered in the design. Operational limitations assume that NDT temperature shifts are accounted for in the reactor operation. The Technical Specifications describe the mechanism for incorporating temperature shifts caused by the neutron flux.

The RCPB is designed, maintained, and tested such that adequate assurance is provided that the boundary will behave in a nonbrittle manner throughout the life of the plant. Therefore, the RCPB is in conformance with Criterion 31.

For further discussion, see the following sections:

a. Design of structures, components, equipment, and systems 3

b. Integrity of RCPB 5.2

c. Reactor vessel 5.3

3.1.2.4.3 Criterion 32 - Inspection of Reactor Coolant Pressure Boundary

> Components which are part of the reactor coolant pressure boundary shall be designed to permit (1) periodic inspection and testing of important areas and features to assess their structural and leaktight integrity, and (2) appropriate material surveillance program for the reactor vessel (GDC 32).

Evaluation Against Criterion 32

The RPV design includes provisions for inservice inspection. Removable plugs in the sacrificial shield and/or removable panels in the insulation provide access for examination of the vessel and its appurtenances. Also, removable insulation is provided on the reactor coolant system safety/relief valves, recirculation system, and on the main steam feedwater systems extending out to and including the first isolation valve outside containment. Inspection of the

RCPB is in accordance with the ASME Code Section XI. Section 5.2.4 defines the Inservice Inspection Program Plan, access provisions, and areas of restricted access.

Vessel material surveillance capsules are located at the 30°, 120°, and 300° azimuth locations in the vessel adjacent to the wall. Specimens of the base metal, weld metal, and heat-affected zone metal are located in each capsule.

The plant testing and inspection program ensures that the requirements of Criterion 32 will be met.

For further discussion, see the following sections:

a.	Design of structures, components, equipment, and systems	3
b.	Inservice inspection and testing of RCPB	5.2.4
c.	Reactor vessel	5.3
d.	Component and subsystem design	5.4

3.1.2.4.4 Criterion 33 - Reactor Coolant Makeup

> A system to supply reactor coolant makeup for protection against small breaks in the reactor coolant pressure boundary shall be provided. The system safety function shall be to assure that specified acceptable fuel design limits are not exceeded as a result of reactor coolant loss due to leakage from the reactor coolant pressure boundary and rupture of small piping or other small components which are part of the boundary. The system shall be designed to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished using the piping, pumps, and valves used to maintain coolant inventory during normal reactor operation (GDC 33).

Evaluation Against Criterion 33

Means are provided for detecting reactor coolant leakage. The leak detection system consists of sensors and instruments to detect, annunciate, and in some cases isolate the RCPB from potential hazardous leaks before predetermined limits are exceeded. Small leaks are detected by temperature and pressure changes, increased frequency of sump pump operation, and by measuring fission product concentration. In addition to these means of detection, large leaks are detected by changes in flow rates in process lines and changes in reactor water level. The allowable leakage rates have been based on predicted and experimentally determined behavior of cracks in pipes, the ability to make up coolant system leakage, the normally expected background leakage due to equipment design, and the detection capability of the various sensors and instruments. The total leakage rate limit is established so that in the absence of normal ac power concomitant with a loss of feedwater supply, makeup capabilities are provided by the RCIC system.

The plant is designed to provide ample reactor coolant makeup for protection against small leaks in the RCPB for anticipated operational occurrences and postulated accident conditions. The design of these systems meets the requirements of Criterion 33.

For further discussion, see the following sections:

a.	Detection of leakage through RCPB	5.2.5
b.	RCIC system	5.4.6
c.	ECCS	6.3
d.	Instrumentation and controls	7
e.	Electric power	8

3.1.2.4.5 Criterion 34 - Residual Heat Removal

> A system to remove residual heat shall be provided. The system safety function shall be to transfer fission product decay heat and other residual heat from the reactor core at a rate such that specified acceptable fuel design limits and the design conditions of the reactor coolant pressure boundary are not exceeded.
>
> Suitable redundancy in components and features, and suitable interconnections, leak detection, and isolation capabilities shall be provided to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished, assuming a single failure (GDC 34).

Evaluation Against Criterion 34

The RHR system provides the means to remove decay heat and residual heat from the nuclear system so that refueling and nuclear system servicing can be performed.

The major equipment of the RHR system consists of heat exchangers cooled by the service water system and main system pumps. The equipment is connected by associated valves and piping, and the controls and instrumentation are provided for proper system operation.

The RHR system consists of two modes of operation that provide the shutdown cooling function. One mode, the normal Shutdown Cooling Mode, is the preferred operational mode. Although preferred, this mode of RHR does not meet the redundancy/single failure requirements of IEEE 279 and 10 CFR 50 Appendix A, GDC 34. As a result, a second shutdown cooling mode, the Alternate Shutdown Cooling Mode, is available and is the shutdown cooling mode credited to meet the requirements of IEEE 279 and GDC 34. This mode is safety related, Quality Class 1, Seismic Category 1, redundant and single failure proof. Since the normal Shutdown Cooling Mode of RHR is preferred for CGS, the components required for the operation of this mode are maintained as safety related, Quality Class 1. Refer to Section 5.4.7 for a complete discussion of the shutdown cooling modes of RHR.

Both normal ac power and the auxiliary onsite power system provide adequate power to operate all the auxiliary loads necessary for plant operation. The power sources for the plant auxiliary power system are sufficient in number and of such electrical and physical independence that no single probable event could interrupt all auxiliary power at one time.

The plant auxiliary buses which supply power to ESF, RPS, and auxiliaries required for safe shutdown are connected by appropriate switching to the standby diesel generators.

Each power source, up to the connection to the auxiliary power buses, is capable of complete and rapid isolation from any other source.

Loads important to plant operation and safety are split and diversified between switchgear sections, and means are provided for detection and isolation of system faults.

The plant layout is designed to physically separate essential bus sections, standby diesel generators, switchgear, interconnections, feeders, power centers, motor control centers, and other system components.

Standby diesel generators are provided to supply electrical power from within the plant that is not dependent on external sources. The standby generators produce ac power at a voltage and frequency compatible with the normal bus requirements for essential equipment within the plant. Each of the diesel generators has sufficient capacity to start and carry the essential loads it is expected to drive.

The RHR system is adequate to remove residual heat from the reactor core to ensure that fuel and RCPB design limits are not exceeded. Redundant reactor coolant circulation paths are available to and from the vessel and RHR system. Redundant onsite electric power systems are provided. The design of the RHR system, including its power supply, meets the requirements of Criterion 34.

For further discussion see the following sections:

a.	RHR system	5.4
b.	ECCS	6.3
c.	ESF systems	7.3
d.	Auxiliary power system	8.3
e.	Water systems	9.2
f.	Accident analyses	15

3.1.2.4.6 Criterion 35 - Emergency Core Cooling

A system to provide abundant emergency core cooling shall be provided. The system safety function shall be to transfer heat from the reactor core following any loss of reactor coolant at a rate such that (1) fuel and clad damage that could interfere with continued effective core cooling is prevented, and (2) clad metal-water reaction is limited to negligible amounts.

Suitable redundancy in components and features, and suitable interconnections, leak detection, isolation, and containment capabilities shall be provided to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished, assuming a single failure (GDC 35).

Evaluation Against Criterion 35

The ECCS, consists of the following:

a. High-pressure core spray system (HPCS),

b. Automatic depressurization system (ADS),

c. Low-pressure core spray (LPCS) system, and

d. Low-pressure coolant injection (LPCI) (an operating mode of the RHR system).

The ECCS are designed to limit fuel cladding temperature over the complete spectrum of possible break sizes in the RCPB including a complete and sudden circumferential rupture of the largest pipe connected to the reactor vessel.

The HPCS system consists of a single motor-driven pump, system piping, valves, controls, and instrumentation. The HPCS system is provided to ensure that the reactor core is adequately cooled to prevent excessive fuel clad temperatures for breaks in the nuclear system that do not result in rapid depressurization of the reactor vessel. The HPCS continues to operate when reactor vessel pressure is below the pressure at which LPCI operation or LPCS operation maintains core cooling. A source of water is available from either the condensate storage tank or the suppression pool.

The automatic depressurization system functions to reduce the reactor pressure so that flow from LPCI and the LPCS enters the reactor vessel in time to cool the core and prevent excessive fuel clad temperature. The ADS uses several of the main steam safety/relief valves (MSRVs) to relieve the high pressure steam to the suppression pool.

The LPCS system consists of (a) a centrifugal pump that can be powered by normal auxiliary power or the standby ac power system, (b) a spray sparger in the reactor vessel, above the core (separate from the HPCS sparger), (c) piping and valves to convey water from the suppression pool to the sparger, and (d) associated controls and instrumentation. In case of low water level in the reactor vessel or high pressure in the drywell, the LPCS automatically sprays water onto the top of the fuel assemblies in time and at a sufficient flow rate to cool the core and prevent excessive fuel temperature. The LPCI system starts from the same signals which initiate the LPCS system and operate independently to achieve the same objective by flooding the reactor vessel.

In case of low water level in the reactor or high pressure in the drywell, the LPCI mode of operation of the RHR system pumps water into the reactor vessel in time to flood the core and prevent excessive fuel temperature. Protection provided by LPCI extends to a small break where the ADS has operated to lower the reactor vessel pressure.

Results of the performance of the ECCS for the entire spectrum of liquid line breaks are discussed in Section 6.3. Peak cladding temperatures are well below the 2200°F design basis.

Also provided in Section 6.3.3 is an analysis to show that the ECCS conform to 10 CFR Part 50, Appendix K. This analysis shows complete compliance with the Final Acceptance Criteria with the following results:

a. Peak clad temperatures are well below the 2200°F NRC acceptability limit,

b. The amount of fuel cladding reacting with steam is nearly an order of magnitude below the 1% acceptability limit,

c. The clad temperature transient is terminated while core geometry is still amenable to cooling, and

d. The core temperature is reduced and the decay heat can be removed for an extended period of time.

The redundancy and capability of the onsite electrical power systems for the ECCS are represented in the evaluation against Criterion 34.

The ECCS provided are adequate to prevent fuel and clad damage that could interfere with effective core cooling and limit clad metal-water reaction to a negligible amount. The design of the ECCS, including their power supply, meets the requirements of Criterion 35.

For further discussion, see the following sections:

a.	RHR system	5.4.7
b.	ECCS	6.3
c.	Onsite power systems	8.3
d.	Water systems	9.2
e.	Accident analyses	15

3.1.2.4.7 Criterion 36 - Inspection of Emergency Core Cooling System

The emergency core cooling system shall be designed to permit appropriate periodic inspection of important components, such as spray rings in the reactor pressure vessel, water injection nozzles, and piping, to assure the integrity and capability of the system (GDC 36).

Evaluation Against Criterion 36

The ECCS discussed in Criterion 36 include inservice inspection considerations. The spray spargers within the vessel are accessible for inspection during each refueling outage. Removable plugs in the reactor shield and/or panels in the insulation provide access for examination of nozzles. Removable insulation is provided on the ECCS piping out to and including the first isolation valve outside the drywell. Inspection of the ECCS is in accordance with the intent of Section XI of the ASME Code. Sections 5.2.4 and 6.6 define the Inservice Inspection Program Plan, access provisions, and areas of restricted access.

During plant operations, the pumps, valves, piping, instrumentation, wiring, and other components outside the primary containment can be visually inspected at any time. Components inside the primary containment can be inspected when the primary containment is open for access. When the reactor vessel is open, for refueling or other purposes, the spargers and other internals can be inspected. Portions of the ECCS which are part of the RCPB are designed to specifications for inservice inspection to detect defects which might affect the cooling performance. Particular attention will be given to the reactor nozzles, core spray, and feedwater spargers. The design of the reactor vessel and internals and the plant testing and inspection program ensure that the requirements of Criterion 36 are met.

For further discussion, see the following sections:

a.	Reactor pressure vessel internals	3.9.5
b.	Inservice inspection and testing of RCPB	5.2.4
c.	Reactor vessel	5.3
d.	ECCS	6.3
e.	Inservice inspection of Class 2 and 3 components	6.6

3.1.2.4.8 Criterion 37 - Testing of Emergency Core Cooling System

> The emergency core cooling system shall be designed to permit appropriate periodic pressure and functional testing to assure (1) the structural and leaktight integrity of its components, (2) the operability and performance of the active components of the system, and (3) the operability of the system as a whole and, under conditions as close to design as practical, the performance of the full operational sequence that brings the system into operation, including operation of applicable portions of the protection system, the transfer between normal and emergency power sources, and the operation of the associated cooling water system (GDC 37).

Evaluation Against Criterion 37

The ECCS consists of the HPCS system, ADS, the LPCI mode of the RHR system, and LPCS system. Each of these systems is provided with test connections and isolation valves to permit periodic pressure testing to ensure the structural and leaktight integrity of its components.

The HPCS, LPCS, LPCI systems, and the ADS are designed to permit periodic testing to ensure the operability and performance of the active components of each system.

The pumps and valves of these systems are tested and flow rate tests conducted periodically to verify operability in accordance with the CGS Inservice Testing Program Plan and Technical Specifications. Flow rate tests are conducted on HPCS, LPCS, and LPCI systems.

The ECCS are tested under conditions as close to design as practical to verify the performance of the full operational sequence that brings each system into operation, in accordance with the Technical Specifications surveillance requirements. The operation of the associated cooling water systems is discussed in the evaluation of Criterion 46. The design of the ECCS meets the requirements of Criterion 37.

For further discussion, see the following sections:

a.	Overpressurization protection	5.2.2
b.	Tests and inspections	6.3.4
c.	ESF systems	7.3
d.	Electric power	8

3.1.2.4.9 Criterion 38 - Containment Heat Removal

> A system to remove heat from the reactor containment shall be provided. The system safety function shall be to reduce rapidly, consistent with the functioning of other associated systems, the containment pressure and temperature following any loss-of-coolant accident and maintain them at acceptably low levels.
>
> Suitable redundancy in components and features, and suitable interconnections, leak detection, isolation, and containment capabilities shall be provided to assure that for onsite electrical power system operation (assuming offsite power is not available) and for offsite electrical power system operation (assuming onsite power is not available) the system safety function can be accomplished, assuming a single failure (GDC 38).

<u>Evaluation Against Criterion 38</u>

In the event of a LOCA within the drywell, the pressure suppression system will rapidly condense the steam to prevent overpressure. An increase in pressure in the drywell from a leak in the RCPB is relieved below the surface of the suppression pool and the steam portion of the gases so vented are condensed by contact with the suppression pool water. Cooling systems remove heat from the core and from the water in the suppression pool during accident conditions and thus provide continuous cooling of the drywell.

The ECCS provides core cooling in the event of a LOCA. Low water level in the reactor vessel or high pressure in the drywell will initiate the ECCS to prevent excessive fuel temperature. Sufficient water is provided in the suppression pool to accommodate the initial energy which can be released from the postulated pipe failure.

The containment heat removal function is accomplished by the RHR system. Following a LOCA, one or both of the following operating modes of the RHR system would be initiated:

a. Containment spray - condenses steam within the containment, and

b. Suppression pool cooling - limits the temperature within the containment by removing heat from the suppression pool water by way of the RHR heat exchangers. Either or both redundant RHR heat exchangers can be manually activated.

The redundancy and capability of the offsite and onsite electrical power systems for the RHR system are presented in the evaluation against Criterion 34.

For further discussion, see the following sections:

a.	RHR system	5.4.7
b.	Containment systems	6.2
c.	ECCS	6.3
d.	ESF systems	7.3
e.	Electric power	8
f.	Standby service water system	9.2.7
g.	Accident analyses	15

3.1.2.4.10 Criterion 39 - Inspection of Containment Heat Removal System

The containment heat removal system shall be designed to permit appropriate periodic inspection of important components, such as the torus, sumps, spray nozzles, and piping to assure the integrity and capability of the system (GDC 39).

Evaluation Against Criterion 39

Provisions are made to facilitate periodic inspections of active components and other important equipment of the drywell pressure reducing systems. During plant operations, the pumps, valves, piping, instrumentation, wiring, and other components outside the drywell can be visually inspected at any time and will be inspected periodically. The testing frequencies of most components will be correlated with the component inspection.

The suppression pool is designed to permit periodic inspection. The containment heat removal system is designed to permit periodic inspection of major components. This design meets the requirements of Criterion 39.

For further discussion, see the following sections:

a.	RHR system	5.4.7
b.	Containment systems	6.2
c.	ECCS	6.3
d.	ESF systems	7.3
e.	Standby service water system	9.2.7

3.1.2.4.11 Criterion 40 - Testing of Containment Heat Removal System

The containment heat removal system shall be designed to permit appropriate periodic pressure and functional testing to assure (1) the structural and leaktight integrity of its components, (2) the operability and performance of the active components of the systems, and (3) the operability of the system as a whole, and under conditions as close to the design as practical, the performance of the full operational sequence that brings the system into operation, including operation of applicable portions of the protection system, the transfer between normal and emergency power sources, and the operation of the associated cooling water system (GDC 40).

Evaluation Against Criterion 40

The containment heat removal function is accomplished by the suppression pool cooling mode of the RHR system.

The RHR system is provided with sufficient test connections and isolation valves to permit periodic pressure testing. The pumps and valves of the RHR system are operated periodically to verify operability in accordance with the Inservice Testing Program Plan. The suppression pool cooling mode is not automatically initiated, and the operation of the components of this system is periodically verified. The operation of associated cooling water systems is discussed in the response to Criterion 46. It is concluded that the requirements of Criterion 40 are met.

For further discussion see the following sections:

a.	RHR system	5.4.7
b.	ECCS	6.3
c.	Electric power	8

3.1.2.4.12 Criterion 41 - Containment Atmosphere Cleanup

> Systems to control fission products, hydrogen, oxygen, and other substances which may be released into the reactor containment shall be provided as necessary to reduce, consistent with the functioning of other associated systems, the concentration and quantity of fission products released to the environment following postulated accidents, and to control the concentration of hydrogen or oxygen and other substances in the containment atmosphere following postulated accidents to assure that containment integrity is maintained.
>
> Each system shall have suitable redundancy in components and features, and suitable interconnections, leak detection, isolation, and containment capabilities to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) its safety can be accomplished, assuming a single failure (GDC 41).

Evaluation Against Criterion 41

A primary containment that provides a barrier to control the release of fission products, hydrogen, oxygen, and other substances to the atmosphere is totally enclosed within the reactor building. Following an accident the reactor building is maintained at a negative pressure with respect to atmosphere to ensure that any leakage from the primary containment does not leak to atmosphere but is retained in the reactor building. The air exhausted from the reactor building to maintain it at a negative pressure is processed through the SGTS, which contains both HEPA and charcoal filters to minimize the release of radioactivity to the environs. The primary containment atmosphere can also be purged through the standby gas treatment for cleanup prior to entry of personnel. The SGTS is composed of two fully redundant trains of filters and fans, with all active components powered from the emergency diesel buses.

The above described systems meet the requirements of Criterion 41.

For further discussion see the following sections:

a.	Principal design criteria	1.2
b.	Containment functional design	6.2.1
c.	Secondary containment functional design	6.2.3
d.	Combustible gas control in containment	6.2.5
e.	Fission product removal and control systems	6.5
f.	Instrumentation and controls	7

3.1.2.4.13 Criterion 42 - Inspection of Containment Atmosphere Cleanup Systems

> The containment atmosphere cleanup systems shall be designed to permit appropriate periodic inspection of important components, such as filter frames, ducts, and piping to assure the integrity and capability of the systems (GDC 42).

<u>Evaluation Against Criterion 42</u>

All components of the SGTS are located in the reactor building and are accessible for inspection during normal plant operation. The design of the system therefore meets the requirements of Criterion 42 (see Section 6.5).

3.1.2.4.14 Criterion 43 - Testing of Containment Atmosphere Cleanup Systems

> The containment atmosphere cleanup systems shall be designed to permit appropriate periodic pressure and functional testing to assure (1) the structural and leaktight integrity of its components, (2) the operability and performance of the active components of the systems such as fans, filters, dampers, pumps, and valves, and (3) the operability of the systems as a whole and, under conditions as close to design as practical, the performance of the full operational sequence that brings the system into operation, including operation of applicable portions of the protection system, the transfer between normal and emergency power sources, and the operation of associated systems (GDC 43).

Evaluation Against Criterion 43

As discussed under the evaluation of Criterion 42, all components of the containment atmosphere cleanup system are accessible for tests and inspections during normal plant operation thereby meeting the requirements of Criterion 43. A detailed discussion of the periodic tests that will be performed to verify system operability is given in Section 6.5.

3.1.2.4.15 Criterion 44 - Cooling Water

> A system to transfer heat from structures, systems, and components important to safety, to an ultimate heat sink shall be provided. The system safety function shall be to transfer the combined heat load of these structures, systems, and components under normal operating and accident conditions.
>
> Suitable redundancy in components and features, and suitable interconnections, leak detection, and isolation capabilities shall be provided to assure that for onsite electrical power system operation (assuming offsite electrical power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished, assuming a single failure (GDC 44).

Evaluation Against Criterion 44

The safety-related cooling water system is the standby service water (SW) system, which supplies cooling for the RHR, HPCS, LPCS, fuel pool cooling and cleanup (FPC) system, and the essential HVAC systems.

The redundant SW systems are open loop systems which transfer heat from structures, systems, and safety-related components to the ultimate heat sink.

The ultimate heat sink consists of two man-made Seismic Category I spray ponds and is designed to withstand extreme natural phenomena.

The piping, valves, pumps, and heat exchangers of the SW system are designed and arranged so that the system safety function can be performed assuming a single failure. Electrical power is supplied to the system from offsite or onsite emergency power sources, with distribution arranged such that a single failure will not prevent the system from performing its safety function.

For further discussion, see the following sections:

a. Principal design criteria 1.2

b. Classification of structures, components, and systems 3.2

c. Wind and tornado loadings 3.3

d. Water level (flood) design 3.4

e. Missile protection 3.5

f. Design of Seismic Category I structures 3.8

g. Ultimate heat sink 9.2.5

h. Standby service water system 9.2.7

i. Electrical power 8

3.1.2.4.16 Criterion 45 - Inspection of Cooling Water System

> The cooling water system shall be designed to permit appropriate periodic inspection of important components, such as heat exchangers and piping, to assure the integrity and capability of the system (GDC 45).

Evaluation Against Criterion 45

The SW system is designed to permit periodic inspection of important components, including pumps, strainers, heat exchangers, and isolation valves, to ensure the integrity and capability of the system.

All important components are located in accessible locations to facilitate periodic inspection during normal plant operation. Suitable manholes, handholes, inspection ports, or other design and layout features are provided for this purpose.

These features meet the requirements of Criterion 45.

For further discussion, see the following sections:

a. Principal design criteria 1.2

b. Inservice inspection of Class 2 and 3 components 6.6

c. Standby service water system 9.2.7

d. Initial test program 14

3.1.2.4.17 Criterion 46 - Testing of Cooling Water System

The cooling water system shall be designed to permit appropriate periodic pressure and functional testing to assure (1) the structural and leaktight integrity of its components, (2) the operability and performance of the active components of the system, and (3) operability of the system as a whole and, under conditions as close to design as practical, the performance of the full operational sequence that brings the system into operation for reactor shutdown and for loss-of-coolant accidents, including operation of applicable portions of the protection system and the transfer between normal and emergency power sources (GDC 46).

Evaluation Against Criterion 46

The SW system is designed to permit system operability testing with simulation of emergency reactor shutdown or LOCA conditions and transfer between normal and emergency power sources.

For further discussion, see the following sections:

a. Principal design criteria 1.2

b. Standby service water system 9.2.7

c. Electric power 8

d. Initial test program 14

3.1.2.5 Group V - Containment (Criteria 50-57)

3.1.2.5.1 Criterion 50 - Containment Design Basis

The reactor containment structure, including access openings, penetrations, and containment heat removal system shall be designed so that the containment structure and its internal compartments can accommodate, without exceeding the design leakage rate and with sufficient margin, the calculated pressure and temperature conditions resulting from any loss-of-coolant accident. This margin shall reflect consideration of (1) the effects of potential energy sources which have not been included in the determination of the peak conditions, such as energy in steam generators and energy from metal-water and other chemical reactions that may result from degraded emergency core cooling functioning, (2) the limited experience and experimental data available for defining accident phenomena and containment responses, and (3) the conservatism of the calculational model and input parameters (GDC 50).

Evaluation Against Criterion 50

The containment structure, including access openings and penetrations, are designed to accommodate, without exceeding the design leak rate, the transient peak pressure and temperature associated with a LOCA up to and including a double-ended rupture of the largest reactor coolant pipe.

The containment structure and ESF systems have been evaluated for various combinations of energy release. The analysis accounts for system thermal energy, chemical energy, and nuclear decay heat energy.

The maximum temperature and pressure reached in the drywell and containment during the worst case accident are shown in Chapters 6 and 15 to be below the design temperature and pressure of this structure.

The cooling capacity of the containment heat removal systems is adequate to prevent overpressurization of the structure and to return the containment to near atmospheric pressure.

For further discussion, see the following sections:

a.	Classification of structures, components, and systems	3.2
b.	Wind and tornado loadings	3.3
c.	Missile protection	3.5
d.	Protection against dynamic effects associated with the postulated rupture of piping	3.6
e.	Seismic design	3.7
f.	Steel containment	3.8
g.	Containment systems	6.2
h.	Containment heat removal systems	6.2.2
i.	Accident analyses	15

3.1.2.5.2 Criterion 51 - Fracture Prevention of Containment Pressure Boundary

> The reactor containment boundary shall be designed with sufficient margin to assure that under operating, maintenance, testing, and postulated accident conditions (1) its ferritic materials behave in a nonbrittle manner and (2) the probability of rapidly propagating fracture is minimized. The design shall reflect consideration of service temperatures and other conditions of the containment boundary material during operation, maintenance, testing, and postulated accident conditions, and the uncertainties in determining (1) material properties, (2) residual, steady-state, and transient stresses, and (3) size of flaws (GDC 51).

Evaluation Against Criterion 51

The containment vessel was designed, fabricated, inspected, and tested to meet the requirements of ASME Code Section III Subsection NE (1971 edition through Summer 1972 Addenda). Material for the containment was qualified by impact testing at a temperature which is at least 30°F below the minimum service temperature of +30°F. Means are provided by an auxiliary boiler to maintain the reactor building and consequently the containment temperature at a suitable level during a shutdown of the unit during cold weather.

To demonstrate that the primary containment pressure boundary meets the requirements of Criterion 51, the Class 1, Class 2, and Class MC components of the CGS containment pressure boundary were reviewed according to the fracture toughness requirements of the Summer 1977 Addenda of Section III for Class 2 components and fracture toughness data presented in NUREG-0577, "Potential for Low Fracture Toughness and Lamellar Tearing of PWR Steam Generator and Reactor Coolant Pump Supports." Based on review of the available fracture toughness data and material fabrication histories, and the use of correlations between metallurgical characteristics and material fracture toughness, it was concluded that the ferritic materials in the CGS containment pressure boundary meet the fracture toughness requirements that are specified for Class 2 components by the 1977 Addenda of Section III of the ASME Code. Compliance with these code requirements provide reasonable assurance that the CGS reactor containment pressure boundary materials will behave in a nonbrittle manner, that the probability of rapidly propagating fracture will be minimized, and that the requirements of Criterion 51 are satisfied.

The preoperational test program and the quality assurance program ensure the integrity of the containment and its ability to meet all normal operating and accident requirements.

The containment design thus meets the requirements of Criterion 51.

For further discussion, see the following sections:

a.	Steel containment	3.8.2
b.	Reactor building ventilation system including spent fuel pool area ventilation system	9.4
c.	Initial test program	14
d.	Quality assurance	17

3.1.2.5.3 Criterion 52 - Capability for Containment Leakage Rate Testing

The reactor containment and other equipment which may be subjected to containment test conditions shall be designed so that periodic integrated leakage rate testing can be conducted at containment design pressure (GDC 52).

Evaluation Against Criterion 52

The containment system is designed and constructed and the necessary equipment is provided to permit periodic integrated leak rate tests during the plant lifetime. The testing program will be conducted in accordance with Appendix J to 10 CFR Part 50.

For further discussion, see the following sections:

a.	Leak rate tests	3.8.2
b.	Containment leakage testing	6.2.6

3.1.2.5.4 Criterion 53 - Provisions for Containment Testing and Inspection

The reactor containment shall be designed to permit (1) appropriate periodic inspection of all important areas, such as penetrations, (2) an appropriate surveillance program, and (c) periodic testing at containment design pressure of the leaktightness of penetrations which have resilient seals and expansions bellows (GDC 53).

Evaluation Against Criterion 53

The reactor containment design permits periodic inspection of the exposed interior surfaces. It also includes provisions for periodic testing of the leaktightness of all penetrations and inserts in the reactor containment pressure boundary.

The containment design therefore meets the requirements of Criterion 53.

For further discussion, see the following sections:

a.	Steel containment	3.8.2
b.	Containment functional design	6.2.1
c.	Containment heat removal systems	6.2.2
d.	Containment leakage testing	6.2.6
e.	Inservice inspection of Class 2 and 3 components	6.6

3.1.2.5.5 Criterion 54 - Piping Systems Penetrating Containment

> Piping systems penetrating primary reactor containment shall be provided with leak detection, isolation, and containment capabilities having redundancy, reliability, and performance capabilities which reflect the importance to safety of isolating these piping systems. Such piping systems shall be designed with a capability to test periodically the operability of the isolation valves and associated apparatus and to determine if valve leakage is within acceptable limits (GDC 54).

Evaluation Against Criterion 54

Piping systems penetrating containment are designed to provide the required isolation and testing capabilities. These piping systems are provided with test connections to allow periodic leak detection tests to be performed.

The ESF actuation system test circuitry provides the means for testing isolation valve operability.

Conformance with Criterion 54 is further discussed in Sections 3.1.2.5.6 (Criterion 55), 3.1.2.5.7 (Criterion 56), and 3.1.2.5.8 (Criterion 57).

3.1.2.5.6 Criterion 55 - Reactor Coolant Pressure Boundary Penetrating Containment

> Each line that is part of the reactor coolant pressure boundary and that penetrates primary reactor containment shall be provided with containment isolation valves as follows, unless it can be demonstrated that the containment isolation provisions for a specific class of lines, such as instrument lines, are acceptable on some other defined basis:

(1) One locked closed isolation valve inside and one locked closed isolation valve outside containment; or

(2) One automatic isolation valve inside and one locked closed isolation valve outside containment; or

(3) One locked closed isolation valve inside and one automatic isolation valve outside containment. A simple check valve may not be used as the automatic isolation valve outside containment; or

(4) One automatic isolation valve inside and one automatic isolation valve outside containment. A simple check valve may not be used as the automatic isolation valve outside containment.

Isolation valves outside containment shall be located as close to the containment as practical and upon loss of actuating power, automatic isolation valves shall be designed to take the position that provides greater safety.

Other appropriate requirements to minimize the probability or consequences of an accidental rupture of these lines or of lines connected to them shall be provided as necessary to assure adequate safety. Determination of the appropriateness of these requirements, such as higher quality in design, fabrication, and testing, additional provisions for inservice inspection, protection against more severe natural phenomena, and additional isolation valves and containment, shall include consideration of the population density, use characteristics, and physical characteristics of the site environs (GDC 55).

Evaluation Against Criterion 55

The RCPB [as defined in 10 CFR Part 50, Section 50.2 (v)] consists of the RPV, pressure retaining appurtenances attached to the vessel, valves, and pipes which extend from the RPV up to and including the outermost isolation valve. The RCPB lines which penetrate the containment have isolation valves capable of isolating the containment to preclude any significant releases of radioactivity. Lines which do not penetrate the containment but which form a portion of the RCPB, can be isolated from the RCPB.

The design of the isolation systems described meet the requirements of Criterion 55.

For further discussion, see the following sections:

a. Integrity of RCPB 5.2

b. Containment isolation system 6.2.4

c. Instrumentation and controls 7

d. Accident analyses 15

3.1.2.5.7 Criterion 56 - Primary Containment Isolation

Each line that connects directly to the containment atmosphere and penetrates primary reactor containment shall be provided with containment isolation valves as follows, unless it can be demonstrated that the containment isolation provisions for a specific class of lines, such as instrument lines, are acceptable on some other defined basis:

(1) One locked closed isolation valve inside and one locked closed isolation valve outside containment; or

(2) One automatic isolation valve inside and one locked closed isolation valve outside containment; or

(3) One locked closed isolation valve inside and one automatic isolation valve outside containment. A simple check valve may not be used as the automatic isolation valve outside containment; or

(4) One automatic isolation valve inside and one automatic isolation valve outside containment. A simple check valve may not be used as the automatic isolation valve outside containment.

Isolation valves outside containment shall be located as close to the containment as practical and upon loss of actuating power, automatic isolation valves shall be designed to take the position that provides greater safety (GDC 56).

Evaluation Against Criterion 56

Criterion 56 requires that lines which penetrate the containment and communicate with the containment interior must have two isolation valves; one inside the containment, the other outside. It should be noted that this criterion does not reflect consideration of the BWR suppression pool design. For instance, those lines which connect to the suppression pool do not have an isolation valve located inside the containment as this would necessitate placement of the valve underwater. In effect, this would result in introducing a potentially unreliable valve in a highly reliable system thereby compromising design.

The manner in which the containment isolation system meets this requirement is described in the following sections:

a.	Containment isolation system	6.2.4
b.	Instrumentation and control systems	7
c.	Primary containment and reactor vessel isolation control system instrumentation and controls	7.3
d.	Accident analyses	15

3.1.2.5.8 Criterion 57 - Closed System Isolation Valves

Each line that penetrates primary reactor containment and is neither part of the reactor coolant pressure boundary nor connected directly to the containment atmosphere shall have at least one containment isolation valve which shall be either automatic, or locked closed, or capable of remote manual operation. This valve shall be outside containment and located as close to the containment as practical. A simple check valve may not be used as the automatic isolation valve (GDC 57).

Evaluation Against Criterion 57

Each line that penetrates the reactor containment and is neither part of the RCPB nor connected directly to the containment atmosphere has at least one containment isolation valve which is automatic, locked closed or capable of remote manual operation, located outside the containment as close to the containment as practical. Simple check valves are not used as automatic isolation valves on these lines.

Details demonstrating conformance with Criterion 57 are provided in the following sections:

a.	Containment isolation system	6.2.4
b.	Instrumentation and controls	7

3.1.2.6 Group VI - Fuel and Reactivity Control

3.1.2.6.1 Criterion 60 - Control of Releases of Radioactive Materials to the Environment

The nuclear power unit design shall include means to control suitably the release of radioactive materials in gaseous and liquid effluents and to handle radioactive solid wastes produced during normal reactor operation, including anticipated operational occurrences. Sufficient holdup capacity shall be provided for retention of gaseous and liquid effluents containing radioactive materials, particularly where unfavorable site environmental conditions can be expected to impose unusual operational limitations upon the release of such effluents to the environment (GDC 60).

Evaluation Against Criterion 60

Waste handling systems have been incorporated in the plant design for processing and/or retention of radioactive wastes from normal plant operations to ensure that the effluent releases to the environment are as low as reasonably achievable (ALARA) and within the limits of 10 CFR Part 20. The plant is also designed to prevent radioactivity releases during accidents from exceeding the limits of 10 CFR Part 100.

The principal gaseous effluents from the plant during normal operation are the noncondensable gases from the air ejectors. These gases are exhausted through a hold up system and offgas treatment system including charcoal adsorbers. The effluent from this system is continuously monitored and controlled and the system will be shutdown and isolated in the event of abnormally high radiation levels.

Ventilation air from the various plant areas is continuously monitored and controlled. In the event of an accident inside containment, noncondensable gases are held within the leaktight containment vessel. Release of these effluents will be by controlled purging of the containment.

Liquid radioactive wastes are collected in waste collector tanks, treated on a batch basis through demineralizers, and then either returned to the plant systems or released in a controlled manner to the environment. All discharges to the environment are routed through a monitoring station that continuously monitors and records the activity of the waste and provides an alarm to the operator in the unlikely event of high activity level.

Solid wastes including spent resins, filter sludges, filter cartridges, evaporator bottoms, contaminated tools, equipment, and clothing are collected, packaged, and shipped offsite in approved shipping containers.

The design of the waste disposal systems meets the requirements of Criterion 60.

For further discussion, see the following sections:

a. Principal design criteria 1.2

b. Detection of leakage through RCPB 5.2.5

c.	Containment functional design	6.2.1
d.	Heating, ventilating, and air conditioning systems	9.4
e.	Liquid waste management systems	11.2
f.	Gaseous waste management systems	11.3
g.	Solid waste management system	11.4
h.	Process and effluent radiological monitoring and sampling systems	11.5
i.	Radiation protection	12
j.	Accident analyses	15

3.1.2.6.2 Criterion 61 - Fuel Storage and Handling and Radioactivity Control

> The fuel storage and handling, radioactive waste, and other systems which may contain radioactivity shall be designed to assure adequate safety under normal and postulated accident conditions. These systems shall be designed (1) with a capability to permit appropriate periodic inspection and testing of components important to safety, (2) with suitable shielding for radiation protection, (3) with appropriate containment, confinement, and filtering systems, (4) with residual heat removal capability having reliability and testability that reflects the importance to safety of decay heat and other RHR, and (5) to prevent significant reduction in fuel storage coolant inventory under accident conditions (GDC 61).

Evaluation Against Criterion 61

3.1.2.6.2.1 New Fuel Storage. New fuel is placed in dry storage in the new fuel storage vault which is located inside the reactor building. The storage vault within the reactor building provides adequate shielding for radiation protection. Storage racks preclude accidental criticality (see evaluation against Criterion 62). The new fuel storage racks do not require any special inspection and testing for nuclear safety purposes.

3.1.2.6.2.2 Spent Fuel Handling and Storage. Irradiated fuel is stored in the spent fuel pool in the reactor building. Fuel pool water is circulated through the FPC system to maintain fuel pool water temperature, purity, water clarity, and water level. Storage racks preclude accidental critically (see evaluation against Criterion 62).

Reliable decay heat removal is provided by the closed loop FPC system. It consists of two circulating pumps, two heat exchangers, two filter-demineralizers, two skimmer surge tanks, and the associated piping, valves, and instrumentation. The pool water is circulated through the system, suction is taken from fuel pool skimmer surge tanks, flow passes in series through the heat exchangers and filters, and it is discharged through diffusers at the bottom of the fuel pool. Normal source of cooling water to the heat exchangers is from the reactor closed cooling water (RCC) system. When the preferred RCC source is unavailable, the SW system provides cooling water and makeup. The FPC system can also be interconnected with the RHR system to increase the cooling capacity of the FPC system. See Section 9.1.3.

The portion of the system required for cooling of spent fuel after refueling is designed to Seismic Category I requirements and can be isolated by automatic redundant, Seismic Category I isolation valves from the Seismic Category II cleanup portion of the system. Safety-grade cooling water from the SW system is available to the shell side of the FPC heat exchangers by remote-manual operation from the control room. Safety-grade makeup water to the fuel pool is also available from the SW system by remote-manual operation from the control room.

High- and low-level switches indicate pool water level changes in the main control room. Fission product concentration in the pool water is minimized by use of the filter-demineralizer. This minimizes the release of radioactivity from the pool to the reactor building environment.

No special tests are required to ensure system operability, except as noted below, because at least one pump, with a heat exchanger, and filter-demineralizer is routinely in operation while fuel is stored in the pool. Routine visual inspection of the system components, instrumentation, and trouble alarms is adequate to verify system operability.

Service water flow to the shell side of the fuel pool cooling heat exchangers, including operation of the valves that interconnect the service water and RCC systems to the FPC system, are tested in conjunction with testing of the service water system. The service water makeup isolation valves are also tested in conjunction with testing of the service water system.

3.1.2.6.2.3 Radioactive Waste Systems. The radioactive waste systems provide all equipment necessary to collect, process, and prepare for disposal of all radioactive liquids, gases, and solid waste produced as a result of reactor operation.

Liquid radwastes are segregated and treated as equipment drain, floor drain, chemical, detergent, sludges, or concentrated wastes. Processing methods include filtration, ion exchange, neutralization, concentration, solidification, analysis, and dilution. Liquid wastes are also decanted and sludge is accumulated for disposal as solid radwaste. Wet solid wastes and concentrates are normally dewatered and packaged in steel or polyethylene containers as required for disposal. Dry solid radwastes are packaged in steel boxes or other suitable containers. Gaseous radwastes are monitored, processed, recorded, and controlled so that radiation doses to persons outside the controlled area are below those allowed by applicable regulations.

Accessible portions of the refueling and radwaste buildings have sufficient shielding to maintain dose rates within the limits set forth in 10 CFR Part 20 and 10 CFR Part 50. The radwaste building is designed to preclude accidental release of radioactive materials to the environs.

The radwaste systems are used on a routine basis and do not require specific testing to ensure operability. Performance is monitored by radiation monitors during operation.

The fuel storage and handling and radioactive waste systems are designed to ensure adequate safety under normal and postulated accident conditions. The design of these systems meets the requirements of Criterion 61.

For further discussion, see the following sections:

a.	RHR system	5.4.7
b.	Containment systems	6.2
c.	New fuel storage	9.1.1
d.	Spent fuel storage	9.1.2
e.	Spent fuel pool cooling and cleanup system	9.1.3
f.	Heating, ventilating, and air conditioning systems	9.4
g.	Radioactive waste management	11
h.	Radiation protection	12

3.1.2.6.3 Criterion 62 - Prevention of Criticality in Fuel Storage and Handling

> Criticality in the fuel storage and handling system shall be prevented by physical systems or processes, preferably by use of geometrically safe configurations (GDC 62).

<u>Evaluation Against Criterion 62</u>

Appropriate plant fuel handling and storage facilities are provided to preclude accidental criticality for new and spent fuel. Criticality in new and spent fuel storage is prevented by the geometrically safe configuration of the new fuel storage racks and by the geometrically safe configuration of the spent fuel racks using fixed poison. There is sufficient spacing between assemblies to ensure that the array when fully loaded is substantially subcritical. Fuel elements are limited to top loading and fuel assembly positions by rack design. The new and spent fuel racks are Seismic Category I components.

Spent fuel and new fuel is stored under water in the spent fuel pool. The racks in which spent fuel assemblies are placed are designed to ensure subcriticality in the storage pool. Fuel is maintained at a subcritical multiplication factor k_{eff} of less than 0.95 under both normal and abnormal conditions. Abnormal criticality conditions may result for abnormal conditions. Abnormal criticality conditions may result from accidental dropping of a fuel assembly to a location adjacent to or on top of a loaded fuel storage rack. Various abnormal loading conditions such as loadings from the operating basis earthquake (OBE) and safe shutdown earthquake (SSE), the dropping of a fuel assembly, or a stuck assembly exerting an upward force on the racks were also considered in the design of the racks.

Refueling interlocks include circuitry which senses conditions of the refueling equipment and the control rods. These interlocks reinforce operational procedures that prohibit making the reactor critical. The fuel handling system is designed to provide a safe, effective means of transporting and handling fuel and to minimize the possibility of mishandling or maloperation.

The use of geometrically safe configurations for new and spent fuel storage, the use of fixed neutron poison in the spent fuel racks, and the design of fuel handling systems precludes accidental criticality in accord with Criterion 62.

For further discussion, see the following sections:

a. Refueling interlocks system instrumentation and controls 7.7.1.13

b. New fuel storage 9.1.1

c. Spent fuel storage 9.1.2

d. Fuel handling system 9.1.4

3.1.2.6.4 Criterion 63 - Monitoring Fuel and Waste Storage

Appropriate systems shall be provided in fuel storage and radioactive waste systems and associated handling areas, (1) to detect conditions that may result in loss of residual heat removal capability and excessive radiation levels and (2) to initiate appropriate safety actions (GDC 63).

Evaluation Against Criterion 63

Appropriate systems have been provided to meet the requirements of this criterion. A malfunction of the FPC system which could result in loss of RHR capability and excessive radiation levels is alarmed in the control room. Alarmed conditions include high/low level in the fuel pool, cooling water pump discharge pressure low, and high/low level in the fuel storage pool skimmer surge tanks. System temperature is also continuously monitored and alarmed in the control room. The area radiation monitors sense radioactivity in this area and initiate an alarm in the control room on abnormal radiation levels.

Area radiation, tank, and sump levels are monitored and alarmed to give indication of conditions that may result in excessive radiation levels in radioactive waste system areas. These systems satisfy the requirements of Criterion 63.

For further discussion, see the following sections:

a. Area radiation monitoring system instrumentation 12.3.4

b. Fuel storage and handling 9.1

c. Radioactive waste management 11

d. Radiation protection 12

3.1.2.6.5 Criterion 64 - Monitoring Radioactivity Releases

Means shall be provided for monitoring the reactor containment atmosphere, spaces containing components for recirculation of loss-of-coolant accident fluids, effluent discharge paths, and the plant environs for radioactivity that may be released from normal operations, including anticipated operational occurrences, and from postulated accidents (GDC 64).

Evaluation Against Criterion 64

Means have been provided for monitoring radioactivity releases resulting from normal and anticipated operational occurrences. The following potential station release paths are monitored:

a. Main plant vent monitor,

b. Turbine building ventilation exhaust,

c. Radwaste building ventilation exhaust,

d. Liquid radwaste system effluent,

e. Standby service water system,

f. Plant service water,

g. Containment monitoring system radiation monitors, and

h. Turbine building sumps.

In addition, offsite monitors are provided.

For further discussion of the means and equipment used for monitoring radioactivity releases, see the following sections:

a.	Detection of leakage through RCPB	5.2.5
b.	Process radiation monitoring system instrumentation and controls	7.6
c.	Process and effluent radiological monitoring systems and sampling systems	11.5
d.	Area radiation and airborne radioactivity monitoring instrumentation	12.3.4

3.2 CLASSIFICATION OF STRUCTURES, COMPONENTS, AND SYSTEMS

Appendix A of 10 CFR Part 50 requires that all nuclear power plants be designed so that, if the safe shutdown earthquake (SSE) occurs, all structures, systems, and components (SSCs) important to safety remain functional.

General Design Criterion 1 (GDC 1) of Appendix A to 10 CFR Part 50 requires that SSCs important to safety be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety function to be performed.

3.2.1 SEISMIC CLASSIFICATION

Plant structures, systems, and components important to safety are designed to withstand the effects of an SSE and remain functional if they are necessary to ensure

a. The integrity of the reactor coolant pressure boundary (RCPB),

b. The capability to shut down the reactor and maintain it in a safe shutdown condition, or

c. The capability to prevent or mitigate the consequences of accidents which could result in potential offsite exposure comparable to the exposure limits of 10 CFR Part 50.67.

Plant structures, systems, equipment, and components, including their foundations and supports, designed to remain functional in the event of an SSE are designated as Seismic Category I, as indicated in Table 3.2-1.

All Seismic Category I SSCs are analyzed under the loading conditions of the SSE and the operating basis earthquake (OBE). Since the two earthquakes vary in intensity, the design of Seismic Category I structures, components, equipment, and systems to resist each earthquake and other loads are based on levels of material stress or load factors, whichever is applicable, with margins of safety appropriate for each earthquake. The margin of safety provided for Seismic Category I structures, components, equipment, and systems for the SSE are sufficient to ensure that their design functions are not jeopardized.

For further details of seismic design criteria refer to

Seismic design	3.7
Design of Category I structures	3.8
Systems and components	3.9

Seismic qualification of Seismic Category I instrumentation and electrical equipment 3.10

All SSCs not analyzed under the loading conditions of the SSE and OBE are classified Seismic Category II. Where applicable, the seismic loading conditions as determined from the Uniform Building Code are used for the design of Seismic Category II SSCs.

The OBE, as defined in 10 CFR Part 100, Appendix A, is not incorporated as part of the seismic classification scheme.

The seismic classification indicated in Table 3.2-1 meets the requirements of NRC Regulatory Guide 1.29, Revision 3, except as otherwise noted in the table.

Seismic Category 1M classification denotes systems, structures or components that are designed and constructed to comply with position C.2 of Regulatory Guide 1.29.

3.2.2 SYSTEM QUALITY GROUP CLASSIFICATIONS

System quality group classifications, as defined in NRC Regulatory Guide 1.26, Revision 3, are determined for each water, steam, or radioactive waste containing component of those applicable fluid systems relied on to ensure

a. The integrity of the RCPB,

b. The capability to shut down the reactor and maintain it in a safe shutdown condition, or

c. The capability to prevent or mitigate the consequences of accidents which could result in potential offsite exposure comparable to the exposure limits of 10 CFR Part 50.67.

The quality group classifications for systems and components which meet these definitions are shown in Table 3.2-1.

System quality group classifications, as indicated in Tables 3.2-1 and 3.2-2, meet the requirements of 10 CFR 50.55a and Regulatory Guide 1.26, Revision 3.

3.2.3 SAFETY CLASSIFICATIONS

Structures, systems, and components were classified as Safety Class 1, Safety Class 2, Safety Class 3, or General Class G in accordance with the ANS-22 definition of their importance to nuclear safety (ANS-22 became ANSI/ANS-52.1-1983). Recognizing that components within a system may be of differing safety importance, a single system may have components in more than one safety class. Supports are designed in accordance with ASME III-Winter 1973 Addenda-Subsection NF and are classified appropriately based on the component supported. Jurisdictional boundaries between NF and other codes were established by the owner as required by the ASME Code. Load combinations and allowable stresses used in support design are in accordance with ASME III-NF as defined in Section 3.9 and are applied to both ASME and the American Institute of Steel Construction (AISC) support members. Modifications and construction practices are controlled by the quality classification of the individual support and by the additional requirements of each applicable code. Table 3.2-1 provides a summary of the safety classes for the principal SSCs of the plant.

Design requirements for components of safety classes are also described in this section. Reference is made to accepted industry codes and standards which define design requirements commensurate with the safety function(s) performed.

3.2.3.1 Safety Class 1

3.2.3.1.1 Definition of Safety Class 1

Safety Class 1 applies to components of the RCPB or core support structure whose failure could cause a loss of reactor coolant at a rate in excess of the normal makeup system or to equipment in which a single failure could cause major fuel damage.

3.2.3.1.2 Design Requirements for Safety Class 1

The design requirements for Safety Class 1 mechanical equipment (i.e., vessel, pipes, pumps, and valves) are described in Section 3.9. The design requirements for the reactor vessel are described in Section 5.3.

The design requirements for Safety Class 1 structures (i.e., reactor pedestal and supports of the RCPB as defined in Section 3.2.3.1.1) are specified in Sections 3.8 and 3.9.

Safety Class 1 SSCs are identified in Table 3.2-1.

3.2.3.2 Safety Class 2

3.2.3.2.1 Definition of Safety Class 2

Safety Class 2 applies to those SSCs, other than service water systems, that are not Safety Class 1 but are necessary to accomplish the safety function of

a. Inserting negative reactivity to shut down the reactor,
b. Preventing rapid insertion of positive reactivity,

c. Maintaining core geometry appropriate to all plant process conditions,
d. Providing emergency core cooling,
e. Providing and maintaining containment, and
f. Removing residual heat from the reactor and reactor core.

3.2.3.2.2 Design Requirements for Safety Class 2

The design requirements for Safety Class 2 mechanical equipment (i.e., vessels including heat exchangers, pipes, pumps, valves, and tanks) are described in Section 3.9.

The design requirements for Safety Class 2 structures (i.e., reactor building including the primary containment) are specified in Section 3.8.

The seismic and environmental design requirements for Safety Class 2 instrumentation and electrical equipment are specified in Sections 3.10 and 3.11.

Safety Class 2 SSCs are identified in Table 3.2-1.

3.2.3.3 Safety Class 3

3.2.3.3.1 Definition of Safety Class 3

Safety Class 3 applies to those SSCs that are not Safety Class 1 nor Safety Class 2 that is relied upon to accomplish a nuclear safety function.

3.2.3.3.2 Design Requirement for Safety Class 3

The design requirements for Safety Class 3 structures, (i.e., standby service water pump houses, diesel generator buildings, radwaste/control building, and spray ponds) are specified in Section 3.8.

The design requirements for Safety Class 3 mechanical equipment (i.e., vessels including heat exchangers, pipes, pumps, valves, and tanks) are described in Section 3.9.

The seismic and environmental design requirements for Safety Class 3 instrumentation and electrical equipment are specified in Sections 3.10 and 3.11.

3.2.3.4 General Class G, Structures, Systems, and Components

3.2.3.4.1 Definition of General Class Structures, Systems, and Components

A boiling water reactor (BWR) has a number of SSCs in the power conversion or other portions of the facility which have no direct safety function but which may be connected to or influenced by the equipment within the safety classes defined above. Such SSCs are designated as General Class G. For example, portions of the service water systems, the turbine generator auxiliaries, and portions of the heating, ventilating, and air conditioning (HVAC) systems are designated as having no safety classification.

3.2.3.4.2 Design Requirements for General Class G Structures, Systems, and Components

The design requirements for equipment classified as General Class G are specified with appropriate consideration of the intended service of the equipment and expected plant and environmental conditions under which it will operate. Design requirements are based on applicable industry codes and standards.

3.2.4 QUALITY ASSURANCE CLASSIFICATION

Structures, systems, and components are classified in Table 3.2-1 using the following quality class designations

a. Quality Class I - Any nuclear system, structure, subassembly, component, or design characteristics that prevent or mitigate the consequences of postulated accidents that could cause undue risk to the health and safety of the public. All engineered safety features fall within this category. All Quality Class I items meet the applicable provisions of 10 CFR Part 50 Appendix B.

b. Quality Class II+ - Assigned to SSCs having no safety-related function but requiring quality augmentation either as a result of NRC requirements or as committed by CGS. Quality augmentation may include such requirements as environmental qualification, seismic qualification, or other quality affecting activities specifically committed. Augmented quality applies to the following categories of SSCs:

   1. Essential fire protection SSCs,

   2. Structures, systems, and components that do not perform a safety-related function but must be seismically supported/mounted (seismic 2-over-1) per Regulatory Guide 1.29,

   3. Structures, systems, and components required for radwaste management that are subject to Regulatory Guide 1.143,

   4. Structures, systems, and components required to cope with a station blackout per Regulatory Guide 1.155 (see Section 8A),

5. Structures, systems, and components required to respond to or mitigate anticipated transients without scram (ATWS) per the requirements of 10 CFR 50.62,

6. Structures, systems, and components required to respond to an electrical separation safe shutdown event (see Section 8.3.1.4.4.1.4),

7. Postaccident monitoring instruments subject to Regulatory Guide 1.97, Category 2, requirements, and

8. Remote shutdown items required in response to control room evacuation (10 CFR 50 Appendix A, GDC 19).

c. Quality Class II - Any system, structure, subassembly, component, or design characteristic that could cause a safety hazard to plant personnel, an extended reduction in unit output, an unscheduled unit trip, or equipment damage. Appropriate quality assurance requirements for these items are assigned in the purchase specifications.

d. Quality Class G - Any nonnuclear system, structure, subassembly, component, or design characteristic to which quality assurance requirements are assigned in accordance with the consequences of failure, operating costs, or procurement costs.

3.2.5 CORRELATION OF SAFETY CLASSES WITH OTHER DESIGN REQUIREMENTS

The design of plant equipment is commensurate with the safety importance of the equipment. Hence, the various safety classes have a gradation of design requirements. The correlation of safety classes with other design requirements is summarized in Table 3.2-3.

3.2.6 IDENTIFICATION OF SAFETY-RELATED SYSTEMS AND COMPONENTS ON FLOW DIAGRAMS AND IN THE MASTER EQUIPMENT LIST

The system classification are shown on the flow diagrams using symbols code group A, B, C, D, D+; Seismic Category I, II; and Quality Class I, II, II+, and G. The Master Equipment List (MEL) uses arabic number designators for quality class and seismic category. The following is a comparison of the classification designators:

Flow Diagram	Table 3.2-1	MEL
Code Group A	Quality Group A	
Code Group B	Quality Group B	
Code Group C	Quality Group C	
Code Group D	Quality Group D	
Code Group D+	Quality Group D+	
Seismic Category I	Seismic Category I*	Seismic Category 1*
		Seismic Category 1M*
Seismic Category II	Seismic Category II*	Seismic Category 2*
Quality Class I	Quality Class I	Quality Class 1
Quality Class II	Quality Class II	Quality Class 2
Quality Class II+	Quality Class II+	Quality Class A
Quality Class G	Quality Class G	Quality Class G

Flow diagrams which include the seismic and code classifications assigned to each piping system are identified in Table 3.2-1. The flow diagrams delineate the boundary of seismic and code classifications for each system and present the as-built classifications. For the appropriate quality class of specific components, refer to the MEL. In some instances, these classifications reflect voluntary upgrades which may exceed Regulatory Guide 1.26, Revision 3, requirements. For inservice inspection requirements, the appropriate levels of inspection are given in Sections 5.2.4 and 6.6.

* Table 3.2-1 may indicate Seismic Category I, 1M or II. Clarification of the Seismic 2 over 1 (i.e., 1M) support/mounting requirements is specified in the notes and/or MEL (see Table 3.2-1).

Table 3.2-1

Equipment Classification

Principal Component[a]	Scope of Supply[b]	Safety Class[c]	Location[d]	Quality Group Classification[e]	Quality Class[f]	Seismic Category[g]	Notes[h]
1. Reactor system							
Reactor vessel	GE	1	C	A	I	I	
Reactor vessel support skirt	GE	1	C	N/A	I	I	
Reactor vessel appurtenances pressure retaining portions	GE	1	C	A	I	I	
Control rod drive housing supports	GE	2	C	N/A	I	I	
Reactor internal structures, engineered safety features	GE	2	C	N/A	I	I	
Reactor internal structures, other	GE	G	C	N/A	N/A	N/A	
Control rods	GE	2	C	N/A	I	I	
Control rod drives	GE	2	C	N/A	I	I	
Core support structure	GE	2	C	N/A	I	I	
Power range detector hardware	GE	2	C	B	I	I	
Fuel assemblies	P	1	C	N/A	I	I	
Reactor vessel stabilizer	GE	2	C	N/A	I	I	
2. Nuclear boiler system (Figure 10.3-2)							
Vessels, level instrumentation condensing chambers	GE	1	C	A	I	I	
Vessels	P	2	C	B	I	I	
Piping, relief valve discharge from relief valve to suppression pool	P	3	C	C	I	I	
Piping, relief valve discharge within suppression chamber and suppression pool	P	2	C	B	I	I	1
Piping, main steam and feedwater within outermost isolation valve	GE/P	1	C,R	A	I	I	
Pipe supports, main steam	GE	1	C,R	A	I	I	
Pipe restraints, main steam	P	2	C,R	N/A	I	I	
Piping, other within outermost isolation valves	P	1	C,R	A	I	I	2
Safety/relief valves	GE	1	C	A	I	I	
Valves, main steam isolation valves	GE		C,R	A	I	I	
Valves, other, isolation valves and within containment	P	1	C	A	I	I	

Table 3.2-1

Equipment Classification (Continued)

Principal Component[a]	Scope of Supply[b]	Safety Class[c]	Location[d]	Quality Group Classification[e]	Quality Class[f]	Seismic Category[g]	Notes[h]
Piping and valves outboard of outside isolation valve to and including first valve capable of timely actuation	P	2	R	B	I	I	2
Mechanical modules, instrumentation (with safety function)	GE	2	C,R	N/A	I	I	
Electrical modules (with safety function)	GE	2	C,R	N/A	I	I	
Cable (with safety function)	P	2	C,R,W	N/A	I	I	
3. Reactor recirculation system (Figure 5.4-7)							
Piping	GE	1	C	A	I	I	2
Pipe suspension, recirculation line	GE	1	C	N/A	I	I	
Pipe restraints, recirculation line	GE	2	C	N/A	I	I	
Pumps	GE	1	C	A	I	I	
Valves	GE	1	C	A	I	I	2
Motor, pump	GE	3	C	N/A	II+	I/IM	37
Electrical modules (with safety function)	GE	2	C,R	N/A	I	I	
Cable (with safety function)	P	2	C,R,W	N/A	I	I	
4. Control rod drive hydraulic system (Figure 4.6-5)							
Valves, scram discharge volume lines	GE	2	R	B	I	I	2
Valves, insert and withdraw lines	GE	2	R	B	I	I	3
Valves, other	P	G	R	D	II	II	2,4
Piping discharge volume lines	P	2	R	B	I	I	
Piping insert and withdraw lines	GE/P	2	C,R	B	I	I	
Piping, other	P	G	R	D	II	II	2,4
Hydraulic control unit	GE	2	R	Special	I	I	5
Electrical modules (with safety function)	GE	2	R	N/A	I	I	
Cables (with safety function)	P	2	C,R,W	N/A	I	I	

Table 3.2-1

Equipment Classification (Continued)

Principal Component[a]	Scope of Supply[b]	Safety Class[c]	Location[d]	Quality Group Classification[e]	Quality Class[f]	Seismic Category[g]	Notes[h]
5. Standby liquid control system (Figure 9.3-14)							6
Standby liquid control tank	GE	2	R	B	II+	I	
Pump	GE	2	R	B	II+	I	
Pump motor	GE	2	R	N/A	II+	I	
Valves, explosive	GE	1	R	A	I	I	
Valves, isolation and within containment	P	1	C	A	I	I	2
Valves, beyond isolation valves	P	2	R	B	II+	I	2
Piping, within isolation valves to reactor vessel	P	1	C,R	A	I	I	2
Piping, beyond isolation valves	P	2	R	B	II+	I	2
Electrical modules (with safety function)	GE	2	C	N/A	I	I	
Cable (with safety function)	P	2	C,R,W	N/A	I	I	
Electrical modules, with boron injection function	GE	2	C	N/A	II+	I	
Cable, with boron injection function	P	2	C,R,W	N/A	II+	I	
6. Neutron monitoring system							
Tubing TIP	GE/P	2/G	C	B/D	I/II	I/II	
Electrical modules, IRM and APRM	GE	2	C,R	N/A	I	I	
Cable, IRM and APRM	P	2	C,R,W	N/A	I	I	
Valves, isolation TIP subsystem	GE/P	2	C,R	B	I	I	
Power range detector hardware	GE	2	R	B	I	I	
7. Reactor protection							
Electrical modules	GE	2	C,R,T	N/A	I	I	
Cable	P	2	C,R,T,W	N/A	I	I	
8. Leak detection system							7
Temperature element	GE	2	C,R	N/A	I	I	
Differential temperature switch	GE	2	C,R	NA/	I	I	
Differential flow indicator	GE	2	C,R	N/A	I	I	
Pressure switch	GE	2	C,R	N/A	I	I	

Table 3.2-1

Equipment Classification (Continued)

Principal Component[a]	Scope of Supply[b]	Safety Class[c]	Location[d]	Quality Group Classification[e]	Quality Class[f]	Seismic Category[g]	Notes[h]
Differential pressure indicator switch	GE	2	C,R	N/A	I	I	
Differential flow summer	GE	2	C,R	N/A	I	I	
9. Process radiation monitors							
Electrical modules, main steam line and building ventilation monitors	GE	2	C,R,T,W	N/A	I	I	
Cable, main steam line and reactor building ventilation monitors	P	2	C,R,T,W	N/A	I	I	
10. Residual heat removal system (Figure 5.4-15)							
Heat exchangers, primary side	GE	2	R	B	I	I	
Heat exchanger, secondary side	GE	3	R	C	I	I	
Piping, within outermost isolation valves, reactor coolant pressure boundary	P	1	C,R	A	I	I	
Piping, other	P	2	R	B	I	I	8
Pumps	GE	2	R	B	I	I	
Water leg pumps	P	2	R	B	I	I	
Main Pump motors	GE	2	R	N/A	I	I	
Valves, isolation reactor coolant pressure boundary	P	1	C,R	A	I	I	
Valves, other	P	2	R	B	I	I	2
Mechanical modules	GE	2	R	B	I	I	
Electrical modules (with safety function)	GE	2	R	N/A	I	I	
Cable (with safety function)	P	2	C,R,W	N/A	I	I	
11. Low-pressure core spray (Figure 6.3-4)							
Piping, within outermost isolation valves to reactor vessel	P	1	C,R	A	I	I	2
Piping, beyond outermost isolation valves	P	2	R	B	I	I	2
Pumps	GE	2	R	B	I	I	
Water leg pumps	P	2	R	B	I	I	
Main Pump motors	GE	2	R	N/A	I	I	

Table 3.2-1

Equipment Classification (Continued)

Principal Component[a]	Scope of Supply[b]	Safety Class[c]	Location[d]	Quality Group Classification[e]	Quality Class[f]	Seismic Category[g]	Notes[h]
Valves, isolation, reactor coolant pressure boundary	P	1	C	A	I	I	2
Valves, other	P	2	C,R	B	I	I	2
Electrical modules (with safety function)	GE	2	R	N/A	I	I	
Cable (with safety function)	P	2	R,W	N/A	I	I	
12. High-pressure core spray (Figure 6.3-4)							
Piping, within outermost isolation valve	P	1	C,R	A	I	I	2
Piping, return test line to condensate storage tank beyond second isolation valve	P	G	R,O	D	II	II	4
Piping, beyond outermost isolation valve, other	P	2	R	B	I	I	2
Pump	GE/P	2	R	B	I	I	
Water leg pumps	P	2	R	B	I	I	
Main Pump motor	GE	2	R	N/A	I	I	
Valves, beyond diesel shutoff valves	P	3	P	C	I	I	
Valves, isolation, reactor coolant pressure boundary	P	1	C	A	I	I	
Valves, beyond isolation valves, motor operated	GE	2	R	B	I	I	2
Valves, other	P	2	R,P	B	I	I	
Electrical modules (with safety function)	GE	2	R	N/A	I	I	
Electrical auxiliary equipment	GE	3	DG	N/A	I	I	
Cable (with safety function)	P	2	W,R	N/A	I	I	
(See 38a - high-pressure core spray standby power supply)							
13. Reactor core isolation cooling system (Figure 5.4-11)							
Piping, within outermost isolation valves reactor coolant pressure boundary	P	1	C,R	A	I	I	2
Piping, beyond outermost isolation valves	P	2	R	B	I	I	2,10
Piping, drip pot discharge valve to condenser	P	G	R	D	II	II	4

Table 3.2-1

Equipment Classification (Continued)

Principal Component[a]	Scope of Supply[b]	Safety Class[c]	Location[d]	Quality Group Classification[e]	Quality Class[f]	Seismic Category[g]	Notes[h]
Piping, condenser to vacuum tank and to the condensate pump discharge; and vacuum pump discharge to the outboard check valve break flange	P	G	R	D	I	I	2
Pump	GE	2	R	B/D	I	I	
Water leg pumps	P	2	R	B	I	I	2
Valves, containment isolation and coolant pressure boundary	P	1	C	A	I	I	
Valves, other (with safety function)	P	2	R	B/D	I	I	
Turbine	GE	2	R	N/A	I	I	9
Electrical modules (with safety function)	GE	2	R	N/A	I	I	
Cable (with safety function)	P	2	R,W	N/A	I	I	
Reactor core isolation cooling electrohydraulic system	GE	2	R	N/A	I	I	
14. Fuel service equipment							
Fuel preparation machine	GE	3	R	N/A	I	I	
General purpose grapple	GE	3	R	N/A	I	I	
New fuel inspection stand	GE	G	R	N/A	II+	1M	
15. Refueling equipment							
Refueling platform	GE	G	R	N/A	II+	1M	
Refueling bellows	P	G	C,R	D	II+	I	11
16. Storage equipment							
Fuel storage racks	GE/P	3	R	N/A	I	I	
17. Radwaste system (Figures 11.2-2, 9.3-9, 9.3-12, 11.2-3, and 11.2-4)							2,4,13, 14,36
Tanks, atmospheric	GE/P	G	W	D+	II+	II	
Heat exchangers	GE/P	G	W	D+	II+	II	

Table 3.2-1

Equipment Classification (Continued)

Principal Component[a]	Scope of Supply[b]	Safety Class[c]	Location[d]	Quality Group Classification[e]	Quality Class[f]	Seismic Category[g]	Notes[h]
Piping and valves forming part of containment boundary	P	2	C,R	B	I	I	
Piping and valves forming part of secondary containment	P	3	R	C,D	I,II+	I	36
Piping, other	P	G	W	D+	II+	II	
Pumps	GE/P	G	W	D+	II+	II	
Valves, flow control and filter system	GE/P	G	W	D+	II+	II	
Valves, other	P	G	W	D+	II+	II	
Mechanical modules	GE/P	G	W	D+	II+	II	
Radioactive equipment and floor drains and other radwaste piping and valves upstream of collector tanks	P	G	R,T,W	D+	II+	II	4
Instrumentation and control boards	GE/P	G	W	N/A	II+	II	
Concentrator	GE	G	W	D+	II+	II	
Plant discharge line	GE/P	G	W	D	II	II	15
18. Reactor water cleanup system (Figure 5.4-22)							
Vessels, filter/demineralizer	GE	G	W	C	II	II	
Heat exchangers	GE	G	R	C	II	II	
Piping, within outermost isolation valves	P	1	C	A	I	I	2
Piping, beyond outermost containment isolation valves	P	G	R,W	C,D,D+	II,II+	II	2,4
Pumps	GE	G	R,W	C,D+	II,II+	II	2,4
Valves, isolation valves reactor coolant pressure boundary	P/GE	1	C,R	A	I	I	2
Valves, beyond outermost containment isolation valves	GE/P	G	R,W	C,D+	II,II+	II	2,4
Mechanical modules	GE	G	R,W	C,D+	II,II+	II	4
19. Fuel pool cooling and cleanup system (Figure 9.1-6)							
a. Cooling							
Vessels	P	3	R	C	II	I	
Heat exchangers	P	3	R	C	II	I	16
Piping	P	3	R	C	II	I	17
Pumps	P	3	R	C	II	I	18
Makeup system (normal)	P	G	R	C	II	II	4,19

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Table 3.2-1

Equipment Classification (Continued)

Principal Component[a]	Scope of Supply[b]	Safety Class[c]	Location[d]	Quality Group Classification[e]	Quality Class[f]	Seismic Category[g]	Notes[h]
RHR connection	P	3	R	C	I	I	19
Makeup system (emergency)	P	3	R	C	I	I	19
b. Cleanup							
Vessels, filter/demineralizer	P	G	W	C	II	II	
Piping	P	G	R,W	C,D+	II,II+	II	4
Pumps	P	G	W	C	II	II	
Piping, suppression pool to outer isolation valves	P	2	R	B	I	I	
20. Control room panels							
Electrical modules (with safety function)	GE	2	W	N/A	I	I	
Cable (with safety function)	GE/P	2	W	N/A	I	I	
21. Local panels and racks							
Electrical modules (with safety function)	GE	2	R	N/A	I	I	
Cable (with safety function)	P	2	R	N/A	I	I	
22. Offgas system (Figure 11.3-2)							4
Tanks	GE	G	T,W	D+	II+	II	
Heat exchangers	GE	G	T,W	D+	II+	II	
Piping	P	G	T,W,O	D+	II+	II	
Pumps	GE	G	T,W	D+	II+	II	
Valves	P	G	T,W	D+	II+	II	
Mechanical modules (with safety function)	GE	G	T,W	C	II	II	
Pressure vessels	GE	G	T,W	D+	II+	II	
23. Standby service water system (Figure 9.2-12)							
Piping	P	3	P,R,DG,O	C	I	I	
Pumps	GE	3	P	C	I	I	
Pump motors	GE	3	P	N/A	I	I	
Valves	P	3	P,R,DG,O	C	I	I	
Electrical modules (with safety function)	P	3	P,R,DG,O,W	N/A	I	I	
Cable (with safety function)	P	3	P,R,DG,O,W	N/A	N/A	I	

Table 3.2-1

Equipment Classification (Continued)

Principal Component[a]	Scope of Supply[b]	Safety Class[c]	Location[d]	Quality Group Classification[e]	Quality Class[f]	Seismic Category[g]	Notes[h]
24. Turbine plant service water (Figure 9.2-1)							
Piping and valves	P	G	A,T,R,O,P, W,S	D	II,II+	II	4
Pumps	P	G	P	D	N/A	II	
25. Reactor building closed cooling water system (Figure 9.2-2)							4
Heat exchangers	P	G	R	D	II	II	
Pumps	P	G	R	D	II	II	
Tanks	P	G	R	D	II	II	
Piping and valves inside containment	P	G	C	C	II	II	
Containment isolation valves and associated piping	P	2	C,R	B	I	I	
Piping and valves in reactor building	P	G	R	D	II	II	
Piping and valves, other	P	G	W	D	II	II	
26. Primary containment cooling system (Figure 9.4-8)							
Piping and valves up to outermost isolation valves, containment purge and exhaust	P	2	C,R	B	I	I	
27. Standby gas treatment system (Figure 3.2-2)							
Filter units	P	2	R	B	I	I	
Fans	P	2	R	B	I	I	
Piping and valves	P	2	R	B	I	I	20
28. Primary containment atmospheric control system (DEACTIVATED) (Figure 3.2-3)							
Piping and valves	P	2	C,R	B	I	I	
Equipment	P	2	R	B	I	I	
29. Other heating, ventilating, and air conditioning (Figures 9.4-1, 9.4-2, and 9.4-7)							
Reactor building (nonessential)	P	G	R	N/A	II	II	4
Reactor building (essential)	P	3	R	N/A	I	I	

Table 3.2-1

Equipment Classification (Continued)

Principal Component[a]	Scope of Supply[b]	Safety Class[c]	Location[d]	Quality Group Classification[e]	Quality Class[f]	Seismic Category[g]	Notes[h]
Turbine building	P	G	G	N/A	II	II	21
Radwaste building	P	G	W	N/A	II	II	21
Control room, critical switchgear area, cable spreading area (nonessential)	P	G	W	N/A	II	II	4
Control room, critical switchgear area, cable spreading area (essential)	P	3	W	N/A	I	I	
Diesel generator building	P	3	DG	N/A	I	I	22
Standby service water pump house	P	3	P	N/A	I	I	22
30. Condensate storage and transfer (Figure 9.2-11)							
Condensate storage tank	P	G	O	C	II	II	23
Piping and valves	P	G	O,T,R,W	D	II	II	4,24
Pumps	P	G	O	D	II	II	
31. Instrument and sample lines							2
32. Fuel storage facilities							
Fuel pool/dryer separator liner	P	3	R	N/A	II	I	
Storage racks and supports	GE	3	R	N/A	I	I	
33. Building cranes							
Reactor building	P	3	R	N/A	I	I	
Turbine building	P	G	T	N/A	II	II	
Radwaste building	P	G	W	N/A	II	II	
Standby service water pump house	P	G	P	N/A	II+	1M	
Miscellaneous	P	G	P,W,T,S	N/A	II	II	
34. Instrument and service air (Figures 9.3-1.1, 9.3-1.2, 9.3-1.3, 9.3-1.4 and 9.3-1.5)							
Piping and valves	P	G	R,W,T,O	D	II	II	4
Compressors	P	G	T	D	II	II	
Vessels	P	G	T	D	II	II	
Compressor cooling system	P	G	T	D	II	II	

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Table 3.2-1

Equipment Classification (Continued)

Principal Component[a]	Scope of Supply[b]	Safety Class[c]	Location[d]	Quality Group Classification[e]	Quality Class[f]	Seismic Category[g]	Notes[h]
35. Containment instrument air system (Figure 9.3-2)							
Piping and valves inside containment to and including outboard isolation valve	P	2	C,R	B	I	I	
Piping and valves to main steam relief valves	P	2	R	B	I	I	
Other piping and valves	P	G	R	D	II	II	4
Receiver	P	G	R	D	II	II	4
Piping and valves outside containment isolation valves to and including solenoid pilot valves controlling supply of nitrogen from the nitrogen bottles	P	3	R	C	I	I	
36. High-pressure core spray standby power systems (Division 3)							
Day tanks	P	3	DG	C	I	I	
Piping and valves, fuel oil system	P	3	DG	C	I	I	
Pumps, fuel oil system	P	3	DG	C	I	I	
Diesel generators	GE	2	DG	N/A	I	I	25
Electrical modules (with safety functions)	GE	2	DG	N/A	I	I	
Cable (with safety functions)	P	3	DG	N/A	I	I	
Diesel fuel storage tanks	P	3	DG	C	I	I	
Diesel generators service water supply	P	3	P	C	I	I	
Diesel starting air	P	3	DG	D	I,II	I	26,27
Diesel intake exhaust piping	P	3	DG	D	I	I	26
Diesel jacket water cooling	GE	3	DG	D	I	I	26
37. Standby ac power systems (Divisions 1 and 2)							
Storage and day tanks	P	3	DG	C	I	I	
Piping and valves, diesel oil	P	3	DG	C	I	I	
Pumps, diesel oil	P	3	DG	C	I	I	
Diesel generators	P	2	DG	N/A	I	I	25
Electrical modules (with safety function)	P	3	DG	N/A	I	I	
Diesel cooling water supply	P	3	DG	C	I	I	
Cable (with safety function)	P	3	DG	N/A	I	I	

Table 3.2-1

Equipment Classification (Continued)

Principal Component[a]	Scope of Supply[b]	Safety Class[c]	Location[d]	Quality Group Classification[e]	Quality Class[f]	Seismic Category[g]	Notes[h]
Diesel intake/exhaust air piping	P	3	DG	D	I	I	26
Diesel starting air	P	3	DG	D	I,II	I	26,27
Diesel jacket water cooling	P	3	DG	D	I	I	26
38. Auxiliary ac power system							
Essential components	P	2	W,R,DG	N/A	I	I	
Nonessential components	P	G	W,R,T,O	N/A	II	II	4
39. Auxiliary 125/250-V dc power system							
Batteries	P	2	W	N/A	I	I	
Battery Chargers	P	3	W	N/A	I	I	
Cables	P	2	W,R	N/A	I	I	
Modules	P	2	W,R	N/A	I	I	
40. 24-V dc power system							
Batteries	P	2	W	N/A	I	I	
Battery chargers	P	2	W	N/A	I	I	
Cables	P	2	W,R	N/A	I	I	
Modules	P	2	W,R	N/A	I	I	
41. 120-V critical power supply system							
Equipment	P	2	W,R	N/A	I	I	
42. Power conversion system (Figures 10.3-1 and 3.2-4)							
Main steam piping from outermost isolation valves up to turbine stop valves	P	2	R,T	B	I	I	28
Main steam branch piping to first valve capable of timely closure	P	2	T	B	I	I	28
Main steam piping downstream of MS-V-146 including turbine bypass piping, and steam piping down stream of first valve capable of timely closure	P	G	T	D	II	II	

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Table 3.2-1

Equipment Classification (Continued)

Principal Component[a]	Scope of Supply[b]	Safety Class[c]	Location[d]	Quality Group Classification[e]	Quality Class[f]	Seismic Category[g]	Notes[h]
Turbine stop valves, turbine control valves, and turbine bypass valves	P	G	T	D	II	II	29,30
Main steam leads from turbine control valve to turbine casing	P	G	T	D	II	II	30
Feedwater and condensate system beyond outermost isolation valve	P	G	R,T	D	II	II	4
Turbine generator	P	G	T	D	II	II	
Condenser	P	G	T	D	II	II	
Air ejection equipment	P	G	T	D	II	II	
Feedwater treatment system	P	G	T	D	II	II	
Turbine bypass system beyond turbine bypass valve	P	G	T	D	II	II	
Turbine gland sealing system components	P	G	T	D	II	II	
Piping, valves, other	P	G	T	D	II	II	
Equipment, other	P	G	T	D	II	II	
43. Circulating water and cooling tower makeup water system(s) (Figure 10.4-4)							
Piping and valves	P	G	P	D	II	II	31
Pumps	P	G	P	D	II	II	
Cooling tower fans	P	G	P	D	II	II	
44. Main steam isolation valves leakage control system (DEACTIVATED) (Figure 3.2-5)							
Piping and valves within primary containment and out through the outermost isolation valves	*P*	*1*	*R,C*	*A*	*I*	*I*	
Piping and valves beyond the outermost isolation valves	*P*	*2*	*R*	*B*	*I*	*I*	
Blowers	*P*	*2*	*R*	*N/A*	*I*	*I*	
45. Containment vessel	P	2	R	B	I	I	

Table 3.2-1

Equipment Classification (Continued)

Principal Component[a]	Scope of Supply[b]	Safety Class[c]	Location[d]	Quality Group Classification[e]	Quality Class[f]	Seismic Category[g]	Notes[h]
46. Buildings							
Reactor building	N/A	2	R	N/A	I	I	
Turbine building	N/A	G	T	N/A	II	II	32
Radwaste control building	N/A	3/G	W	N/A	I,II	I/II	33
Diesel generator building	N/A	3	DG	N/A	I	I	
Spray ponds and standby service water pump house	N/A	3	P	N/A	I	I	
Service building	N/A	G	S	N/A	II	II	
Cooling towers	N/A	G	O	N/A	II	II	
Makeup water pump house	N/A	G	O	N/A	I	II	31
Circulation water pump house	N/A	G	O	N/A	II	II	
Air intake structures No. 1 and No. 2	N/A	3	O	N/A	I	I	
47 Containment/drywell atmosphere monitoring system	P	3	R	A	I	I	
48. Drywell insulation							
Insulation on piping which is within the drywell	P	G	R	N/A	I	I	
49. Instrumentation and control equipment							
Safety-related instrumentation and control systems	P	1	C,R,T,DG	A	I	I	34
50. Postaccident sampling system (Figure 3.2-6)							
Piping within outermost reactor coolant boundary isolation valves	P	1	C,R	A	I	I	
Piping within outermost containment isolation valves	P	2	C,R	B	I	I	
Piping within the outermost RHR system isolation valves	P	2	R	B	I	I	
Piping beyond the outermost reactor coolant boundary, containment, or RHR system isolation valves	P	G	R,W	D	II	II	
Sample station	GE	G	R,W	D	G	II	
All other	P	G	R,W	D	II	I	

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Table 3.2-1

Equipment Classification (Continued)

Principal Component[a]	Scope of Supply[b]	Safety Class[c]	Location[d]	Quality Group Classification[e]	Quality Class[f]	Seismic Category[g]	Notes[h]
51. Hydrogen Water Chemistry System							
Hydrogen and air injection piping and valves near and inside TGB (Figure 10.4-9.1)	GE/P	G	O,T	D	II	II	
52. Hydrogen Storage and Supply Facility (HSSF) (Figures 10.4-9.2, 10.4-9.3)							
Tubing and valves	P	G	H	D	II	II	
Buried piping from HSSF to TGB	P	G	H,O	D	II	II	
Liquid Hydrogen Storage Vessel	P	G	H	D	II+	1M	35
Vessels, other	P	G	H	D	II	II	

[a] A module is an assembly of interconnected components which constitute an identifiable device or piece of equipment. For example, electric modules include sensors, power supplies, and signal processors; and mechanical modules include turbines, strainers, and orifices.

[b] GE - General Electric; P - Plant Owner (Energy Northwest)

[c] 1, 2, 3, G - Safety classes defined in Section 3.2.3.

[d]

A - Auxiliary building	M - Any other location
C - Part of or within primary containment	O - Outdoors onsite
L - Offsite locale	P - Pump house
R - Reactor building	W - Radwaste and control building
S - Service building	DG - Diesel generator building
T - Turbine building	H - Hydrogen Storage and Supply Facility

Table 3.2-1

Equipment Classification (Continued)

[e] A, B, C, D - NRC quality groups defined in Regulatory Guide 1.26, Revision 3. The equipment is constructed in accordance with the codes listed in Table 3.2-2, as minimum requirements.

N/A - Quality Group classification not applicable to this equipment.

[f] Quality Classes defined in Section 3.2.4.

[g] I - constructed in accordance with the seismic requirements of Seismic Category I structures and equipment as described in Section 3.7.

II - constructed in accordance with the requirements of Seismic Category II structures and equipment as described in Section 3.7. The approaches outlined in the Uniform Building Code will be followed where applicable.

N/A - seismic requirements as described for Seismic Category I and II structures and equipment are not applicable to this structure or equipment.

1M - non-safety-related components required to be seismically supported/mounted. This was alternatively called Seismic II⊕ which is Seismic II piping supported on hangars designed to Seismic I loads.

[h] The notes clarify system/component boundaries, design requirements, and/or alternate quality classification applications.

Reactor pressure boundary components (RPBC) and their original applicable code cases are listed in the attachments to Table 3.2-1. Current practices regarding code cases that are adopted for use at CGS (and are approved by Regulatory Guides 1.147, 1.84, 1.85, or other approval authority) requires that they are specified in the component's design specification as required by ASME Section III, NA-3250.

Table 3.2-1

Equipment Classification (Continued)

NOTES

1. This piping has been upgraded from Safety Class 3 to Safety Class 2 and from Quality Group Classification C to Quality Group Classification B.

2. a. Lines 0.75-in. and smaller which are part of the RCPB are Quality Group B or higher and Seismic Category I.

   b. All instrument lines which are connected to the RCPB and are used to actuate and monitor safety systems are Quality Group B from the outer isolation valve or the process shutoff valve (root valve) to the bulkhead of the instrument rack, if rack mounted, or the sensing instrumentation, if locally mounted.

   c. All instrument lines which are connected to the RCPB and are not used to actuate and monitor safety systems are Quality Group D from the outer isolation valve or the process shutoff valve (root valve) to the sensing instrument

   d. All other instrument lines:

      1. Through the root valve are of the same classification or higher as the system to which they are attached;

      2. Beyond the root valve to the instrument rack bulkhead, if rack mounted, or to the sensing instrumentation, if locally mounted, if used to actuate a safety system, are of the same classification as the system to which they are attached;

      3. Beyond the root valve, if not used to actuate a safety system are Quality Group D.

      4. All sample lines from the outer isolation valve or the process root valve through the remainder of the sampling system are Quality Group D.

Table 3.2-1

Equipment Classification (Continued)

NOTES (Continued)

3. The control rod drive insert and withdraw lines from the drive flange up to and including the first valve on the hydraulic control unit (HCU) is in Safety Class 2.

4. Supports for Quality Class II (nonessential) piping systems, HVAC, cable trays, and system components in the reactor building, primary containment, the control building, diesel generator building, the standby service water (SW) pump houses, and the radwaste building corridor are designed and constructed to withstand an SSE per position C.2 of Regulatory Guide 1.29. These supports are constructed to Quality Class II requirements, as a minimum.

5. The HCU is a GE factory assembled engineered module of valves, tubing, piping, and stored water which controls a single control rod drive by the application of precisely timed sequences of pressure and flows to accomplish slow insertion or withdrawal of the control rods for power control and rapid insert ion for reactor scram.

   Although the HCU is field installed and connected to process piping, many of its internal parts differ markedly from process piping components because of the more complex functions they must provide. Thus, although the codes and standards invoked by Quality Groups A, B, C, D pressure integrity quality levels clearly apply to all levels to the interfaces between the HCU and the connecting conventional piping components (e.g., pipe nipples, fittings, simple hand valves, etc.), it is considered that they do not apply to the specialty parts (e.g., solenoid valves, pneumatic components, and instruments).

   The design and construction specifications for the HCU do invoke such codes and standards as can be reasonably applied to individual parts in developing required quality levels, but these codes and standards are supplemented with additional requirements for these parts and for the remaining parts and details. For example, (a) all welds are LP inspected, (b) all socket welds are inspected for gap between pipe and socket bottom, (c) all welding is performed by qualified welders, and (d) all work is done per written procedures. Quality Group D is generally applicable because the codes and standards invoked by that group contain clauses which permit the use of manufacturer's standards and proven design techniques which are not explicitly defined within the codes of Quality Groups A, B, or C. This is supplemented by the quality control techniques described above.

Table 3.2-1

Equipment Classification (Continued)

NOTES (Continued)

6. The standby liquid control (SLC) system is not a safety-related backup to a scram function (RPS/CRD). However, these components of the SLC system are designed to engineering standard greater than those normally applied to Safety Class 2 systems.

7. Only equipment associated with a safety action (e.g., isolation) need conform to a safety function.

8. These lines meet the requirements of Quality Group B except that hydrostatic testing of the containment spray piping is not required.

9. The RCIC turbine does not fall within the applicable design codes. To ensure that the turbine is fabricated to the standards commensurate with its safety and performance requirements, GE has established specific design requirements for this component which are as follows:

   a. All welding is qualified in accordance with Section IX, ASME Boiler and Pressure Vessel Code;

   b. All pressure containing castings and fabrications are hydrotested to 1.5 times design pressure;

   c. All high-pressure castings are radiographed according to:

      1. ASTM E-94
      2. E-142 maximum feasible volume
      3. E-71, 186 or 280 severity level 3;

   d. All-cast surfaces are magnetic particulate or liquid penetrant tested according to ASME, Section III, Paragraph NB-2575 or NB-2576;

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Table 3.2-1

Equipment Classification (Continued)

NOTES (Continued)

e. Wheel and shaft forgings are ultrasonically tested according to ASTM-A-388;

f. Butt-welds are radiographed according to ASME Section III, IX-3300, and magnetic particle or liquid penetrant tested according to ASME Section III, IX-3500, or IX-3600;

g. Notification is made on major repairs and records maintained thereof;

h. Record system and traceability according to ASME Section III, NA-4900;

i. Control and identification according to ASME Section III, NA-4400;

j. Procedures conform to ASME Section III, NA-4400;

k. Inspection personnel are qualified according to ASME Section III, Appendix IX, paragraph IX-400.

10. The RCIC turbine exhaust line from the isolation valve to the suppression pool meets all the requirements of Quality Group B except that hydrostatic testing of this portion of piping is not required.

11. Although the refueling bellows are designed to withstand the SSE without rupture, they may be plastically deformed.

12. DELETED.

13. Equipment, piping, and valves that are part of the radwaste system but not used for processing radioactive fluids are designed to Quality Group D standards.

Table 3.2-1

Equipment Classification (Continued)

NOTES (Continued)

14. Equipment, piping, and valves that are part of the radwaste solids handling system are designed to Quality Group D standards.

15. Up to and including the last stop valve, this line meets requirements of Quality Group C.

16. The fuel pool cooling heat exchangers have been upgraded to 270 psig (shell side) and Seismic Category I through several modifications. These heat exchangers are Quality Class II (with the modifications installed as Quality Class I) and ASME Section III Class 3. See Note 13 in Figure 9.1-6.

17. Piping and valves of the cooling portion of the fuel pool cooling system have been upgraded to Seismic Category I and are ASME Section III Class 3, Code Group C (with exception of valves listed in Note 16 in Figure 9.1-4). See Note 13 in Figure 9.1-6 for Quality Classification.

18. The fuel pool cooling circulation pumps have been upgraded to Seismic Category I. Their respective motors have been upgraded to Quality Class I Seismic Category I. The fuel pool cooling pumps are ASME Section III Class 3. See Note 13 in Figure 9.1-6 for Quality Classification.

19. The fuel pool cooling system (see Section 9.1.3) normally receives makeup from the Seismic Category II condensate storage tank. Should this normal supply be unavailable, makeup is available from the Seismic Category I SW system. Likewise, the normal source of cooling water to the FPC heat exchangers is from the Seismic Category II RCC system. When this normal supply is unavailable, the Seismic Category I SW system provides the cooling water and makeup. In addition, by means of removable spool-pieces, the RHR system is also available as the supplementary source of cooling during cold shutdown under core offload conditions. The above complies with Regulatory Guides 1.26, Revision 3, and 1.29, Revision 3.

Table 3.2-1

Equipment Classification (Continued)

NOTES (Continued)

The cleanup portion is automatically isolated from the cooling portion of the system by Seismic Category I valves on low fuel pool level (see Section 9.1.3).

20. Piping and valves downstream of valves SGT-V-5A1, SGT-V-5A2, SGT-V-5B1, and SGT-V-5B2 meet the requirements of Quality Group B except that a pneumatic test is not performed.

21. Lavatory exhaust systems are designed to Quality Assurance Class G.

22. Part of the HVAC components are non-safety-related. Non-safety-related equipment and components required to be seismically supported/mounted according to Regulatory Guide 1.29 are designated Seismic Category 1M and Quality Assurance Class II+. Non-safety-related equipment and components not required to be seismically supported/mounted according to Regulatory Guide 1.29 are designated Seismic Category II and Quality Assurance Class II.

23. The condensate storage tanks are designed, fabricated, and tested to meet ASME Code, Section III, Subsection ND-3800. In addition, the specification for this tank requires 100% surface examination of the side wall to bottom joint, and 100% volumetric examination of the side-wall weld joints.

24. The high-pressure core spray (HPCS) suction piping from the condensate storage tank provides the initial source of makeup water to the HPCS system for safety injection. Consequently, this piping has been upgraded by full volumetric examination of every weld.

25. The auxiliary piping systems on the engines are built to the guidelines of ANSI B31.1.

26. These piping systems are liquid penetrant or magnetic particle examined to the acceptance standards of ASME Section III, Class 3, if they are over 4 in. IPS as required by the 1973 Winter Addenda except for the diesel cooling water piping that is on engine skid.

Table 3.2-1

Equipment Classification (Continued)

NOTES (Continued)

27. Piping upstream from the check valves at the inlet of the air receivers is Quality Class II, ANSI B31.1, Seismic Category I. All other piping is Quality Class 1.

28. The piping is supported to Seismic Category I requirements, but is not housed in a Seismic Category I structure. The power conversion system structures are constructed in accordance with applicable codes for steam power plants. The turbine building, interacting with main steam lines and branch lines, is designed as a modified Seismic Category II structure as described in Section 3.8.4.1.3.

29. All cast pressure retaining parts of a size and configuration for which volumetric examination methods are required are examined by radiographic methods by qualified personnel. Ultrasonic examination to equivalent standards may be used as an alternative to radiographic methods. Examination procedures and acceptance standards are at least equivalent to those specified as supplementary types of examination in ANSI B31.1.0 Code, Paragraph 136.4.3.

30. The following qualification is met with respect to the certification requirements:

    a. The manufacturer of the turbine stop valves, turbine governor valves, turbine bypass valves, and mainsteam leads from turbine control valve to turbine casting utilized quality control procedures, and

    b. A certification has been obtained from the manufacturer of these valves and steam leads that the quality control program so defined has been accomplished.

Table 3.2-1

Equipment Classification (Continued)

NOTES (Continued)

31. The makeup water pump house is designed to withstand the design-basis tornado. The design also considers the possible effects of tornado-generated missiles. The tower makeup water piping, valves, and cabling located underground are provided with adequate earth cover to be resistant to tornado-generated missiles or are protected by tornado-resistant structures. The circulating water system also supports the post-tornado operation of the standby service water system. In the event of the loss of the spray headers in the SW spray ponds, tornado-protected underground lines can provide a flow path to provide makeup water from TMU, and to return water to the CW basin. See FSAR Section 9.2.5.3 for additional discussion.

32. Portions of the turbine building that support or interact with main steam piping are designed to Seismic Category I.

33. Those portions of the radwaste and control building that house systems or components necessary for safe shutdown of the reactor are designed to Quality Class I and Seismic Category I requirements. Those portions of the radwaste building housing equipment containing significant quantities of radioactive material are designed to Seismic Category I requirements.

34. Safety-related instrument and control systems are identified in Table 7.1-1.

35. The HSSF liquid hydrogen storage tank, foundations, anchorage (i.e., anchor bolts, slide plates, and the baseplate welding) and the underlying soil are not safety-related, and are designated as Quality Class II+. However, these are designed for Seismic Category I loads and ground motion as defined by Regulatory Guide 1.60. In addition, they were designed to remain in place for both design basis tornado characteristics and maximum probable flood.

36. Piping located between secondary containment isolation valves EDR-V-394, EDR-V-395, FDR-V-219, FDR-V-220, FDR-V-221 and FDR-V-222 is classified as Seismic Category I, Quality Class II+ and Quality Group D.

37. Seismic Category I applies to the motor rotor, bearings and bearing load path parts. The motor is mounted Seismic Category IM.

Table 3.2-1 Attachment 1

Nuclear Boiler System

Component	Code Case	Comments	Component	Code Case	Comments
PI(1)-ST-X70d			PI(1)-4S-X114		
PI(1)-4S-X71a			PI(1)-ST-X114		
PI(1)-ST-X71a			PI(1)-4S-X115		
PI(1)-4S-X71b			PI(1)-ST-X115		
PI(1)-ST-X71b			PI(1)-ST-(IR-64)-4		
PI(1)-4S-X72a			PI(1)-ST-(IR-64)-5		
PI(1)-ST-X72a			PI(1)-ST-MS-PT-2		
PI(1)-4S-X75c			RFW(1)-4A	1 thru 11, K,L,M,S,T	
PI(1)-ST-X75c			RFW(1)-4B	1 thru 11, K,L,M,S,T	
PI(1)-4S-X75d					
PI(1)-ST-X75d					
PI(1)-4S-X106					
PI(1)-ST-X106					
PI(1)-4S-X107					
PI(1)-ST-X107					
PI(1)-4S-X108					
PI(1)-ST-X108					
PI(1)-4S-X109					
PI(1)-ST-X109					
PI(1)-4S-X110					
PI(1)-ST-X110					
PI(1)-4S-X111					
PI(1)-ST-X111					
PI(1)-4S-X112					
PI(1)-ST-X112					
PI(1)-4S-X113					
PI(1)-ST-X113					

Table 3.2-1 Attachment 1

Nuclear Boiler System (Continued)

Component	Code Case	Comments	Component	Code Case	Comments
MS(9)-4	1, 4 thru 11 K,L,M,S,T		PI(1)-4S-X39a		
MS(18)-2-1	1, 4 thru 11 K,L,M,S,T		PI(1)-ST-X39a		
MS(18)-2-2	1 thru 11 K,L,M,S,T		PI(1)-4S-X39b		
MS(18)-2-3	1, 3 thru 11 K,L,M,S,T		PI(1)-ST-X39b		
MS(18)-2-4	1, 4 thru 11, K,L,M,S,T		PI(1)-4S-X42a		
MS(18)-2-5	1, 3 thru 11 K,L,M,S,T		PI(1)-ST-X42a		
MS(18)-2-6	1, 4 thru 11 K,L,M,S,T		PI(1)-4S-X42b		
MS(18)-2-7	1, 4 thru 11 K,L,M,S,T		PI(1)-ST-X42b		
MS(18)-2-8	1, 4 thru 11,K,L,M,S,T		PI(1)-4S-X42e		
MS(18)-2-9	1, 3 thru 11 K,L,M,S,T		PI(1)-ST-X42e		
MS(18)-2-10	1, 3 thru 11 K,L,M,S,T		PI(1)-4S-X61c		
MS(18)-2-11	1, 3 thru 11 K,L,M,S,T		PI(1)-ST-X61c		
MS(18)-2-12	1, 3 thru 11 K,L,M,S,T		PI(1)-4S-X62b		
MS(18)-2-13	1, 4 thru 11 K,L,M,S,T		PI(1)-ST-X62b		
MS(18)-2-14	1, 4 thru 11 K,L,M,S,T		PI(1)-4S-X69a		
MS(18)-2-15	1, 3 thru 11 K,L,M,S,T		PI(1)-ST-X69a		
MS(18)-2-16	1, 3 thru 11 K,L,M,S,T		PI(1)-4S-X69b		
MS(18)-2-17	1, 4 thru 11 K,L,M,S,T		PI(1)-ST-X69b		
MS(18)-2-18	1, 4 thru 11 K,L,M,S,T		PI(1)-4S-X69f		
B22-G001A	1 thru 3		PI(1)-ST-X69f		
B22-G001B	1 thru 3		PI(1)-4S-X70a		
B22-G001C	1 thru 3		PI(1)-ST-X70a		
B22-G001D	1 thru 3		PI(1)-4S-X70b		
PI(1)-4S-X38a			PI(1)-ST-X70b		
PI(1)-ST-X38a			PI(1)-4S-X70c		
PI(1)-4S-X38b			PI(1)-ST-X70c		
PI(1)-ST-X38b			PI(1)-4S-70d		

Table 3.2-1 Attachment 2

Reactor Recirculation System

Component	Code Case	Comments	Component	Code Case	Comments
PI(1)-4S-X40c			PI(1)-ST-X44Ah		
PI(1)-ST-X40c			PI(1)-4S-X44Aj		
PI(1)-4S-X40d			PI(1)-ST-X44Aj		
PI(1)-ST-X40d			PI(1)-4S-X44Ak		
PI(1)-4S-X40e			PI(1)-ST-X44Ak		
PI(1)-4S-X40f			PI(1)-4S-X44A1		
PI(1)-4S-X41c			PI(1)-ST-X44A1		
PI(1)-ST-X41c			PI(1)-4S-X44Am		
PI(1)-4S-X41d			PI(1)-ST-X44Am		
PI(1)-ST-X41d			PI(1)-4S-X44Ba		
PI(1)-4S-X41e			PI(1)-ST-X44Ba		
PI(1)-4S-X41f			PI(1)-4S-X44Bb		
PI(1)-4S-X44Aa			PI(1)-ST-X44Bb		
PI(1)-ST-X44Aa			PI(1)-4S-X44Bc		
PI(1)-4S-X44Ab			PI(1)-ST-X44Bc		
PI(1)-ST-X44Ab			PI(1)-4S-X44Bd		
PI(1)-4S-X44Ac			PI(1)-ST-X44Bd		
PI(1)-ST-X44Ac			PI(1)-4S-X44Be		
PI(1)-4S-X44Ad			PI(1)-ST-X44Be		
PI(1)-ST-X44Ad			PI(1)-4S-X44Bf		
PI(1)-4S-X44Ae			PI(1)-ST-X44Bf		
PI(1)-ST-X44Ae			PI(1)-4S-X44Bg		
PI(1)-4S-X44Af			PI(1)-ST-X44Bg		
PI(1)-ST-X44Af			PI(1)-4S-X44Bh		
PI(1)-ST-X44Ag			PI(1)-ST-X44Bh		
PI(1)-ST-X44Ag			PI(1)-4S-X44Bj		
PI(I)-4S-X44Ah			PI(1)-ST-X44Bj		

Table 3.2-1 Attachment 3

Control Rod Drive Hydraulic System

Component	Code Case	Comments	Component	Code Case	Comments
PI(1)-4S-X44Bk			PI(1)-ST-X75b		
PI(1)-ST-X44Bk			PI(1)-4S-X75e		
PI(1)-4S-X44Bl			PI(1)-ST-X75e		
PI(1)-ST-X44B1			PI(1)-4S-X75f		
PI(1)-4S-X44Bm			PI(1)-ST-X75f		
PI(1)-ST-X44Bm			PI(1)-4S-X78f		
PI(1)-4S-X61a			PI(1)-ST-X78f		
PI(1)-ST-X61a			PI(1)-1-RRC-SPV-85A		
PI(1)-4S-X61b			PI(1)-1-RRC-SPV-85B		
PI(1)-ST-X61b			RRC(5)-4S-A	1, 3 thru 11 K,L,M,S,T	
PI(1)-4S-X62c			RRC(5)-4S-B	1, 3 thru 11 K,L,M,	
PI(1)-ST-X62c			RRC(51)-1	1, 3 thru 11 K,L,M	
PI(1)-4S-X62d			RRC(51)-4	1, 3 thru 11 K,L,M,S,T	
PI(1)-ST-X62d			B35-G001A	1, 3	
PI(1)-4S-X69e			B35-G001B	1, 3	
PI(1)-ST-X69e					
PI(1)-4S-X70e					
PI(1)-4S-X70f					
PI(1)-4S-X74a					
PI(1)-ST-X74a					
PI(1)-4S-X74e					
PI(1)-ST-X74e					
PI(1)-4S-X74f					
PI(1)-ST-X74f					
PI(1)-4S-X75a					
PI(1)-ST-X75a					
PI(1)-4S-X75b					

Table 3.2-1 Attachment 3

Control Rod Drive Hydraulic System (Continued)

Component	Code Case	Comments	Component	Code Case	Comments
CRD-I/W	G				
CRD-SCRAM/INST	5, 13, G				
Hy(1)-65-A	1, 3 thru 11 K,L,M,S				
Hy(1)-65-B	4 thru 11 K,L,M,S,T				

Table 3.2-1 Attachment 4

Standby Liquid Control System

Component	Code Case	Comments	Component	Code Case	Comments
SLC(1)-1S	1, 3 thru 11 K,L,M				
SLC(2)-3S	1, 3 thru 11 K,L,M,S,T				
SLC(2)-4S	1, 3 thru 11 K,L,M,S,T				
PI(1)-ST-SLC-FT-1					
PI(1)-ST-SLC-PT-4					

Table 3.2-1 Attachment 5

Residual Heat Removal System

Component	Code Case	Comments	Component	Code Case	Comments
PI(1)-4S-X37e			PI(1)-ST-(H22-PO21)-A10		
PI(1)-ST-X37e			PI(1)-ST-(H22-PO21)-B1		
PI(1)-4S-X37f			PI(1)-ST-(H22-PO21)-B3		
PI(1)-ST-X37f			PI(1)-ST-(H22-PO21)-B4		
PI(1)-4S-X39d			PI(1)-ST-(H22-PO21)-B5		
PI(1)-ST-X39d			PI(1)-ST-(H22-PO21)-B6		
PI(1)-4S-X39e			PI(1)-ST-(IR-69)-2		
PI(1)-ST-X39e			PI(1)-ST-(IR-69)-3		
PI(1)-4S-X42d			PI(1)-ST-(IR-69)-7		
PI(1)-ST-X42d			PI(1)-ST-(IR-71)-1		
PI(1)-4S-X54Bf			PI(1)-ST-(IR-71)-2		
PI(1)-ST-X54Bf			PI(1)-ST-RHR-DPIS-9A		
PI(1)-4S-X61f			PI(1)-ST-RHR-DPIS-9B		
PI(1)-ST-X61f			PI(1)-ST-RHR-DPIS-9C		
PI(1)-1-X62f			PI(1)-ST-RHR-FT-1		
PI(1)-1-X69c			PI(1)-ST-RHR-FT-13		
PI(1)-4S-X74b			PI(1)-ST-RHR-LT-8A		
PI(1)-ST-X74b			PI(1)-ST-RHR-LT-8B		
PI(1)-ST-(H22-P018)-A5			PI(1)-ST-RHR-PI-2A		
PI(1)-ST-(H22-P018)-A6			PI(1)-ST-RHR-PI-2B		
PI(1)-ST-(H22-P018)-A7			PI(1)-ST-RHR-PI-2C		
PI(1)-ST-(H22-P018)-A9			PI(1)-ST-RHR-PS-18		
PI(1)-ST-(H22-P018)-A10			RHR(1)-2A	1, 3 thru 12 K,L,M,S,T	
PI(1)-ST-(H22-P021)-A5			RHR(1)-2B	1 thru 12 K,L,M,S,T	
PI(1)-ST-(H22-P021)-A6			RHR(1)-2C	1, 3 thru 11 K,L,M,S,T	
PI(1)-ST-(H22-P021)-A7			RHR(1)-4A	1, 3 thru 11 K,L,M,S,T	
PI(1)-ST-(H22-P021)-A9			RHR(1)-4B	1, 3 thru 11 K,L,M,S,T	

Table 3.2-1 Attachment 5

Residual Heat Removal System (Continued)

Component	Code Case	Comments	Component	Code Case	Comments
RHR(1)-4C	1, 3 thru 11 K,L,M,S,T				
RHR(3)-1C	1, 3 thru 12 K,L,M,S,T				
RHR(3)-2A	1, 3 thru 11 K,L,M,S,T				
RHR(3)-2B	1, 3 thru 11 K,L,M,S,T				
RHR(4)-1A	1, 3 thru 12 K,L,M,S,T				
RHR(4)-1B	1, 3 thru 12 K,L,M,S,T				
RHR(4)-1C	1, 3 thru 11 K,L,M,S,T				
RHR(9)-1	1, 3 thru 11 K,L,M,S,T				
RHR(1)-4B1	1, 4 thru 11 K,L,M,S,T				
RHR(1)-4A1	1, 3 thru 11 K,L,M,S,T				

Table 3.2-1 Attachment 6

Low-Pressure Core Spray

Component	Code Case	Comments	Component	Code Case	Comments
LPCS(1)-2	1 thru 11 K,L,M,P,S,T				
LPCS(1)-4	1, 4 thru 11 K,L,M,S,T				
LPCS(2)-1	1, 3 thru 11 K,L,M,S,T				
LPCS(3)-1	1, 3 thru 11 K,L,M,S,T				
PI(1)-4S-X78b					
PI(1)-ST-X78b					
PI(1)-1-X78d					
PI(1)-ST-X78d					
PI(1)-ST-(H22-POO1)-A2					
PI(1)-ST-(H22-POO1)-A3					
PI(1)-ST-(H22-POO1)-A4					
PI(1)-ST-(H22-POO1)-A5					
PI(1)-ST-(H22-POO1)-A6					
PI(1)-ST-(H22-POO1)-A7					
PI(1)-ST-LPCS-PI-1					

Table 3.2-1 Attachment 7

High-Pressure Core Spray

Component	Code Case	Comments	Component	Code Case	Comments
PI(1)-4S-X73a					
PI(1)-ST-X73a					
PI(1)-1-X78e					
PI(1)-ST-X78e					
PI(1)-ST-(H22-PO24)-A1					
PI(1)-ST-(H22-PO24)-A2					
PI(1)-ST-(H22-PO24)-A5					
PI(1)-ST-(H22-PO24)-A6					
PI(1)-ST-(H22-PO24)-A8					
PI(1)-ST-HPCS-PS-3					
HPCS(1)-4CL1	1, 3 thru 11 K,L,M,S,T				
HPCS(1)-4CL2	1, 3 thru 11 K,L,M,S,T				
HPCS(2)-1	1, 3 thru 11 K,L,M,S,T				
HPCS(3)-1	1, 4 thru 11 K,L,M,S,T				

Table 3.2-1 Attachment 8

Reactor Core Isolation Cooling System

Component	Code Case	Comments	Component	Code Case	Comments
RCIC(1)-4CL1	1, 4 thru 11 K,L,M,S,T		PI(1)-ST-(H22-PO17)-A5		
RCIC(1)-4CL2	1, 4 thru 12 K,L,M,S,T		PI(1)-ST-(H22-PO17)-A6		
RCIC(2)-1	1 thru 11 K,L,M,S,T		PI(1)-ST-(H22-PO17)-A7		
RCIC(12)-4CL1	1, 2, 4 thru 11 K,L,M,S,T		PI(1)-ST-(H22-PO17)-A8		
RCIC(12)-4CL2	1, 3 thru 11 K,L,M,S,T		PI(1)-ST-(H22-PO17)-B1		
RCIC(13)-4CL2	1 thru 11 K,L,M,S,T		PI(1)-ST-(H22-PO17)-B2		
RCIC(16)-1	1, 3 thru 11 K,L,M,S,T		PI(1)-ST-(H22-PO17)-B3		
RCIC(19)-1	1, 3 thru 11 K,L,M,S,T		PI(1)-ST-(H22-PO17)-B6		
RCIC(50)-1	1 thru 11 K,L,M,S,T		PI(1)-ST-(H22-PO17)-B7		
PI(1)-4S-X38c			PI(1)-ST-(H22-PO18)-A13		
PI(1)-ST-X38c			PI(1)-ST-(H22-PO21)-A13		
PI(1)-4S-X38d			PI(1)-ST-(H22-PO29)-A5		
PI(1)-ST-X38d			PI(1)-ST-(H22-PO29)-A6		
PI(1)-4S-X38e			PI(1)-ST-(IR-62)-2		
PI(1)-ST-X38e			PI(1)-ST-(IR-62)-4		
PI(1)-4S-X38f			PI(1)-ST-(IR-62)-5		
PI(1)-ST-X38f			PI(1)-ST-(IR-63)-15		
PI(1)-1-X54Aa			PI(1)-ST-(IR-63)-16		
PI(1)-ST-X54Aa			PI(1)-ST-(IR-67)-4		
PI(1)-4S-X71c			PI(1)-ST-RCIC-PCV-15		
PI(1)-ST-X71c			PI(1)-ST-RCIC-PS-1		
PI(1)-4S-X71d			PI(1)-ST-RCIC-PS-9A		
PI(1)-ST-X71d			PI(1)-ST-RCIC-PS-9B		
PI(1)-4S-X71e			PI(1)-ST-RCIC-PS-34		
PI(1)-ST-X71e					
PI(1)-4S-X71f					
PI(1)-ST-X71f					

Table 3.2-1 Attachment 9

Radwaste System

Component	Code Case	Comments	Component	Code Case	Comments
PI(1)-ST-(IR-61)-6					
PI(1)-ST-(IR-65)-3					
PI(1)-ST-(IR-61)-4					
PI(1)-ST-(IR-65)-4					
PI(1)-ST-(IR-62)-17					
PI(1)-ST-(IR-62)-18					
EDR(48)-1	1, 3 thru 11 K,L,M,S,T				
FDR(48)-1	1, 4 thru 11 K,L,M,S,T				
MWR(62)-15	1, 3				

Table 3.2-1 Attachment 10

Reactor Water Cleanup System

Component	Code Case	Comments	Component	Code Case	Comments
RWCU(1)-3A	1 thru 11 K,L,M,S,T				
RWCU(1)-3B	1, 3 thru 12 K,L,M,S,T				
RCWU(1)-4	1, 3 thru 11 K,L,M,S,T				
RWCU(3)-4	1 thru 11 K,L,M,S,T				
PI(1)-4S-X78c					
PI(1)-4S-X79a					
PI(1)-ST-X79a					
PI(1)-4S-X79b					
PI(1)-ST-X79b					
PI(1)-ST-(H22-P002)-A8					
PI(1)-ST-(H22-P002)-A9					
PI(1)-ST-(H22-P002)-A12					
PI(1)-ST-(H22-P002)-A13					

Table 3.2-1 Attachment 11

Standby Service Water System

Component	Code Case	Comments	Component	Code Case	Comments
PI(1)-ST-(IR-69)-1					
PI(1)-ST-(IR-69)-8					
PI(1)-ST-(IR-69)-10					
PI(1)-ST-(IR-62)-3					
PI(1)-ST-(IR-71)-4					
PI(1)-ST-(IR-71)-5					
PI(1)-ST-FPC-FT-16					
PI(1)-ST-FPC-FT-17					
PI(1)-ST-FPC-LIS-1A					
PI(1)-ST-FPC-LIS-1B					
PI(1)-ST-FPC-SPV-1					
FPC(1)-1	1 thru 11 K,L,M				
FPC(2)-1A	1, 3 thru 11 K,L,M				
FPC(2)-1B	1, 3 thru 11 K,L,M,S,T				
FPC(5)-2	1, 3 thru 11 K,L,M,S,T				
FPC(7)-1	1				
FPC(12)-1	1, 4 thru 11 K,L,M,N,S,T				

Table 3.2-1 Attachment 12

Standby Service Water System

Component	Code Case	Comments	Component	Code Case	Comments
SW(1)-2	1, 3 thru 11 K,L,M,S,T		PI(1)-ST-SW-PS-11B		
SW(1)-2-UG	1, 3 thru 11 E,K,L,M		PI(1)-ST-SW-PS-11A		
SW(2)-2	1, 3 thru 11 K,L,M,S,T		PI(1)-ST-(CMS-SR-14)-C2		
SW(2)-2-UG	1, 4 thru 11 E,K,L,M		PI(1)-ST-(IR-22)-1		
SW(21)-2	1 thru 11 K,L,M,S,T		PI(1)-ST-(IR-22)-2		
SW(21)-2-UG	1, 3 thru 11 E,K,L,M,S,T		PI(1)-ST-(IR-22)-3		
SW(22)-2	1, 3 thru 11 K,L,M,S,T		PI(1)-ST-(IR-69)-15		
SW(22)-2-UG	1, 3 thru 11 E,K,L,M,S,T		PI(1)-ST-(SW-SR-43)-A		
SW(70)-1-HPCS	1, 3 thru 11 K,L,M,S,T		PI(1)-ST-(SW-SR-43)-B		
SW(71)-1-HPCS	1, 3 thru 11 K,L,M		P1(1)-ST-SW-FIS-9		
PI(1)-ST-(CMS-SR-13)-C1			PI(1)-ST-SW-FIS-12		
PI(1)-ST-(CMS-SR-13)-C2			PI(1)-ST-SW-FIS-15		
PI(1)-ST-(H22-PO18)-A11					
PI(1)-ST-(H22-PO18)-A12					
PI(1)-ST-(H22-PO21)-A11					
PI(1)-ST-(H22-PO21)-A12					
PI(1)-ST-(IR-21)-1					
PI(1)-ST-(IR-21)-2					
PI(1)-ST-(IR-21)-3					
PI(1)-ST-(IR-24)-1					
PI(1)-ST-(IR-24)-2					
PI(1)-ST-(IR-71)-9					
PI(1)-ST-(SW-SR-42)-A					
PI(1)-ST-(SW-SR-42)-B					
PI(1)-ST-(CMS-SR-14)-C1					
PI(1)-ST-SW-FT-8A					
PI(1)-ST-SW-FT-8B					

Table 3.2-1 Attachment 13

Reactor Building Closed Cooling Water System

Component	Code Case	Comments	Component	Code Case	Comments
RCC(3)-1	1 thru 11 K,L,M,S,T				
RCC(4)-2	3 thru 11 K,L,M,S,T				
RCC(5)-2	1, 3 thru 11 K,L,M,S,T				
RCC(36)-1	1 thru 11 K,L,M,S,T				

Table 3.2-1 Attachment 14

Primary Containment Cooling System

Component	Code Case	Comments	Component	Code Case	Comments
CSP(1)-1B	1, 3 thru 11 K,L,M,S,T		PI(1)-4S-X73e		
CEP(1)-1A	1, 2, 4 thru 11 K,L,M,S,T		PI(1)-ST-X73e		
CEP(1)-1B	1 thru 11 K,L,M,S,T		PI(1)-4S-X82b		
PI(1)-4S-X29c			PI(1)-ST-X82b		
PI(1)-4S-X29d			PI(1)-4S-X82c		
PI(1)-ST-X29d			PI(1)-ST-X82c		
PI(1)-4S-X29e			PI(1)-4S-X84a		
PI(1)-ST-X29e			PI(1)-ST-X84a		
PI(1)-4S-X30a			PI(1)-4S-X84b		
PI(1)-ST-X30a			PI(1)-ST-X84b		
PI(1)-4S-X30d			PI(1)-4S-X85c		
PI(1)-ST-X30d			PI(1)-ST-X85c		
PI(1)-4S-X72b			PI(1)-4S-X85d		
PI(1)-ST-X72b			PI(1)-ST-X85d		
PI(1)-4S-X72c			PI(1)-4S-X85e		
PI(1)-ST-X72c			PI(1)-ST-X85e		
PI(1)-4S-X72d			PI(1)-4S-X86a		
PI(1)-ST-X72d			PI(1)-ST-X86a		
PI(1)-4S-X72e			PI(1)-4S-X86b		
PI(1)-ST-X72e			PI(1)-ST-X86b		
PI(1)-4S-X72f			PI(1)-4S-X87a		
PI(1)-ST-X72f			PI(1)-ST-X87a		
PI(1)-4S-X73c			PI(1)-4S-X87b		
PI(1)-ST-X73c			PI(1)-ST-X87b		
PI(1)-4S-X73d			PI(1)-ST-(IR-62)-14		
PI(1)-ST-X73d			PI(1)-ST-(IR-62)-15		
			PI(1)-ST-(IR-62)-16		

Table 3.2-1 Attachment 14

Primary Containment Cooling System (Continued)

Component	Code Case	Comments	Component	Code Case	Comments
PI(1)-ST-(IR-62)-19					
PI(1)-ST-(IR-63)-1A					
PI(1)-ST-(IR-63)-1B					
PI(1)-ST-(IR-63)-2					
PI(1)-ST-(IR-63)-3					
PI(1)-ST-(IR-63)-4					
PI(1)-ST-(IR-63)-10					
PI(1)-ST-(IR-64)-1A					
PI(1)-ST-(IR-64)-1B					
PI(1)-ST-(IR-64)-2					
PI(1)-ST-(IR-64)-38					
PI(1)-ST-(IR-64)-7					
PI(1)-ST-(IR-64)-9					
PI(1)-ST-(IR-65)-1					
PI(1)-ST-(IR-65)-2					
PI(1)-ST-(IR-65)-7					
PI(1)-ST-(IR-65)-8					
PI(1)-ST-(IR-65)-9					
PI(1)-ST-(IR-66)-3					
PI(1)-ST-(IR-67)-2					
PI(1)-ST-(IR-67)-3					
PI(1)-ST-(IR-68)-1					
PI(1)-ST-(IR-68)-2					
PI(1)-ST-(CMS-SR-13)-DT					
PI(1)-ST-(CMS-SR-14)-DT					

Table 3.2-1 Attachment 15

Standby Gas Treatment System

Component	Code Case	Comments	Component	Code Case	Comments
PI(1)-ST-SGT-FT-1A1					
PI(1)-ST-SGT-FT-1A2					
PI(1)-ST-SGT-FT-1B1					
PI(1)-ST-SGT-FT-1B2					

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Table 3.2-1 Attachment 16

Primary Containment Atmospheric Control System (DEACTIVATED)

Component	Code Case	Comments	Component	Code Case	Comments
PI(1)-ST-CAC-FT-1A					
PI(1)-ST-CAC-FT-1B					
PI(1)-ST-CAC-FT-2A					
PI(1)-ST-CAC-FT-2B					
PI(1)-ST-CAC-FT-3A					
PI(1)-ST-CAC-FT-3B					
PI(1)-ST-CAC-FT-4A					
PI(1)-ST-CAC-FT-4B					
CAC(1)-1	1, 3 thru 11 K,L,M,S,T				
CAC(2)-1	1, 3 thru 11 K,L,M,S,T				
CAC(11)-1	1 thru 11 K,L,M,S,T				
CAC(12)-1	1, 2, 3, 7				
CAC(21)-1A	1				
CAC(21)-1B	4 thru 11 K,L,M				

Table 3.2-1 Attachment 17

Other Heating, Ventilating, Air-Conditioning

Component	Code Case	Comments	Component	Code Case	Comments
PI(1)-ST-(IR-67)-10					
PI(1)-ST-(IR-67)-13					
PI(1)-ST-(IR-68)-15					
PI(1)-ST-(IR-69)-16					
PI(1)-ST-(IR-69)-17					
PI(1)-ST-(IR-71)-12					
PI(1)-ST-(IR-71)-13					

Table 3.2-1 Attachment 18

Condensate Storage and Transfer

Component	Code Case	Comments	Component	Code Case	Comments
COND(98)-1	1, 4 thru 11 K,L,M,N				

Table 3.2-1 Attachment 19

Instrument and Service Air

Component	Code Case	Comments	Component	Code Case	Comments
CAS(5)-1	1, 3 thru 11 K,L,M				
SA(1)-1	1, 4 thru 11 K,L,M,S,T				

Table 3.2-1 Attachment 20

Containment Instrument Air

Component	Code Case	Comments	Component	Code Case	Comments
CIA(3)-2	1, 3 thru 11 K,L,M				
CIA(5)-1A	1, 3 thru 11 K,L,M				
CIA(5)-1B	1 thru 11 K,L,M				
CIA(5)-2A	1, 3 thru 11 K,L,M				
CIA(5)-2B	1, 3 thru 11 K,L,M				
PI(1)-1-X56					
PI(1)-ST-(IR-67)-8					
PI(1)-ST-(IR-67)-9					
PI(1)-ST-(IR-68)-8					
PI(1)-ST-(IR-68)-9					
PI(1)-ST-(IR-68)-10					
PI(1)-ST-(IR-71)-17					
PI(1)-ST-(IR-71)-18					
PI(1)-ST-(IR-72)-1					
PI(1)-ST-(IR-74)-4					
PI(1)-ST-CIA-PIS-5					
PI(1)-ST-CIA-PIS-6					
PI(1)-ST-CIA-PIS-7					
PI(1)-ST-CIA-PIS-8					

Table 3.2-1 Attachment 21

Diesel Generator System

Component	Code Case	Comments	Component	Code Case	Comments
DO(1)-1A	1, 3 thru 11 K,L,M				
DO(1)-1B	1, 4 thru 11 K,L,M				
DO(1)-1-HPCS	1, 3 thru 11 J,K,L,M				
DO(9)-1A	1				
DO(9)-1B	1				
DO(9)-1-HPCS	1				
PI(1)-ST-DO-LS-11A					
PI(1)-ST-DO-LS-11B					
PI(1)-ST-DO-LS-13					

Table 3.2-1 Attachment 22

Power Conversion System

Component	Code Case	Comments	Component	Code Case	Comments
PI(1)-ST-(IR-81)-1					
PI(1)-ST-(IR-81)-3					
PI(1)-ST-(IR-82)-1					
PI(1)-ST-(IR-82)-3					
PI(1)-ST-(IR-83)-1					
PI(1)-ST-(IR-83)-3					
PI(1)-ST-(IR-84)-1					
PI(1)-ST-(IR-84)-3					
PI(1)-ST-MS-PTD-1A					
PI(1)-ST-MS-PTD-1B					
PI(1)-ST-MS-PT-54A					
PI(1)-ST-MS-PT-54B					
PI(1)-ST-MS-PS-56A					
PI(1)-ST-MS-PS-56B					
PI(1)-ST-MS-PS-56C					
PI(1)-ST-MS-PS-56D					
MS(1)-4A	1, 3 thru 12 K,L,M,S,T				
MS(1)-4B	1, 3 thru 11 K,L,M				
MS(1)-4C	1, 3 thru 12 K,L,M,S,T				
MS(1)-4D	1, 3 thru 12 K,L,M,S,T				

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Table 3.2-1 Attachment 23

Main Steam Isolation Valves Leakage Control System (DEACTIVATED)

Component	Code Case	Comments	Component	Code Case	Comments
MSLC(2)-1	1 thru 11 K,L,M,S,T				
MSLC(4)-1	1, 3 thru 11 K,L,M				
PI(1)-ST-(IR-21)-7					
PI(1)-ST-(IR-21)-9					

Table 3.2-1 Attachment 24

Containment Vessel

Component	Code Case	Comments	Component	Code Case	Comments
Containment Vessel	F				
Containment System	2, 14, H	Pen. Assem. & Stiffener			

Table 3.2-1 Attachment 25

Postaccident Sampling System

Component	Code Case	Comments	Component	Code Case	Comments
PI(1)-4S-X73f					
PI(1)-4S-X77Ac					
PI(1)-4S-X77Ad					
PI(1)-4S-X80b					
PI(1)-4S-X82d					
PI(1)-4S-X82f					
PI(1)-4S-X83a					
PI(1)-4S-X83f					
PI(1)-4S-X84e					
PI(1)-4S-X84f					
PI(1)-4S-X88					

Table 3.2-1 Attachment 26

Key to Code Cases

The Following Code Cases Were Used as Noted on the Attachments

Regulatory Guide 1.85		Regulatory Guide 1.84	
1.	N-242-1	A.	N-154
2.	1567	B.	N-122
3.	1713	C.	N-318
4.	1644 thru 1644-6	D.	N-316
5.	N-71-7 (1644-7)	E.	N-328
6.	N-71-8 (1644-8)	F.	N-362
7.	N-71-9 (1644-9)	G.	1606-1
8.	N-249	H.	N-58 (1614)
9.	N-249-1	J.	N-192
10.	1728	K.	1718
11.	N-225	L.	N-111 (1729)
12.	N-224-1	M.	N-247
13.	1571	N.	N-240
14.	N-274	P.	N-272
		R.	N-252
		S.	1683-1
		T.	N-175 (1818)
		U.	N-411-1

Table 3.2-2

Code Group Designations - Industry Codes and Standards for Mechanical Components[a,b]

Quality Group Classification	ASME Section III Code Classes	ASME Section III Code Applicable Subsection				
		Pressure Vessels and Heat Exchangers	Pumps, Valves, and Piping	Metal Containment Components	Storage Tanks 0-15	Storage Tanks Atmospheric
A	1	NA and NB TEMA C	NA and NB[c]	-	-	-
B	2 or MC	NA and NC TEMA C	NA and NC[c]	- NA and NE	NA and NC API-620[d]	NA and NC or D100, B96.1 or API-620[e]
C	3	NA and ND TEMA C	NA and ND[c]	- -	NA and ND or API-620[f]	NA and ND or D100, B96.1 or API-650[f]
D		ASME Section VIII Division 1 TEMA C	Piping and valves B31.1[g] Pumps[h]	-	API-620 or equivalent[i]	API-650 AWWA-D100 ANSI B96.1 or equivalent[i]

[a] With options and additions as necessary for service conditions and environmental requirements.

Table 3.2-2

Code Group Designations - Industry Codes and Standards
for Mechanical Components[a,b] (Continued)

[b] Components of the reactor coolant pressure boundary meet the requirements of 10 CFR 50, Section 50.55a, "Codes and Standards," except as shown in Table 5.2-1 and discussed in Section 5.2.1.1. All components satisfy codes and addenda in effect at the time of component order or later.

[c] For pumps classified A, B, or C; applicable subsections NB, NC, or ND; respectively, in ASME Code Section III is used as a guide in calculating the thickness of pressure-retaining portions of the pump and in sizing cover bolting.

[d] 100% volumetric examination of the sidewall and roof weld joints for plates over 3/16 in. thick and 100% surface examination of weld joints for plates 3/16 in. thick or less of the sidewall to bottom and sidewall to roof joints. These examination requirements are performed in accordance with the rules of ASME Section III, Class 2.

[e] 100% volumetric examination of the sidewall weld joints for plates over 3/16 in. thick and 100% surface examination sidewall to bottom joints. These examination requirements are performed in accordance with the rules of ASME Code Section III, Class 2.

[f] Nondestructive tests examination requirements per ASME Code, Section VIII, Division 1.

[g] Welds not totally conforming to B31.1 are evaluated and dispositioned on a case-by-case basis considering (a) the function of the systems, (b) the risk of failure, and (c) the consequences of failure for safety and plant availability.

[h] For pumps classified Group D, and operating above 150 psig and 212°F Section VIII, Division 1 is used as a guide in calculating the wall thickness for pressure-retaining parts and in sizing the cover bolting.

[i] Tanks are designed to meet the intent of API, AWWA, and/or ANSI 96.1 standards as applicable.

Table 3.2-3

Summary of Safety Class Design Requirements (Minimum)

	Safety Class			
Design Requirements	1	2	3	G
Quality group classification[a]	A	B	C	D
Quality class[b]	I	I, II+	I, II+	II, G
Seismic category[c]	I	I, 1M	I, 1M	II

[a] The equipment is constructed in accordance with the indicated code group listed in Table 3.2-1 and defined in Table 3.2-2.

[b] I (1 in MEL) - the equipment is constructed in accordance with the quality assurance requirements of 10 CFR 50, Appendix B, and Nuclear Energy Division Boiling Water Reactor Quality System Summary.

II+ (A in MEL) - Non-safety-related equipment for which Columbia Generating Station has made specified commitments to the NRC or others relating to the quality. Some 10 CFR 50, Appendix B, criteria are applied.

II (2 in MEL), G - The equipment is constructed in accordance with the quality assurance requirements as specified in contract documents as described in Section 3.2.3.

[c] I - the equipment of these safety classes is constructed in accordance with the seismic requirements for the safe shutdown earthquake as described in Section 3.7.

1M - The equipment does not perform a safety-related function but must be seismically supported/mounted per Regulatory Guide 1.29.

II - The seismic requirements for the safe shutdown earthquake are not applicable to the equipment of this classification. The approaches outlined in the Uniform Building Code are followed where applicable.

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3.3 WIND AND TORNADO LOADINGS

3.3.1 WIND LOADINGS

3.3.1.1 Design Wind Velocity

All Seismic Category I structures are designed to withstand a basic wind velocity (fastest mile), including gusts, of 100 mph at an elevation of 30 ft above the site grade. This wind velocity exceeds the basic wind velocity having a statistically derived probable period of recurrence of 100 years in this geographical area, as specified in the American Society of Civil Engineers Task Committee Report (Reference 3.3-1).

The Hanford region experiences high wind speeds due to squall lines, frontal passages, strong pressure gradients, and thunderstorms. The Hanford Reservation has experienced only one recorded tornado (June 1948) and has not been known to be affected by typhoons. No complete statistics are readily available which present frequency of occurrence of high winds produced or accompanied by a particular meteorological event. However, the highest winds produced by any cause are tabulated for the Hanford Meteorological Station (HMS) in Tables 2.3-5 and 2.3-6. Figure 2.3-4 indicates the return probability of any peak wind gust again due to any cause. The 100-mph design wind speed is conservative for the CGS site for the following reasons:

a. Peak wind gusts measured at the 50-ft HMS tower level, as reported in Tables 2.3-5 and 2.3-6, have never exceeded 80 mph, and

b. The statistically derived 100-year return period peak gust (Figure 2.3-4) at an elevation of 50 ft is 86 mph based on HMS records.

Recurrence intervals, data sources, and the history of occurrence of high winds, hurricanes, and tornadoes in the vicinity of the site are discussed in the CGS Environmental Report.

3.3.1.2 Determination of Applied Forces

The basic wind velocity of 100 mph is applied in accordance with Table 1(a) of Reference 3.3-1, including the variation in wind velocity with height and drag coefficients. Considering a gust factor of 1.0 and drag coefficient of C_D and V as the specific wind velocity at a particular height above grade given in Table 1(a) of Reference 3.3-1, the total combined average wind pressure p is given by

$$p = C_D \times 0.002558 V^2$$

For example, at a height of 30 ft with V = 100 mph, C_D = 1.3, p = 33 psf

The above chosen gust factor is adequate for the CGS site for the following reasons. In Reference 3.3-1, the wind gust factor is defined as gust velocity divided by the velocity of the fastest mile of wind. Whereas the gust factor can not be less than 1.0, it is implied in Section 3.3.1.1 that the factor is unity since the 100-mph fastest mile wind accounts for wind gusts. A factor greater than 1.0 is not warranted at CGS since the statistically derived peak wind gust considered is less than the assumed site basic wind velocity. Additional considerations regarding gust factors and fastest mile velocities are presented in Section 2.3.1.2.1.4.

The following wind velocities are used in the design of structures at various elevations:

Height Above Grade (ft)	Wind Speed, V (mph)*
Less than 50	100
50 to 149	120
150 to 400	140

The wind pressures are applied as static forces. The translation of wind velocities into applied static forces, the wind force distribution, and the drag coefficients are in accordance with published values such as Reference 3.3-1. *The italicized information is historical and was provided to support the application for an operating license. The magnitude and distribution of the applied static forces originally calculated for structures are as follows:*

Height Above Grade	*Windward Side 0.8q (psf)*	*Leeward Side 0.5q (psf)*
Less than 50 ft	*20*	*13*
50 to 150 ft	*30*	*18*
150 to 400 ft	*0*	*25*

The term q, as described in Reference 3.3-1, is designated as the dynamic pressure of a free wind stream at a point on the surface of a structure immersed in the wind stream and when multiplied by the pressure coefficient, Cp, characteristic of the building types defined in Reference 3.3-1, gives the pressure on the structure's surfaces.

The original reactor building membrane roofing system has been replaced with a modern elastic sheet membrane system in accordance with the Factory Mutual system for Class I Fire and I-90 windstorm rating. Local wind uplift forces at the roof perimeter and corners would result in local failure of the existing controlled release fasteners during a gale or squall windstorm.

* *Table 1(a), Reference 3.1-1.*

3.3.2 TORNADO LOADINGS

3.3.2.1 Applicable Design Parameters

The tornado design criteria were revised for CGS based on design basis tornado characteristics in NUREG-1503 (Reference 3.3-2). In January 1996, the revised criteria were found acceptable by the NRC (References 3.3-3 and 3.3-4). The original tornado design criteria provided for wind speeds of 300 mph rotational and 60 mph translational velocity, with a pressure drop of 3 psi to occur in 3 sec.

The design basis tornado is one having a maximum horizontal component of tangential (or peripheral rotational) wind velocity of 160 mph and a constant translational velocity of 40 mph. The resultant wind velocity (the sum of the maximum horizontal component of tangential velocity and the translational velocity) is 200 mph. The atmospheric pressure at the center of the tornado is 0.9 psi below ambient. The 0.9 psi external pressure drop is assumed to occur at a rate of 0.3 psi/sec. The nonventing structures are designed for the worst combination of wind velocity and associated atmospheric pressure drop in accordance with the load combinations contained in Section 3.3.2.2. The venting structures are designed for the worst combination of wind velocity and associated difference of the pressure within and the atmospheric pressure outside. The effects of a design basis tornado are considered in combination with other loads, including tornado-generated missiles, in Section 3.5.

3.3.2.2 Determination of Forces on Structures

Design static pressures, drag coefficients, and wind pressures are selected in accordance with published values such as Reference 3.3-1. The provisions for gust factors and variation of wind velocity, noted therein, are not applied for the following reasons.

The wind velocity may vary with the height of the structures but, for conservatism, the wind force due to tornado loadings is applied as a uniform static load invariant with the height above grade.

The total wind velocity occurs only in a localized area but is used in the design over the full height of the projected area of the structure. The total wind velocity is in effect the gust wind velocity, since by definition a tornadic gust wind velocity is a high localized wind velocity of very short duration. Therefore, no additional gust factor is applied.

The procedure used to transform the design-basis tornado wind velocity into an effective pressure applied to exposed surfaces of structures is described in Reference 3.3-1.

The following is an example of how Reference 3.3-1 was applied in the original calculations:

The same procedure as that used to transform the basic design-basis wind velocity in Section 3.3.1.2 is used with the exception that the velocity and velocity pressure are assumed not to vary with height.

The equivalent uniform tornado wind velocity used on the structure due to a tangential component of 300 mph and a transitional component of 60 mph is 360 mph. The pressure loads are calculated on the basis of a uniform 360 mph wind velocity.

a. *The dynamic pressure on the structure is:*

   $q = 0.002558 \times (360)^2 = 331.5$ *psf*

b. *The applied static pressures are:*

   1. *Windward pressure on walls:*

      $p = 0.8 \times 331.5 = 265$ *psf*

   2. *Leeward suction on walls:*

      $p = 0.5 \times 331.5 = 166$ *psf*

   3. *Total design pressure on the structure is the sum of 265 psf and 166 psf, or 431 psf.*

The differential pressure loading is calculated using the following pressure-time function: The differential pressure is assumed to vary from zero to 0.9 psi at a rate of 0.3 psi/sec and then return to zero at 0.3 psi/sec.

The procedure used for transforming the tornado-generated missile loadings into effective static loads is described in Section 3.5.3.

Mathematical models for the design-basis tornado take into consideration the phase relationship between the wind load and the differential pressure effects.

The tornado load, W′, in the load combinations in Tables 3.8-9 and 3.8-10 constitutes the combined effect of the three separate loads a, b, and c generated by the design basis tornado. The three loads are combined in the following manner to obtain the combined effect:

a. $W' = W_w$

b. $W' = W_p$

c. $W' = W_m$

d. $W' = W_w + 0.5W_p$

e. $W' = W_w + W_m$

f. $W' = W_w + 0.5W_p + W_m$

where:

W' = Total tornado load

W_w = Tornado wind load

W_p = Tornado differential pressure load

W_m = Tornado-generated missile load

3.3.2.3 Additional Design Features

Except for the steel superstructure atop the refueling floor, the reactor building remains sealed through the tornado event and a differential pressure of 0.9 psi across the exterior and interior is bounded by the design. All other Seismic Category I structures are provided with adequate openings to relieve a differential pressure of 0.9 psi in 3 sec or are designed to withstand an external pressure drop of 0.9 psi.

The structural steel frame superstructure atop the refueling floor of the reactor building is designed to withstand the design basis tornado. However, all the siding and roof decking enclosing the steel superstructure is designed for a maximum differential pressure of approximately 0.5 psi. The siding and girts are designed to blow off the steel frame when a differential pressure of approximately 0.5 psi is exceeded. The roof decking and roof purlins are designed to blow off the steel frame when a differential pressure of approximately 0.5 psi is exceeded. This value considers the dead weight loading from the roof membrane, roofing insulation, roof decking, and roof purlins. This is ensured by the use of controlled release type fasteners connecting the girts to the columns and roof purlins to the roof trusses. The release of the girts, siding, roof purlins, and roof decking from the structural steel frame will not affect the ability to shut down the reactor, the integrity of the primary containment or other Seismic Category I structures, or the capability of the essential heat removal systems to perform their intended design functions.

The design of the reactor building crane and its support system considers tornado effects in addition to normal loads to eliminate the possibility of generating internal missiles which may endanger the primary and secondary containment structures. The trolley is provided with latches to engage tornado racks attached to the bridge girder to prevent horizontal movement of the trolley due to tornado loadings. The bridge trucks are provided with latches to engage tornado racks attached to the crane runway girders to prevent horizontal movement of the bridge due to tornado loadings. See Figure 3.8-44.

Based on the General Electric publication, (Reference 3.3-5), there is no credible mechanism by which a significant amount of water could be sucked from the fuel pool by a tornado.

The design considers the possible effects of tornado-generated missiles discussed in Section 3.5. Primary containment and components, equipment, and systems essential to a safe shutdown are protected from tornado-generated missiles by enclosing structures.

The diesel generator building and the radwaste and control building are designed to withstand the effects of tornado-generated missiles that might be released, such as girts and roof purlins. The results of analyses determining the effects of missiles are discussed in Section 3.5.

Piping and cabling required for safe shutdown and which penetrate exterior walls of the tornado-resistant structures are located below grade or are protected, as in the case of the standby service water (SW) piping at the service water pump houses, by tornado-resistant structures. Piping and cabling required for safe shutdown, which is provided with less than adequate earth cover for tornado protection, is protected with a tornado-resistant concrete slab or other structure. For information concerning protection provided for the SW system see Section 9.2.5.

3.3.2.4 Effect of Failure of Structures or Components Not Designed for Tornado Loads

The two reinforced-concrete spray pond structures have the capability to tolerate tornado generated missiles. Each pond is made up of four perimetral walls, 1 ft 3 in. at the top and 2 ft 0 in. at the bottom, and a floor slab of 7-in. thickness placed on top of a 2-in.-unreinforced-concrete subgrade leveling slab, as described in Section 3.8.4.1.5. Finish grade around the perimetral walls of the spray ponds is 1 ft 0 in. below the top of the walls, and the normal pond water level is at 6 in. below the top of the walls. The walls and slab are bounded by Quality Class I high relative density backfill. On this basis, missile protection is provided for the pond structures. A direct hit by a tornado-generated design basis missile resulting in localized floor and wall penetration is unlikely because of the protection provided by the backfill and the water in the pond.

Damage to the spray pond concrete structure due to tornado-generated missiles would be localized cracking in the area of impact. The structure will remain intact and any leakage will be made up by the cooling tower makeup system which pumps water directly from the river.

The makeup water pump house is a non-Seismic Category I structure impervious to tornado damage. The pump house and structures associated with the cooling tower makeup system, such as valve boxes, are designed to withstand the design basis tornado, including tornado-generated missiles. Piping, valves, and electrical equipment are protected in a similar way. Exterior wall and roof penetrations are designed and anchored to withstand the design basis tornado and the effects of tornado-generated missiles. Tornado-generated missiles are discussed in Section 3.5. Section 9.2.5.3 provides a discussion on the makeup water system and ultimate heat sink interaction if tornado damage to the spray headers require a feed-and-bleed mode of operation.

The availability of essential electrical power to the makeup water pump house systems is ensured. The electrical lines are underground with sufficient earth cover to resist tornado-generated missiles.

The electrical lines are installed in such manner as to provide two redundant electrical systems from the power source to the makeup water pump house. The two electrical systems are physically separated to provide adequate missile protection of one system from the other. At one end of each system, redundant power source transformers, associated switchgear, and cabling are provided on the ground floor of the turbine building where, for the trajectories required to cause damage to this equipment, they are protected against missile impact and spalling by the exterior walls of the turbine building and the floor slabs overhead. The terminal ends and transformers at the makeup water pump house are enclosed within the tornado-resistant pump house. Manholes within each system are also designed to withstand tornado-generated missiles.

The spray pond piping and supports are designed to withstand the effects of the design basis tornado. The piping system cannot be protected from the impact of tornado-generated missiles. In the event of missile damage to one of the pond spray headers, the alternate spray system which is 100% redundant is placed in operation. (In the event that both spray systems are rendered inoperative, the cooling tower makeup water system is placed into operation to provide continuous water supply to the spray ponds using Columbia River water which based on historical data has not exceeded 70°F.) The cooling tower makeup water system is provided with sufficient protection to prevent its loss of function in the event of a design-basis tornado passing over the project site. Since the makeup water flow rate exceeds that of the SW systems, and since the makeup water temperature is lower than the SW system design temperature of 85°F, the continuous availability of cooling water is ensured. Procedures for alternate spray pond use is described in Section 9.2.5.

Failure of nontornado-resistant cooling towers due to tornado loads does not endanger Seismic Category I structures since the plant arrangement provides sufficient distance between the cooling towers and Seismic Category I structures.

The liquid nitrogen storage tank located at the corner of the diesel generator building will not fail due to tornado wind loads. However, if the design-basis tornado missile were to strike the tank straight on near the top of the tank, it could be toppled. Toppling of the liquid nitrogen storage tank due to the impact of a tornado missile can cause the entire contents of the tank to be rapidly emptied in the vicinity of the inerting system skid. There is no safety-related equipment in the vicinity of the tank that would be affected by the cryogenic temperatures associated with liquid nitrogen. In addition, due to the turbulent mixing produced in close proximity to a tornado, no oxygen deficiency condition could be sustained outdoors at the diesel generator intake structures.

Failure of nontornado-resistant structures and components will not affect the ability to shut down the reactor, the integrity of the primary containment or other Seismic Category I structures, the capability of the essential heat removal systems to perform their intended design functions, nor result in the release of radioactivity.

3.3.2.5 Conformance with Regulatory Guide 1.76, Revision 0

The discussion of conformance with Regulatory Guide 1.76, Revision 0, is in Section 1.8.3.

3.3.3 REFERENCES

3.3-1 Task Committee Report, Wind Forces on Structures, Transactions of the American Society of Civil Engineers, Paper No. 3269, Vol. 126, Part II, 1961.

3.3-2 Nuclear Regulatory Commission, Final Safety Evaluation Report Related to the Certification of the Advanced Boiling Water Reactor Design, NUREG-1503, Vol. 1, July 1994.

3.3-3 Letter from J. V. Parrish, Supply System, to Document Control Desk, NRC, Subject: WNP-2 Request for Approval to Revise Tornado Design Criteria, October 10, 1995 (G02-95-212).

3.3-4 Letter from J. W. Clifford, NRC, to J. V. Parrish, Supply System, Subject: Revision of Tornado Design Criteria for the Supply System WNP-2, dated January 24, 1996 (G12-96-032).

3.3-5 Miller, D. R., and Williams, W. A., Tornado Protection for the Spent Fuel Storage Pool, General Electric Company, APED-5696, November 1968.

3.4 WATER LEVEL (FLOOD) DESIGN

3.4.1 FLOOD PROTECTION

3.4.1.1 External Flood Levels

3.4.1.1.1 Design Basis Flood

Flood levels and conditions are defined in Section 2.4. From the flooding conditions considered in Section 2.4, the design basis flood (DBF) condition arrived at for use in the protection of Seismic Category I structures and safety-related systems and components is that flood which results from the probable maximum precipitation (PMP) event. The PMP event results in a flood elevation of 433.3 ft mean sea level (msl). This event includes an additional 1.9 ft to account for wind wave action.

3.4.1.1.2 Breach of the Grand Coulee Dam

Floods associated with breaches of the Grand Coulee Dam are discussed in Section 2.4.4. The associated highest flood level is at el. 424 ft msl with wave action included.

3.4.1.1.3 Acceptance Criteria

The facility design and equipment locations are in accordance with General Design Criterion 2, as related to systems and components withstanding flood conditions, and Regulatory Guide 1.59, Revision 1, dated April 1976.

3.4.1.2 Groundwater Levels

3.4.1.2.1 Design Basis Groundwater

The design basis groundwater conditions are defined in Section 2.4.13. From the groundwater conditions considered in Section 2.4.13, the design-basis groundwater condition for use in the protection of Seismic Category I structures and safety-related systems and components is a rise of the present groundwater table with the possible construction of Ben Franklin Dam as discussed in Section 2.4.13.

The design-basis groundwater is el. of 420 ft msl. The normal groundwater table is at approximately el. 380 ft msl.

3.4.1.2.2 Breach of the Grand Coulee Dam

The effects on the groundwater table due to a breach of the Grand Coulee Dam is discussed in Section 2.4.4. This flood would peak for a short time and would not affect the design-basis groundwater level of 420 ft msl due to its short duration.

3.4.1.2.3 Design Basis Flood Probable Maximum Precipitation

As stated in Section 3.4.1.1.1, the worst DBF condition results from the PMP event. Due to the short duration of this flood and its confines, groundwater level at the site would not be affected.

3.4.1.3 Identification of Structures, Systems, and Components

Seismic Category I structures and safety-related systems and components are identified in Section 3.2. Figures 1.2-1 through 1.2-24 show locations and elevations for systems and components. See Section 2.4.2 for discussion of flood protection of safety-related systems and components.

3.4.1.4 Description of Structures, Systems, and Components

3.4.1.4.1 Flood Protection Requirements

3.4.1.4.1.1 External Flood Protection Requirements. The plant site elevation at Seismic Category I structures and safety-related systems and components is approximately 441 ft msl, except at the spray ponds where the finish grade elevation is 434 ft msl and the top of spray ponds walls is 435 ft msl. These elevations are sufficient to protect the plant site and, therefore, Seismic Category I structures and the safety-related systems and components housed therein against the DBF. Exterior and access openings to all Seismic Category I structures are located above the plant site grade and, therefore, above the DBF level. Flood protection measures are not provided since they are not required.

3.4.1.4.1.2 Internal Flood Protection Requirements. Section 3.4.1.5.2 discusses internal flood protection measures provided for safety-related systems, equipment, and components.

Figures 1.2-2, 1.2-7 through 1.2-17, and 1.2-22 illustrate plant arrangement and layout and show that safety-related equipment is located within individual rooms or compartments. The pump rooms located on the 422 ft 3 in. elevation are enclosed by reinforced-concrete walls. Penetrations and doors in these walls are provided with seals that minimize the effects of flooding between rooms should a break occur in one of the rooms.

The potential flooding and environmental effect attributable to postulated through-wall leakage cracks in moderate-energy fluid piping systems and postulated rupture of high-energy fluid piping systems are evaluated and discussed in Section 3.6.1.

Section 6.3 addresses single failure of the emergency core cooling systems (ECCS) piping, including leak detection requirements for ECCS passive failures, ECCS passive failures during long-term cooling, and potential flooding attributable thereto.

Section 9.3.3 discusses the design bases, system descriptions, safety evaluation testing and inspection requirements, and the instrumentation requirements relative to equipment and floor drainage systems. The design bases used ensure equipment and floor drainage systems integrity during normal plant operation and preclude any danger to health and safety of plant personnel, the environs and the general public. Five independent sumps are provided in the reactor building at floor el. 422.25 ft. These sumps serve pump rooms as shown on Figures 1.2-6 and 1.2-7. There is a single equipment drain sump for the reactor building (see Figure 9.3-9) located in the CRD/Condensate pump room. This sump also connects to the RCIC pump room through an unisolable drain header. The other four sumps are floor drain sumps serving the ECCS pump rooms. A single floor drain sump serves no more than two rooms. A sump located in one room is connected to a second room by a drain header containing a single isolation valve. Each sump is equipped with a level monitoring system, which automatically actuates the sump pump when high water level is reached in the sump. Each sump pump discharges to the Radwaste System. The isolation valve located in the header between connected rooms automatically closes on high water level in the sump to isolate the room with the leak.

The sumps collect water from such typical sources as equipment drains from the drywell and from other equipment carrying low purity water and from the drywell floor drain and other floor and pit drains. In the event of a pipe break in one of the pump rooms of sufficient size to exceed the capacity of the sump pump and overflow a sump in one room, the effects of common mode flooding between pump rooms are minimized by the following:

a. The reactor building equipment drain sump, located in the CRD/Condensate pump room, serves only the RCIC and the CRD/Condensate pump rooms. Although both rooms will be affected by a flood occurring in one room, analysis ensures that sufficient safe shutdown equipment remains unaffected by the flood, preserving the ability to safely shutdown the plant. Existing equipment drains located in the RHR pump rooms A and B are capped and, thus, do not connect to each other or to a common equipment drain sump. See Section 9.3.3.2.1.

b. The floor drain sumps serving more than one pump room have a single isolation valve in the drain header between pump rooms. This valve is fail-safe and automatically closes on a high water level in the sump. Accordingly, flooding in any pump room, exceeding the capacity of the sump pump, will be confined to the room in which the leakage occurs with only minimal leakage through wall penetrations and door seals into other rooms or areas. Analysis is performed to ensure that sufficient safe shutdown equipment remains unaffected by the flood, preserving the ability to safely shutdown the plant. The high sump level also annunciates in the main control room. See Section 9.3.3.2.2.1.

c. Wall-mounted Class 1E level switches are also located in each of the ECCS and RCIC pump rooms and are mounted just above floor level. These level switches ensure that if a failure of the sump pump, the sump header isolation valve, or sump alarm system should occur in any of these rooms during a flood event, prompt operator notification of the event would be received allowing sufficient time for mitigating actions and safe plant shutdown. See Sections 6.3.2.5 and 9.3.3.2.2.1.

Administrative controls ensure that separation criteria is maintained by ensuring the appropriate doors and hatches are closed.

3.4.1.4.2 Groundwater Protection Requirements

Seismic Category I structures house safety-related systems and components and Seismic Category I components. The elevation of the lowest floor surface of these structures is as follows:

Structures	Elevation of Top of Floor
Reactor building	422 ft 3 in.
Radwaste and control room areas of radwaste control building	437 ft 0 in.
Diesel-generator building	441 ft 0 in.
Spray ponds 1A and 1B	420 ft 0 in.
Standby service water pump houses 1A and 1B	408 ft 3 in.
Retaining area for the condensate storage tanks	441 ft 0 in.

Seismic Category I structures and safety-related systems and components are located above the present groundwater el. 380 ft msl and are not subject to any force effects of buoyancy and static water from this groundwater elevation. Uplift and increased lateral hydrostatic pressure are considered in the design of all Seismic Category I structures and safety-related systems and components, to ensure their safety in the event of a rise in the groundwater table to 420 ft msl. Standby service water pump houses 1A and 1B are designed to resist the increased hydrostatic pressure which would result from the rise in the groundwater to el. 420 ft msl. The lowest floor surface in the reactor building is the top of the foundation mat at el. 422 ft 3 in. msl. Since this is above the design basis groundwater level, the structure is unaffected by the force effects of buoyancy and static water due to groundwater at el. 420 ft msl. Groundwater el. 420 ft msl was compared with foundation levels of Seismic Category I structures and it was determined that waterproofing is not required. Seismic Category I piping and electric conduit penetrations that are below grade are above the design basis groundwater table, and sealing against groundwater pressure is therefore not required. However, all pipes penetrating exterior walls are waterproofed sealed by boots installed on both sides of the wall penetration; all electrical conduit penetrations are through-wall waterproof sealed using silicon foam.

The only materials underlying the site that might exhibit unfavorable response to seismic or other events under saturated soil conditions are the loose to medium dense, fine to coarse sand with scattered gravel, in the upper approximate 40 ft of the soil profile. These were removed and recompacted as structural fill, as described in Sections 2.5.4.8 and 2.5.4.12. Structural fill supports the Seismic Category I structures, including the turbine generator building and service building, in the central plant complex. Structural fill, as required, is also utilized below the other Seismic Category I and safety-related structures including underground piping and electrical duct banks. The structural fill is compacted to a minimum of 75% relative density and an average relative density of not less than 85%. The compacted backfill will eliminate the possibility of liquefaction and provide satisfactory foundation performance should the groundwater level at the Columbia Generating Station site rise.

To evaluate the possible effects of a gross rise in groundwater levels, Shannon and Wilson, soil consultants, performed a series of repetitive triaxial tests in identical soils in the dry and saturated states and concluded that saturation would not necessitate changes in allowable bearing pressures or settlement calculations as discussed in Section 2.5.4.10.

To provide conservatism in the design of structures, the seismic dynamic response of the structures and components was examined over a range of soil shear moduli, as discussed in Section 2.5.4.7.

The possibility of soil liquefaction under the effects of the safe shutdown earthquake was evaluated for the conditions found to exist at the Columbia Generating Station site. Soils having high (75% plus) relative densities have been found in the past to be safe against liquefaction. In the soil underlying the foundations at Columbia Generating Station, liquefaction potential can only be assessed on the basis of soil type and general characteristics.

The foundations of all critical structures lie on compacted fill of fine to coarse sand with gravels. The phenomenon of the liquefaction has never been observed in such soils as this, and therefore, liquefaction was not postulated at the Columbia Generating Station site.

3.4.1.5 Flood Protection Measures

3.4.1.5.1 External Flood Protection Measures

As discussed in Section 3.4.1.4.1, external flood protection measures are not provided since they are not required. Equipment is located so that it is not vulnerable. Protection is not required to cope with potential leakage from such phenomena as cracks in structure walls. Since Seismic Category I structures are located at sufficient grade and distance from the Columbia River, as described in Section 2.4, the effects of wind wave action, including spray, do not require flood protection measures.

3.4.1.5.2 Internal Flood Protection Measures

The primary or safety-related functions fulfilled by the reactor building include the capability to withstand the effects of flooding from internal sources. Measures provided are discussed in Sections 3.8.4.1.1, 3.8.4.3.3, and 3.8.4.4.1.1. ECCS and RCIC pump rooms located in the reactor building basement at elevation 422 ft 3 in. are designed to withstand the effects of flooding between and including the top of the foundation mat at el. 422 ft 3 in. and el. 466 ft. Although not watertight, the doors and penetrations in these pump room walls are provided with seals that will minimize flooding between rooms (except for the RCIC and CRD pump rooms that are connected by an unisolable sump pipe) even with significant hydrostatic pressure generated from flooding water levels up to 466 ft. These pump rooms, comprised of exterior walls, interior walls, biological shield wall, isolation valves (when provided), penetrations, and doors are designed so that any one compartment at any one time, flooded to an elevation of 466 ft, can withstand the effects of seismically induced water sloshing loads. Section 3.8.4.3.3 discusses the CRD/Condensate pump room and railroad bay south exterior wall design hydrostatic pressure. Section 3.8.4.1.1 discusses the equipment hatch in the vehicle air lock (railroad bay) floor at el. 441 ft. See Section 3.4.1.4.1.2 for additional nonstructural related internal flood protection measures.

3.4.1.6 Emergency Flood Protection

Emergency flood protection procedures for bringing the reactor to a safe shutdown are unnecessary because flood conditions do not impact safe shutdown operation.

3.4.2 ANALYSIS PROCEDURES

Seismic Category I structures are located at sufficient grade and distance from the Columbia River so that dynamic water forces are precluded. Seismic Category I structures are designed for the force effects of buoyancy and hydrostatic pressure due to the design basis groundwater in conjunction with loading combinations in Section 3.8.4. Seismic Category I structures are checked for stability and foundation pressures. In all cases, the dead loads of structures maintain stability and a positive soil bearing pressure. Design loads due to groundwater force application are applied to all Seismic Category I structures as follows:

a. A vertical hydrostatic pressure due to the water head below el. 420 ft msl on subgrade surfaces of structures and a lateral hydrostatic pressure on subgrade surfaces of structures. The lateral pressure is treated as an additional triangular loading increasing at the rate of 62.4 lb/ft^2 per vertical foot from el. 420 ft to the bottom of the structure, and

b. A buoyant force equal to the weight of water displaced by the structure.

The loading combinations used in conjunction with hydrostatic pressure due to groundwater, taken from Table 3.8-9, are

a. $U = 1.4D + 1.7L + 1.7\ P_o + 1.4F + 1.7Q$

b. $U = 1.4D + 1.7L + 1.4\ T_o + 1.7\ P_o + 1.4F + 1.7Q$

c. $U = 0.9D + 1.4F + 1.7Q$

d. $U = 0.9D + 1.4\ T_o + 1.4F + 1.7Q$

e. $U = 1.1D + 1.3L + 1.3\ P_o + 1.1F + 1.3Q + 1.3W$

f. $U = 1.1D + 1.3L + 1.1\ T_o + 1.3\ P_o + 1.1F + 1.3Q + 1.3W$

g. $U = 1.4D + 1.7L + 1.7\ P_o + 1.9E + 1.4F^* + 1.7Q$

h. $U = 1.4D + 1.4L + 1.4\ T_o + 1.4\ P_o + 1.4E + 1.4F^* + 1.4Q^*$

In the above load combinations, Q denotes lateral pressure on subgrade surfaces of structures due to either dry or saturated soils; and Q* denotes lateral pressure, including seismic effects, on subgrade surfaces of structures due to either dry or saturated soils. In applying the above loading combinations in conjunction with lateral hydrostatic pressure due to groundwater, the value for saturated soil is used.

In the above load combinations, the vertical hydrostatic pressure due to the water head below el. 420 ft msl and acting on subgrade surfaces of structures is included in D, dead loads, with consideration given to the direction of the hydrostatic pressure. The loading combinations used in conjunction with hydrostatic pressure due to internal flooding are given in Table 3.8-9. The definitions of load terms used in the load combinations are given in Section 3.8.4.3.3.

3.5 MISSILE PROTECTION

The CGS missile protection design basis conforms to 10 CFR 50, Appendix A, General Design Criteria for Nuclear Power Plants, Criterion 4, Environmental and Missile Design Bases. The objectives of missile protection design are to ensure that the plant can be brought to and kept in a safe shutdown mode and to prevent offsite radiological consequences assuming an additional single component failure.

The primary provisions incorporated into the design of the CGS facility to preclude damage to the systems necessary to achieve and maintain a safe plant shutdown are separation and redundancy. These provisions provide that in the event of an accident plus an additional active component failure, where a system required for safe (cool) shutdown is rendered unavailable, enough systems are left available to bring the plant to a safe (cool) shutdown without allowing any offsite radiological consequences. This redundancy and separation is obtained by the deliberate routing of systems, by the presence of structural floors, walls, structural steel members, and adjacent equipment which serve as barriers.

Design against generated missiles involves an initial selection process to define postulated missiles, an evaluation of postulated missile credibility, then a damage assessment to evaluate the effects of credible missiles, and finally, if necessary, to ensure safe shutdown, the provision of barriers or physical modifications of systems and components to preclude damage.

Structures housing systems and components essential for safe shutdown are designed to withstand externally generated missiles so that essential systems and components are not damaged by such missiles or by the secondary effects of such missiles.

3.5.1 MISSILE SELECTION AND DESCRIPTIONS

3.5.1.1 Internally Generated Missiles (Outside Containment)

3.5.1.1.1 Systems Available for Safe Shutdown

Systems available outside containment to facilitate safe shutdown include: high-pressure core spray (HPCS), low-pressure core spray (LPCS), residual heat removal (RHR), standby service water (SW), reactor core isolation cooling (RCIC), control rod drives (CRD), and the reactor feedwater system (RFW). These systems and their function are described in Sections 4.6, 5.4.9, 7.3, and 7.4.

Figures 3.5-1 through 3.5-15 illustrate the location of these systems. Seismic categories, quality group classifications, and reference sections are provided in Table 3.5-1.

3.5.1.1.2 Missiles Due to Rotating Equipment Failure

The systems located outside the primary containment have been reviewed to identify potential rotating equipment missiles. The design objective is to prevent the generation of missiles and their effects.

All rotating equipment (e.g., pumps, turbines, fans, and compressors) outside the primary containment have been evaluated to determine missile generation potential (postulated missiles), missile credibility, and an analysis of credible missile effects was completed. Credible missiles outside containment, missile sources, safety-related systems requiring protection (if any), and the extent of damage to safety-related systems (if any) are listed in Table 3.5-2. All emergency core cooling systems (ECCS) rotating equipment outside the primary containment are grouped by division in different rooms or areas of the plant, separated by walls or barriers, so that a single missile cannot damage redundant systems. The walls or barriers are designed to contain all missiles.

The RCIC turbine is prevented from reaching a runaway speed, where component failure could occur, by overspeed tripping devices. However, the RCIC turbine, similar to all plant rotating equipment, is also evaluated for credible missile generation at normal full speed operation. In addition, as with the ECCS systems, the RCIC turbine is located in a separate compartment.

3.5.1.1.3 Missiles Due to Pressurized Component Failure

The potential of the following equipment to generate missiles was investigated:

a. High energy piping

Pressurized piping in systems where the service temperature exceeded 200°F and/or the service pressure exceeded 275 psig was evaluated for potential generation of missiles. High energy piping pipe whip is discussed in Section 3.6.

b. Valve bonnets

1. Pressurized seal bonnets

Valves with an American National Standards Institute (ANSI) rating of 900 psig and above are pressurized seal bonnet type valves, constructed in accordance with the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (B&PV) Code Section III. Valve bonnets, on pressure seal bonnet type valves, are prevented from becoming missiles by the retaining ring, which would have to fail in shear, and by the yoke, which would capture the bonnet or reduce bonnet energy.

The bonnet bolts preload the pressure seal gasket to seal the valve initially. When pressurized, the valve is sealed by process fluid pressure and the bonnet bolts are under no load. All ASME III, Class I, 900# bonnet seal type valves were analyzed per ASME B&PV Code Section III requirements. Valve design pressures used in these analyses were given by the ASME B&PV Code Section III, Division 1, Subsection NB, Figure NB-3545.1-2, for weld-end valves. Using a typical pressure seal valve, the total thrust load on the retaining ring and valve body was calculated. The results demonstrated that both the retaining ring and valve body meet the ASME B&PV Code Section III, Division 1, NB-3227, requirements for pressures much higher than the normal operating pressure of the valve.

The majority of valves inside containment have massive valve operators which are supported by the yoke. For these valves, the valve operators act as an additional limitation to the yoke becoming a missile.

For a yoke clamp to fail, it must be assumed that the retaining ring fails completely and instantaneously so that the bonnet could strike the yoke. The yoke is normally under no load and complete failure of the yoke clamp is not considered credible.

Because of the highly conservative design of the retaining ring of these valves, bonnet ejection is highly improbable, and hence bonnets are not considered credible missiles.

2. Bolted bonnets

Most valves of ANSI rating 600 psig and below are valves with bolted bonnets. Valve bonnets are prevented from becoming missiles by limiting stresses in the bonnet-to-body bolting material by requirements set forth in the ASME B&PV Code Section III and by designing flanges in accordance with applicable code requirements. Even if bolt failure were to occur, the likelihood of all bolts experiencing simultaneous complete severance failure is remote. A study of bolted valve bonnets was made in which 25% of the connective bolts in the circular pattern were assumed missing. The stresses occurring under operating conditions with these bolts missing were found to be within acceptable limits. The widespread use of valves with bolted bonnets and the lack of historical incidence of complete severance failures of bonnets confirms that bolted valve bonnets need not be considered credible missiles.

3. Screwed-typed bonnets

Some valves in the 1 in. to 1.5 in. size range have coarse threaded bonnets which screw into the valve body. These valves were analyzed and found to have low stress intensities in the bonnet retaining threads. The valve design stress intensities were found to be a minimum of 4.5 times the stress intensities that will be experienced by the valves. Because of the highly conservative design of these valves, they are not considered credible missiles.

c. Valve stems

Valve stems are not considered credible missiles if at least one feature in addition to the stem threads is included in their design to prevent ejection; for example, valves with backseats.

d. Thermowells and sample probes

Temperature or other detectors installed on piping or in wells are evaluated as potential missiles if failure of a single circumferential weld would cause their ejection. This is highly improbable since a complete and sudden failure of a circumferential weld is needed for a detector to become a missile. These circumferential welds were analyzed and found to have yield stress values from six to 20 times the stress intensities that will be experienced in service. Because of their highly conservative design, thermowells and sample probes are not considered credible missiles.

e. Nuts and bolts

Nuts, bolts, nut and bolt combinations, and nut and stud combinations are unlikely to fail because of the low stress intensities for these parts. The ASME and ANSI Codes limit the allowable stresses in bolts and studs to 20% to 30% of yield. These low stress intensities are ensured by measuring the torque of all bolts, studs, and nuts during installation. Because of their highly conservative design, nuts and bolts are not considered credible missiles.

f. Blind flanges

Bolted blind flanges are not considered credible missiles because of the extremely unlikely occurrence of all bolts experiencing simultaneous complete severance failure as discussed in Item b.2 above.

g. Nitrogen tanks and bottles

Nitrogen tanks and bottles in the reactor building provide nitrogen for CRDs, charging of main steam safety/relief valve (SRV), isolation valve accumulator tanks, and instrument nitrogen inside containment. These tanks and bottles have design pressures considerably in excess of their operating pressures. Because of their highly conservative design, installed nitrogen tanks and bottles are not considered credible missiles.

No credible missiles are in a position to impact any of the nitrogen tanks or bottles.

3.5.1.1.4 Evaluation of Postulated Missiles

a. Assessment of postulated missile credibility

Postulated missiles are analyzed to determine if a credible failure mode resulting in a missile exists. Failure modes determined to be credible are then assessed for impact on plant safe shutdown.

b. Assessment of potential credible missile damage

The ability of the plant to achieve safe shutdown is ensured by physical separation and redundancy of safety-related systems. The adequacy of the physical separation and redundancy of safety-related systems was evaluated using the following procedure:

1. Target determination

Based on the missile location and orientation, the target areas are predicted. Trajectories are selected to encompass the most adverse conditions. The essential systems within that region are assumed damaged and not available for a safe shutdown;

2. Evaluation of system damage

The essential systems which are available after the worst postulated missile accident and the most critical additional single failure are determined. An evaluation is then made to determine whether these remaining systems are sufficient to achieve safe reactor shutdown; and

3. Protection of systems

When the separation and redundancy of the essential systems is not adequate, or when a redundant system is not available, one or more of the following measures are taken to ensure safe shutdown:

(a) The orientation of the credible missile is changed so that systems necessary for safe shutdown are not damaged,

(b) Missile barriers are provided, and

(c) It is shown that the essential components will not be damaged by the credible missile.

c. Determination of missile energies

One of the following methods is used to calculate the extent of the damage caused by a credible missile:

1. Piston-type missiles

The velocity of a piston-type missile is calculated by assuming that the work done will be converted into kinetic energy of the missile with no losses of energy due to friction or air resistance.

Work is the integral of force times displacement, while the kinetic energy of the missile is one-half the product of the missile mass times the square of the missile velocity. Assuming the applied force constant (PA_o), the kinetic energy is equated to the work done (Reference 3.5-1). Subsequently, the missile velocity is obtained by the expression:

$$V = \left[\frac{2\ PA_o\ L}{m}\right]^{1/2}$$ (Reference 3.5-1)

where:

V = the initial velocity at the end of a piston stroke (ft/sec)

P = pressure of the fluid (psi)

A_o = cross-sectional area of the piston (in.^{2})

L = length of the stroke in ft

m = mass of missile (lb-sec^2/ft)

2. Jet-propelled missiles

Postulated jet-propelled missiles are propelled by fluid escaping from a pressurized system in which there is essentially no lateral containment of the fluid. The escaping jet will not only impinge on the missile, but will also flow around and past the missile.

The velocity of this type of postulated missile is estimated by (Reference 3.5-1):

$$\left(1-\frac{V}{V_f}\right) - \text{Log}_e\left(1-\frac{V}{V_f}\right) = K_1 - \frac{K_2}{N_o + X \text{ Tan } B}$$

where:

$$K_1 = \left(1-\frac{V_o}{V_f}\right) - \text{Log}_e\left(1-\frac{V_o}{V_f}\right) + \frac{K_2}{N_o}$$

$$K_2 = \frac{A_o A_m P_b}{m\pi(\text{Tan } B)}$$

V = missile velocity at distance X (fps)

V_f = jet velocity = (fps)

N_o = radius of throat (ft)

P_f = density of the jet fluid (lb-sec^2/ft^4)

X = distance traveled (ft)

B = angle of jet expansion, degrees from normal

V_o = initial velocity of missiles

A_o = throat area (ft^2)

A_m = cross-sectional area of missile (ft^2)

m = mass of missile ($lb\text{-}sec^2/sec$)

3. Stored strain energy missiles

Stored strain energy missiles are assumed to convert all the strain energy at which they fail into kinetic energy. The velocity is calculated from the following formula (Reference 3.5-1):

$$V = \left[\frac{g}{EW}\right]^{1/2} S$$

where:

V = missile velocity (ft/sec)

E = modulus of elasticity (lb/ft^2)

W = specific weight of missile (lb/ft^3)

S = ultimate stress in the missile before failure (lb/ft^2)

g = acceleration of gravity (ft/sec^2)

4. Rotating Machinery

A variety of missiles from rotating machinery can be treated by considering each as a rotating block. Because it is part of a rotating structure, the block is considered to be initially rotating about its axis of revolution at a speed, ω, radians per second. The kinetic energy (KE) of the block is given by (Reference 3.5-4):

$$KE = \frac{1}{2}\left[Rcg^2 + K^2\right]\left(\frac{w}{g}\right)(\omega)^2, \text{ ft} - \text{lb}$$

where:

Rcg = radius to the center of gravity (CG) of the rotating block, measured from the initial axis of rotation in the machinery, ft

K = radius of gyration of the rotating block, about the cg axis of the rotating block

w = block weight, lb

g = acceleration of gravity, ft/sec^2

ω = angular velocity, rad/sec

In this expression the Rcg term gives the kinetic energy due to translation, while the K term gives the kinetic energy due to rotation of the block about its cg axis.

3.5.1.1.5 Example of Postulated Missile Evaluation

a. Assessment of postulated missile credibility

The reactor protection system motor generator sets in the critical dc switchgear rooms in the reactor building (el. 467 ft 0 in.) were analyzed to determine their credibility as missiles. A structural failure of the 1800 rpm flywheel during the normal operation of the motor generator would produce high energy missiles with the potential to damage systems, components, or structures in their paths. Motor generator set modification to eliminate or contain the flywheel missiles was evaluated, but a feasible modification was not practical. The flywheels were, therefore, credible missiles.

b. Assessment of potential credible missile damage

The flywheel missiles were postulated to exit the motor generator sets along a plane perpendicular to the motor generator set, with the missile exiting a maximum of 10 degrees from the perpendicular plane.

1. Target determination

   The potential targets for the flywheel missiles were determined by reviewing the applicable drawings and visual inspection of the target area in the switchgear rooms. It was determined that safety-related cables in these rooms could potentially be damaged.

2. Evaluation of system damage

   The safety-related cables which could be damaged by the flywheel provide dc power to instrument panels in the control room and to isolation valves inside containment. Damage to these cables was determined to be unacceptable.

3. Protection of systems

   It was determined that there was not a feasible method of providing that the cables would not be damaged by the flywheel missiles. To preclude damage to the safety-related cables, the following alternatives were investigated:

   (a) The motor generator sets were analyzed to determine if a change in orientation was feasible. This was not a feasible alternative.

   (b) The feasibility of constructing a barrier around the flywheel was investigated. This was a feasible alternative.

c. Barrier design

A missile barrier proved to be the only feasible alternative. The barrier was designed to contain the highest energy missile that could be produced by the flywheel. The barrier was constructed of steel and energy-absorbing aluminum honeycomb material and firmly anchored to the concrete floor. This eliminated the effects of the credible missile. A tabulation of plant systems protected by missile barriers is provided in Table 3.5-3.

3.5.1.2 <u>Internally Generated Missiles (Inside Containment)</u>

3.5.1.2.1 Systems Available for Safe Shutdown

Figures 3.5-16 through 3.5-32 describe the mechanical and instrumentation locations of systems available for a safe shutdown. Each system (LPCS, HPCS, RHR, ADS, CRD, and primary containment) is color coded to specify the location of structures, systems, or components. In addition, the reactor protection system and containment isolation valves inside containment are available for safe shutdown of the plant and to prevent offsite radiological consequences. Information pertaining to applicable seismic category, quality group classification, and reference sections where these systems are described is provided in Table 3.5-4. The evaluation of credible missile kinetic energies and missile target determinations is discussed in Section 3.5.1.1.4. Target and barrier damage evaluations are discussed in Section 3.5.3.

3.5.1.2.2 Missiles Due to Rotating Equipment

Rotating equipment inside containment consists of the following:

a. Recirculation pump and motor

The most substantial piece of nuclear steam supply system (NSSS) rotating machinery is the recirculation pump and motor. This potential missile source is discussed in Reference 3.5-4.

It is concluded in Reference 3.5-4 that destructive pump overspeed is highly improbable. If it occurred, it could result in failure of certain pump and motor components having the potential to become missiles. A careful examination of the pump and motor structure shows that rotor or shaft failure will not result in ejection of motor-generated missiles, and impeller missiles cannot penetrate the pump case. Reference 3.5-4 concludes that in the unlikely event of impeller failure resulting in ejection of missiles through ruptured pipe, penetration of containment by missile fragments is highly improbable. Evaluation of the effects on safety-related systems of impeller fragments that might be ejected from openings in ruptured pipe is not evaluated because of the extreme improbability of this event and because the effects would not be more severe than the assumed consequences of jet impingement due to pipe inside containment as discussed in Section 3.6. The recirculation pump and motor are, therefore, not considered to be credible missile sources.

b. Fans as potential missiles

The fans inside primary containment are designed such that the casing will restrain any possible missile. Therefore, fans and parts thereof are not considered as possible sources of missiles.

3.5.1.2.3 Missiles Due to Pressurized Component Failure

A discussion of the potential for missile generation from the failure of pressurized components, e.g., valve stems, valve bonnets, and temperature element assemblies, is presented in

Section 3.5.1.1.3. That discussion is also applicable to pressurized components inside containment. In addition, SRV and main steam isolation valve (MSIV) accumulators are particular to inside containment.

Pressurized ASME III vessels, such as SRV and MSIV accumulators, are not considered credible missiles. These vessels have low stresses and operate in the “moderate energy” range and, therefore, any failures would be a crack-type and not of concern for missile generation.

All potential sources of postulated missiles inside the primary containment were analyzed to determine missile credibility utilizing the criteria discussed above and in Section 3.5.1.1.3, as required by General Design Criterion 4, “Environmental and Missile Design Basis.” It was determined that all postulated missiles inside the primary containment incorporated design features that eliminated their credibility as potential sources of missiles.

3.5.1.2.4 Falling Objects

Structural elements, equipment, and components inside containment which could be considered as potential falling objects are evaluated in accordance with Section 3.7.2.8.

3.5.1.2.5 Secondary Missiles Generated by Postulated Credible Primary Missiles

Secondary missiles are not considered credible missiles due to their low probability of occurrence and their low kinetic energy levels. The probability of damage due to a secondary missile is the probability of occurrence of a primary missile times the probability of hitting a part that can become a secondary missile times the probability that the part will actually become a missile. This probability is very low.

The level of stored kinetic energy in a secondary missile will be low because of the large energy required to produce a secondary missile. In addition, no reliable method to predict secondary missile characteristics is known, other than those characteristics in common with primary missiles.

3.5.1.3 Turbine Missiles

In April 1992, the original Westinghouse shrunk-on-disc type LP rotors were removed and new Westinghouse Fully Integral LP rotors were installed. The LP turbine missile analysis is based on Fully Integral rotors using a probabilistic method (Reference 3.5-27).

It is concluded that the probability of damage to safety-related systems by turbine missiles is acceptably low, due to (a) the protection provided by reinforced-concrete structural barriers, (b) the calculated probability of turbine missile generation, and (c) periodic testing and inspection of turbine overspeed protection systems with associated corrective action as required.

3.5.1.3.1 Safety-Related Targets

Target areas which are evaluated for capability to protect safety-related equipment, components, and systems from postulated turbine missiles consist of the following:

a. Vertical targets

1. Reactor building north exterior wall,

2. Control room north wall, and

3. North wall of vertical cable chase, between reactor building and control room.

b. Horizontal targets

1. Reactor building refueling floor,

2. Roof over vertical cables chase, and

3. Floor slab above control room.

3.5.1.3.2 Turbine Placement and Orientation

Figure 3.5-33 shows the turbine generator layout relative to safety-related plant structures and turbine missile target areas. Also shown on this drawing is the reinforced-concrete shield wall which acts as a barrier for protection of some safety-related targets from postulated low trajectory turbine missiles. A cross-sectional view through the turbine building and reactor building is shown in Figure 3.5-34 to indicate relative elevations of the turbine and target areas. See Figures 1.2-4, 1.2-8 and 1.2-16 for a general arrangement drawing of the turbine building, reactor building, and control building at the turbine operating floor elevation. CGS has an "unfavorable oriented" turbine generator in relation to the identified safety-related targets.

3.5.1.3.3 Missile Identification and Characteristics

3.5.1.3.3.1 <u>High Pressure Turbine</u>. Postulated missiles from the high pressure turbine (HPT) are shown in Reference 3.5-22 to have insufficient energy to penetrate the casing at normal operating speed. At 20% overspeed (120% of normal, or rated speed), HPT missiles are postulated to penetrate the casing, but at velocities too low to reach safety-related targets. The minimum bursting speed of the high pressure turbine rotor, based on minimum specified mechanical properties of the rotor material, is 300% of the rated speed (Reference 3.5-6).

The maximum speed at which the unit may rotate is 3200 rpm, which is 178% of rated speed 1800 rpm (Reference 3.5-27). At this speed the highest stressed low pressure turbine disc would fracture, damaging the turbine to the extent that additional overspeed would not be possible. Therefore, high pressure turbine missiles are not considered.

3.5.1.3.3.2 Low Pressure Turbine. Each low pressure turbine consists of a double-flow rotor assembly, an outer cylinder, two inner cylinders, and stationary blade rings. The rotor is a fully integral, single forging consisting of the shaft, disks, and couplings without any shrink fits or keyways and is made with low stress corrosion cracking (SCC) susceptible materials.

Westinghouse Report "Analysis of the Probability of the Generation of Missiles from Fully Integral Nuclear Low Pressure Rotors" (Reference 3.5-27) states

> To assess the probability of missile generation resulting from a burst of a fully integral nuclear low pressure rotor, five potential failure mechanisms are considered:
>
> a. Ductile burst,
>
> b. Fracture resulting from high cycle fatigue cracking,
>
> c. Fracture resulting from low cycle fatigue cracking-startup/shutdown cycles,
>
> d. Fracture resulting from stress corrosion cracking, and
>
> e. Destructive overspeed.
>
> All the listed failure mechanisms demonstrate that the new Fully Integral rotor design significantly reduces the likelihood of missile generation, except for the destructive overspeed mechanism.
>
> The probability of reaching a destructive overspeed for the Fully Integral LPT rotor is primarily dependent on the DEH control system, which functions to avoid this condition.

3.5.1.3.4 Strike and Damage Probability

A probabilistic approach is adopted to assess the possibility of damage to systems required for safe shutdown or of accidents which could result in potential offsite exposure due to high trajectory missiles. The probability of this occurring is represented by combined probabilities of

$$P_4 = P_1 \times P_2 \times P_3$$

where:

P_1 = missile generation probability

P_2 = probability of a missile striking a structure or component required for safe shutdown or whose failure could result in release of radioactivity

P_3 = probability of significant damage to the structure or component

P_4 = combined overall probability

The terms and assumptions applicable to this analysis follow the procedures outlined by S. H. Bush in Reference 3.5-7.

3.5.1.3.4.1 Missile Generation Probability. Missile generation probability is based on the failure probability of the Fully Integral rotor design (installed 1992) and the turbine overspeed protection system. The dominant failure mechanisms of the rotor are SCC and turbine overspeed. The P1 values for SCC were provided by Westinghouse (References 3.5-27 and 3.5-28) and meet the requirements of the NRC Safety Evaluation Report (SER) for specific turbine inspection and testing intervals. However, the SER states the NRC position that the LP rotor operation, maintenance, and inspection programs be based on the P1 values being less than the 1×10^{-5} probability requirement for loading the turbine without inspection or inservice restriction. This value is applicable to the CGS unfavorably oriented turbines.

The P1 values for the CGS low pressure turbine Fully Integral rotors for stress corrosion cracking are

Inspection Interval (years)	P1 at Rated Running Speed	P1 at 20% Overspeed
4	4.0×10^{-14}	1.4×10^{-14}
8	8.8×10^{-11}	2.0×10^{-11}
12	4.3×10^{-9}	7.7×10^{-10}
16	5.1×10^{-8}	7.9×10^{-9}
32	8.3×10^{-6}	8.9×10^{-7}

The probability for SCC at 20% overspeed (120% of normal or rated speed) is lower than the SCC at rated speed condition, and therefore not considered in determining the overall strike and damage probability of turbine-generated missiles (Reference 3.5-28).

As with previous rotor designs (including the CGS original built-up rotors and current Fully Integral rotors), the potential exists for SCC to be the dominant failure mechanism depending on inspection interval (Reference 3.5-29). However, in Fully Integral rotor designs, the probability of failure by the SCC mechanism has been reduced drastically and the analysis shows that at the normal inspection interval established by the preventive maintenance (PM) program of 80,000 to 100,000 running hours (translates to 10-12 years), the contribution from SCC to P1 would be approximately 4.3×10^{-9}. Considering typical use factors for nuclear turbines, and considering that the crack locations are readily observable during normal turbine maintenance, it is concluded that, notwithstanding the normal PM inspection program, periodic inspections are not required within the expected life of the turbine before exceeding the NRC safety criteria of P1 $\leq 1 \times 10^{-5}$.

The methodology developed to calculate the probability of missile generation due to overspeed events identifies three turbine overspeed events that can result from the failure of the turbine valves to close following a system separation or a total loss of load (References 3.5-30 and 3.5-31). These overspeed events are design overspeed (approximately 120% of rated turbine speed), intermediate overspeed (approximately 130%), and destructive overspeed (runaway speed in excess of approximately 180%). Design overspeed is assumed when a system separation occurs and a turbine trip does not occur at event initiation, one or more governor valves or two or more interceptor valves fail to close immediately, and a successful overspeed trip closes the throttle valves and reheat stop valves. Intermediate overspeed is assumed to occur when there is a system separation and one or more alignment of reheat stop valves and interceptor valves fail to close. Destructive overspeed is assumed to occur when a system separation occurs and at least one governor valve and one throttle valve in the same steam chest fail to close.

Missile generation probability analyses (References 3.5-30 and 3.5-31), indicate that the design and intermediate overspeed failure probabilities are not major contributors to turbine missile ejection probability for plants with fully integral low pressure rotors.

The P1 value for turbine overspeed is derived from the probability of failure of the turbine overspeed protection system in a manner that results in destructive overspeed and includes consideration for periodic turbine valve testing. Additionally, the P1 value includes conservatism to account for design and intermediate overspeed missile generation. The overspeed protection system is a fault tolerant, redundant, digital electro-hydraulic turbine control and overspeed protection system (installed-May 2007). The system has the capability of failure detection and on-line repair of critical components. The missile generation probability analysis for potential turbine overspeed yields a missile generation probability of 8.2×10^{-7} for P1 based on a three month turbine valve testing interval.

Combining the SCC mechanism (assuming inspection at 12 year and 32 year intervals) and the destructive overspeed mechanism to establish a combined P1 value for missile generation, yields a probability of turbine missile generation of 8.2×10^{-7} and 9.1×10^{-6}, respectively.

3.5.1.3.4.2 Strike Probability (P2) and Damage Probability (P3). Rather than performing elaborate calculations for site-specific strike and damage probabilities as previously performed for the disc type rotors, generally accepted industry typical values were used for P1 and P3. For CGS, the product of P2 * P3 is 1×10^{-2} for unfavorably oriented turbine generators (Reference 3.5-28).

3.5.1.3.4.3 Combined Overall Probability (P4). Assuming the longest turbine inspection interval of 32 years and a destructive overspeed probability assuming a three month turbine valve test interval, the highest overall damage probability for postulated missiles is then calculated by

$$
\begin{aligned}
P4 &= P1 * P2 * P3 \\
&= (9.1 \times 10^{-6}) * (1 \times 10^{-2}) \\
&= 9.1 \times 10^{-8} \text{ (which is acceptably less than the NRC safety criteria of } P4 \leq 1 \times 10^{-7})
\end{aligned}
$$

3.5.1.3.5 Turbine Overspeed Protection System and Testing

A single failure in the overspeed sensing and turbine trip systems will not prevent overspeed protection from operating. The turbine generator is equipped with a fault tolerant and redundant DEH control system.

The DEH control system, with its overspeed protection features and inspection and testing requirements, is described in Sections 7.7.1.5 and 10.2.

3.5.1.3.6 Turbine Valve Testing

Turbine valve testing is discussed in Section 10.2.

3.5.1.3.7 Turbine Characteristics

For information characterizing the Westinghouse turbines used at CGS, refer to Westinghouse report covering the effects of a high pressure turbine rotor fracture (Reference 3.5-6) and low pressure turbine forged integral rotor fractures, (Reference 3.5-27). Therein, the low-pressure disc materials, manufacturing processes, and operating conditions are stated.

3.5.1.4 Missiles Generated by Natural Phenomena

The consideration of potential missiles injected or suspended in a tornado wind stream is based on References 3.5-8 and 3.5-12.

All Seismic Category I and safety-related structures and components are designed to include the effects of missiles generated by the design basis tornado described in Section 3.3.2 except as noted in Sections 3.5.1.4.1 and 3.5.2. Missiles are categorized as either external or internal missiles. External missiles are materials and/or items usually found outside and in the immediate vicinity of the buildings, whereas internal missiles are materials and/or items found inside the buildings. Descriptions, properties, and impact velocities of the design-basis tornado-generated external missiles are shown in Table 3.5-5. Tornado missile protection provided by plant structures is described in Section 3.5.1.4.1. Those structures have adequate thickness to prevent penetration, perforations, and backface spalling. The basis for evaluating missile penetration is discussed in Section 3.5.3. The protection provided for external and internal missiles is discussed in Sections 3.5.1.4.1 and 3.5.1.4.2, respectively.

Figures 1.2-1 through 1.2-24 indicate the location of structures, equipment, and components protected against tornado-generated missiles.

3.5.1.4.1 Tornado-Generated External Missiles

Structures that house systems, equipment, and components essential to safe shutdown are designed to withstand the effects of design-basis tornado-generated missiles described in Section 3.5.1.4. These structures provide protection by the following means:

a. Reactor building

The location of the reactor building with respect to the other plant structures is illustrated in Figure 1.2-1. Portions of the reactor building exterior walls are protected by adjacent structures against direct impact of tornado generated missiles, as indicated in Figure 3.5-35. The exterior walls of the reactor building, up to the refueling floor at el. 606 ft 10.5 in., are capable of withstanding the impact of the design basis tornado generated missiles. The exterior walls are constructed of 4 ft thick reinforced concrete to el. 471 ft 0 in. which is 30 ft above plant finish grade. From el. 471 ft 0 in. to the refueling floor at el. 606 ft 10.5 in., the exposed exterior walls are constructed of reinforced concrete, 18 in. minimum in thickness. The reactor building exterior wall thickness from plant grade to the refueling floor at el. 606 ft 10.5 in. is adequate to prevent design basis missile penetration and spalling of concrete. (See Figures 3.5-36 through 3.5-38.)

The refueling floor at el. 606 ft 10.5 in. comprises the spent fuel storage pool, dryer/separator pool, and various other items of refueling equipment. The reactor building walls and roof above this floor are constructed of insulated metal siding and insulated metal roof decking erected on a superstructure consisting of a structural steel frame as indicated in Figures 1.2-11 and 1.2-12. The superstructure also supports the overhead bridge crane. The refueling floor at el. 606 ft 10.5 in. is constructed of reinforced concrete of various thicknesses, with a minimum thickness of 18 in. The walls and floor of the pools have a minimum thickness of 5 ft. The equipment on the refueling floor is not required for safe shutdown. The refueling floor and the walls and floor of the pools are sufficiently thick to withstand the effects of the design basis missiles, and to prevent secondary missile effects caused by spalling of concrete. A missile impacting the spent fuel pool does not have sufficient energy to damage the equipment and spent fuel located in the pool, as discussed in Reference 3.5-12.

b. Diesel generator building

The exposed exterior walls of the structure are constructed of reinforced concrete with a minimum thickness of 2 ft 8 in. The roof has a minimum thickness of 1 ft 6 in. The thicknesses of walls and roof are sufficient to withstand the effects of the design-basis tornado-generated missiles. Figures 1.2-22 and 3.5-42 through 3.5-44 illustrate this structure.

c. Radwaste and control building

The exposed exterior concrete walls and roofs, housing safety-related systems, equipment, and components are designed to withstand the effects of the design-basis tornado-generated missiles. Figures 1.2-13 through 1.2-17 illustrate the radwaste and control building and their relative location in the plant complex. The exterior walls that house safety-related equipment have a minimum thickness of 2 ft. The roof of the portion of the building that houses safety-related equipment is 1 ft 6 in. thick (see Figure 3.5-45).

d. Standby service water pump houses and spray ponds

The exterior walls of both pump houses are constructed of reinforced concrete and are 2 ft 4 in. thick, minimum. The roofs of the pump houses are 1 ft thick. These thicknesses are adequate to withstand design-basis tornado-generated missiles. In addition, the two pump houses are redundant to each other. In the event that one pump house is inoperable, the other is capable of providing sufficient service water for safe shutdown. See Figures 3.5-46 and 3.5-47.

The ability of the spray ponds to tolerate the design-basis tornado-generated missiles is discussed in Section 3.3.2.3.

Figure 1.2-20 illustrates the pump houses and spray ponds.

e. Makeup water pump house

The exterior walls and roof of the makeup water pump house are of reinforced concrete and are sufficiently thick to withstand the effects of the design-basis tornado-generated missiles, as discussed in Section 3.3.2.3. The exterior walls, with the exception of the equipment access opening, are 2 ft 4 in. thick and the roof slab is 1 ft 4 in. thick. Figures 1.2-1, 1.2-23 and 1.2-24 furnish its location and arrangement. The 8 ft 0 in. by 10 ft 0 in. equipment access opening is protected by a 14-in.-thick concrete door.

f. Turbine building

Safety-related components in the turbine building are located in areas where tornado missile protection is provided by reinforced-concrete exterior walls, a minimum of 18 in. in thickness, and two reinforced-concrete slabs overhead, at el. 471 ft and 501 ft. Figures 1.2-2 through 1.2-6 illustrate the turbine building.

g. Safety-related systems outside of tornado-hardened structures

The protection provided to the pipe lines and electrical lines located underground between the spray ponds, SW pump houses, makeup water pump houses, reactor building, and the diesel generator building is described in Sections 3.5.2 and 3.5.3. The protection provided to the critical electrical trays in the corridor between the turbine generator building and the radwaste and control building is also described in Sections 3.5.2 and 3.5.3.

3.5.1.4.2 Tornado-Generated Internal Missiles

The tornado-generated internal missiles as mentioned in Section 3.5.1.4 are materials and/or items attached to or found inside a building, but subjected to the design basis tornado described in Section 3.3.2 as a result of a loss of a building exterior wall or roof. The materials and/or items considered as potential tornado-generated internal missiles are discussed below.

a. The reactor building steel framed superstructure uses girts and roof purlins fastened to the building frame by means of controlled release fasteners. The steel girts and purlins are considered to become free falling tornado-generated internal missiles which can strike the roof of the diesel generator building, the radwaste and control building, and main steam corridor slabs, in the event a tornado blows the roofing and/or siding off of the building frame. Structures housing safety-related systems, equipment, and components are designed to withstand the effects of these missiles.

b. In the event that a tornado blows the roof purlins, roof decking, girts, and siding panels off the reactor building frame, the reactor building crane is then exposed to the design basis tornado. The reactor building crane is designed with provisions which preclude it, or any part thereof, from becoming a missile (see Section 3.3.2.3).

3.5.1.4.3 Flood Generated Missiles

The design basis flood el. discussed in Section 3.4 and defined in Section 2.4, exceeds the flood levels associated with breaches of the Grand Coulee Dam. The final plant grade level is higher than the design basis flood. Therefore, flood-generated missiles are not considered in the design of the Seismic Category I safety-related structures and installations.

3.5.1.4.4 Protection and Design

Systems protected from missiles generated by natural phenomena, and barrier design are described in Sections 3.5.2 and 3.5.3 respectively.

3.5.1.5 <u>Missiles Generated by Events Near the Site</u>

Hazards due to missiles postulated in the design basis explosions or accidents at nearby industrial plants, military facilities, pipe lines, or storage facilities can be discounted as discussed in Section 2.2.

The Hydrogen Storage and Supply Facility (HSSF) contains a liquid hydrogen storage tank, ASME tubes (gaseous hydrogen), trailer tubes (gaseous hydrogen) and a hydrogen pipeline to the plant. An analysis shows that an explosion and subsequent missile generation from a random tank rupture at normal pressure would not affect the plant due to the remote distance of the facility. Another analysis shows that an overpressurization event and subsequent rupture of the liquid hydrogen tank is a credible event. However, the total annual probability of impact for any of the missiles generated is less than 10^{-7} and does not meet the threshold for consideration as a design basis event. The hydrogen storage containers have relief valves to prevent overpressurization.

3.5.1.6 Aircraft Hazards

NUREG-0800, Standard Review Plan, Section 3.5.1.6, "Aircraft Hazards," provides guidance to ensure that the risks from aircraft hazards are low enough for nuclear power plants. If the distance at which aircraft activity occurs meets all the criteria provided in the following, the probability of an aircraft accident resulting in radiological consequences greater than the exposure limits in 10 CFR 50.67 can be considered to be less than about 10^{-7} per year and no additional analysis is required (Reference 3.5-16):

a. The plant-to-airport distance (D) is between 5 and 10 statute miles, and the projected annual number of operations is less than 500 D^2, or D is greater than 10 statute miles and the number of operations is less than 1000 D^2,

b. The plant is at least 5 statute miles from the edge of military training routes, including low-level training routes, except for those associated with a use greater than 1000 flights per year or where activities (such as practice bombing) may create an unusual stress situation, and

c. The plant is at least 2 statute miles beyond the nearest edge of a federal airway, holding pattern, or approach pattern.

Additionally, aviation crashes associated with general aviation (light planes, $<$12,500 lb) may be excluded from this analysis, as tornado design-basis requirements establish structural requirements sufficient to protect safety-related structures, systems, and components against these types of events (Reference 3.5-17).

3.5.1.6.1 Airports

There are no active airports located within 10 statute miles of the CGS site, but there are three commercial airports and six private airports which are active within 20 miles of the site. The annual number of operations of flights from these airports of less than 1000 D^2 is satisfied for all aviation uses at the commercial airports, including general, airtaxi, freight, and commercial traffic.

3.5.1.6.2 Military Airspace Use

The airspace over the Hanford site is periodically used as a marshaling area for military aircraft participating in training missions at the Yakima Training Center. A low-level military training route associated with this training passes above the western edge of the Hanford site about 18 miles west of CGS (Reference 3.5-18). Thus Criterion 2 associated with military use is also satisfied.

3.5.1.6.3 Federal Airways and Airport Approaches

CGS does not satisfy the third criterion identified in Section 3.5.1.6 of being at least 2 statute miles beyond the nearest federal airway, holding pattern, or approach pattern. Therefore, the probability of an aircraft accident resulting in an exposure greater than 10 CFR 50.67 limits has been evaluated.

Figure 2.2-3 shows the commercial airports, low-level federal airways, and airport instrument approaches in the vicinity of the CGS site. Airway V187 (with a minimum altitude of 3500 ft mean sea level) passes over the CGS site. The 14 nautical mile (NM) instrumented approach pattern is also important to the CGS site. Aircraft tend to maintain significant altitude over the Hanford site because of a request by the Federal Aviation Administration (FAA) (Reference 3.5-18).

The Tri-City Airport traffic supervisor provided a conservative estimate of 7500 flights per year that can be expected through airway V187 and the 14 NM approach (Reference 3.5-19). The number obtained is for flights by aircraft that are under instrument flight rule (IFR) flight plans. Aircraft that leave the airport in the general direction of V187 and that are under visual flight rules do not have to file a flight plan and are not required to follow the airway. For the purpose of this evaluation, the total number of aircraft flying through the airspace over CGS is assumed to be those constrained to the airway V187 and those using the 14 NM instrument-approach path (i.e., 7500 per year). Other flight paths are either sufficiently far away (more than 2 miles) with sufficiently low volume or are not currently active.

The aircraft using the Richland Airport instrumented approach path (localized approach) may also pass over the CGS airspace. However, the only aircraft over 12,500 lb expected to use this approach would be ambulance (Life-Flight) aircraft using the Richland Airport under adverse weather conditions. Other aircraft activities possibly impacting the CGS site include activities conducted by/for the Department of Energy or their contractors on the Hanford site. These typically consist of less than 10 low-level flights per year over the entire Hanford site airspace (Reference 3.5-19). The additional risk posed by these operations is expected to be very low and is sufficiently bound by the conservatisms used in the above estimates.

The probability per year (P_{PY}) of an aircraft crashing into the plant can be expressed as (Reference 3.5-16) equation 3.5.1.6-1:

$$P_{PY} = C \text{ x } N \text{ x } A_{eff}/w$$

where:

C = In-flight crash rate per mile for aircraft using airway

N = Number of flights per year along the airway

w = Width of airway in miles

A_{eff} = Effective area of the plant in square miles.

The effective area of the facility is the ground surface area covered by all flight crash trajectories that could impact the surface structure. This area is larger than the facility itself because of the possibility of the aircraft skidding across the ground before hitting the facility as well as the possibility of a direct hit on the structure before striking the ground. The effective area is the sum of the structure's area, the shadow area (behind the structure), and the aircraft skid area (in front of the structure).

3.5.1.6.4 Summary

The probability per year, estimated per Reference 3.5-16, is 9.07 x 10^{-8}. Thus, the probability of an aircraft accident resulting in radiological consequences exceeding 10 CFR 50.67 limits is below 10^{-7}/year. Therefore, aircraft are discounted as credible missiles and accidents involving aircraft are not considered design basis events.

3.5.2 SYSTEMS TO BE PROTECTED

The structures, systems, and components necessary for bringing the plant to a safe shutdown and the protection provided for these structures, systems, and components from missiles is discussed in Sections 3.5.1 and 3.5.2.

Protection provided for the safety-related structures located outdoors against tornado-generated missiles is described in Section 3.5.1.4 and by turbine missiles in Section 3.5.1.3.

The plant structures, systems, equipment, and components that are required to bring the plant to a safe shutdown condition, or whose failure could lead to offsite radiological consequences under accident conditions, are protected from external (outdoor) missiles by barrier structures or redundant systems as follows:

a. The exposed exterior walls of safety-related structures are designed to protect internal structures, systems, and components. Structures with such exposed exterior walls are described in Section 3.8.4 and shown in figures referred to therein. Figure 3.5-35 illustrates exterior walls subject to tornado-generated missile impact.

b. All openings for heating, ventilation, and air conditioning system fresh air intakes (FAI) and exhausts (EXH), in buildings housing safety-related equipment, are protected against externally generated missiles as indicated in Table 3.5-6. Examples are the louvered openings above the floor at el. 572 ft 0 in. in the north and south walls of the reactor building. These openings are protected by a labyrinth of missile shield walls inside the opening.

c. The SW pipelines and electrical lines between the SW pump houses, the reactor building, and the diesel generator building are located below grade and are protected from external missiles by sufficient Quality Class I earth cover of high relative density (described in Section 3.5.3). The TMU system is required for safe shutdown only when both spray ring headers are lost to tornado missiles (see Section 3.3.2).

d. The two SW pump houses, the tower makeup water pump house, and the valve boxes along the cooling tower makeup water lines are tornado-hardened (see Section 3.5.3).

e. As described in Sections 3.8.4.1.5 and 3.3.2.3, the spray ponds (except spray headers) are below grade and are protected from external missiles by sufficient Quality Class I earth backfill of high relative density. The spray headers are not required to be protected from tornado missiles.

f. The critical electrical trays in the corridor between the turbine generator building and the radwaste and control building are protected from possible missiles by the cantilever wall on column line 17 (Figure 3.5-48) and the additional 10 ft 0 in. high cantilever barrier wall immediately east of it. The 10 ft 0 in. high barrier wall provides missile protection for the door at el. 441 ft 0 in. on column line 17. Although the trays are above the door level and do not require missile protection provided by the 10 ft 0 in. high wall, the protection is provided, as shown in Figure 3.5-48.

g. The 18 in. CW piping used to provide service water return flow to the CW basin following a tornado has been analyzed to withstand the effects of all postulated tornado missiles, assuming that it is embedded in soil with the surface of the soil flush with the top of the pipe.

h. The tower makeup water (TMU) pipe lines and electrical lines between the TMU pump house and the SW pump houses are located below grade and are protected from external missiles by sufficient earth cover.

i. Portions of the following underground and aboveground piping associated with the diesel oil system are safety related. However, tornado missile protection is not provided to that piping. In the event of damage to the fill and vent connections due to tornado missiles, there are tank pump out connections and unused flanged connections on the storage tank which can be used as fill and vent openings.

1. Diesel oil day tank overflow and drain to storage tanks,
2. Diesel oil storage tank fill and vent lines, and
3. Filter-polisher unit recirculation piping.

3.5.3 BARRIER DESIGN

The barrier design objectives emphasize missile containment and structural integrity without secondary missile generation.

The overall response of barriers subject to impact are investigated by the use of general energy equations given in Reference 3.5-9. On determination of penetration depth and duration of impact, an effective dynamic force is computed. The additional calculation of the natural period of the target structure and the selection of a ductility ratio facilitates the determination of the required structural resistance. In this manner, missile impact is translated to an equivalent static load in an effort to quantify bending moments and shear. The detailed method used for predicting the overall response of missile barriers, including the forcing function method of determining ductility in structural elements and the basis for the ductility ratios used in the calculations, is provided in Appendix C of Reference 3.5-13 that was approved by the NRC.

3.5.3.1 Concrete Barriers

Concrete missile barriers are designed in accordance with penetration equations such as the modified Petry equation (Reference 3.5-2), the Ballistic Research Laboratory formula (Reference 3.5-1), or the modified NDRC formula (Reference 3.5-14). In all cases, except for barriers exposed to turbine missiles, a minimum concrete thickness of 2.2 times the penetration thickness determined for an infinitely thick slab is provided to prevent perforation, spalling, or scabbing. For discussion of turbine-generated missiles, see Section 3.5.1.3.

3.5.3.2 Steel Barriers

The Ballistic Research Laboratories Formula (Reference 3.5-1) is used to determine penetration depths of missiles into steel barriers.

3.5.3.3 Earth Barriers

When the protective barrier is of earthen origin, the soil penetration studies are based on alternate techniques. Buried safety-related piping and electrical systems required for a safe shutdown are ensured adequate protection from tornado-generated missiles. An embedment depth of 5 ft was calculated to provide acceptable protection for the original design-basis tornado/missile parameters. The current design-basis tornado/missile parameters are less severe and are used for any required evaluations for protection from postulated missiles.

3.5.3.4 Applications

Examples of barrier design are as follows:

Steel covers for manholes containing cabling for safety-related equipment required for safe shutdown are designed to withstand tornado-generated missile impact and associated wind pressure. These 2 ft 9 in. circular steel plates are designed using conventional elastic analysis and design methods for determining stress and strain. The design adopted uses two 1-1/8-in. plates of ASTM A 514 steel plate to prevent penetration and blowout.

The reactor building vehicle air lock (railroad bay) exterior doors and the SW pump house exterior equipment doors are designed and certified by the manufacturer to withstand the effects of tornado-generated exterior missiles as described in Section 3.5.1.4.

All other doors in Seismic Category I and safety-related structures are not designed to withstand the effects of the missiles described in Section 3.5.1. These doors are backed up, wherever missile protection is required, with reinforced-concrete walls forming a labyrinth behind the door. Similarly, louvers in exterior walls, which are vulnerable to missile penetration, are backed up by reinforced-concrete plenums or walls.

Based on the selection and description of missiles cited in Section 3.5.1, the interaction of missiles with structural elements is determined and the results are given in Section 3.5.1.4.1. The tabulations assume the missiles to impact at the most vulnerable point of a structure or component (e.g., at the center of a slab).

The reactor protection system motor generator sets flywheels located in the critical dc switchgear rooms at el. 467 ft 0 in. in the radwaste building were analyzed and determined to be credible missile sources, with the potential consequences affecting the safe shutdown of the plant. Barriers were constructed around these flywheels of steel and aluminum honeycomb material, which were designed to contain the credible missiles (see Section 3.5.1.1.5).

The SW piping between the SW pump houses, the reactor building, and the diesel generator building is provided with sufficient cover for protection from tornado-generated missiles (see Section 3.5.3.3). The SW piping exits the pump houses at a centerline el. of 435 ft 3 in. and immediately turns down at a 45 degree angle to el. 432 ft, where the piping is routed to the reactor building in high relative density Quality Class I backfill. Grade level is at 440 ft 6 in., providing an embedment depth of over 7 ft from the top of the pipe. Where the pipe exits the pump houses, a 1.5-in. asphaltic concrete road with a 6-in. base coarse and 2-in. leveling coarse bed provides additional protection from tornado-generated missiles. Additionally, the two SW loops are separated by at least 20 ft to preclude loss of redundancy.

The tower makeup water piping to the river is embedded in sufficient soil cover for protection of tornado-generated missiles (see Section 3.5.3.3).

The control room remote air intake structure piping is embedded a minimum of 5 ft for protection from tornado-generated missiles. The remote air intake structures are also missile-hardened.

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* Applicable for original HPT and LPT Rotors, not the replacement Fully Integral (FI) LPT rotors. Superseded by 3.5-27.

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Table 3.5-1

Systems Description Outside Containment

Systems Available for a Safe Shutdown	Function	Section	Figures	Seismic/Quality Class
RCIC	Maintain RPV water inventory	5.4.6, 7.4.1.1	3.5-9 through 3.5-14	I/I
HPCS	Maintain RPV water inventory	6.3, 7.3.1.1.1.1	3.5-9 through 3.5-14	I/I
SW	Heat rejection	7.3.1.1.6	3.5-9 through 3.5-14	I/I
RHR A B C	Maintain water inventory and decay heat removal	5.2, 7.3.1.1.1.4, 6.3, 5.4.7	3.5-1 through 3.5-8	I/I
CRD	Reactivity control	7.7.1.2	3.5-15	I/I
RFW	Maintain RPV water inventory	5.4.9	3.5-9 through 3.5-14	--
LPCS	Maintain RPV water inventory	6.3, 7.3.1.1.1.3	3.5-1 through 3.5-8	I/I

Note: Identification of missiles to be protected against, their sources, and bases for selection are discussed in Section 3.5.1.1.3. The ability of the structures, systems, and components to withstand the effects of selected internally generated missiles is discussed in Section 3.5.1.1.2.

Table 3.5-2

Internally Generated Missiles Outside Containment

Missile	Equipment Number	Description	Seismic/ Quality Class	Resolution Code Notes
F-7	RRA-FN-6	RCIC pump room fan coil fan	I/I	(a)
F-9	RRA-FN-2	MS fan coil fan	I/I	(a)
F-11	RRA-FN-7	RR lock fan coil fan	II/II	(b)
F-27	RRA-FN-16	Air conditioning unit fan	II/II	(a)
F-33	REA-FN-2A	Exhaust fan	II/II	(b)
F-35	REA-FN-2B	Exhaust fan	II/II	(b)
F-37	REA-FN-1A	Exhaust fan	II/II	(b)
F-39	REA-FN-1B	Exhaust fan	II/II	(b)
F-41	ROA-FN-1A	Supply fan	II/II	(b)
F-43	ROA-FN-1B	Supply fan	II/II	(b)
F-57	RRA-FN-3	RHR pump room fan coil fan	I/I	(b)
F-59	REA-FN-15	Exhaust fan	II/II	(b)
F-61	RRA-FC-19	FPC heat exchanger and pump room cooler	I/I	(b)
F-62	RRA-FC-20	FPC heat exchanger and pump room cooler	I/I	(b)
P-1	LPCS-P-2	LPCS water leg pump	I/I	(d)
P-3	HPCS-P-3	HPCS water leg pump	I/I	(d)
P-5	CRD-P-1A	CRD pump C12-C001A	II/II	(d)
P-6	CRD-P-1B	CRD pump C12-C001B	II/II	(d)

Table 3.5-2

Internally Generated Missiles
Outside Containment (Continued)

Missile	Equipment Number	Description	Seismic/ Quality Class	Resolution Code Notes
P-7	RHR-P-3	RHR water leg pump	I/I	(d)
P-9	COND-P-3	Reactor building condensate supply pump	II/II	(d)
P-10	RCIC-P-4	RCIC condensate pump	I/I	(d)
P-11	RCIC-P-2	RCIC vacuum pump	I/I	(d)
P-12	RCIC-DT-1	RCIC turbine drive	I/I	(d)
P-13	RCIC-P-1	RCIC pump	I/I	(d)
P-14	COND-P-4	Radwaste building condensate supply pump	II/II	(c)
P-15	COND-P-5	Condensate filter demineralizer backwash pump	II/II	(c)
P-16	RCIC-P-3	RCIC water leg pump	I/I	(d)
P-17	FPC-P-3	Suppression pool cleanup pump	II/II	(b)
P-18	RHR-P-2A	RHR pump	I/I	(b)
P-19	RHR-P-2B	RHR pump	I/I	(b)
P-20	RWCU-P-1A	Cleanup circulation pump	II/II	(b)
P-21	RWCU-P-1B	Cleanup circulation pump	II/II	(b)
P-24	RCC-P-1A	RBCC water pump	II/II	(b)
P-25	RCC-P-1B	RBCC water pump	II/II	(b)
P-26	RCC-P-1C	RBCC water pump	II/II	(b)
P-27	RCC-P-2	RBCC chemical metering pump	II/II	(b)

Table 3.5-2

Internally Generated Missiles
Outside Containment (Continued)

Missile	Equipment Number	Description	Seismic/ Quality Class	Resolution Code Notes
P-28	FPC-P-1A	Fuel pool cooling pump	I/(II/I)	(b) (g)
P-29	FPC-P-1B	Fuel pool cooling pump	I/(II/I)	(b) (g)
P-30	PWC-P-4A	Potable water pump	II/G	(b)
P-31	PWC-P-4B	Potable water pump	II/G	(b)
P-32	EDR-P-5A EDR-P-5B	Reactor building equipment drain sump pumps	II/II	(b)
P-33	FDR-P-1A	Reactor building floor drain sump pump	II/II	(b)
P-34	FDR-P-1B	Reactor building floor drain sump pump	II/II	(b)
P-35	FDR-P-2	Reactor building floor drain sump pump	II/II	(b)
P-36	FDR-P-3	Reactor building floor drain sump pump	II/II	(b)
P-37	FDR-P-4A	Reactor building drywell floor sump pump	II/II	(b)
P-38	FDR-P-4B	Reactor building drywell floor sump pump	II/II	(b)
P-39	ROA-P-1A	Air washer pump	II/II	(b)
P-40	ROA-P-1B	Air washer pump	II/II	(b)
F-70	DEA-FN-51	Fan (in MS tunnel)	I/I	(b)
R-1	RPS-M/GEN-1	RPS motor generator set	II/II	(e)
R-2	RPS-M/GEN-2	RPS motor generator set	II/II	(e)
F-71	RRA-FN-8	MS tunnel fan coil fan	I/I	(b)

Table 3.5-2

Internally Generated Missiles
Outside Containment (Continued)

Missile	Equipment Number	Description	Seismic/ Quality Class	Resolution Code Notes
F-72	RRA-FN-9	MS tunnel fan coil fan	I/I	(b)
R-5	DSA-C-1C	Starting air compressor HPCS diesel generator	I/I	(b)
R-6	DSA-C-2C	Starting air compressor HPCS diesel generator	I/I	(b)
R-7	DSA-C-1A1	Starting air compressor diesel generator A1	I/I	(f)
R-8	DSA-C-1A2	Starting air compressor diesel generator A2	I/I	(f)
R-9	DSA-C-1B1	Starting air compressor diesel generator B1	I/I	(f)
R-10	DSA-C-1B2	Starting air compressor diesel generator B2	I/I	(f)
R-11	DLO-P-3A1	Motor driven lube oil pump diesel generator A1	I/I	(b)
R-12	DLO-P-3A2	Motor driven lube oil pump diesel generator A2	I/I	(b)
R-13	DLO-P-3B1	Motor driven lube oil pump diesel generator B1	I/I	(b)
R-14	DLO-P-3B2	Motor driven lube oil pump diesel generator B2	I/I	(b)
R-15	DO-P-3A1	Motor driven lube oil pump	I/I	(b)
R-16	DO-P-3A2	Motor driven lube oil pump	I/I	(b)
R-17	DO-P-3B1	Motor driven lube oil pump	I/I	(b)
R-18	DO-P-3B2	Motor driven lube oil pump	I/I	(b)
R-19	DO-P-6	dc motor driven HPCS fuel pump	I/I	(b)

Table 3.5-2

Internally Generated Missiles
Outside Containment (Continued)

Missile	Equipment Number	Description	Seismic/ Quality Class	Resolution Code Notes
F-73	DEA-FN-13	Exhaust fan loop A pump room	IM/I	(b)
F-74	DEA-FN-23	Exhaust fan loop B pump room	IM/I	(b)
F-75	DEA-FN-33	Exhaust fan HPCS oil pump room	IM/I	(b)
F-76	DEA-FN-32	Exhaust fan day tank room	IM/I	(b)
F-77	DEA-FN-12	Exhaust fan day tank room	IM/I	(b)
F-78	DEA-FN-22	Exhaust fan day tank room	IM/I	(b)
F-79	DEA-FN-34	Exhaust fan diesel generator room	I/I	(b)
F-80	DRA-EUH-11	Electric unit heater loop A oil pump room	II/II	(b)
F-81	DRA-EUH-21	Electric unit heater loop B oil pump room	II/II	(b)
F-82	DRA-EUH-31	Electric unit heater HPCS oil pump room	II/II	(b)
F-83	DRA-EUH-32	Electric unit heater fire deluge equipment room	II/II	(b)
F-84	DRA-EUH-33	Electric unit heater fire deluge equipment room	II/II	(b)
F-85	DMA-AH-32	Air handling HPCS diesel generator room	I/I	(b)
F-86	DMA-AH-12	Air handling Division 1 diesel generator room	I/I	(b)
F-87	DMA-AH-22	Air handling Division 2 diesel generator room	I/I	(b)
F-88	DEA-FN-11	Exhaust fan Division 1 diesel generator room	I/I	(b)

Table 3.5-2

Internally Generated Missiles
Outside Containment (Continued)

Missile	Equipment Number	Description	Seismic/ Quality Class	Resolution Code Notes
F-89	DEA-FN-21	Exhaust fan Division 2 diesel generator room	I/I	(b)
F-90	DEA-FN-31	Exhaust fan HPCS diesel generator room	I/I	(b)

RESOLUTION CODE NOTES:

[a] These postulated missiles originate from air conditioning fan coil units. These fan coil units are contained in housings capable of containing any potential fan missile. The air outlets have grating installed to prevent potential missiles from exiting via this route. The air inlets are into the fan impeller eye and thus the fan inlets are not a missile exit path. Missiles originating from this equipment cannot get beyond the protective housing.

[b] These postulated missiles were determined to be credible. A safe shutdown analysis was performed, which determined that the failure of all equipment in the missile path envelope would have no effect on the ability of the plant to safely shut down in the event of an accident.

[c] These postulated missiles are impeller fragments originating from the radwaste building condensate supply pump and the condensate filter demineralizer backwater pump. These missiles were assumed to be credible. A safe shutdown analysis was performed, which determined that the failure of all equipment in the missile path would have no effect on the ability of the plant to shut down in the event of an accident, with the exception of impacts on one 20-in. standby service water pipe with standard (3/8-in.) wall thickness. A worst case missile, which would transmit the maximum energy to a point of contact with the nearest standby service water pipe was analyzed.

This worst-case missile and impact could not penetrate the standby service water pipes. The target pipe would sustain acceptable deformation in which the pressure integrity of the target pipe is not affected.

This analysis demonstrated that these pumps could not generate a missile capable of compromising the safe shutdown functions of the service water pipes.

Table 3.5-2

Internally Generated Missiles
Outside Containment (Continued)

Resolution Code Notes (Continued):

[d] These postulated missiles are pump impeller and turbine blade fragments. Analysis has shown that the impeller and turbine casings have sufficient thickness to contain each postulated missile. Postulated missiles originating from this equipment cannot be ejected beyond the protective casings.

[e] These postulated missiles are flywheel fragments from the RPS motor-generator sets. A safe shutdown analysis was performed which showed critical Division 1 and 2 electrical power and control cables were in the potential missile path that were required to safely shut down the plant in the event of an accident. Missile barriers were designed and installed around both motor-generator flywheels. These barriers ensure that potential missiles originating from the motor generators will have no effect on the ability of the plant to shut down in the event of an accident.

[f] These postulated missiles originate from the air compressors that provide starting air for the diesel generator sets. These four compressors are of the reciprocating three-cylinder piston type. The three cylinder heads cover the upper portion of the crankshaft and preclude any missile exiting the compressor from the upper half of the crankcase. Credible missiles were postulated to exit out the bottom half of the crankcase. A safe shutdown analysis was performed, which determined that the failure of all equipment in the missile path envelope would have no effect on the ability of the plant to safely shut down in the event of an accident.

[g] Pumps purchased or installed prior to Jan 1, 1980 shall be Quality Class II. Pumps purchased or installed after Jan 1, 1980 shall be Quality Class 1.

Table 3.5-3

Plant Systems Protected By Missile Barriers

The missile barrier for RPS motor-generator set 1, located in the Division 1 essential switchgear room in the radwaste building, el. 467 ft 0 in., protects electric power and control cables servicing the following equipment:

Power distribution panel DP-S1-1A

Power panel PP-7A-F

Inverter IN-3

Motor control center MC-S1-1D

Motor control center MC-7A

Power panel PP-7A

24 V dc power supply system DP-SO-A

The missile barrier for RPS motor-generator set 2, located in the Division 2 essential switchgear room in the radwaste building, el. 467 ft 0 in., protects electric power and control cables servicing the following equipment:

24 V dc power supply system PP-DP-SO-8

125 V dc power distribution panel DP-S1-2D

125 V dc power distribution panel DP-S1-2E

125 V dc power distribution panel DP-S1-2A

125 V dc power distribution panel MC-S1-2D

Motor control center MC-8A

Motor control center MC-8F

Instrument and control power panel PP-8A-F

Instrument and control power panel PP-8A-E

Motor control center MC-8B

Table 3.5-4

Systems Description Inside Containment

Systems Available for a Safe Shutdown	Function	Section	Figures	Seismic/Quality Class
RPS	Reactor protection	7.2.1.1, 4.6	3.5-29	I/I
CRD	Reactivity control	7.7.1.2	3.5-19 through 3.5-22, 3.5-25, 3.5-27, 3.5-28	I/I
Inboard isolation valves	Containment isolation	5.4.5, 6.2.4	--	I/I
HPCS	Maintain RPV water inventory	7.3.1.1.1.1, 6.3	3.5-16 through 3.5-18, 3.5-20, 3.5-25, 3.5-27, 3.5-29, 3.5-31	I/I
LPCS	Maintain RPV water inventory	7.3.1.1.1.3, 6.3	3.5-16 through 3.5-19, 3.5-26, 3.5-28, 3.5-29, 3.5-31	I/I
ADS	Depressurize RPV	5.2, 7.3.1.1.1.2	3.5-17, 3.5-29	I/I
RHR A B C	Maintain RPV water inventory and decay, heat removal	6.3, 5.4.7, 7.3.1.1.1.4	3.5-16 through 3.5-18, 3.5-20 3.5-26	I/I
Containment	Containment integrity	3.8.2	3.5-1 through 3.5-50, 3.8-49	I/I

Note: Identification of missiles to be protected against, their sources, and bases for selection are discussed in Section 3.5.1.2.3. The ability of the structures, systems, and components to withstand the effects of selected internally generated missiles is discussed in Section 3.5.1.2.2.

Table 3.5-5 Not Available For Public Viewing

Table 3.5-6

Location of and Missile Protection
Provided for Fresh Air Intakes (FAI) and Exhausts (EXH)

Building Location	Exterior Walls: Opening Type	Exterior Walls: Size	Opening Centerline Location: Elevation	Opening Centerline Location: Plan	Figure	Protection
Reactor						
South	FAI	6 ft 2-7/8 in. W x 17 ft 8 in. H	581 ft 8 in.	5 ft 8 in. west of col. line 6	3.5-36 3.5-37	Shielding wall
North	EXH	72 in. diameter	595 ft 8 in.	6 ft 0 in. west of col. line 6	3.5-36 3.5-38	Shielding wall
Radwaste and control						
West	FAI	4 ft 0 in. x 4 ft 0 in.	526 ft 6 in.	4 ft 6 in. north of col. line K.1	3.5-45 3.5-49 3.5-50	Shielding wall, slab and hood
Roof	EXH	3 ft 0 in. x 3 ft 0 in.	546 ft 0 in.	5 ft 0 in. north of col. line K.1 4 ft 0 in. east of col. line 15.1	3.5-39 3.5-40	Concrete shielding hood
Diesel Generator						
South	FAI	10 ft 11 in. W x 12 ft 8 in. H	449 ft 4 in.	5 ft 7.5 in. east of col. line 9.4	3.5-41 3.5-42	Shielding wall
South	FAI	10 ft 11 in. W x 12 ft 8 in. H	449 ft 4 in.	7 ft 5.5 in. east of col. line 7.4	3.5-41 3.5-42	Shielding wall
South	FAI	10 ft 11 in. W x 12 ft 8 in. H	449 ft 4 in.	6 ft 5.5 in. west of col. line 3.8	3.5-41 3.5-42	Shielding wall
South	EXH	36 in. diameter	480 ft 2.5 in.	3 ft 9 in. east of col. line 8.4	3.5-43 3.5-42	Penthouse
South	EXH	36 in. diameter	480 ft 2.5 in.	3 ft 9 in. west of col. line 7.4	3.5-42 3.5-43	Penthouse

Table 3.5-6

Location of and Missile Protection
Provided for Fresh Air Intakes (FAI)
and Exhausts (EXH) (Continued)

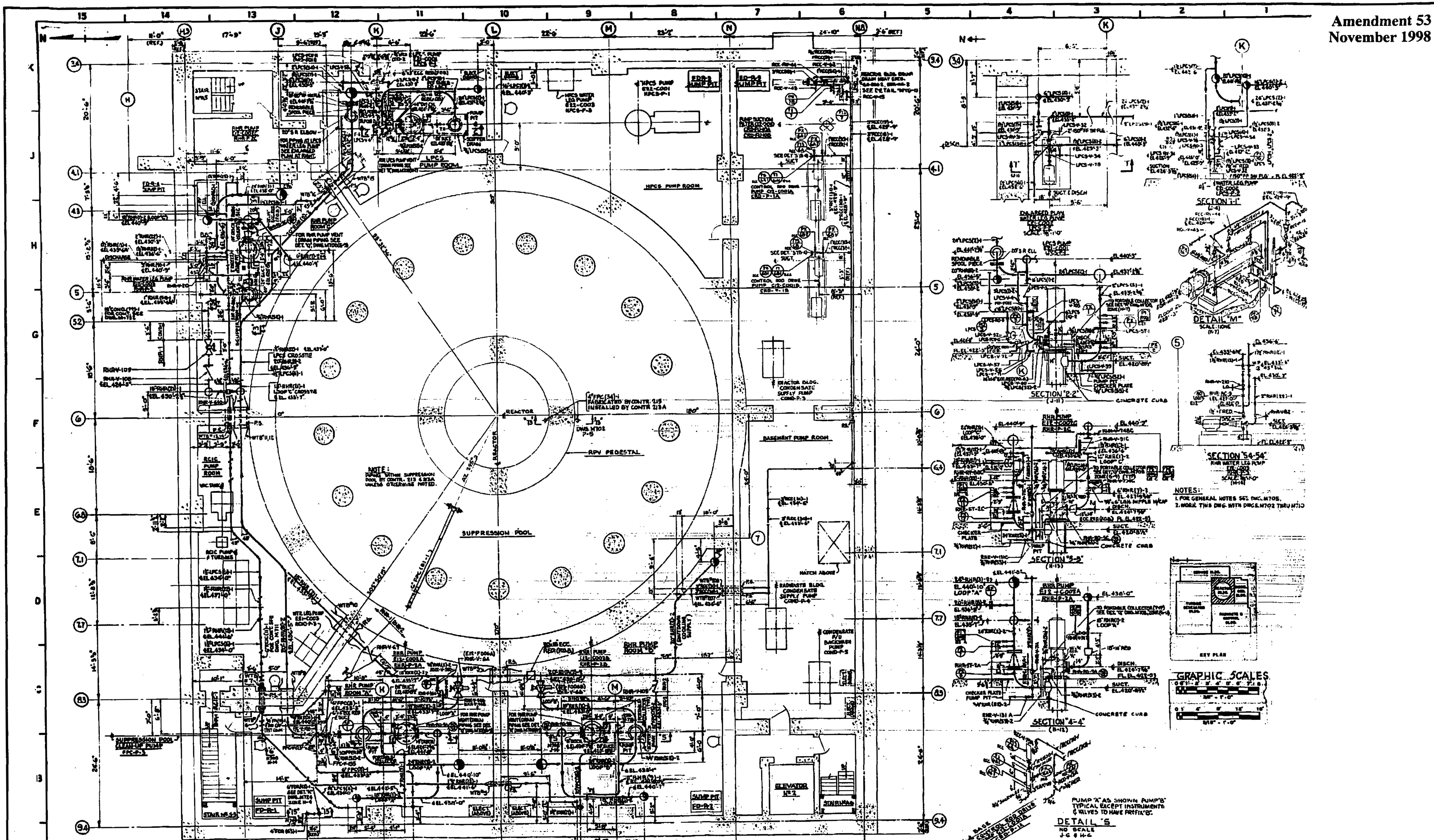
Building Location	Exterior Walls: Opening Type	Exterior Walls: Size	Opening Centerline Location: Elevation	Opening Centerline Location: Plan	Figure	Protection
Diesel Generator (Continued)						
South	EXH	36 in. diameter	480 ft 2.5 in.	3 ft 9 in. east of col. line 6.6	3.5-42 3.5-43	Penthouse
South	EXH	36 in. diameter	480 ft 2.5 in.	3 ft 9 in. west of col. line 5.5	3.5-42 3.5-43	Penthouse
South	EXH	36 in. diameter	480 ft 2.5 in.	2 ft 9 in. east of col. line 5.5	3.5-42 3.5-43	Penthouse
East	EXH	1 ft 0.5 in. x 1 ft 0.5 in.	452 ft 6 in.	12 ft 8 in. south of col. line P.1	3.5-44 3.5-42	Shielding beam MB5
East	FAI	1 ft 0.5 in. x 1 ft 0.5 in.	452 ft 6 in.	3 ft 2 in. north of col. line Q	3.5-44 3.5-42	Shielding beam MB5
East	EXH	1 ft 0.5 in. x 1 ft 0.5 in.	452 ft 6 in.	12 ft 10 in. south of col. line Q	3.5-44 3.5-42	Shielding beam MB5
East	FAI	1 ft 0.5 in. x 1 ft 0.5 in.	452 ft 6 in.	8 ft 2 in. north of col. line R	3.5-44 3.5-42	Shielding beam MB5
East	EXH	1 ft 0.5 in. x 1 ft 0.5 in.	452 ft 6 in.	7 ft 10 in. south of col. line R	3.5-44 3.5-42	Shielding beam MB5
East	FAI	1 ft 0.5 in. x 1 ft 0.5 in.	452 ft 6 in.	10 in. north of col. line P.1	3.5-44 3.5-42	Shielding beam MB5

Table 3.5-6

Location of and Missile Protection
Provided for Fresh Air Intakes (FAI)
and Exhausts (EXH) (Continued)

	Exterior Walls		Opening Centerline Location			
Building Location	Opening Type	Size	Elevation	Plan	Figure	Protection
SSWPH[a] 1A						
East	FAI	2 ft 6 in. x 2 ft 6 in.	437 ft 6 in.	3 ft 3 in. south of col. line A	3.5-46	Wall and floor
West	EXH	3 ft 5 in. x 3 ft 5 in.	451 ft 11 in.	3 ft 4.5 in. north of col. line E	3.5-46 3.5-47	Wall
SSWPH[a] 1B						
South	FAI	2 ft 6 in. x 2 ft 6 in.	437 ft 6 in.	3 ft 3 in. west of col. line A	3.5-46	Wall and floor
North	EXH	3 ft 5 in. x 3 ft 5 in.	451 ft 11 in.	3 ft 4.5 in. east of col. line E	3.5-46 3.5-47	Wall
FAIS[b]						
Roof	FAI	3 ft 3 in. x 3 ft 5 in.	441 ft 9 in.	Centerline of structure	3.8-49	Slabs, walls

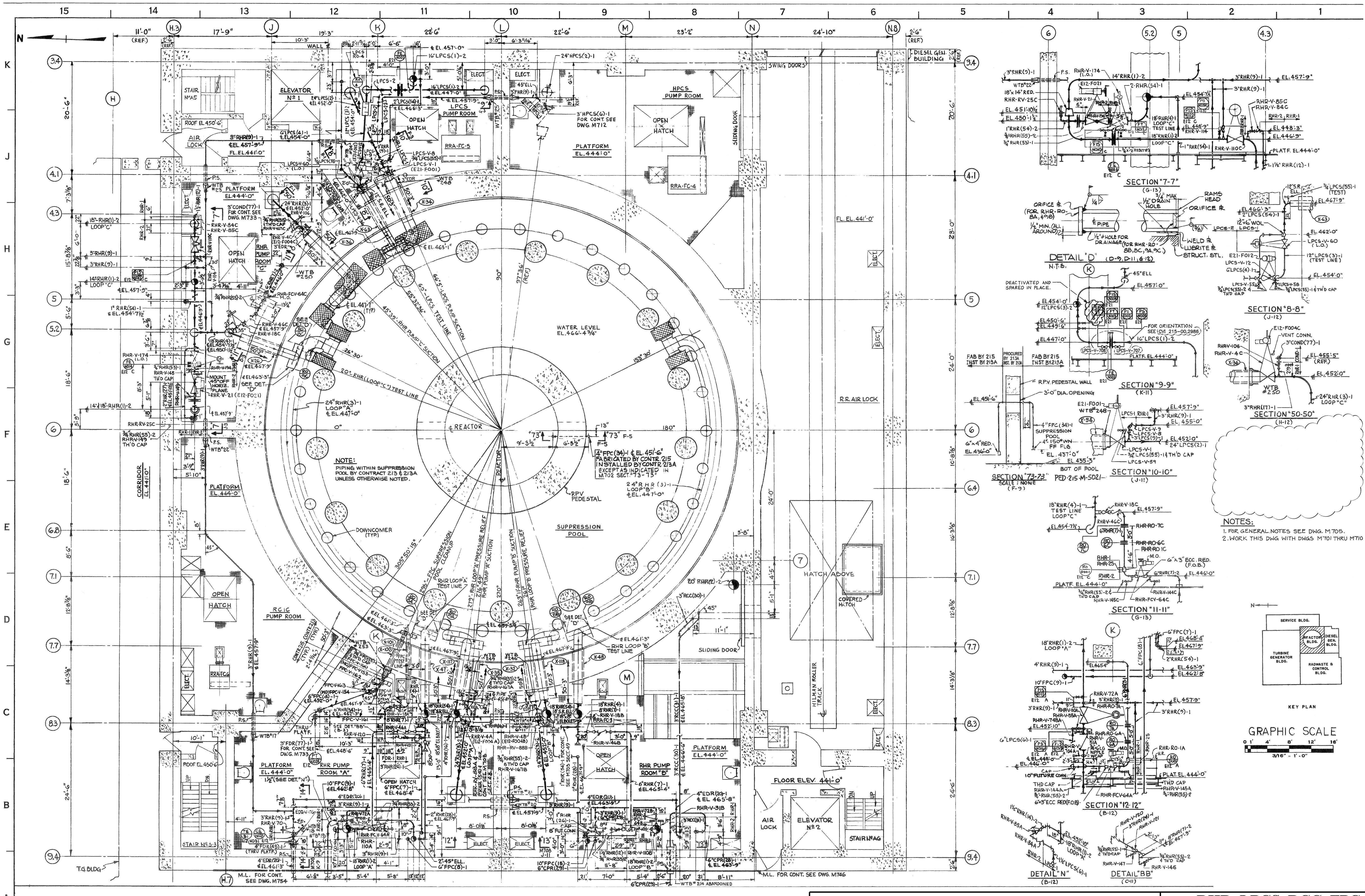
[a] Standby service water pump house
[b] Fresh air intake structures



Columbia Generating Station
Final Safety Analysis Report

RHR, LPCS, RCC, FPC, SLC, and CPR Plan, Sections and Details - El. 422 ft 3 in. - Reactor Building

Draw. No.	M701	Rev.	31	Figure	3.5-1

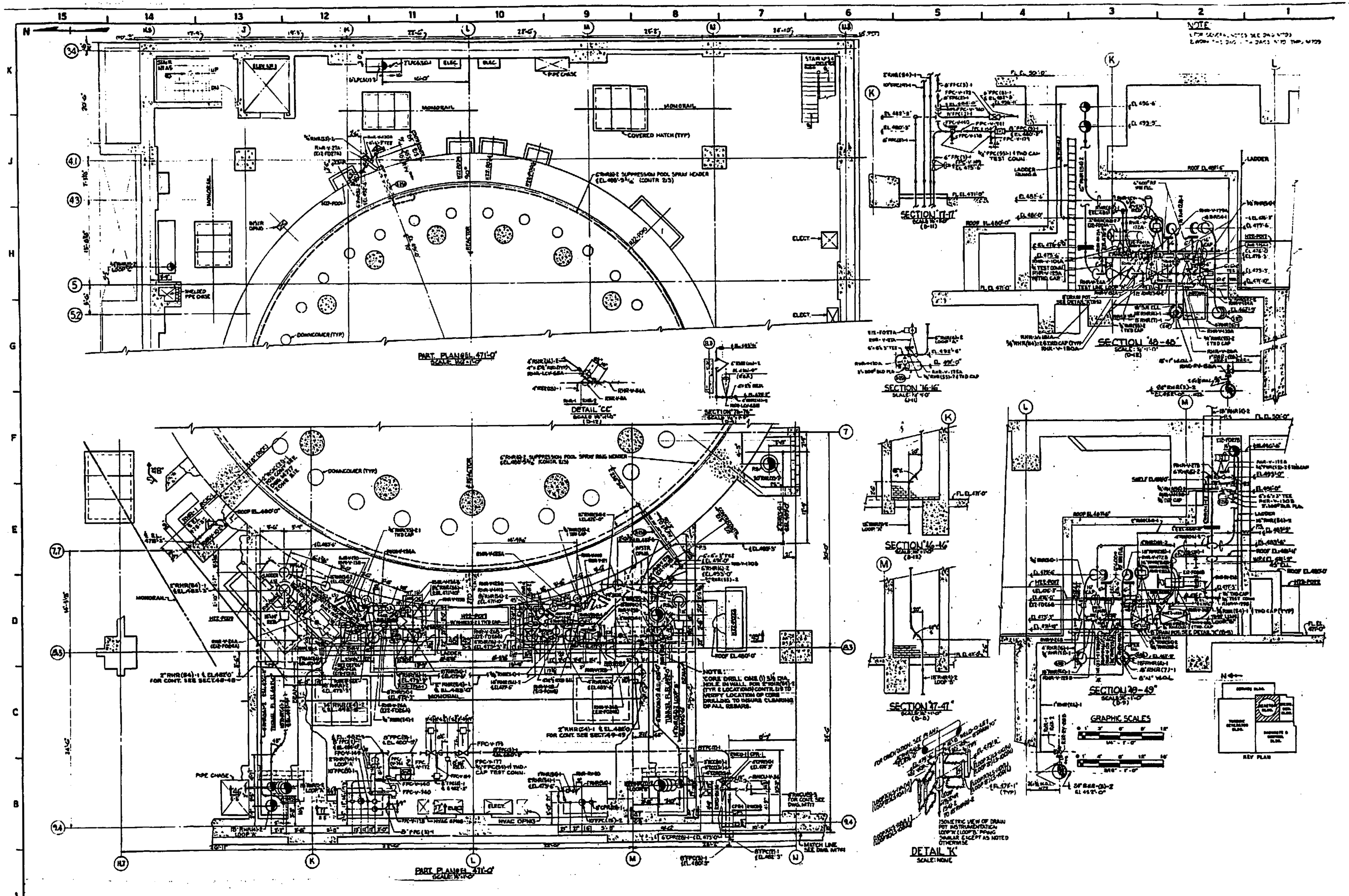


Columbia Generating Station
Final Safety Analysis Report

RHR, LPCS, RCC, FPC, and CPR Plan, Sections and Details - El. 444 ft 0 in. and 441 ft 0 in. Reactor Building

Draw. No. M702	Rev. 35	Figure 3.5-2

Amendment 53
November 1998



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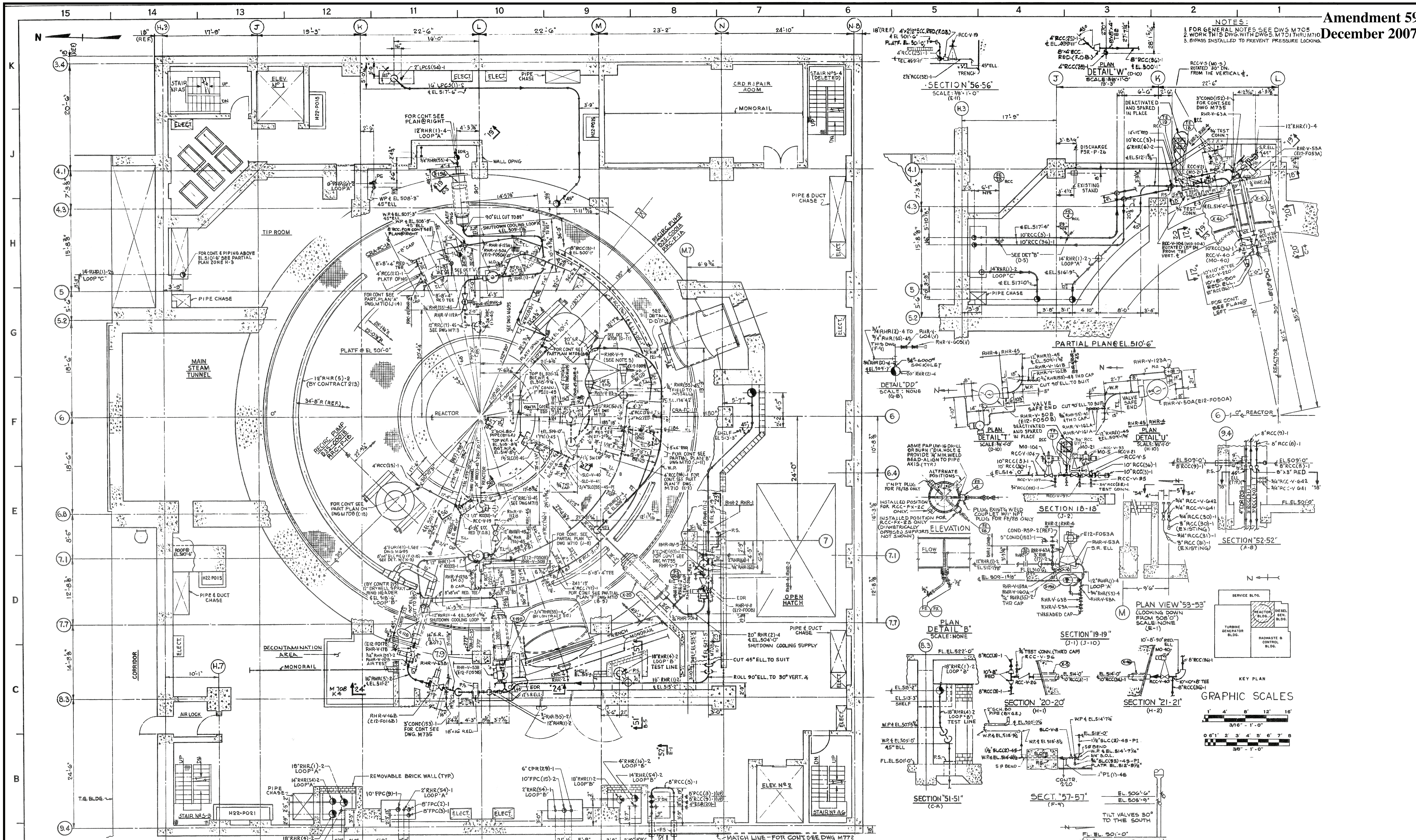
RHR, LPCS, RCC, FPC, SLC, and CPR Partial Plan, Sections and Details - El. 471 ft 0 in. - Reactor Building

Draw. No.	M703	Rev.	25	Figure	3.5-3



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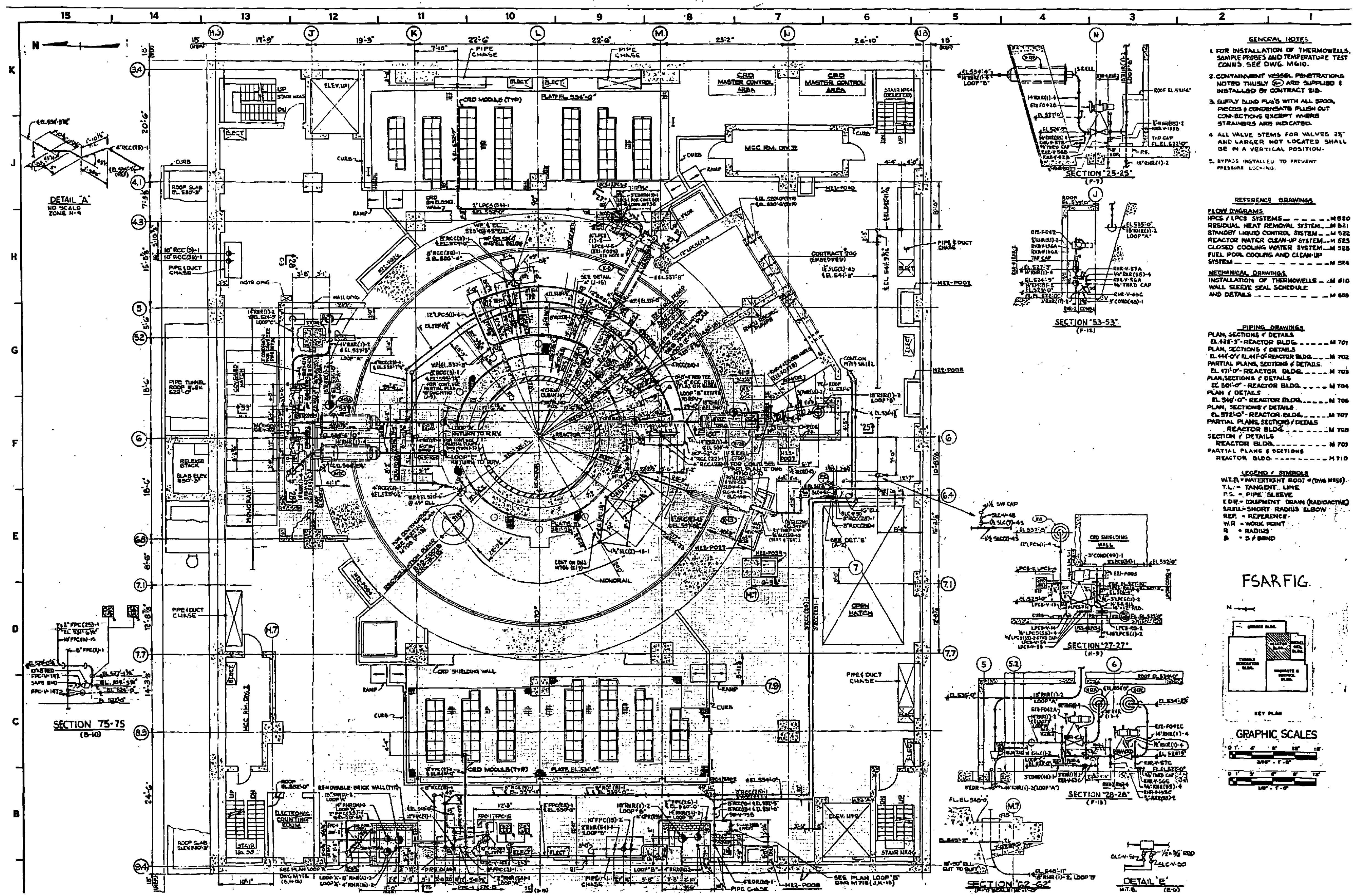
Final Safety Analysis Report

RHR, LPCS, RCC, FPC, SLC, and CPR Plan, Sections and Details – El. 501 ft 0 in. - Reactor Building

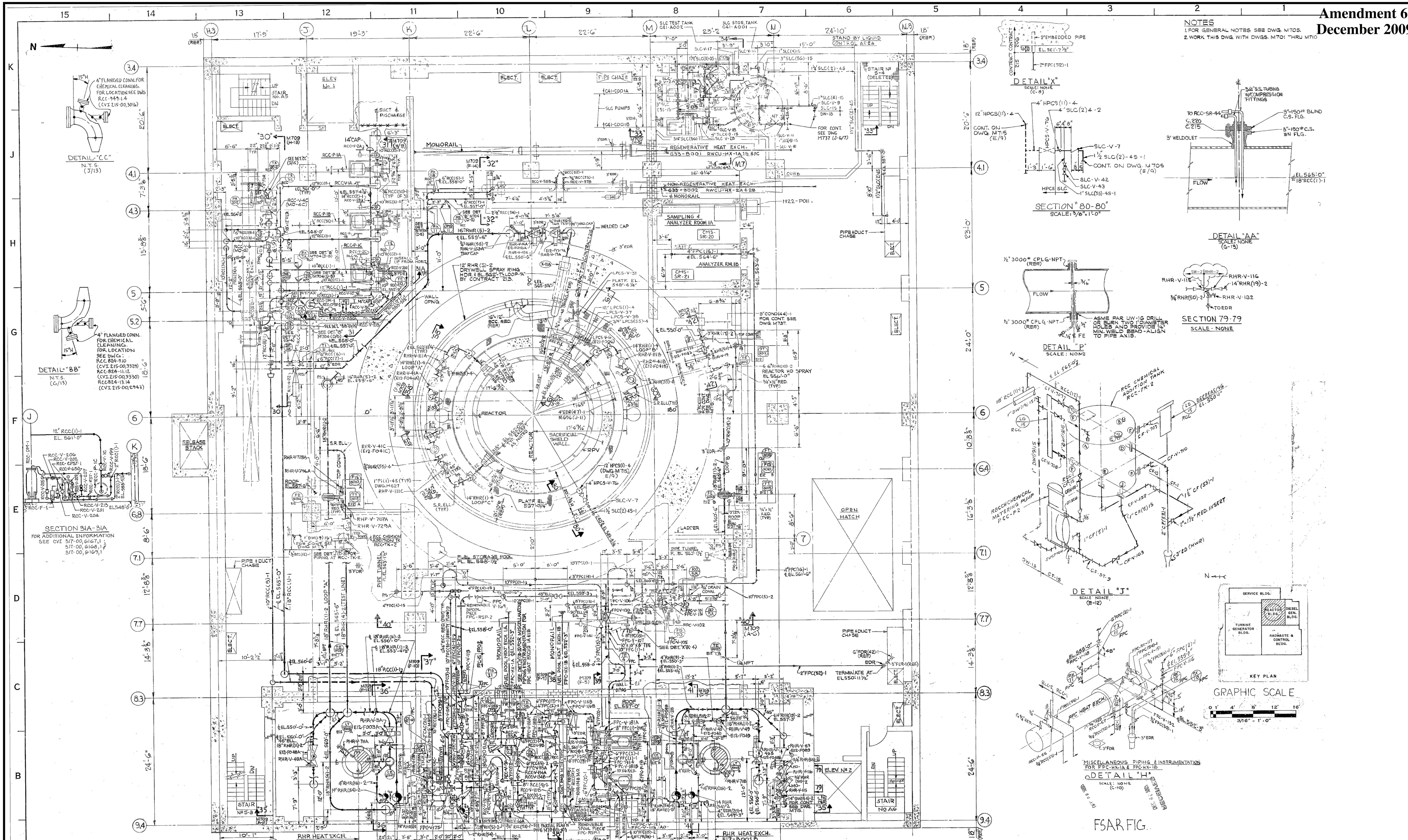
Draw. No. M704 | Rev. 35 | Figure 3.5-4



Columbia Generating Station
Final Safety Analysis Report

RHR, LPCS, RCC, FPC, SLC, and CPR Plan, Sections and Details - El. 522 ft 0 in. - Reactor Building

Draw. No. M705	Rev. 33	Figure 3.5-5



Columbia Generating Station
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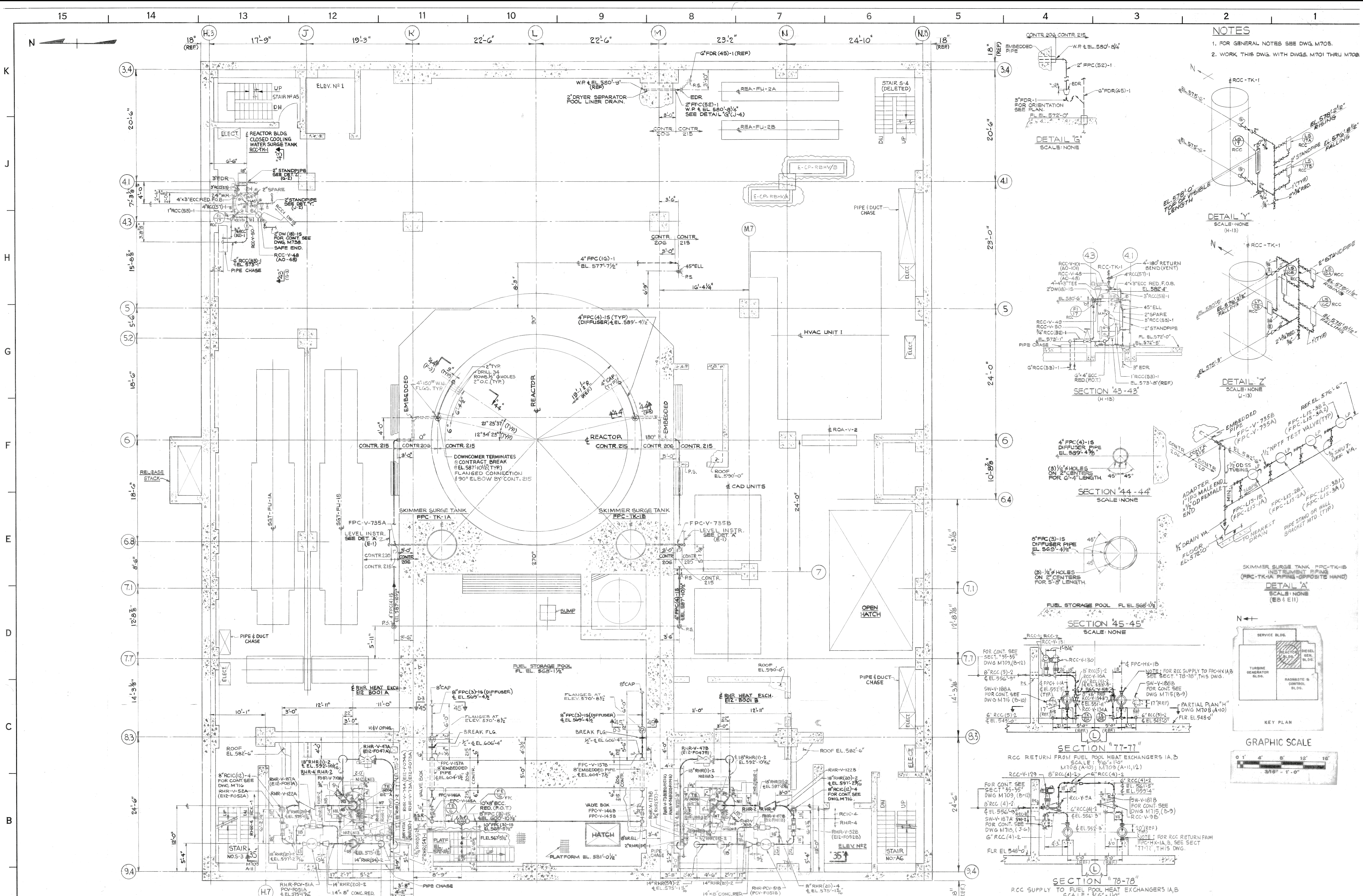
Sections and Details - El. 548 ft 0 in. - Reactor Building

Draw. No. M706 | Rev. 50 | Figure 03.5-06

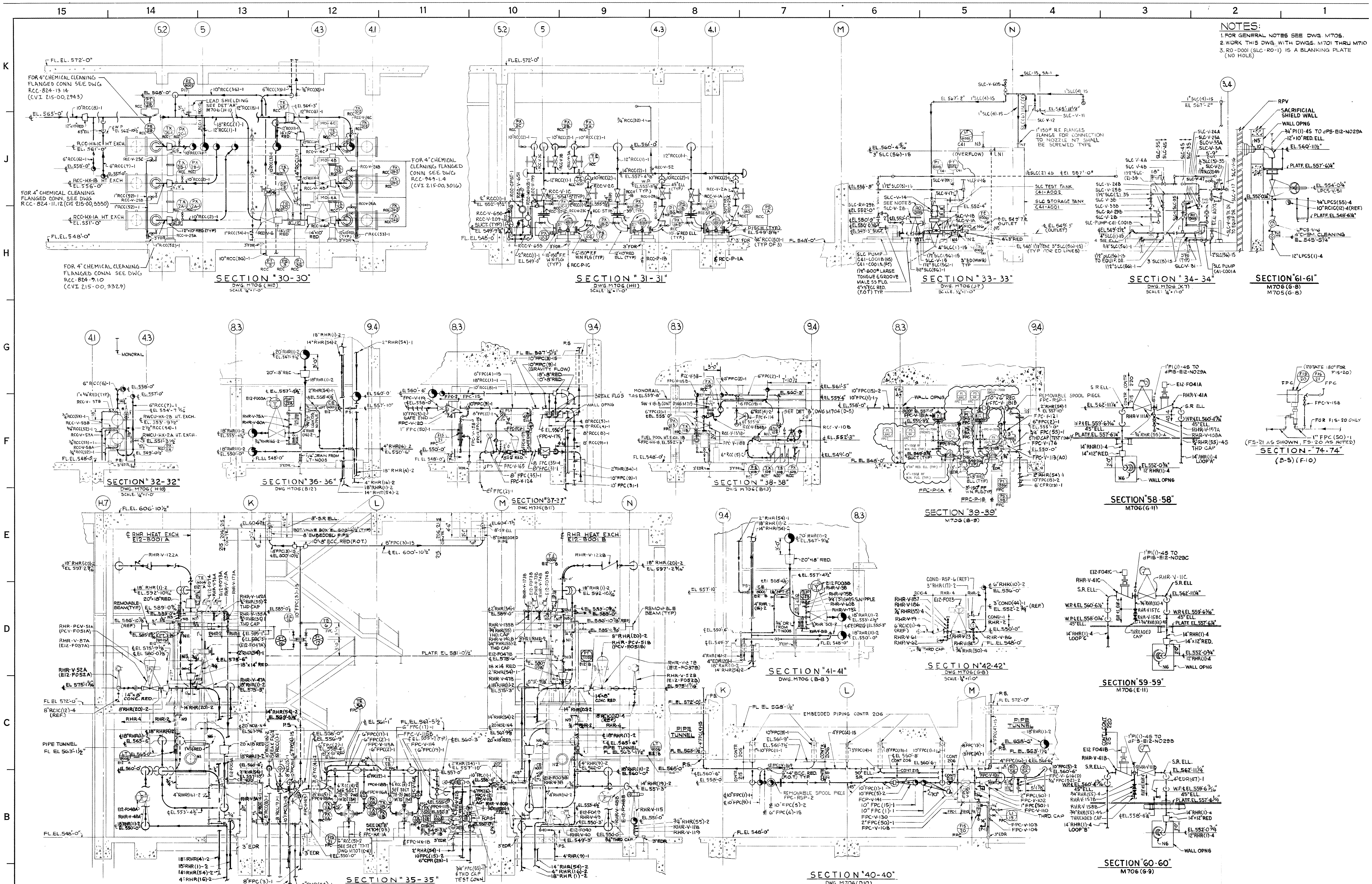


Columbia Generating Station
Final Safety Analysis Report

RHR, LPCS, RCC, FPC, SLC, and CPR Plan, Sections and Details - El. 572 ft 0 in. - Reactor Building

Draw. No. M707	Rev. 19	Figure 3.5-7

Amendment 61
December 2011



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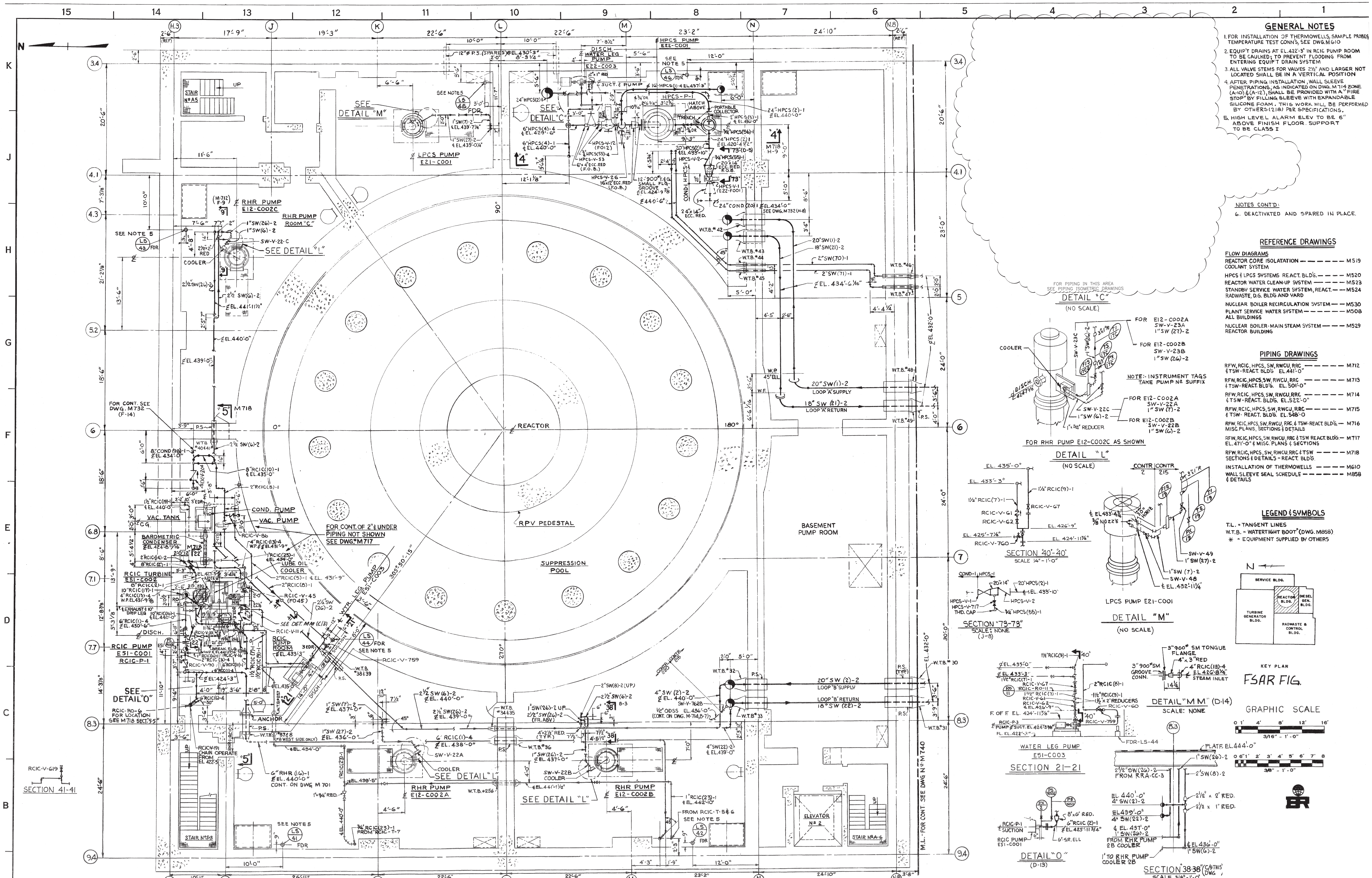
Final Safety Analysis Report

RHR, LPCS, RCC, FPC, SLC, and
CPR Plan, Sections and Details - Reactor Building

Draw. No. M709 | Rev. 44 | Figure 3.5-8

Form No. 960690ai

Amendment 61
December 2011



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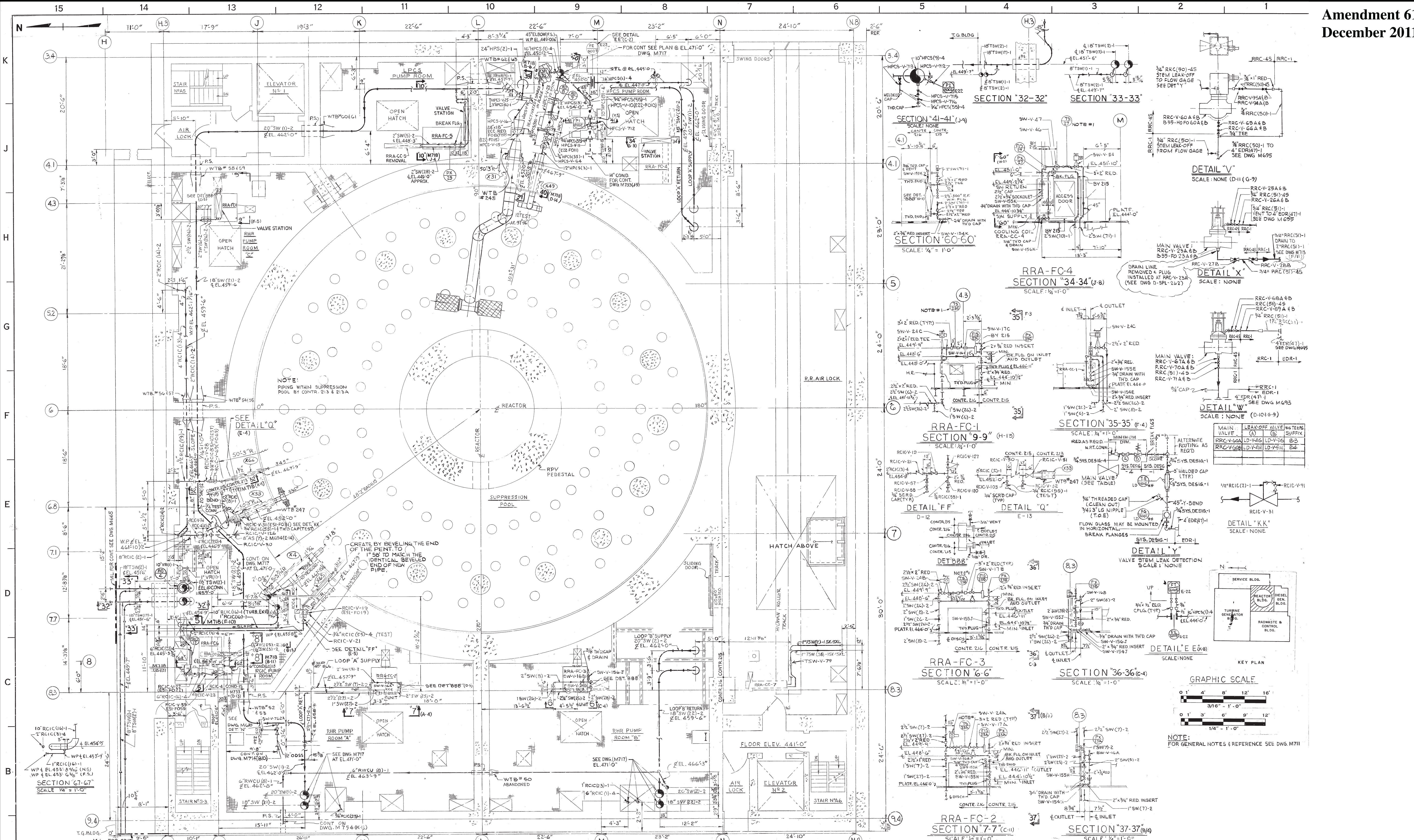
Final Safety Analysis Report

RFW, RCIC, HPCS, SW, RWCU, RRC, and TSW - El. 422 ft 3 in. - Reactor Building

Draw. No. M711	Rev. 33	Figure 3.5-9

Form No. 960690ai

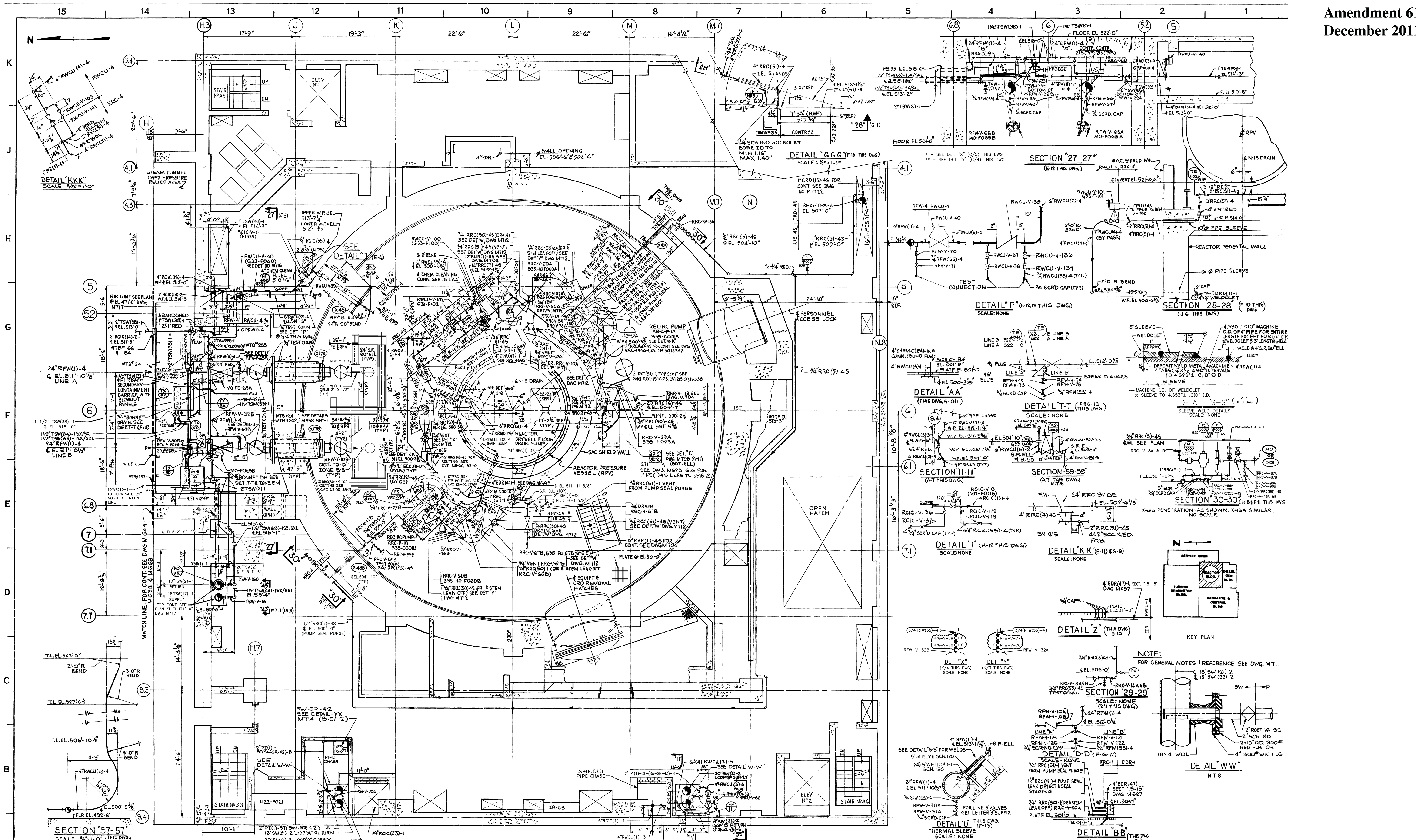


Columbia Generating Station
Final Safety Analysis Report

RFW, RCIC, HPCS, SW, RWCU, RRC, and TSW - El. 441 ft 0 in. - Reactor Building

Draw. No. M712	Rev. 46	Figure 3.5-10

Amendment 61
December 2011



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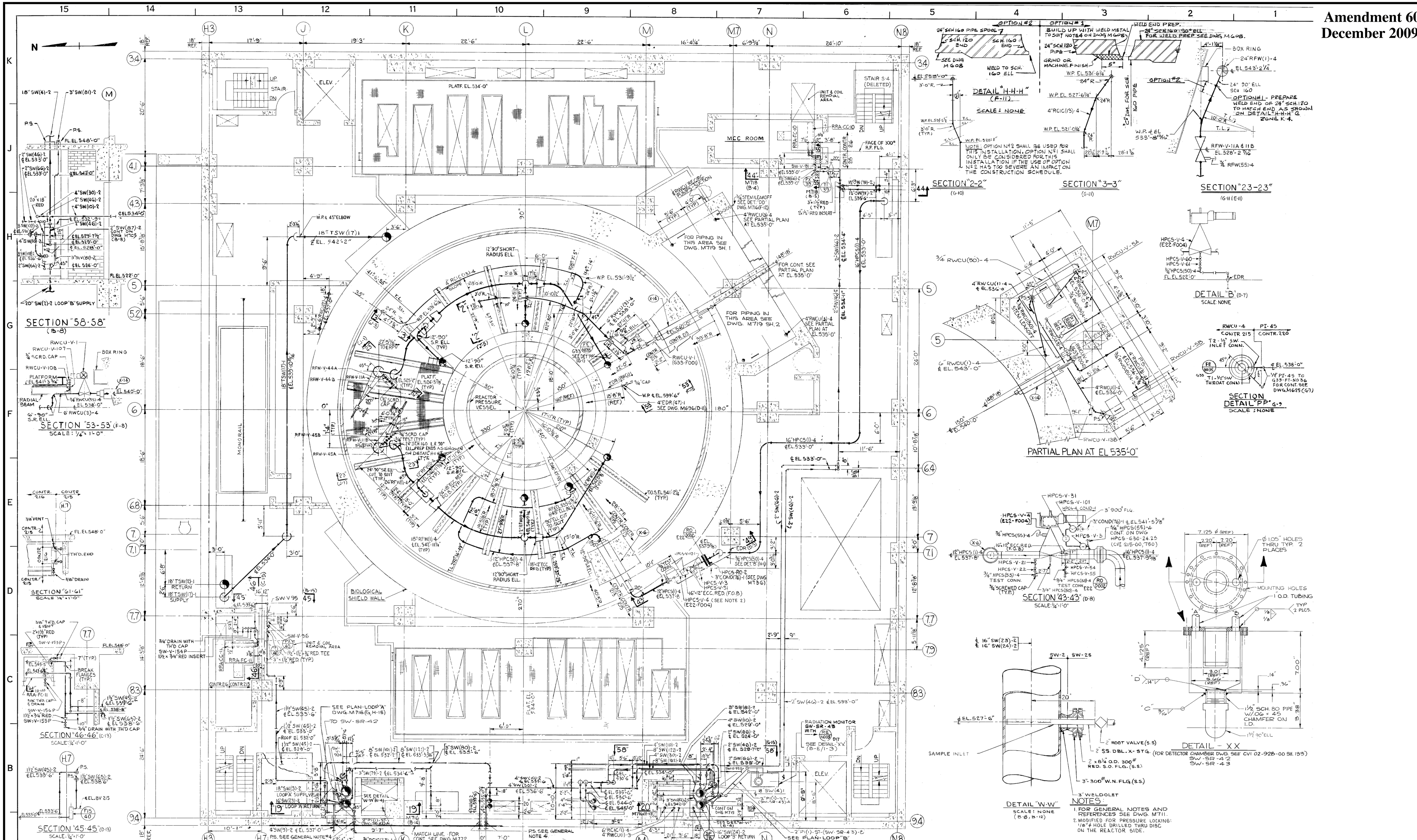
Final Safety Analysis Report

RFW, RCIC, HPCS, SW, RWCU, RRC, and TSW - El. 501 ft 0 in. - Reactor Building

Draw. No. M713 | Rev. 54 | Figure 3.5-11

Form No. 960690ai



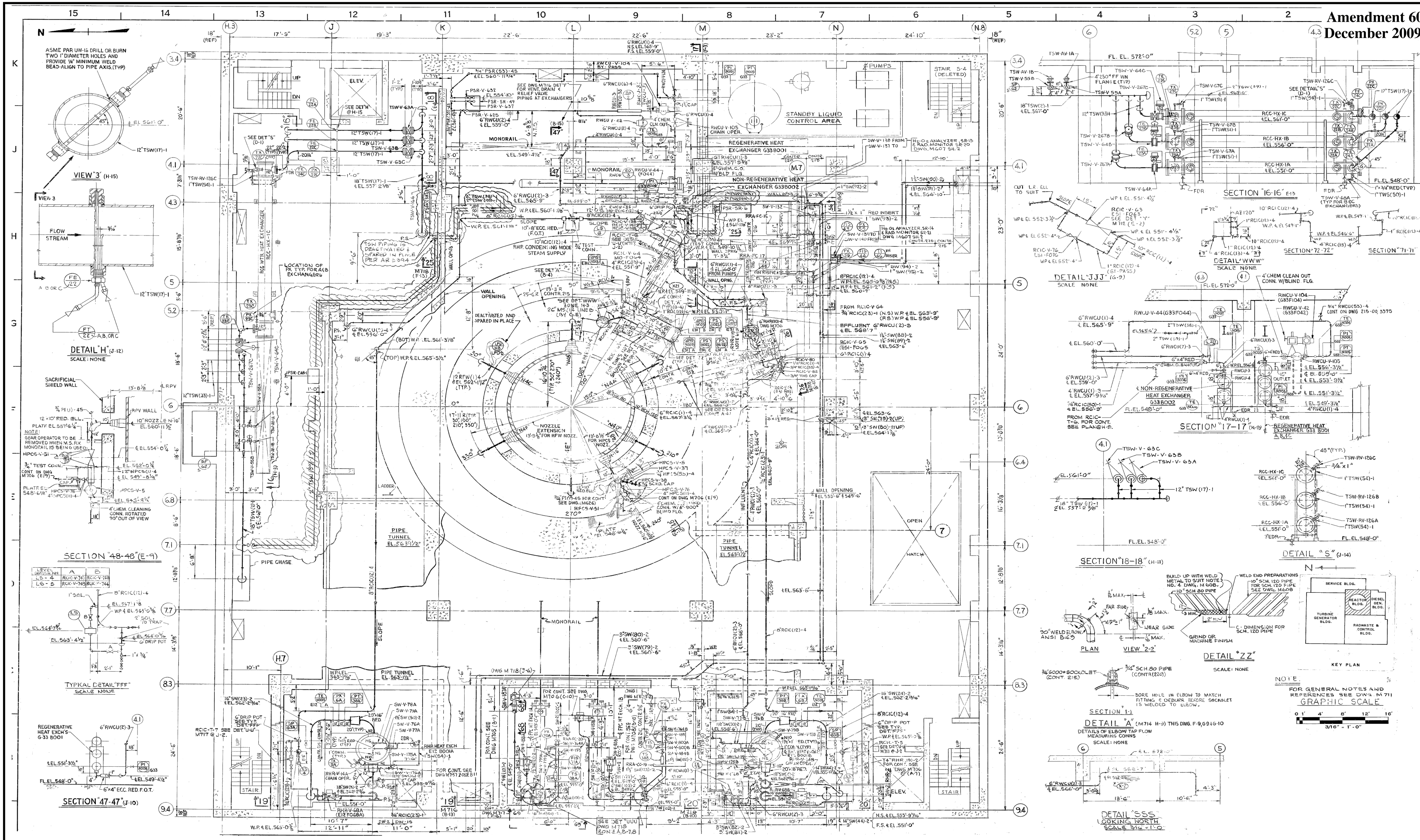
Columbia Generating Station
Final Safety Analysis Report

RFW, RCIC, HPCS, SW, RWCU,RRC, and TSW - El. 522 ft 0 in. - Reactor Building

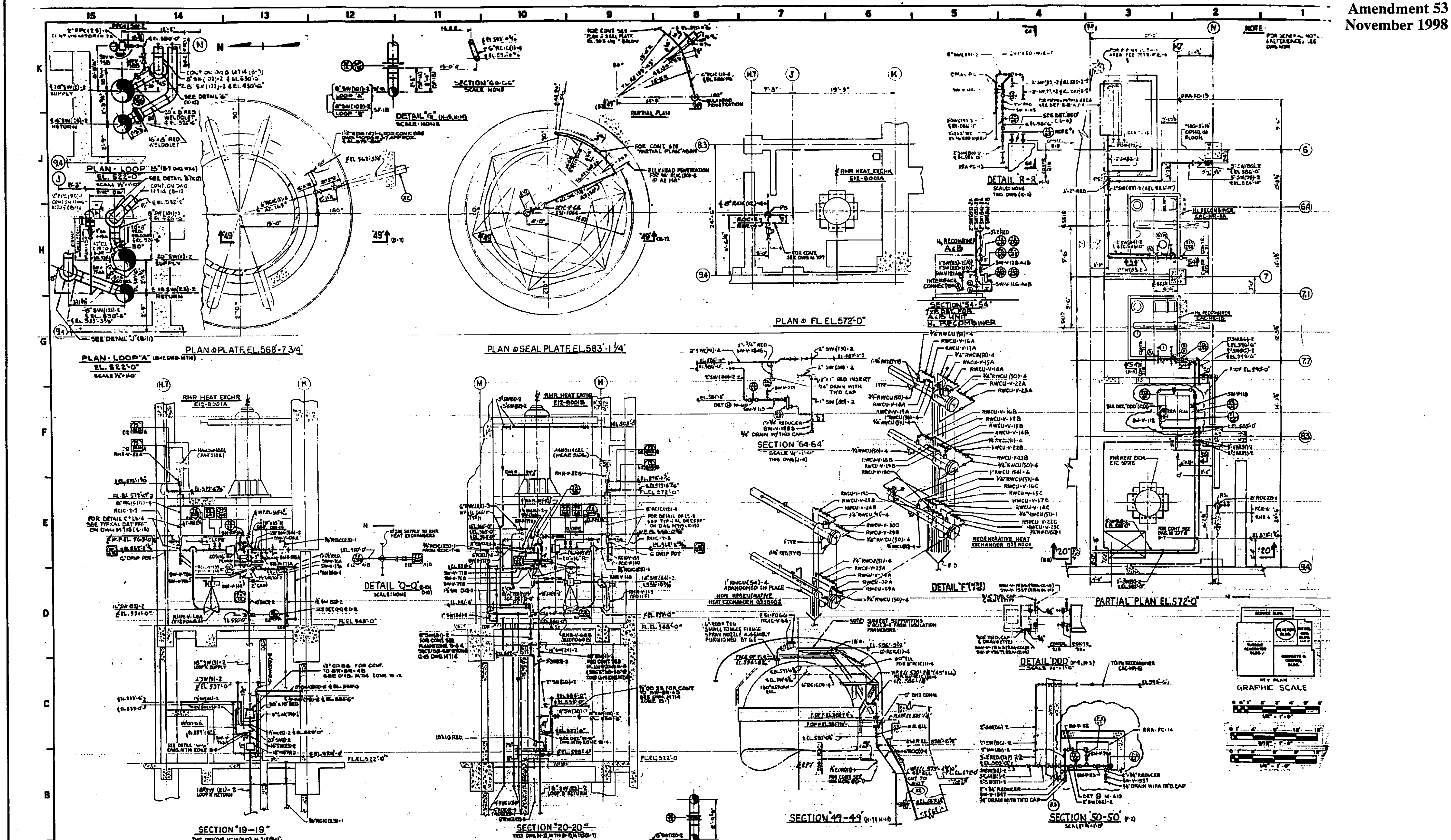
Draw. No. M714 | Rev. 40 | Figure 03.5-12

Form No. 960690ai

Amendment 60
December 2009



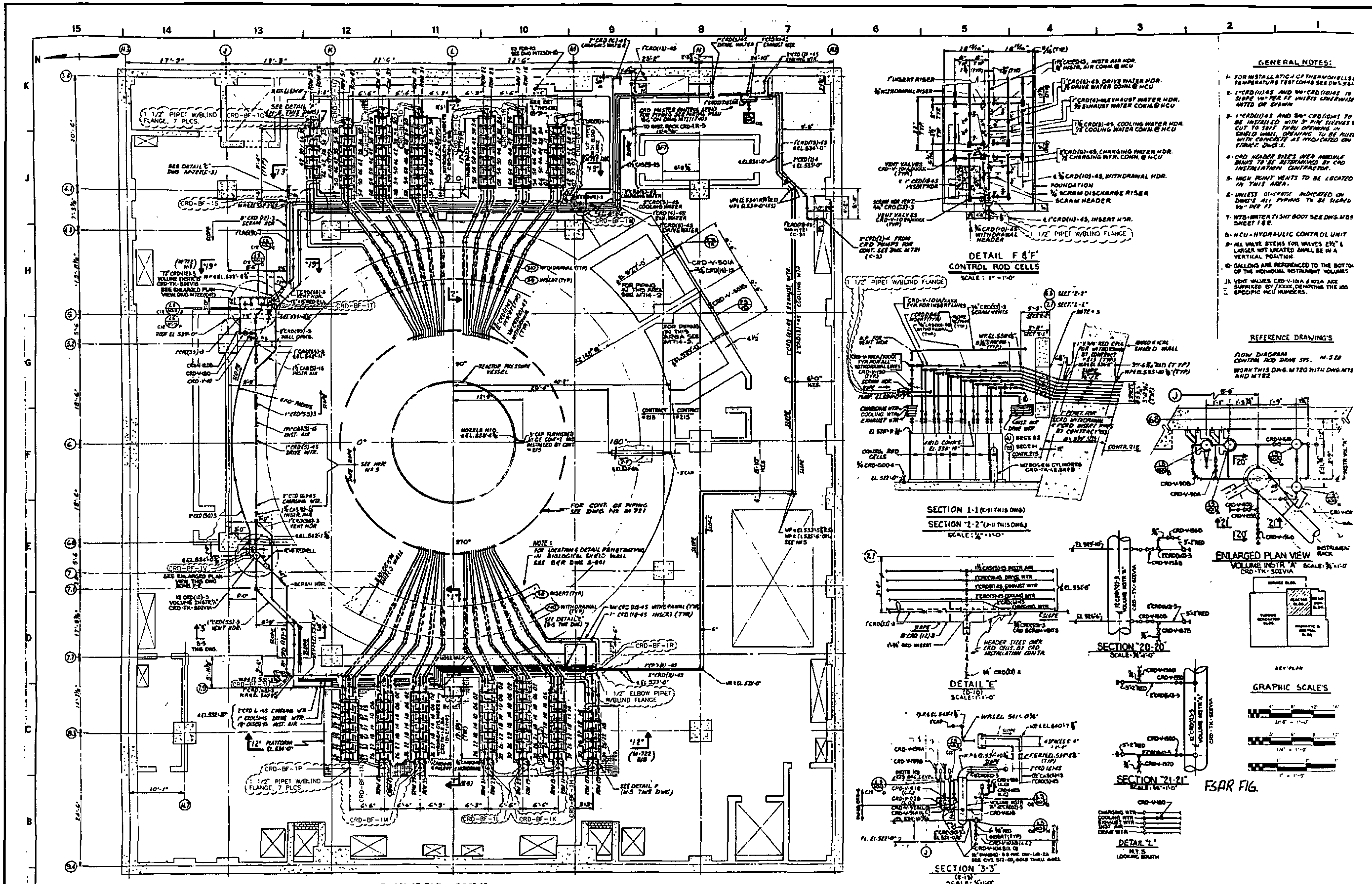
Columbia Generating Station Final Safety Analysis Report	RFW, RCIC, HPCS, SW, RWCU, RRC, and TSW - El. 548 ft 0 in. - Reactor Building
	Draw. No. M715 — Rev. 47 — Figure 03.5-13



Columbia Generating Station
Final Safety Analysis Report

RFW, RCIC, HPCS, SW, RWCU, RRC, and TSW - Miscellaneous Plan, Sections and Details - Reactor Building

Draw. No.	M716	Rev.	32	Figure	3.5-14

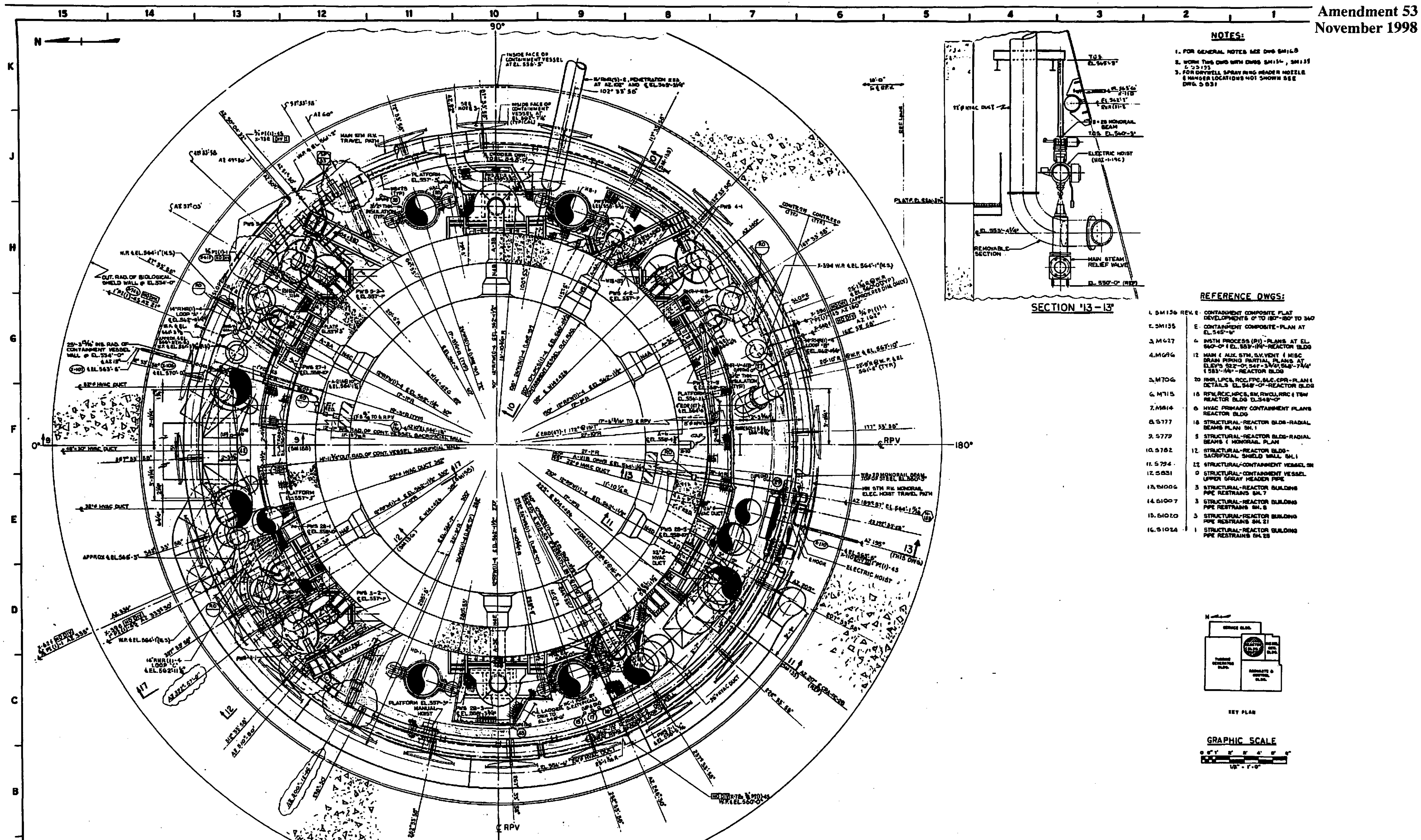


Columbia Generating Station
Final Safety Analysis Report

Control Rod Drive Piping Plan and Sections – El. 522 ft 0 in. – Reactor Building

Draw. No. M720 | Rev. 20 | Figure 3.5-15

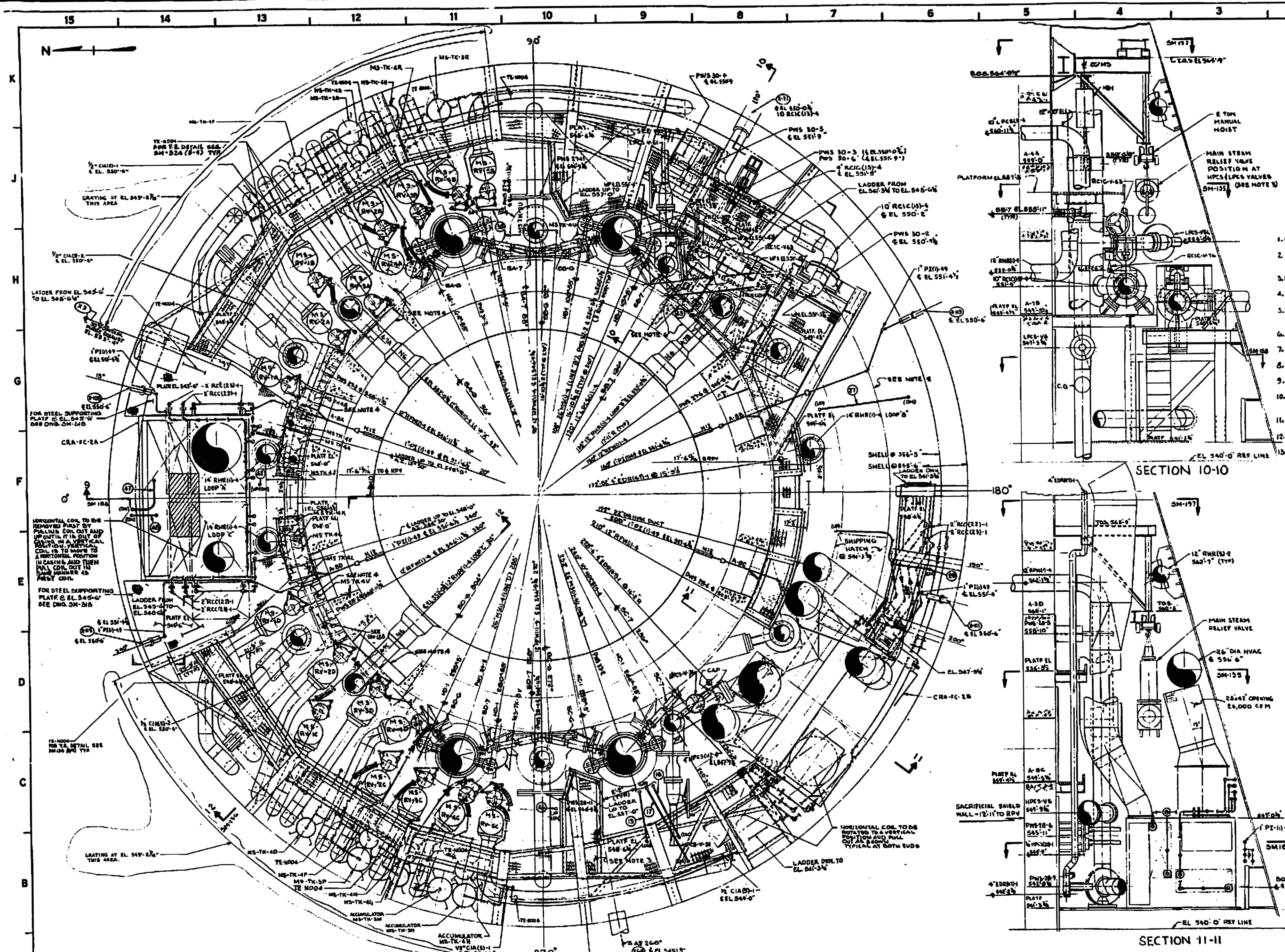
Form No. 960690



Columbia Generating Station
Final Safety Analysis Report

Containment Composite Plan and Sections - El. 556 ft 5 in.

Draw. No. SM197 | Rev. E | Figure 3.5-16

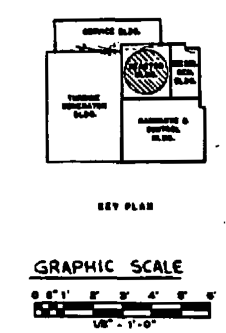


NOTES

1. FOR A COMPLETE LIST OF GENERAL NOTES SEE DWG SM-140
2. WORK THIS DWG WITH DWGS SM-136 SM-183 & SM-197
3. LAYDOWN AREA FOR RIGGING RELIEF VALVES TO CLEAR LPCS-V51 & HPCS-V56
4. TO OPEN DOORS A-7A, A-8A, A-7C (A-8B) THE RELIEF VALVES (A&1D, 2A&2D, 3A&3D, 4A&4D, AND PWS-21-2 & 28-2 WHIP RESTRAINT STRAPS, WILL HAVE TO BE REMOVED.
5. RELIEF VALVE REMOVAL PATHS FOR MANUAL & ELECTRIC HOISTS. (SEE DWG SM-320 FOR MONORAIL AND TURNTABLES)
6. DOOR HALF AT A-7B (B) WILL HAVE TO BE SHIPPED TO ALLOW REMOVAL
7. WORK THIS DRAWING WITH S1186-1, 2, 3.

REFERENCE DRAWINGS

1. M626	REV. 1	INSTM PROCESS (PI) - PLAN AT EL 541'-3⅝" - REACTOR BLDG
2. M696	17	MAIN & AUX STM., S.V. VENT & MISC DR PIPING - PARTIAL PLANS AT ELEV'S 522'-0", 541'-3¼", 564'-7⅝" & 588'-1¼" - R.B.
3. M706	20	RHR, LPCS, RCC, FPC, SLC, CPR - PLAN & DETAILS, EL 548'-0" - REACTOR BLDG
4. M715	18	RFW, RCIC, HPCS, SW, RWCU, RRC & TSW EL 548'-0" REACTOR BLDG.
5. M736	12C	COMP AIR, FIRE PROT., COND., DEMIN WTR, & MISC PIPING - PLANS & SECTIONS EL 522'-0" - REACTOR BLDG.
6. M814	6	HVAC - PRIMARY CONTAINMENT PLANS - REACTOR BLDG.
7. S777	18	STRUCTURAL REACTOR BUILDING RADIAL BEAMS - PLAN SH 1 (PLATF'S)
8. S779	5	STRUCTURAL REACTOR BUILDING RADIAL BEAMS & MONORAIL PLANS
9. S783	9	STRUCTURAL REACTOR BUILDING SACRIFICIAL SHIELD WALL SH. 2
10. S784	6	STRUCTURAL REACTOR BUILDING SACRIFICIAL SHIELD WALL SH 3 (SAC WALL DOORS)
11. S831	0	STRUCTURAL CONTAINMENT VESSEL UPPER SPRAY RING HEADER.
12. S1000	16	STRUCTURAL REACTOR BUILDING PIPE RESTRAINTS SCHEDULE
13. S795	35	STRUCTURAL REACTOR BUILDING CONTAINMENT VESSEL SH 2 (PENET)

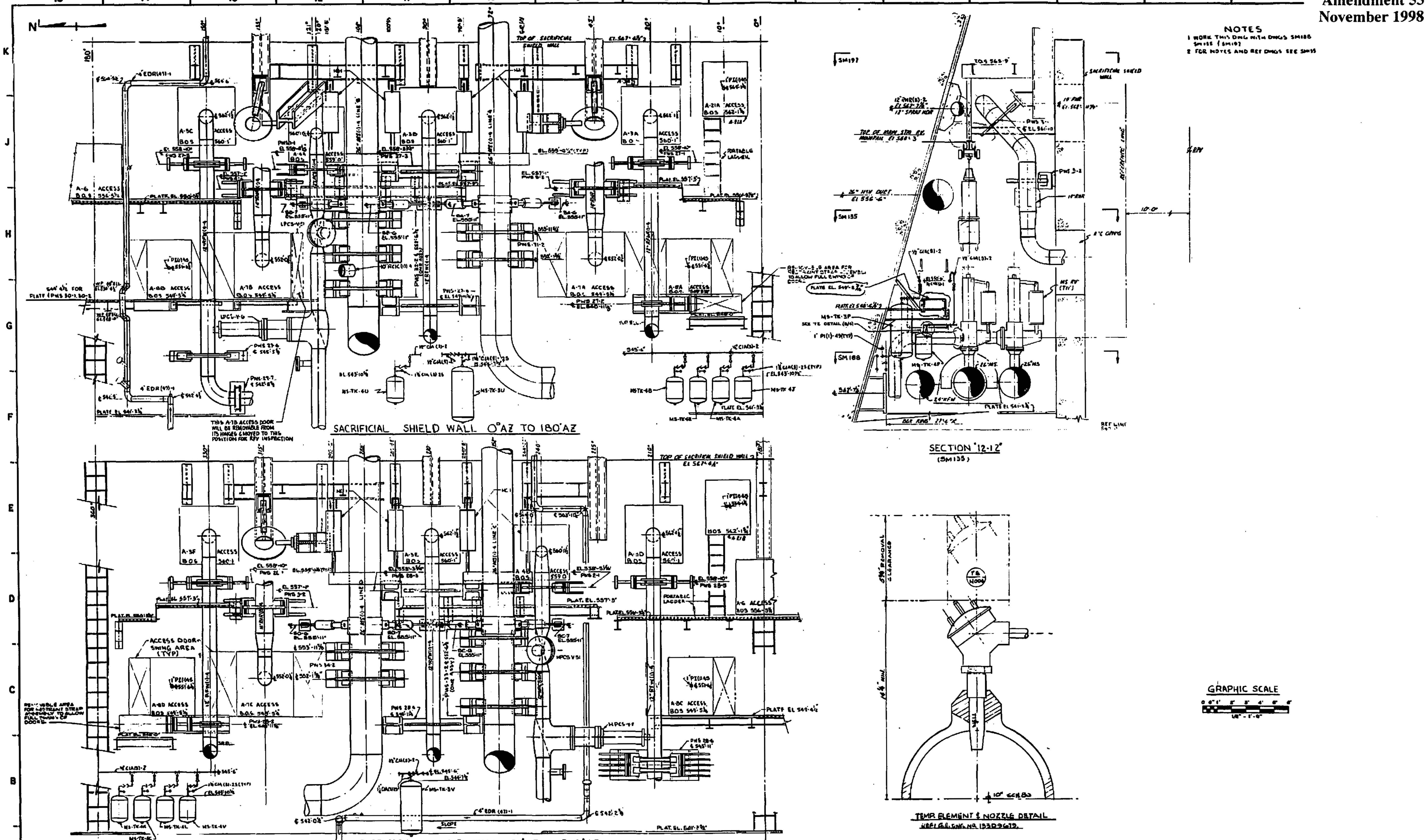


Columbia Generating Station
Final Safety Analysis Report

Containment Composite Plan - El. 545 ft 6 in.

Draw. No.	SM135	Rev.	G	Figure	3.5-17

Form No. 960690



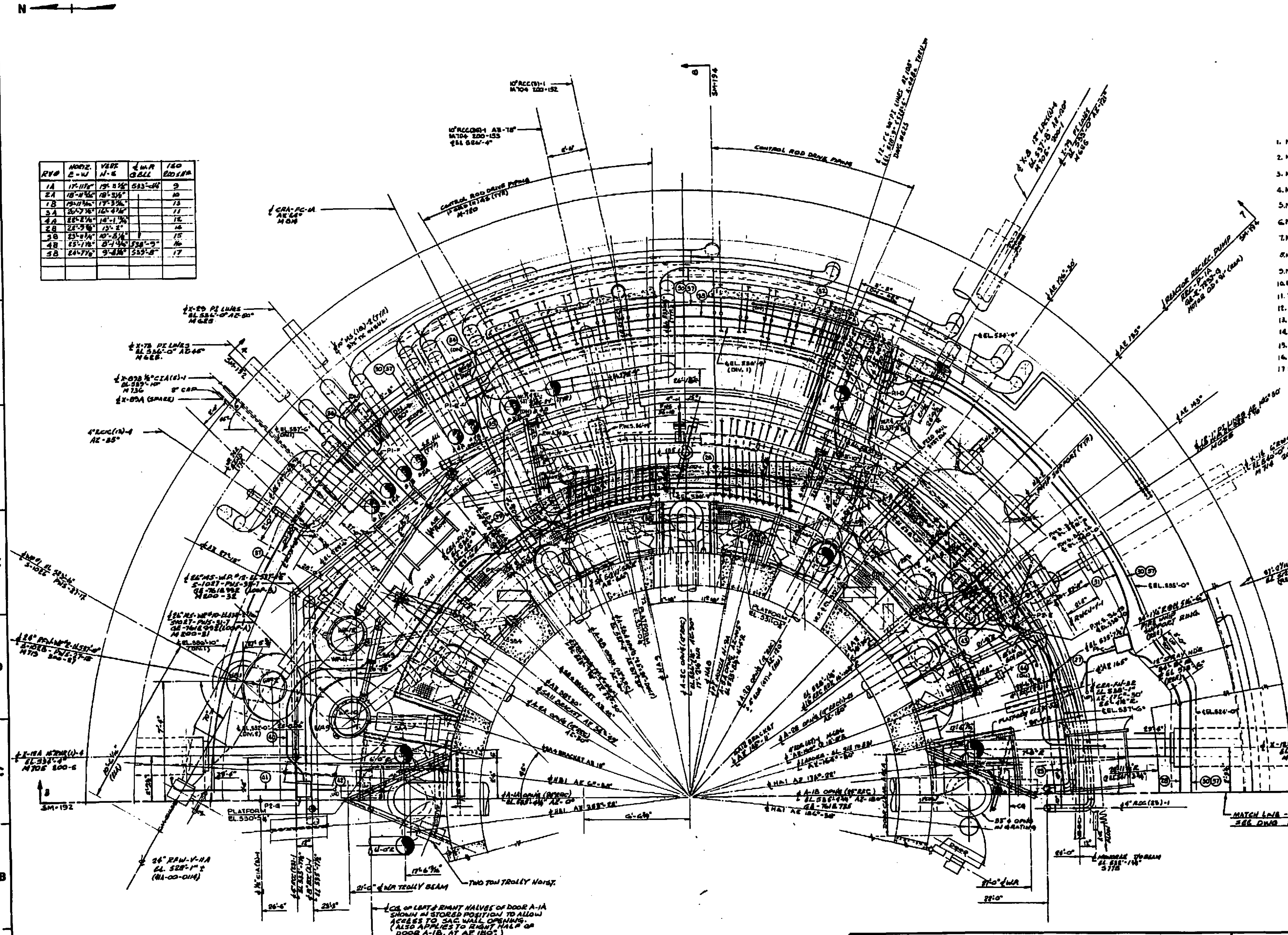
Columbia Generating Station
Final Safety Analysis Report

Containment Composite Plan and Section Developments - El. 541 ft 3 3/4 in. and El. 567 ft 4 1/2 in.

Draw. No.	SM136	Rev. F	Figure	3.5-18

Form No. 960690

Amendment 53
November 1998



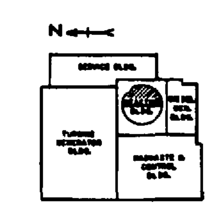
RV#	HORIZ. E-W	VERT. N-S	℄ W.R. BELL	ISO BOOSTER
1A	17'-11⅛"	19'-8⅛"	533'-0⅝"	9
2A	18'-11 [illegible]"	18'-5⅜"		10
1B	19'-11 [illegible]"	17'-3 [illegible]"		13
3A	20'-7 [illegible]"	16'-4 [illegible]"		11
4A	22'-2 [illegible]"	14'-1 [illegible]"		12
2B	22'-7 [illegible]"	13'-2"		14
3B	23'-11⅛"	10'-8⅛"		15
4B	25'-1⅛"	8'-4 [illegible]"	534'-9"	16
5B	24'-7⅛"	9'-8 [illegible]"	533'-8"	17

NOTES:

1. FOR CONTAINMENT COMPOSITE DWG LIST SEE SM168
2. FOR GENERAL NOTES & REFERENCE DWG'S SEE SM160
3. WORK THIS DWG WITH SM183 AND SM188, 189, 190, 184 & 185
4. FOR GENERAL ELECTRIC CO RPA DWGS SEE SM ITEM 15 THRU 20.
5. SNUBBER NO. SA-12 AND SUPPORT BRACKET MUST BE REMOVED BEFORE MONORAIL IS USED TO SERVICE PUMP RRC-P-1A.

REFERENCE DRAWINGS:

1. M696	REV. 12	MAIN & AUX. STM, SK VENT & MISC. DRAIN PIPING.
2. M704	11	RHR, LPCS, RCC, FPC, SLC & CPR PIPING.
3. M705	13A	(SAME AS M704).
4. M713	19	RFW, RCIC, HPCS, SW, RWCU, RRC & TSW PIPING.
5. M714	18	(SAME AS M713).
6. M720	7	CONTROL ROD DRIVE PIPING.
7. M736	11	COMP. AIR, FIRE PROT., COND., DEMIN. WATER & MISC. PIPING.
8. M624	3	INSTR. PROCESS (RI.) PIPING.
9. M625	3	(SAME AS M624).
10. H814	8	HVAC - PRIM. CONTAINMENT.
11. S782	12	SACRIFICIAL SHIELD WALL. SH-1
12. S789	6	REACTOR PEDESTAL SH-1
13. S794	12	CONTAINMENT VESSEL SH-1
14. S778	22	RADIAL BEAMS-PLAN SH-8
15. S1000	16	PIPE RESTRAINTS SH-1
16. S823	21	RADIAL BEAMS SH-7.
17. SM219	2	CONTAINMENT-STUDY OF RECIRC SYS FLOW CONTROL VALVE REMOVAL @ AZ 135°



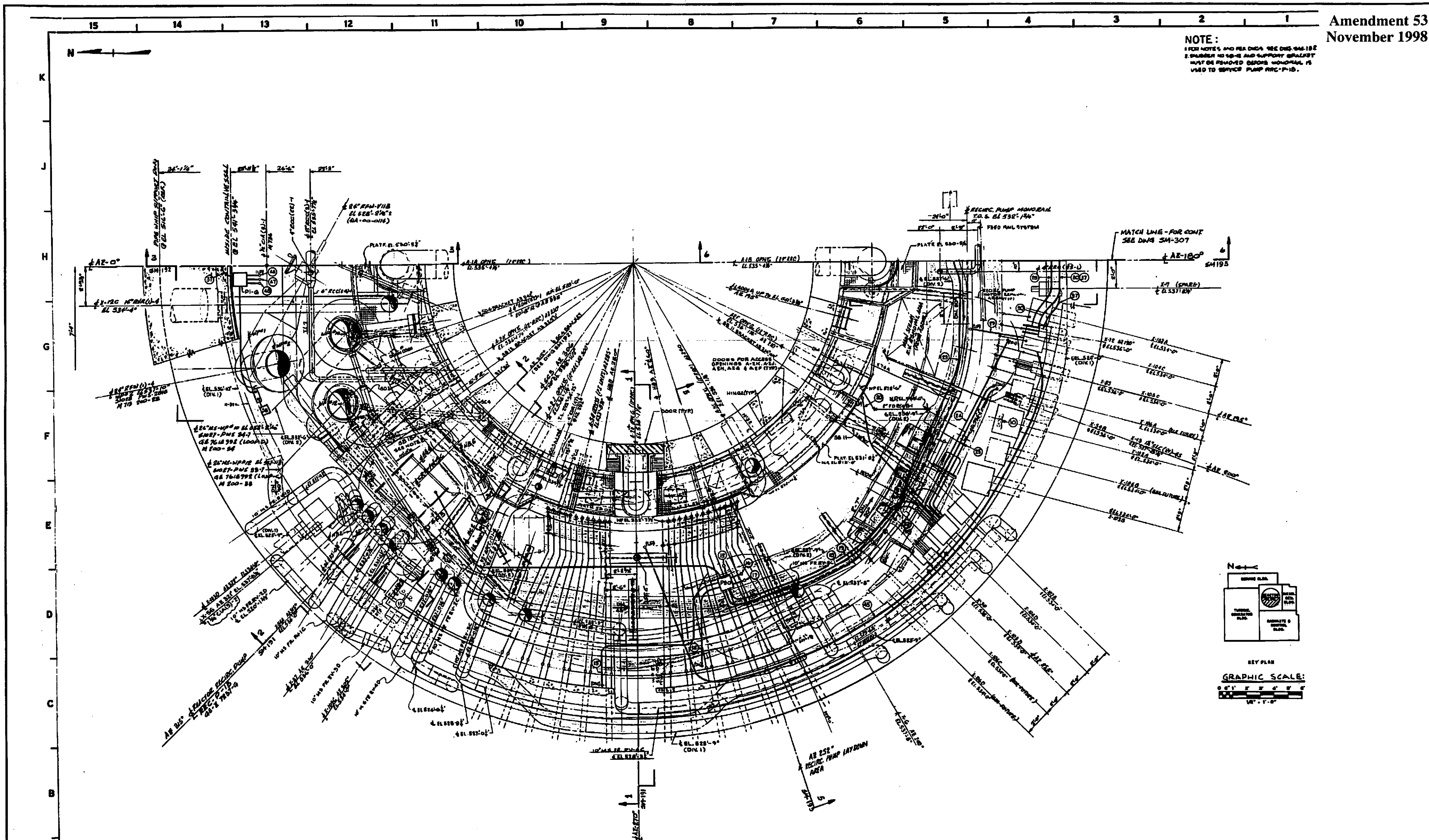
KEY PLAN

GRAPHIC SCALE:

Columbia Generating Station
Final Safety Analysis Report

Containment Composite Plan - El. 524 ft 3 7/8 in. - Azimuth 0° to 180°

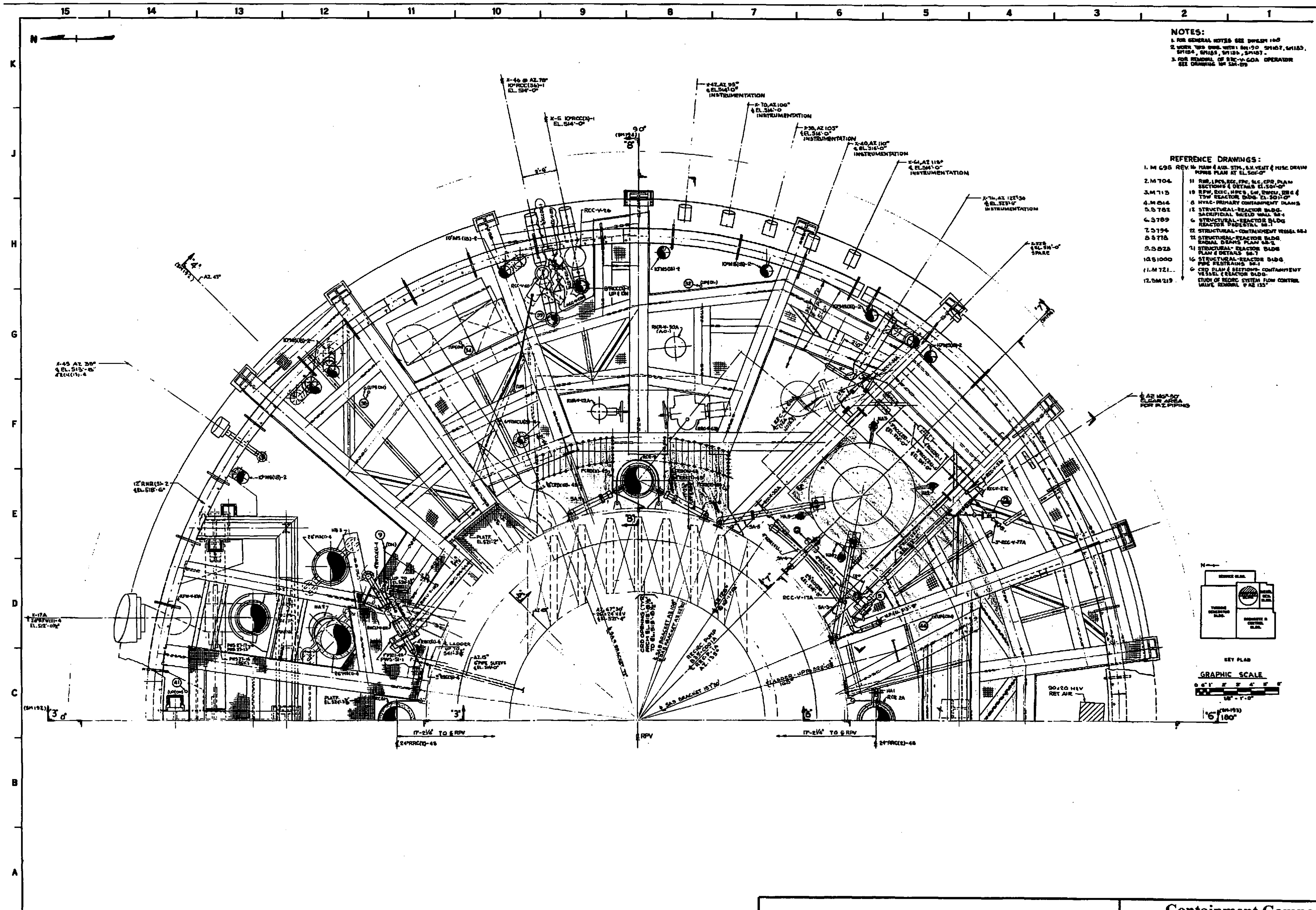
Draw. No.	SM182	Rev.	G	Figure	3.5-19



Columbia Generating Station
Final Safety Analysis Report

Containment Composite Plan - El. 524 ft 3 7/8 in. - Azimuth 180° to 360°

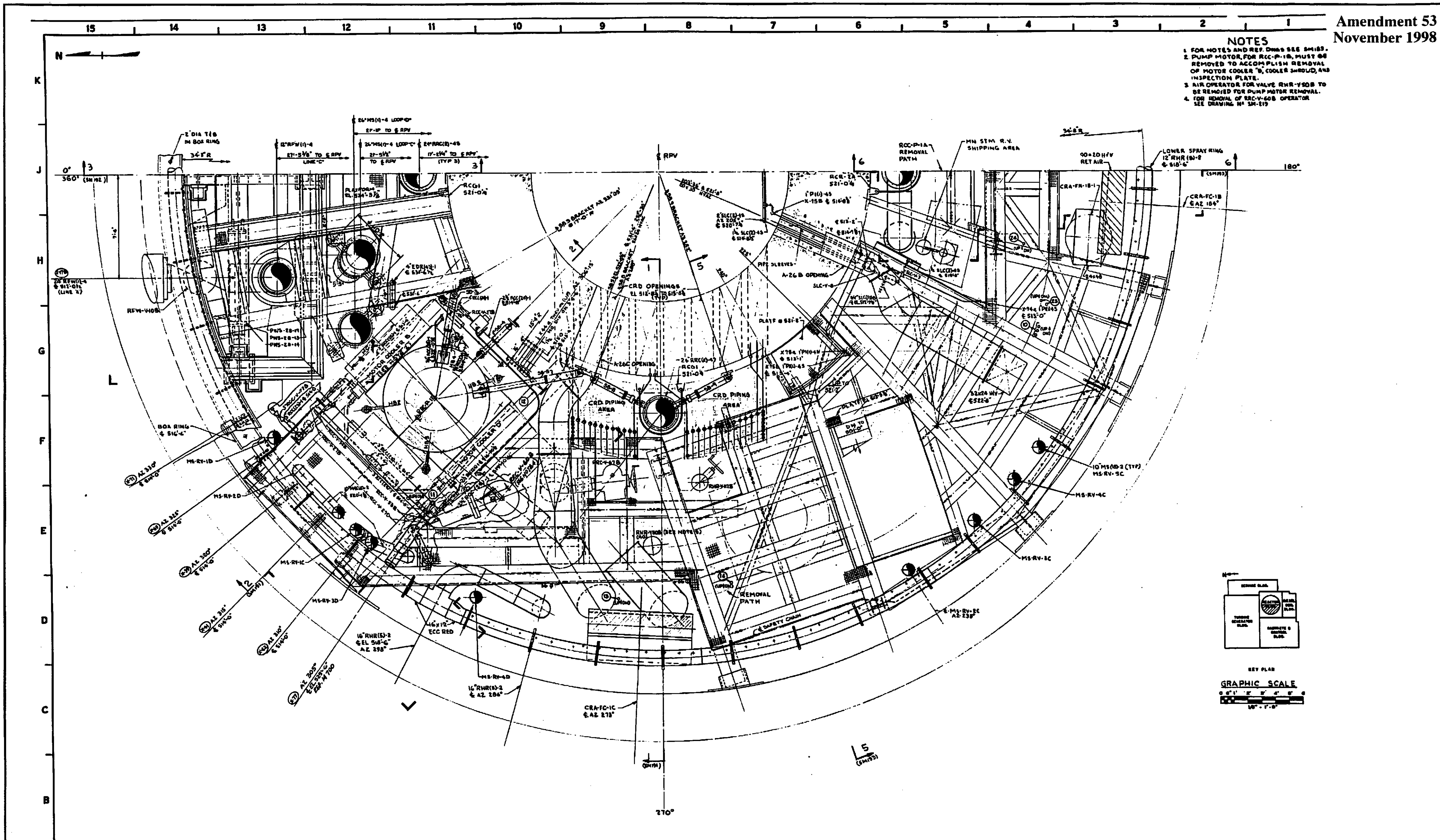
Draw. No.	SM183	Rev.	F	Figure	3.5-20



Columbia Generating Station
Final Safety Analysis Report

Containment Composite Plan - El. 512 ft 9 1/2 in. - Azimuth 0° to 180°

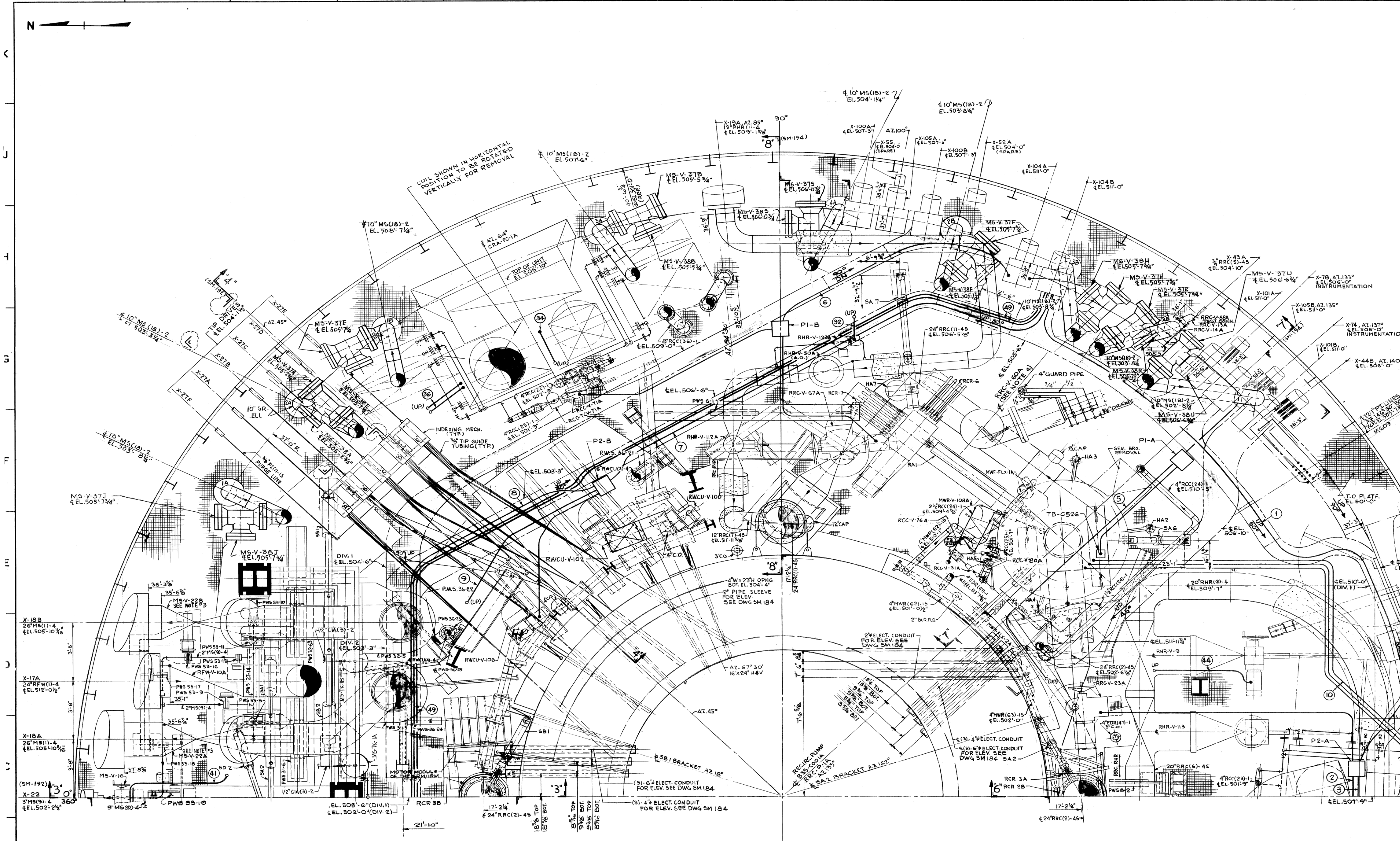
Draw. No.	SM189	Rev.	J	Figure	3.5-21



Columbia Generating Station
Final Safety Analysis Report

Containment Composite Plan - El. 512 ft 9 1/2 in. - Azimuth 180° to 360°

Draw. No.	SM190	Rev.	F	Figure	3.5-22

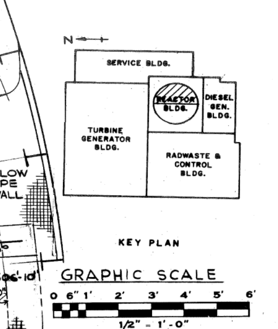


NOTES:

1. FOR GENERAL NOTES SEE DWG. SM168
2. WORK THIS DWG. WITH DWGS: SM187, SM168 SM189 & SM190
3. FOR REMOVAL OF MS-V-22A,B,C&D OPERATORS SEE DWG'S SM-178 & 179
4. FOR REMOVAL OF RRC-V-60A SEE DWG SM-219.

REFERENCE DRAWINGS:

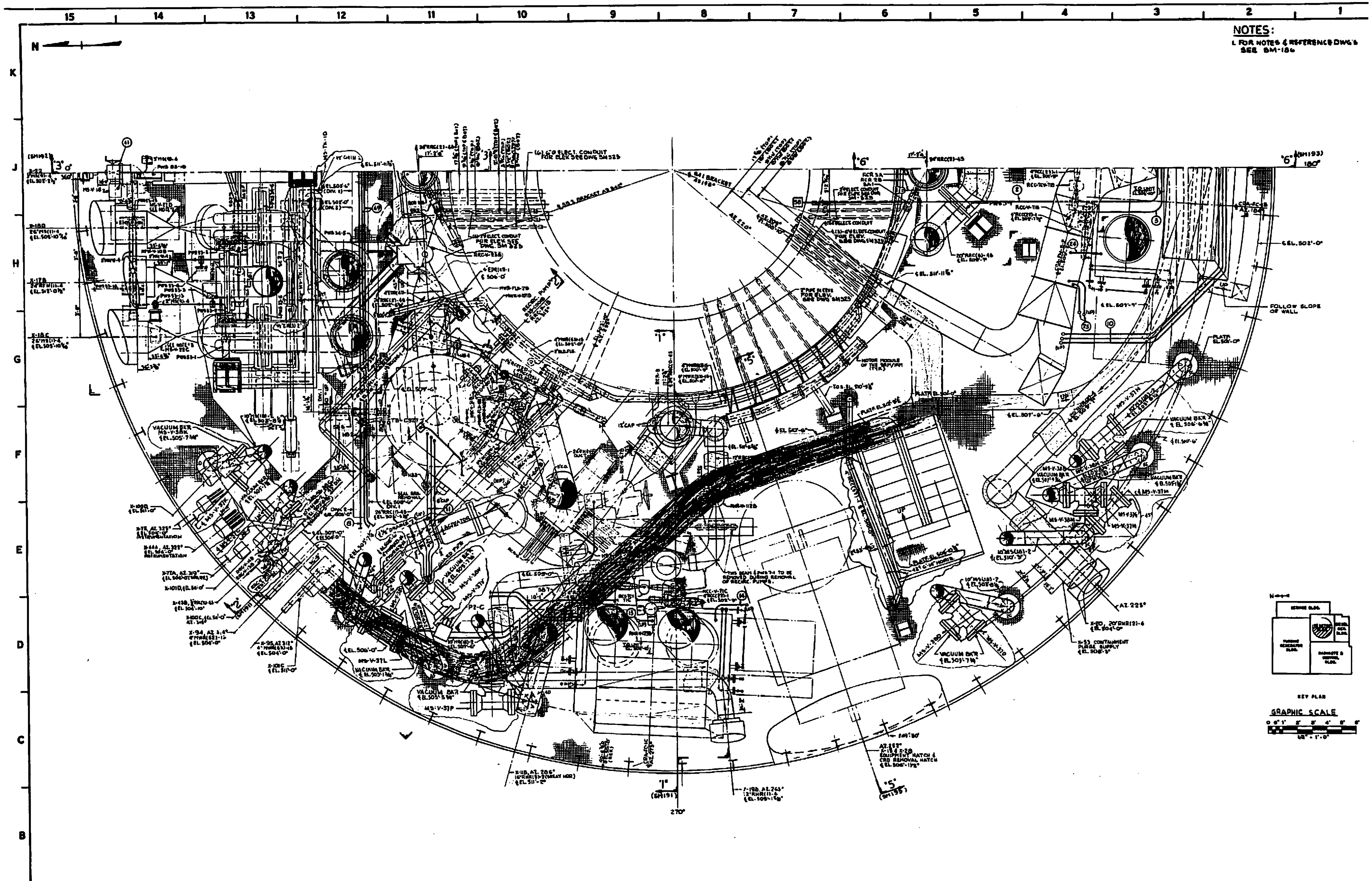
1. M695	REV. 16	MAIN & AUX. STM., S.V. VENT & MISC. DRAIN PIPING PLAN AT EL. 501'-0".
2. M704	11	RHR, LPCS, RCC, FPC, SLC, CPR PLAN SECTIONS & DETAILS EL. 501'-0".
3. M710	2	RHR, LPCS, RCC, FPC, SLC, CPR PARTIAL PLANS & SECTIONS.
4. M713	19	RFW, RCIC, HPCS, SW, RWCU, RRC & TSW REACTOR BLDG. EL. 501'-0".
5. M735	14	COMPR. AIR, FIRE PROTECTION, COND, DEMIN. WATER & MISC. PIPING - PLAN, SECTIONS & DETAILS AT EL. 501'-0".
6. M814	8	HVAC - PRIMARY CONTAINMENT - PLANS.
7. S773	8	STRUCTURAL - REACTOR BLDG. - EL. 499'-6" - MISC. DETAILS. SH. 1.
8. S782	12	STRUCTURAL - REACTOR BLDG. - SACRIFICIAL SHIELD WALL SH. 1.
9. S789	6	STRUCTURAL - REACTOR BLDG. - REACTOR PEDESTAL SH. 1.
10. S794	22	STRUCTURAL - CONTAINMENT VESSEL. SH. 1.
11. S828	9	STRUCTURAL - REACTOR BLDG. - EL. 499'-6" MISC. DETAILS. SH-2.
12. S829	8	STRUCTURAL - REACTOR BLDG. EL. 499'-6" MISC. DETAILS. SH. 3.
13. S839	8	STRUCTURAL - REACTOR BLDG. - RADIAL BEAMS PLAN & DETAILS SH. 8.
14. S1000	16	STRUCTURAL - REACTOR BLDG. PIPE RESTRAINTS SH. 1.
15. GE-Z-7237		REACTOR RECIRC. PUMP.
16. GE-761E735		RECIRCULATION LOOP PIPING.
17. GE-761E992		MAIN STEAM PIPING.
18. GE-762E823		RECIRC. LOOP PIPE WHIP RESTRAINT.
19. GE-762E538		RECIRC. LOOP SUSPENSION.
20. GE-762E493		STEAM PIPE SUSPENSION.
21. M603	REV. 1	UNDER VESSEL NEUTRON MONITORING SYSTEM - PLAN & SECTION
22. M604	3	UNDER VESSEL NEUTRON MONITORING SYSTEM - PLAN, SECTION & DETAILS
23. M609	3	INSTRUMENTATION PROCESS (PI) PLAN AT FLOOR EL. 501'-0 (0° TO 180° AZ)
24. M623	1	INSTRUMENTATION PROCESS (PI) PLAN AT FLOOR EL. 501'-0 (180° TO 360° AZ)
25. SM178		MAIN STEAM VALVE REMOVAL STUDY - PLANS CONTAINMENT.
26. SM179		MAIN STEAM VALVE REMOVAL STUDY - SECTIONS CONTAINMENT.
27. SM 219		CONTAINMENT - STUDY OF RECIRC. SYS. FLOW CONTROL VALVE REMOVAL @ AZ 135°.



Columbia Generating Station
Final Safety Analysis Report

Containment Composite Plan - El. 501 ft 0 in. - Azimuth 0 degrees to 180 degrees

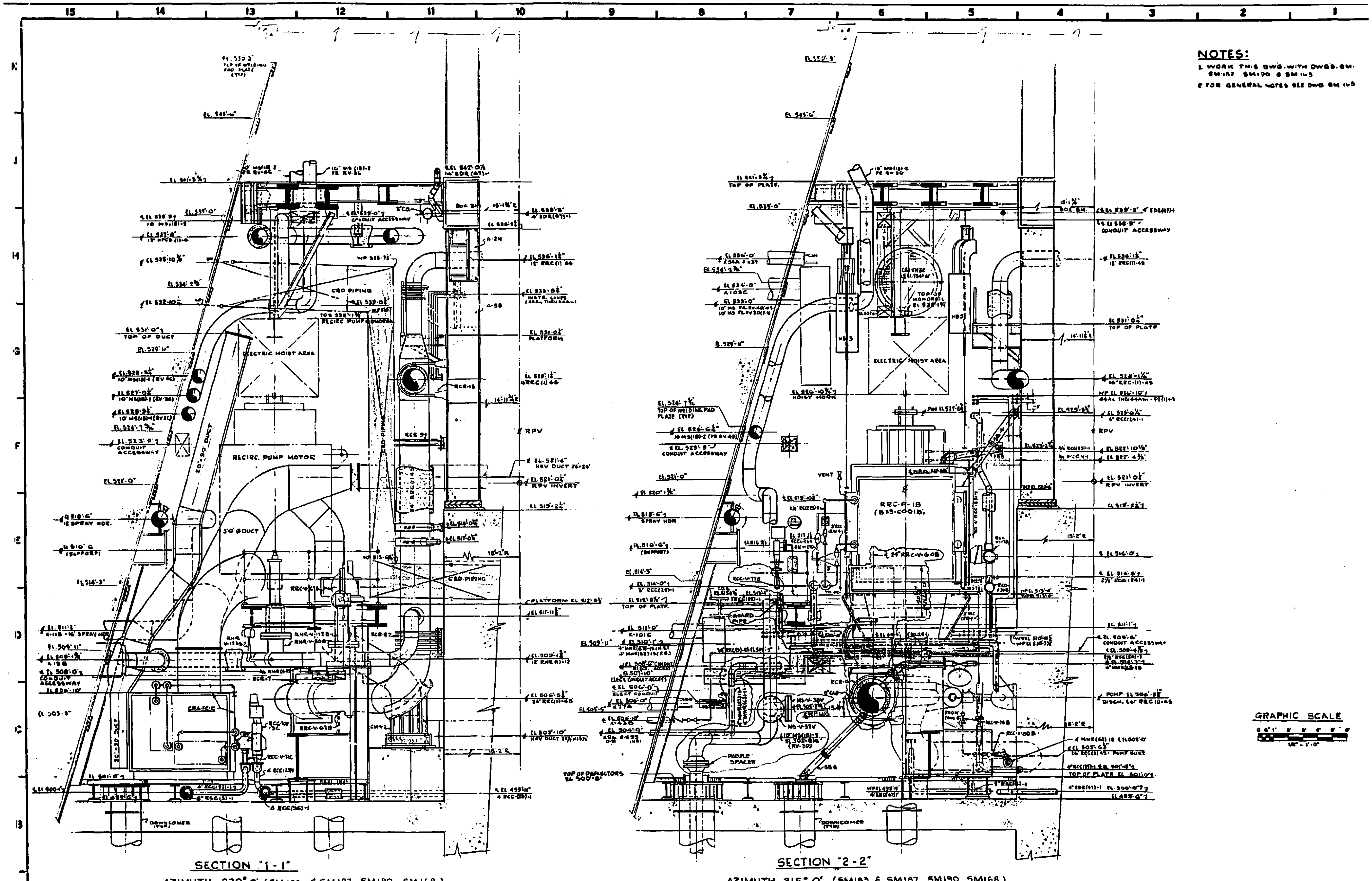
Draw. No. SM186	Rev. 0	Figure 03.5-23



Columbia Generating Station
Final Safety Analysis Report

Containment Composite Plan - El. 501 ft 0 in. - Azimuth 180° to 360°

Draw. No.	SM187	Rev.	F	Figure	3.5-24

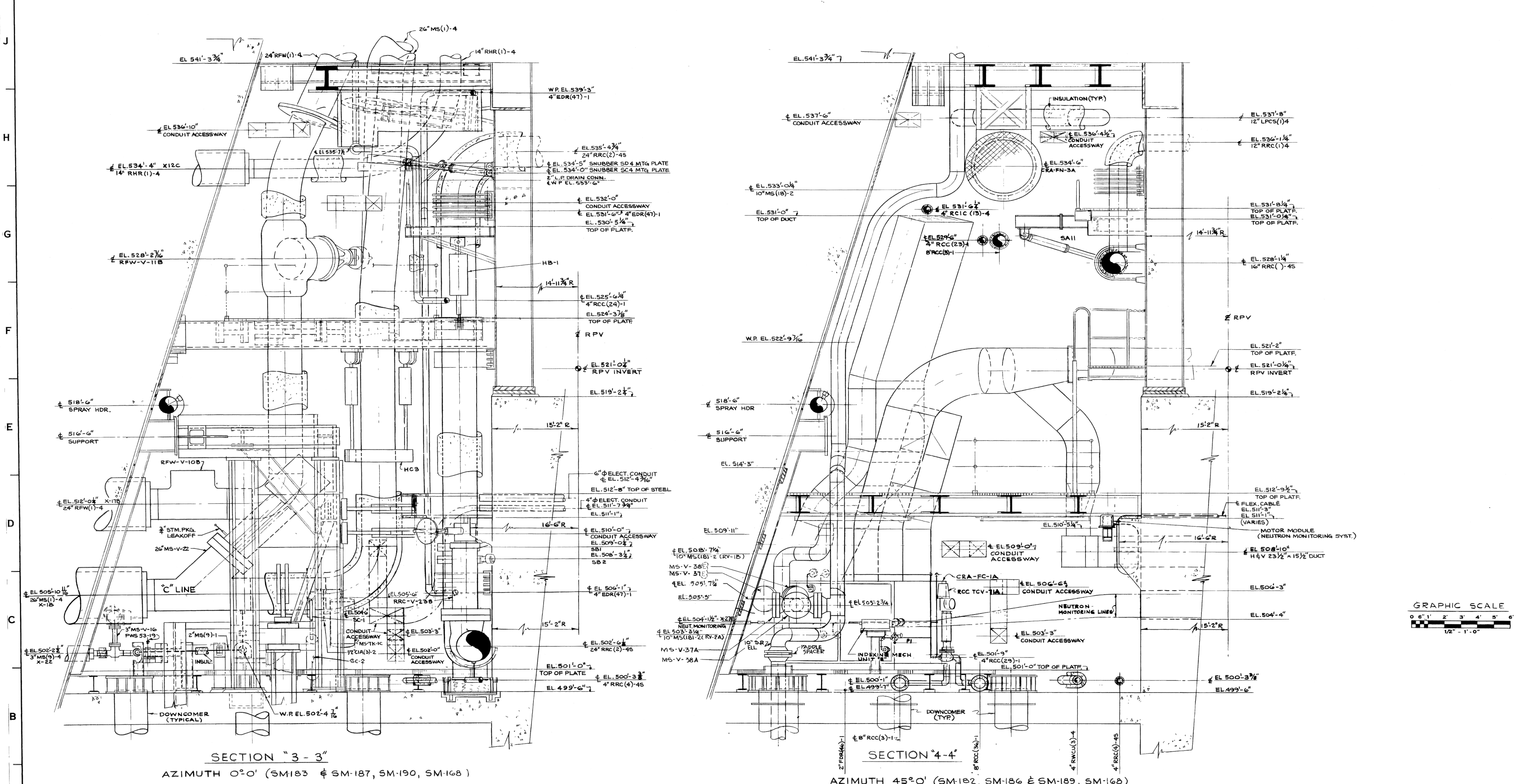


Columbia Generating Station
Final Safety Analysis Report

Containment Composite Sections "1-1" and "2-2"

Draw. No.	SM191	Rev.	F	Figure	3.5-25

NOTES:
1. WORK THIS DWG WITH DWGS SM-182 SM-183, SM-186, SM-187, SM-183, SM-190 & SM-168
2 FOR GENERAL NOTES SEE DWG SM315.



Columbia Generating Station
Final Safety Analysis Report

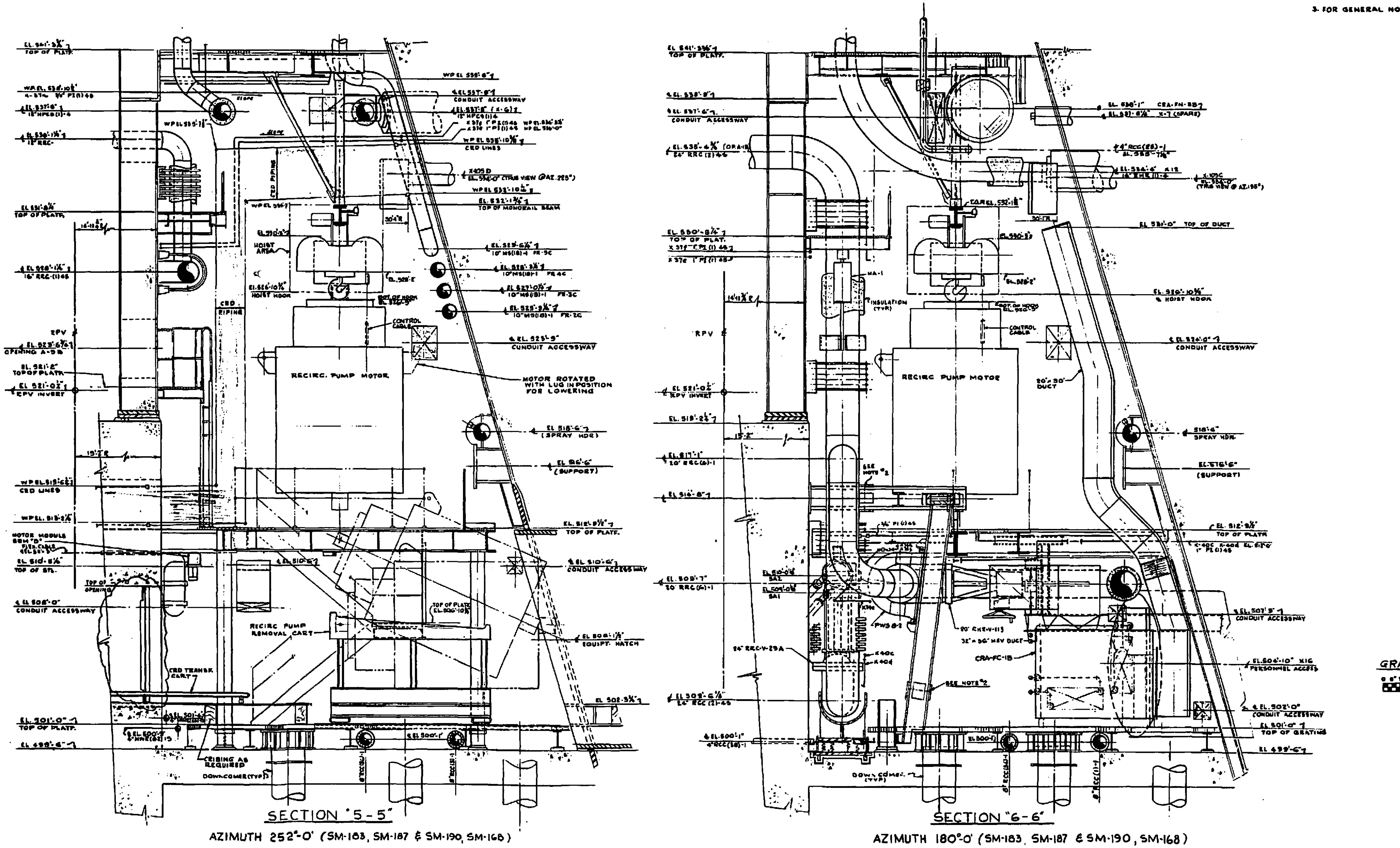
Containment Composite Sections "3-3" and "4-4"

Draw. No. SM192 | Rev. 0 | Figure 03.5-26

Form No. 960690ai

NOTES:

1. WORK THIS DWG. WITH DWGS SM-183, SM-187, SM-190 & SM-168
2. STEEL INSTALLED WITH SPLICES TO FACILITATE DISMANTLING OF STEEL DURING REMOVAL OF RECIRC. PUMP MOTOR.
3. FOR GENERAL NOTES SEE DWG. SM 315.



GRAPHIC SCALE

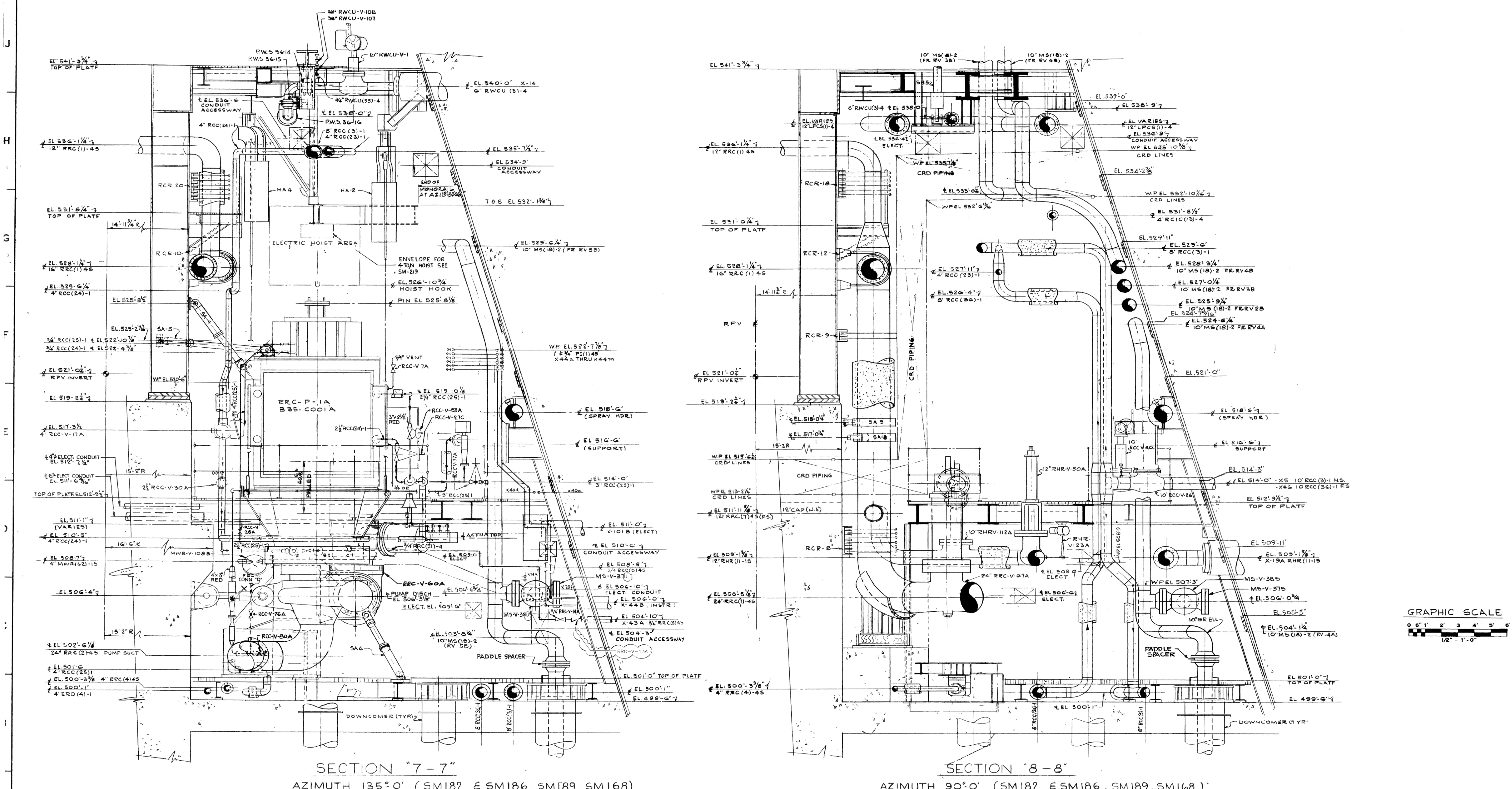
Columbia Generating Station
Final Safety Analysis Report

Containment Composite Sections "5-5" and "6-6"

Draw. No.	SM193	Rev.	F	Figure	3.5-27

Amendment 60
December 2009

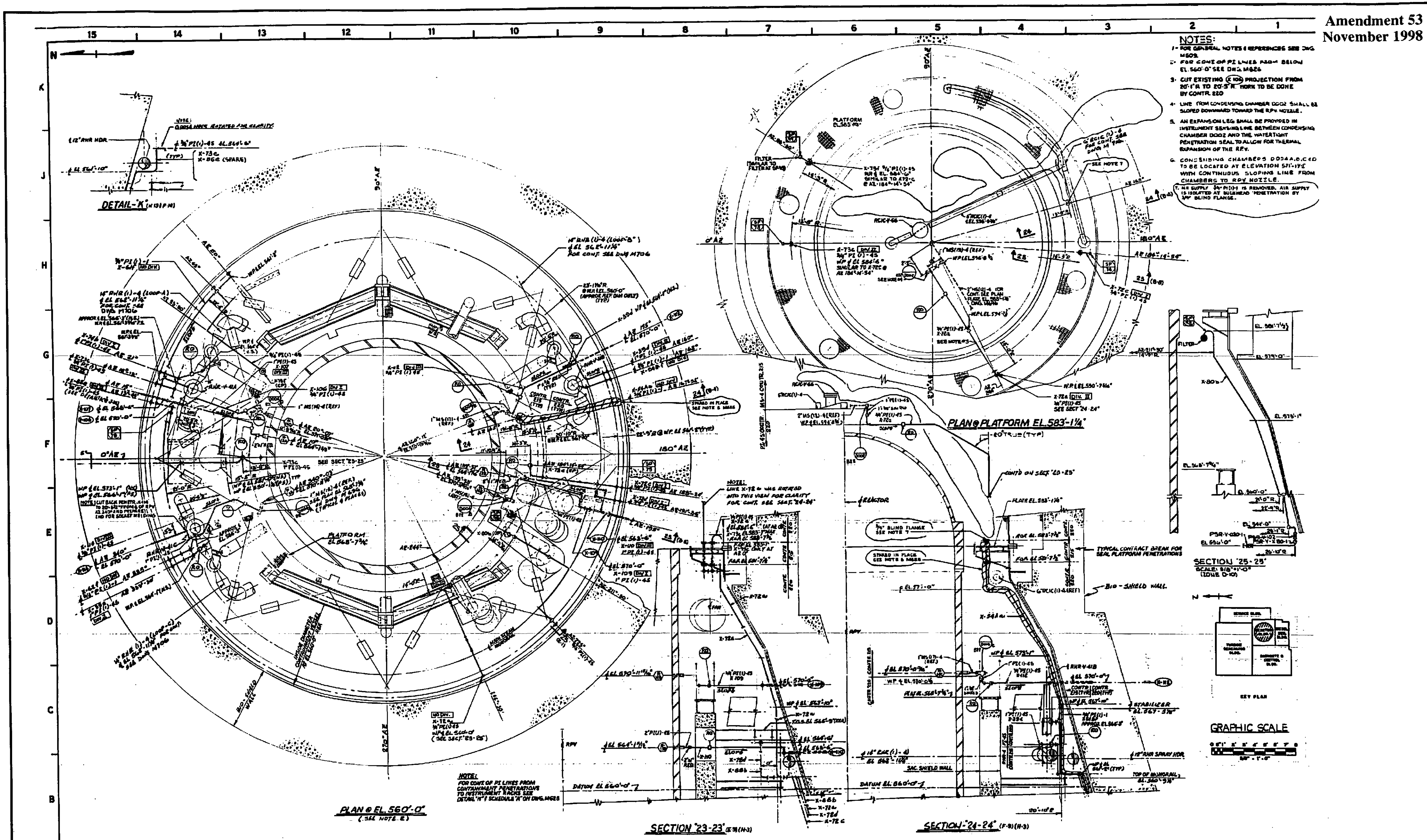
NOTES:
1. WORK THIS DWG WITH DWGS SM182, SM186, SM189, & SM168.
2 FOR GENERAL NOTES SEE DWG. SM168.



Columbia Generating Station
Final Safety Analysis Report

Containment Composite Sections "7-7" and "8-8"

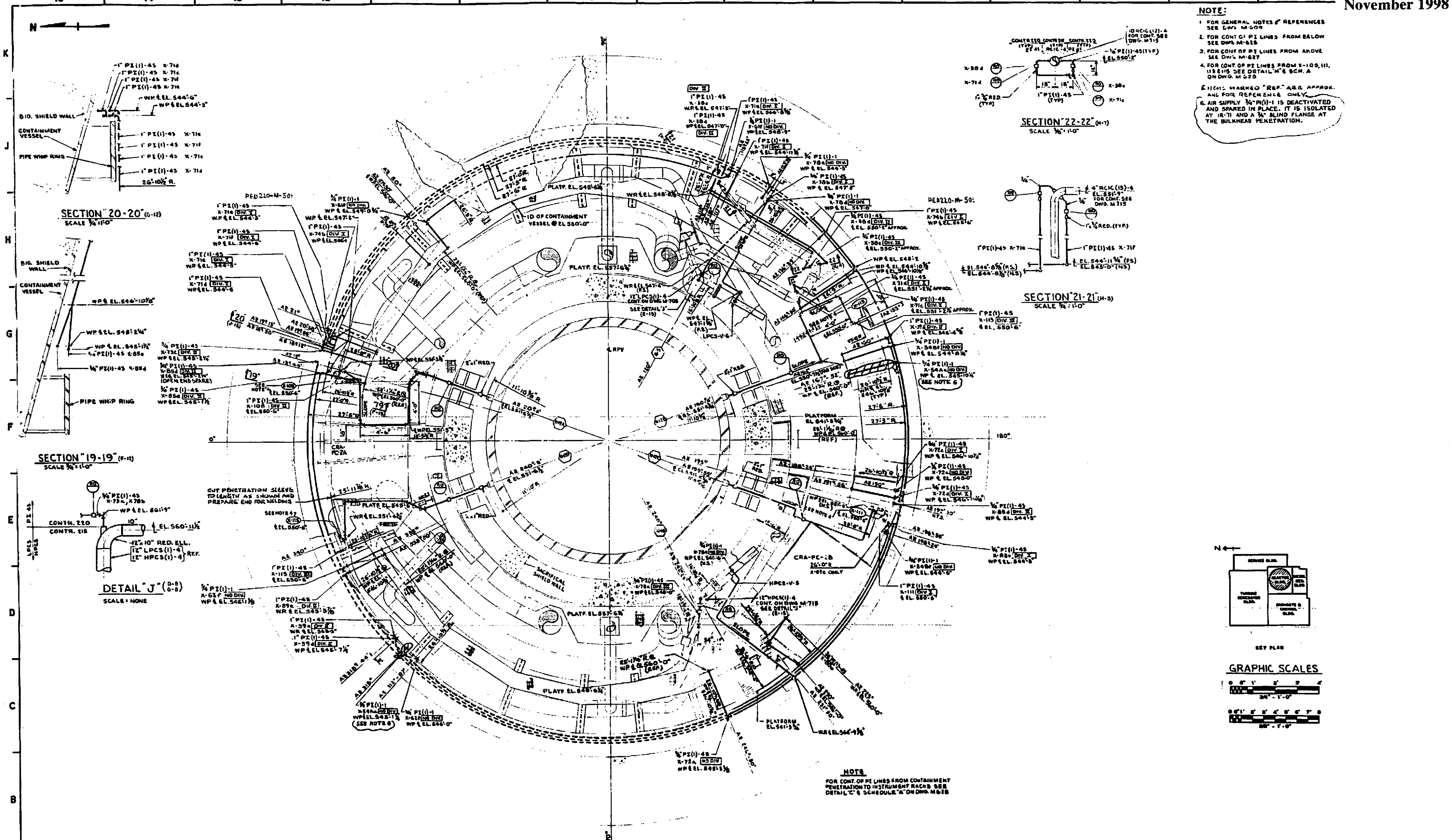
Draw. No. SM194 | Rev. 0 | Figure 03.5-28



Columbia Generating Station
Final Safety Analysis Report

Instrumentation Lines Plan - El. 560 ft 0 in. and El. 583 ft 1-1/4 in. - Reactor Building

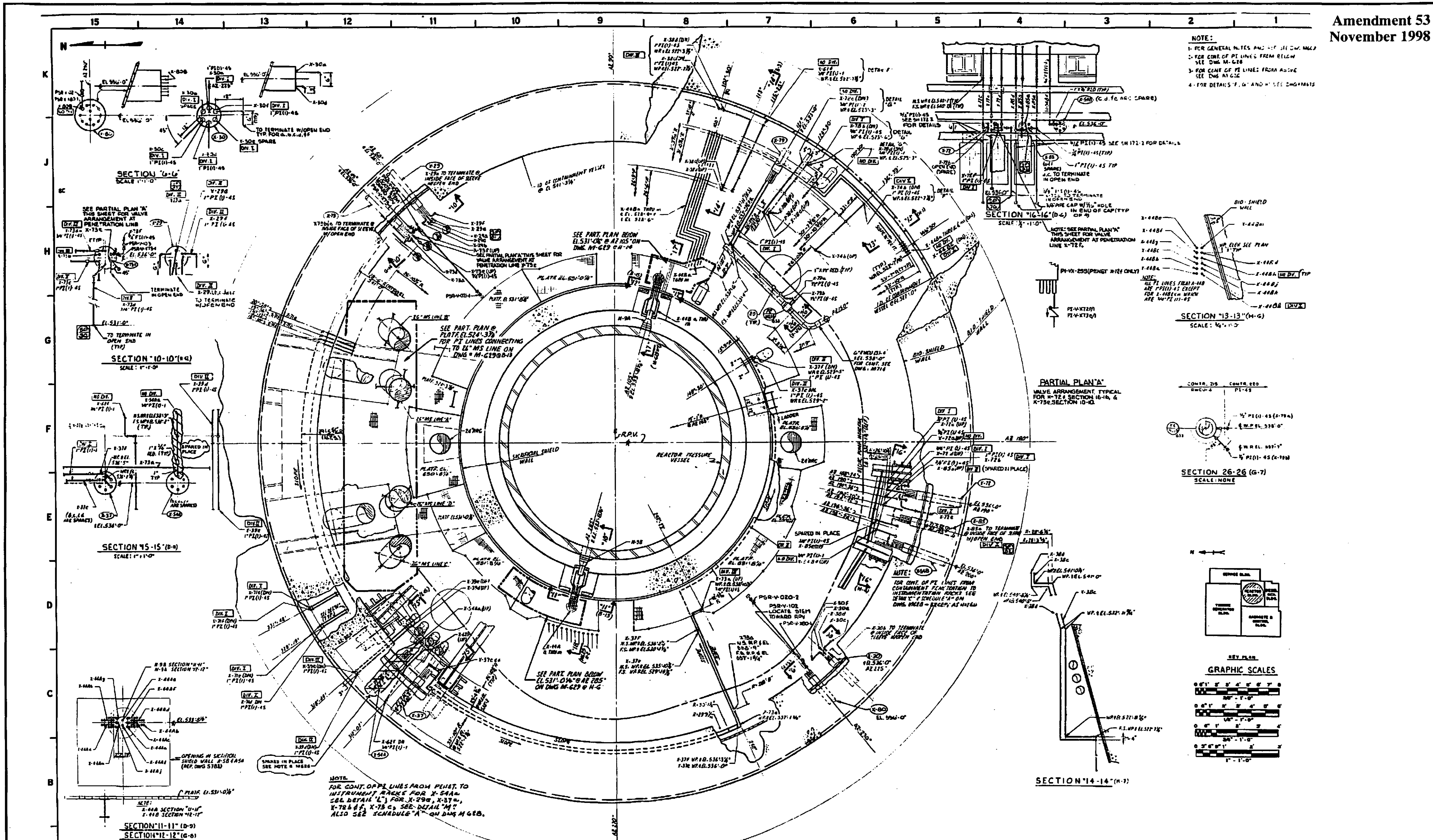
Draw. No.	M627	Rev.	16	Figure	3.5-29



Columbia Generating Station
Final Safety Analysis Report

Instrumentation Lines Plan - Platform El. 541 ft 3-3/4 in. - Reactor Building

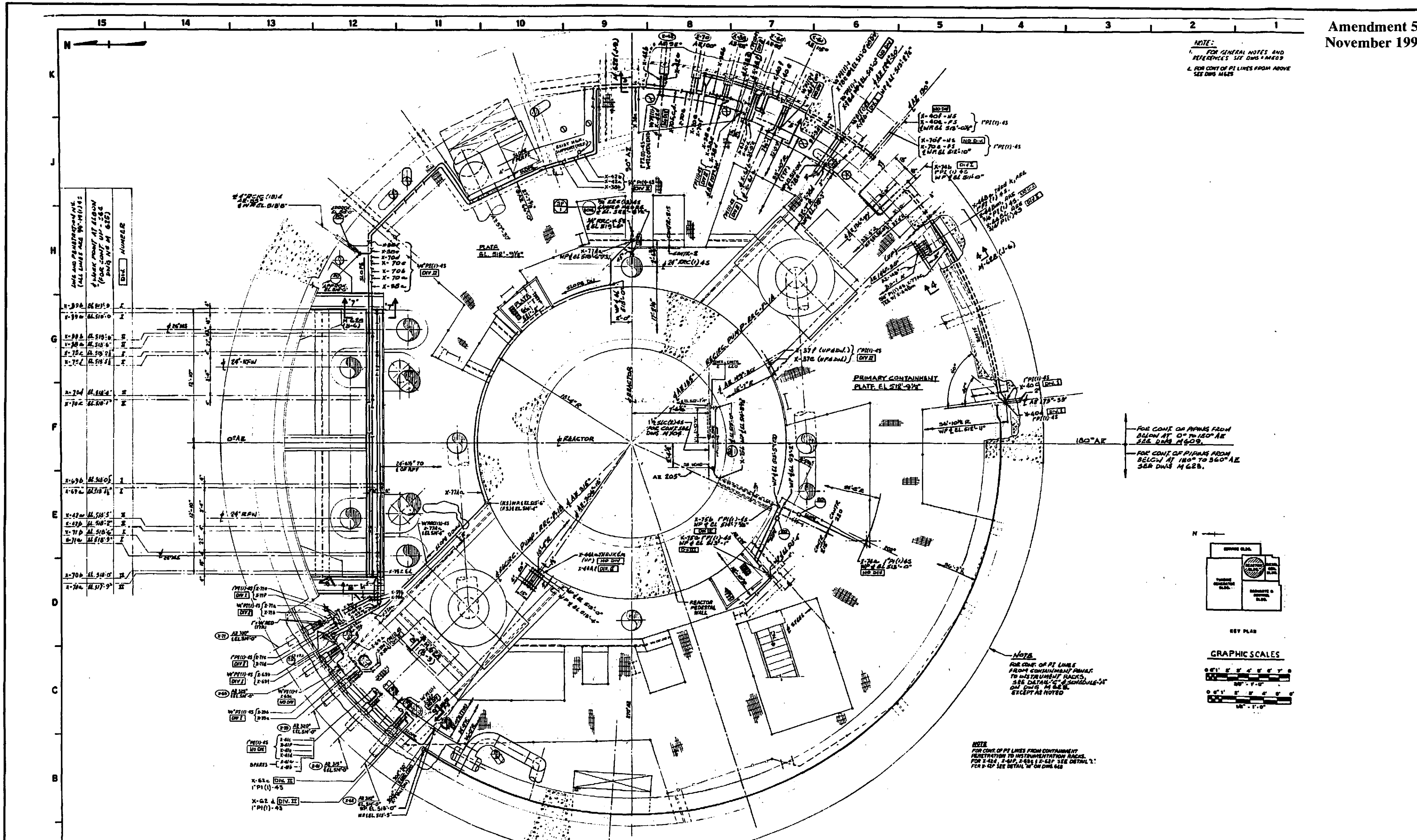
Draw. No.	M626	Rev.	10	Figure	3.5-30



Columbia Generating Station
Final Safety Analysis Report

Instrumentation Process (PI) Plan - Floor El. 522 ft 0 in. - Reactor Building

Draw. No.	M625	Rev.	16	Figure	3.5-31



Columbia Generating Station
Final Safety Analysis Report

Instrumentation Process (PI) Plan - Platform
El. 512 ft 9-1/2 in. - Reactor Building

Draw. No. M624 | Rev. 11 | Figure 3.5-32

Figure Not Available For Public Viewing

Figure Not Available For Public Viewing

Figure Not Available For Public Viewing

Figure Not Available For Public Viewing