3.1 Emergency Response Facilities

Design Description

The technical support center (TSC) is a facility from which management and technical support is provided to main control room (MCR) personnel during emergency conditions. The operations support center (OSC) provides an assembly area where operations support personnel report in an emergency.

1. The TSC has floor space of at least 75 ft^2 per person for a minimum of 25 persons.
2. The TSC has voice communication equipment for communication with the MCR, emergency operations facility, OSC, and the U.S. Nuclear Regulatory Commission (NRC).
3. The plant parameters listed in Table 2.5.4-1, minimum inventory table, in subsection 2.5.4, Data Display and Processing System (DDS), with a "Yes" in the "Display" column, can be retrieved in the TSC.
4. The OSC has voice communication equipment for communication with the MCR and TSC.
5. The TSC and OSC are in different locations in the annex building. The TSC is adjacent to the passage from the annex building to the nuclear island control room.

Inspections, Tests, Analyses, and Acceptance Criteria

Table 3.1-1 specifies the inspections, tests, analyses, and associated acceptance criteria for the emergency response facilities.

Table 3.1-1
Inspections, Tests, Analyses, and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. The TSC has floor space of at least 75 ft^2 per person for a minimum of 25 persons.	An inspection will be performed of the TSC floor space.	The TSC has at least 1875 ft^2 of floor space.
2. The TSC has voice communication equipment for communication with the MCR, emergency operations facility, OSC, and the NRC.	An inspection and test will be performed of the TSC voice communication equipment.	Communications equipment is installed, and voice transmission and reception are accomplished.
3. The plant parameters listed in Table 2.5.4-1, minimum inventory table, in subsection 2.5.4, DDS, with a "Yes" in the "Display" column, can be retrieved in the TSC.	An inspection will be performed for retrievability of the plant parameters in the TSC.	The plant parameters listed in Table 2.5.4-1, minimum inventory table, in subsection 2.5.4, DDS, with a "Yes" in the "Display" column, can be retrieved in the TSC.
4. The OSC has voice communication equipment for communication with the MCR and TSC.	Inspection will be performed of the OSC voice communication equipment.	Communications equipment is installed, and voice transmission and reception are accomplished.
5. The TSC and OSC are in different locations in the annex building. The TSC is adjacent to the passage from the annex building to the nuclear island control room.	An inspection will be performed of the location of the TSC and OSC.	The TSC and OSC are in different locations in the annex building. The TSC is adjacent to the passage from the annex building to the nuclear island control room.

3.2 Human Factors Engineering

Design Description

The AP600 human-system interface (HSI) will be developed and implemented based upon a human factors engineering (HFE) program. Figure 3.2-1 illustrates the HFE program elements. The HSI scope includes the design of the operation and control centers system (OCS) and each of the HSI resources. For the purposes of the HFE program, the OCS includes the main control room (MCR), the remote shutdown room (RSR), the local control stations, and the associated workstations for each of these centers. The HSI resources include the wall panel information system, alarm system, plant information system (nonsafety-related displays), qualified data processing system (safety-related displays), and soft and dedicated controls. Minimum inventories of controls, displays, and visual alerts are specified as part of the HSI for the MCR and the remote shutdown workstation (RSW).

The MCR provides a facility and resources for the safe control and operation of the plant. The MCR includes a minimum inventory of displays, visual alerts and fixed-position controls. Refer to item 8.a and Table 2.5.2-5 of subsection 2.5.2 for this minimum inventory.

The RSR provides a facility and resources to establish and maintain safe shutdown conditions for the plant from a location outside of the MCR. The RSW includes a minimum inventory of displays, controls, and visual alerts. Refer to item 2 and Table 2.5.4-1 of subsection 2.5.4 for this minimum inventory. As stated in item 8.b of subsection 2.5.2, the protection and safety monitoring system (PMS) provides for the transfer of control capability from the MCR to the RSW.

The mission of local control stations is to provide the resources, outside of the MCR, for operations personnel to perform monitoring and control activities.

Implementation of the HFE program includes activities 1 through 5 listed below. The MCR includes design features specified by items 6 through 8 below. The RSW includes the design features specified by items 9 through 12 below. Local control stations include the design feature of item 13.

1. The integration of human reliability analysis with HFE design is performed in accordance with the implementation plan. Critical human actions (if any) and risk-important tasks are identified and used as an input to the task analysis activities.

2. Task analysis is performed in accordance with the task analysis implementation plan. Task analysis identifies the information and control requirements for the operators to execute the tasks allocated to them.

3. The HSI design is performed for the OCS in accordance with the HSI design implementation plan. The HSI design includes the functional design of the operation and control centers and the HSI resources, the specification of design guidelines, the HSI resource design specifications, and the man-in-the-loop concept testing.

4. An HFE program verification and validation implementation plan is developed in accordance with the programmatic level description of the AP600 human factors verification and validation plan. The implementation plan establishes methods for conducting evaluations of the HSI design.

5. The HFE verification and validation program is performed in accordance with the HFE verification and validation implementation plan and includes the following activities:

   a) HSI Task support verification
   b) HFE design verification
   c) Integrated system validation
   d) Issue resolution verification
   e) Plant HFE/HSI (as designed at the time of plant startup) verification

6. The MCR includes reactor operator workstations, supervisor workstation(s), safety-related displays, and safety-related controls.

7. The MCR provides a suitable workspace environment for use by MCR operators.

8. The HSI resources available to the MCR operators include the alarm system, plant information system (nonsafety-related displays), wall panel information system, and nonsafety-related controls (soft and dedicated).

9. The RSW includes reactor operator workstation(s) from which licensed operators perform remote shutdown operations.

10. The remote shutdown room (RSR) provides a suitable workspace environment, separate from the MCR, for use by the RSW operators.

11. The HSI resources available at the RSW include the alarm system displays, the plant information system, and the controls.

12. The RSW and the available HSI permit execution of tasks by licensed operators to establish and maintain safe shutdown.

13. The capability to access displays and controls is provided (controls as assigned by the MCR operators) for local control and monitoring from selected locations throughout the plant.

Inspections, Tests, Analyses, and Acceptance Criteria

Table 3.2-1 specifies the inspections, tests, analyses, and associated acceptance criteria for the HFE program, MCR, RSW, and local control stations.

Table 3.2-1
Inspections, Tests, Analyses, and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. The integration of human reliability analysis with HFE design is performed in accordance with the implementation plan.	An evaluation of the implementation for the integration of human reliability analysis with HFE design will be performed.	A report exists and concludes that critical human actions (if any) and risk important tasks were identified and examined by task analysis, and used as input to the HSI design, procedure development, staffing, and training.
2. Task analysis is performed in accordance with the task analysis implementation plan.	An evaluation of the implementation of the task analysis will be performed.	A report exists and concludes that function-based task analyses were conducted in conformance with the task analysis implementation plan and include the following functions: – Control reactivity – Control reactor coolant system (RCS) boron concentration – Control fuel and cladding temperature – Control RCS coolant temperature, pressure, and inventory – Provide RCS flow – Control main steam pressure – Control steam generator inventory – Control containment pressure and temperature – Provide control of main turbine

Table 3.2-1 (cont.) Inspections, Tests, Analyses, and Acceptance Criteria		
Design Commitment	**Inspections, Tests, Analyses**	**Acceptance Criteria**
		A report exists and concludes that operational sequence analyses (OSAs) were conducted in conformance with the task analysis implementation plan. OSAs performed include the following: – Plant heatup and startup from post-refueling to 100% power – Reactor trip, turbine trip, and safety injection – Natural circulation cooldown (startup feedwater with steam generator) – Loss of reactor or secondary coolant – Post-loss-of-coolant accident (LOCA) cooldown and depressurization – Loss of RCS inventory during shutdown – Loss of the normal residual heat removal system (RNS) during shutdown – Manual automatic depressurization system (ADS) actuation – Manual reactor trip via PMS, via diverse actuation system (DAS) – ADS valve testing during mode 1

Table 3.2-1 (cont.) Inspections, Tests, Analyses, and Acceptance Criteria		
Design Commitment	**Inspections, Tests, Analyses**	**Acceptance Criteria**
3. The HSI design is performed for the OCS in accordance with the HSI design implementation plan.	An evaluation of the implementation of the HSI design will be performed.	A report exists and concludes that the HSI design for the OCS was conducted in conformance with the implementation plan and includes the following documents: – Operation and Control Centers System Specification Document – Functional requirements and design basis documents for the alarm system, plant information system, wall panel information system, controls (soft and dedicated), and the qualified data processing system – Design guideline documents (based on accepted HFE guidelines, standards, and principles) for the alarm system, displays, controls, and anthropometrics – Design specifications for the alarm system, plant information system, wall panel information system, controls (soft and dedicated), and the qualified data processing system. – Man-in-the-loop concept test reports

Table 3.2-1 (cont.)
Inspections, Tests, Analyses, and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
4. An HFE program verification and validation implementation plan is developed in accordance with the programmatic level description of the AP600 human factors verification and validation plan.	An inspection of the HFE verification and validation implementation plan will be performed.	A report exists and concludes that the HFE verification and validation implementation plan was developed in accordance with the programmatic level description of the AP600 human factors verification and validation plan and includes the following activities: – HSI task support verification – HFE design verification – Integrated system validation – Issue resolution verification – Plant HFE/HSI (as designed at the time of plant startup) verification

Table 3.2-1 (cont.)
Inspections, Tests, Analyses, and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
5. The HFE verification and validation program is performed in accordance with the HFE verification and validation implementation plan and includes the following activities: a) HSI Task support verification b) HFE design verification c) Integrated system validation d) Issue resolution verification e) Plant HFE/HSI (as designed at the time of plant startup) verification	a) An evaluation of the implementation of the HSI task support verification will be performed. b) An evaluation of the implementation of the HFE design verification will be performed. c) (i) An evaluation of the implementation of the integrated system validation will be performed.	A report exists and concludes that: a) Task support verification was conducted in conformance with the implementation plan and includes verification that the information and controls provided by the HSI match the display and control requirements generated by the function-based task analyses and the operational sequence analyses. b) HFE design verification was conducted in conformance with the implementation plan and includes verification that the HSI design is consistent with the AP600 specific design guidelines (compiled as specified in the third acceptance criteria of design commitment 3) developed for each HSI resource. c) (i) The test scenarios listed in the implementation plan for integrated system validation were executed in conformance with the plan and noted human deficiencies were addressed.

Table 3.2-1 (cont.) Inspections, Tests, Analyses, and Acceptance Criteria		
Design Commitment	**Inspections, Tests, Analyses**	**Acceptance Criteria**
	c) (ii) Tests and analyses of the following plant evolutions and transients, using a facility that physically represents the MCR configuration and dynamically represents the MCR HSI and the operating characteristics and responses of the AP600 design, will be performed: – Normal plant heatup and startup to 100% power – Normal plant shutdown and cooldown to cold shutdown – Transients: reactor trip and turbine trip – Accidents: - Small-break LOCA - Large-break LOCA - Steam line break - Feedwater line break - Steam generator tube rupture	c) (ii) The test and analysis results demonstrate that the MCR operators can perform the following: – Heat up and start up the plant to 100% power – Shut down and cool down the plant to cold shutdown – Bring the plant to safe shutdown following the specified transients – Bring the plant to a safe, stable state following the specified accidents
	d) An evaluation of the implementation of the HFE design issue resolution verification will be performed.	d) HFE design issue resolution verification was conducted in conformance with the implementation plan and includes verification that human factors issues documented in the design issues tracking system have been addressed in the final design.
	e) An evaluation of the implementation of the plant HFE/HSI (as designed at the time of plant startup) verification will be performed.	e) The plant HFE/HSI, as designed at the time of plant startup, is consistent with the HFE/HSI verified in 5.a) through 5.d).

Table 3.2-1 (cont.)
Inspections, Tests, Analyses, and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
6. The MCR includes reactor operator workstations, supervisor workstation(s), safety-related displays, and safety-related controls.	An inspection of the MCR workstations and control panels will be performed.	The MCR includes reactor operator workstations, supervisor workstation(s), safety-related displays, and safety-related controls.
7. The MCR provides a suitable workspace environment for use by the MCR operators.	i) See Tier 1 Material, subsection 2.7.1, Nuclear Island Nonradioactive Ventilation System. ii) See Tier 1 Material, subsection 2.2.5, MCR Emergency Habitability System. iii) See Tier 1 Material, subsection 2.6.3, Class 1E dc and UPS System. iv) See Tier 1 Material, subsection 2.6.5, Lighting System. v) See Tier 1 Material, subsection 2.3.19, Communication System.	i) See Tier 1 Material, subsection 2.7.1, Nuclear Island Nonradioactive Ventilation System. ii) See Tier 1 Material, subsection 2.2.5, MCR Emergency Habitability System. iii) See Tier 1 Material, subsection 2.6.3, Class 1E dc and UPS system. iv) See Tier 1 Material, subsection 2.6.5, Lighting System. v) See Tier 1 Material, subsection 2.3.19, Communication System.
8. The HSI resources available to the MCR operators include the alarm system, plant information system (nonsafety-related displays), wall panel information system, and nonsafety-related controls (soft and dedicated).	An inspection of the HSI resources available in the MCR for the MCR operators will be performed.	The HSI (at the time of plant startup) includes an alarm system, plant information system (nonsafety-related displays), wall panel information system, and nonsafety-related controls (soft and dedicated).
9. The RSW includes reactor operator workstation(s) from which licensed operators perform remote shutdown operations.	An inspection of the RSW will be performed.	The RSW includes reactor operator workstation(s).

Table 3.2-1 (cont.)
Inspections, Tests, Analyses, and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
10. The RSR provides a suitable workspace environment, separate from the MCR, for use by the RSW operators.	i) See Tier 1 Material, subsection 2.7.1, Nuclear Island Nonradioactive Ventilation System. ii) See Tier 1 Material, subsection 2.6.5, Lighting System. iii) See Tier 1 Material, subsection 2.3.19, Communication System.	i) See Tier 1 Material, subsection 2.7.1, Nuclear Island Nonradioactive Ventilation System. ii) See Tier 1 Material, subsection 2.6.5, Lighting System. iii) See Tier 1 Material, subsection 2.3.19, Communication System.
11. The HSI resources available at the RSW include the alarm system displays, the plant information system, and the controls.	An inspection of the HSI resources available at the RSW will be performed.	The as-built HSI at the RSW includes the alarm system displays, the plant information system, and the controls.
12. The RSW and the available HSI permit execution of tasks by licensed operators to establish and maintain safe shutdown.	Test and analysis, using a workstation that physically represents the RSW and dynamically represents the RSW HSI and the operating characteristics and responses of the AP600, will be performed.	A report exists and concludes that the test and analysis results demonstrate that licensed operators can achieve and maintain safe shutdown conditions from the RSW.
13. The capability to access displays and controls is provided (controls as assigned by the MCR operators) for local control and monitoring from selected locations throughout the plant.	An inspection of the local control and monitoring capability is provided.	The capability for local control and monitoring from selected locations throughout the plant exists.

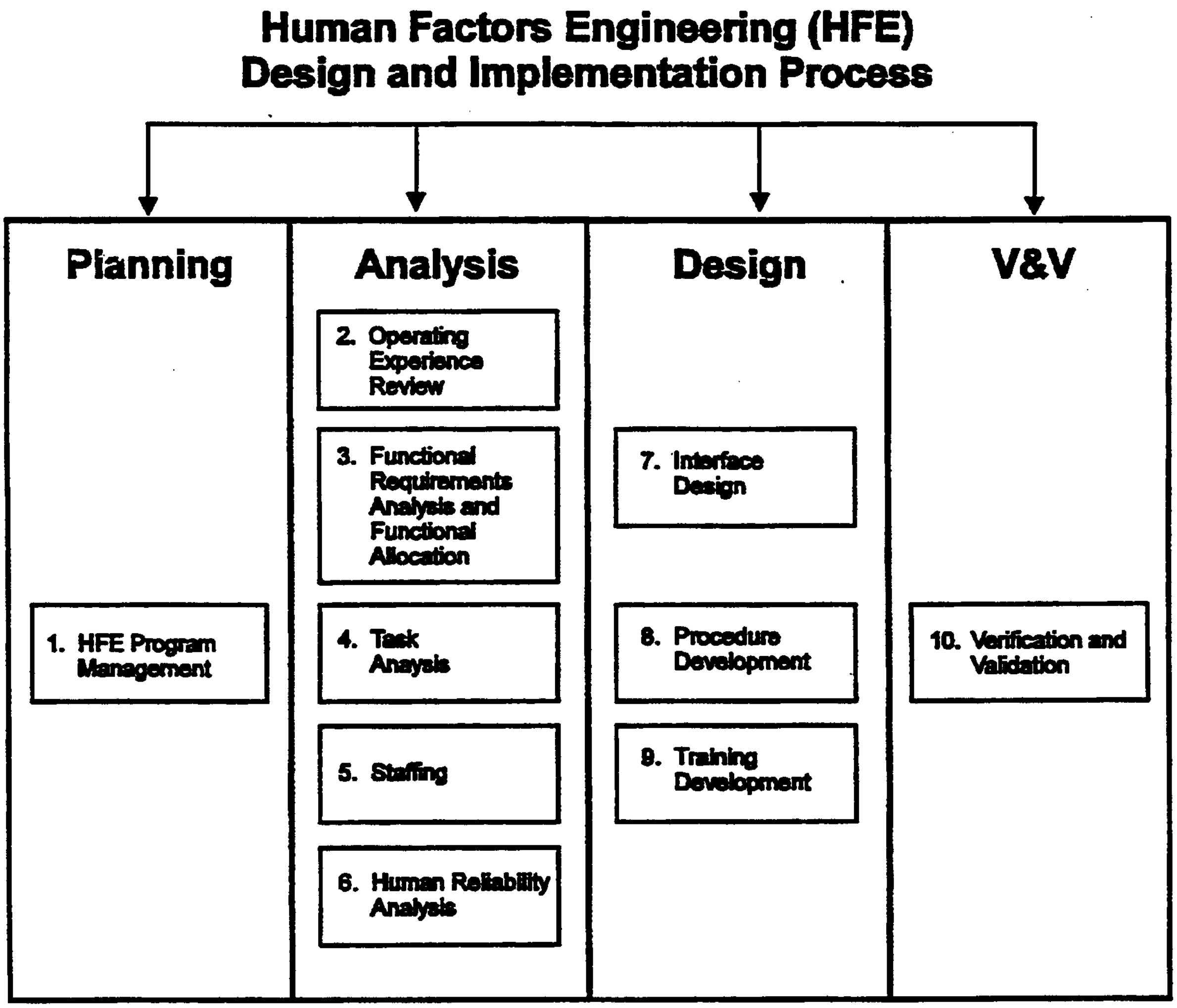

Figure 3.2-1
Human Factors Engineering (HFE)
Design and Implementation Process

3.3 Buildings

Design Description

The nuclear island structures include the containment (the steel containment vessel and the containment internal structure) and the shield and auxiliary buildings. The containment, shield and auxiliary buildings are structurally integrated on a common basemat which is embedded below the finished plant grade level. The containment vessel is a cylindrical welded steel vessel with elliptical upper and lower heads, supported by embedding a lower segment between the containment internal structures concrete and the basemat concrete. The containment internal structure is reinforced concrete with structural modules used for some walls and floors. The shield building is reinforced concrete and, in conjunction with the internal structures of the containment building, provides shielding for the reactor coolant system and the other radioactive systems and components housed in the containment. The shield building roof is a reinforced concrete structure containing an integral, steel lined passive containment cooling water storage tank. The auxiliary building is reinforced concrete and houses the safety-related mechanical and electrical equipment located outside the containment and shield buildings.

The portion of the annex building adjacent to the nuclear island is a structural steel and reinforced concrete seismic Category II structure and houses the technical support center, non-1E electrical equipment, and hot machine shop.

The radwaste building is a steel framed structure and houses the low level waste processing and storage.

The turbine building is a non-safety related structure that houses the main turbine generator and the power conversion cycle equipment and auxiliaries. There is no safety-related equipment in the turbine building. The turbine building is located on a separate foundation. The turbine building structure is adjacent to the nuclear island structures.

The diesel generator building is a non-safety related structure that houses the two standby diesel engine powered generators and the power conversion cycle equipment and auxiliaries. There is no safety-related equipment in the diesel generator building. The diesel generator building is located on a separate foundation at a distance from the nuclear island structures.

The plant gas system (PGS) provides hydrogen, carbon dioxide, and nitrogen gases to the plant systems as required. The component locations of the PGS are located either in the turbine building or the yard areas.

1. The physical arrangement of the nuclear island structures and the annex building is as described in the Design Description of this Section 3.3, and as shown on Figures 3.3-1 through 3.3-14. The physical arrangement of the radwaste building, the turbine building, and the diesel generator building is as described in the Design Description of this Section 3.3.

2. a) The nuclear island structures, including the critical sections listed in Table 3.3-7, are seismic Category I and are designed and constructed to withstand design basis loads, as specified in the Design Description, without loss of structural integrity and the safety-related functions. The design bases loads are those loads associated with:

    - Normal plant operation (including dead loads, live loads, lateral earth pressure loads, and equipment loads, including hydrodynamic loads, temperature and equipment vibration);
    - External events (including rain, snow, flood, tornado, tornado generated missiles and earthquake); and
    - Internal events (including flood, pipe rupture, equipment failure, and equipment failure generated missiles).

   b) Site grade level is located relative to floor elevation 100'-0" per Table 3.3-5. Floor elevation 100'-0" is defined as the elevation of the floor at design plant grade.

   c) The containment and its penetrations are designed and constructed to ASME Code Section III, Class MC.(1)

   d) The containment and its penetrations retain their pressure boundary integrity associated with the design pressure.

   e) The containment and its penetrations maintain the containment leakage rate less than the maximum allowable leakage rate associated with the peak containment pressure for the design basis accident.

   f) The key dimensions of the nuclear island structures are as defined on Table 3.3-5.

   g) The containment vessel greater than 7 feet above the operating deck provides a heat transfer surface. A free volume exists inside the containment shell above the operating deck.

3. Walls and floors of the nuclear island structures as defined on Table 3.3-1, except for designed openings and penetrations, provide shielding during normal operations.

4. Walls and floors of the annex building as defined on Table 3.3-1, except for designed openings and penetrations, provide shielding during normal operations.

5. a) Exterior walls and the basemat of the nuclear island have a water barrier up to site grade.

   b) The boundaries between mechanical equipment rooms and the electrical and instrumentation and control (I&C) equipment rooms of the auxiliary building as identified in Table 3.3-2 are

1. Containment isolation devices are addressed in subsection 2.2.1, Containment System.

designed to prevent flooding of rooms that contain safety-related equipment up to the maximum flood level for each room defined in Table 3.3-2.

c) The boundaries between the following rooms, which contain safety-related equipment – passive core cooling system (PXS) valve/accumulator room A (11205), PXS valve/accumulator room B (11207), and chemical and volume system (CVS) room (11209) – are designed to prevent flooding between these rooms.

6. a) The radiologically controlled area of the auxiliary building between floor elevations 66'-6" and 82'-6" contains adequate volume to contain the liquid volume of faulted liquid radwaste system (WLS) storage tanks. The available room volumes of the radiologically controlled area of the auxiliary building between floor elevations 66'-6" and 82'-6" exceeds the volume of the liquid radwaste storage tanks (WLS-MT-05A, MT-05B, MT-06A, MT-06B, MT-07A, MT-07B, MT-07C, MT-11).

   b) The radwaste building packaged waste storage room has a volume greater than or equal to 1293 cubic feet.

7. a) Class 1E electrical cables, fiber optic cables associated with only one division, and raceways are identified according to applicable color-coded Class 1E divisions.

   b) Class 1E divisional electrical cables and fiber optic cables associated with only one division are routed in their respective divisional raceways.

   c) Separation is maintained between Class 1E divisions in accordance with the fire areas as identified in Table 3.3-3.

   d) Physical separation is maintained between Class 1E divisions and between Class 1E divisions and non-Class 1E cables.

   e) Class 1E fiber optic cables which interconnect two divisions are routed and separated such that the Protection and Safety Monitoring System voting logic is not defeated by the loss of any single raceway or fire area.

8. Equipment labeled as essential targets in Table 3.3-4 and located in rooms identified in Table 3.3-4 are protected from the dynamic effects of postulated pipe breaks.

9. The reactor cavity sump has a minimum concrete thickness as shown on Table 3.3-5 between the bottom of the sump and the steel containment.

10. The shield building roof and the passive containment cooling system (PCS) storage tank support and retain the PCS water. The passive containment cooling system tank has a stainless steel liner which provides a barrier on the inside surfaces of the tank. Leak chase channels are provided over the tank boundary liner welds.

11. The construction approach for soft soil sites having unconsolidated deposits with shear wave velocities in the range from 1,000 to 2,000 feet per second includes two limits:

    - Concrete will not be placed in the shield building or containment internal structure above floor elevation 84'-0" before the walls of the auxiliary building are completed to floor elevation 82'-6", and

    - Concrete will not be placed in the auxiliary building above floor elevation 117'-6" before the shield building is completed to floor elevation 82'-6".

12. The extended turbine generator axis intersects the shield building.

13. Separation is provided between the structural elements of the turbine, annex, and radwaste buildings and the nuclear island structure. This separation permits horizontal motion of the buildings in a safe shutdown earthquake without impact between structural elements of the buildings.

14. Protected Area/Vital Area walls that are accessible and unmonitored are security hardened.

Inspections, Tests, Analyses, and Acceptance Criteria

Table 3.3-6 specifies the inspections, tests, analyses, and associated acceptance criteria for the buildings.

Table 3.3-1
Definition of Wall Thicknesses for Nuclear Island Buildings and Annex Building[1]

Wall or Section Description	Column Lines	Floor Elevation or Elevation Range	Concrete Thickness[2][3]	Applicable Radiation Shielding Wall (Yes/No)
Containment Building Internal Structure				
Shield Wall between Reactor Vessel Cavity and RCDT Room	E-W wall parallel with column line 7	From 71'-6" to 83'-0"	3'-0"	Yes
West Reactor Vessel Cavity Wall	N-S wall parallel with column line N	From 83'-0" to 98'-0"	7'-6"	Yes
North Reactor Vessel Cavity Wall	E-W wall parallel with column line 7	From 83'-0" to 98'-0"	9'-0"	Yes
East Reactor Vessel Cavity Wall	N-S wall parallel with column line N	From 83'-0" to 98'-0"	7'-6"	Yes
West Refueling Cavity Wall	N-S wall parallel with column line N	From 98'-0" to 135'-3"	4'-0"	Yes
North Refueling Cavity Wall	E-W wall parallel with column line 7	From 98'-0" to 135'-3"	4'-0"	Yes
East Refueling Cavity Wall	N-S wall parallel with column line N	From 98'-0" to 135'-3"	4'-0"	Yes
South Refueling Cavity Wall	E-W wall parallel with column line 7	From 98'-0" to 135'-3"	4'-0"	Yes
South wall of west steam generator compartment	Not Applicable	From 103'-0" to 135'-3"	2'-6"	No
West wall of west steam generator compartment	Not Applicable	From 103'-0" to 135'-3"	2'-6"	No
North wall of west steam generator compartment/south wall of pressurizer compartment	Not Applicable	From 103'- 0" to 135'-3" and 158'-0"	2'-6"	Yes
West wall of pressurizer compartment	Not Applicable	From 107'-2" to 158'-0"	2'-6"	Yes
North wall of pressurizer compartment	Not Applicable	From 107'-2" to 158'-0"	2'-6"	Yes
East wall of pressurizer compartment	Not Applicable	From 118'-6" to 158'-0"	2'-6"	Yes
North-east wall of in-containment refueling water storage tank	Parallel to column line N	From 103'-0" to 135'-3"	2'-6"	No
West wall of in-containment refueling water storage tank	Not applicable	From 103'-0" to 135'-3"	5/8" steel plate with stiffeners	No
South wall of east steam generator compartment	Not Applicable	From 87'-6" to 135'-3"	2'-6"	Yes

1. The column lines and floor elevations are identified and included on Figures 3.3-1 through 3.3-13.
2. These wall thicknesses have a construction tolerance of ± 1 inch, except for exterior walls below grade where the tolerance is +12 inches, - 1 inch.
3. For walls that are part of structural modules, the concrete thickness also includes the steel face plates.

Table 3.3-1 (cont.)
Definition of Wall Thicknesses for Nuclear Island Buildings and Annex Building(1)

Wall or Section Description	Column Lines	Floor Elevation or Elevation Range	Concrete Thickness(2)(3)	Applicable Radiation Shielding Wall (Yes/No)
East wall of east steam generator compartment	Not Applicable	From 94'-0" to 135'-3"	2'-6"	Yes
North wall of east steam generator compartment	Not Applicable	From 87'-6" to 135'-3"	2'-6"	Yes
Shield Building				
Shield Building Cylinder	Not Applicable	From 100'-0" to 241'-0"	3'-0"	Yes
Columns between air inlets	Not Applicable	From 241'-0" to 246'-0"	3'-0"	Yes
Tension Ring	Not Applicable	From 246'-0" to 250'-0"	3'-0"	Yes
Conical Roof	Not Applicable	From 250'-0" to 289'-0"	1'-6" cast-in-place concrete over 6" pre-cast concrete ribbed conical sections	Yes
PCS Tank External Cylindrical Wall	Not Applicable	From 276'-0" to 308'-3"	2'-0"	Yes
PCS Tank Internal Cylindrical Wall	Not Applicable	From 289'- to 308'-3"	1'-6"	Yes
PCS Tank Roof	Not Applicable	308'-3"	1'-3"	No
Auxiliary Building Walls/Floors				
Column Line 1 wall	From I to N	From 66'-6" to 100'-0"	3'-0"	No
Column Line 1 wall	From I to N	From 100'-0" to 180'-0"	2'-3"	Yes
Column Line 2 wall	From I to K-2	From 66'-6" to 135'-3"	2'-6"	Yes
Column Line 2 wall	From K-2 to L-2	From 66'-6" to 135'-3"	5'-0"	Yes
Column Line 2 wall	From L-2 to N	From 98'-0" to 135'-3"	2'-6"	Yes
Column Line 2 wall	From I to J-1	From 135'-3" to 153'-0"	2'-0"	Yes
Column Line 3 wall	From J-1 to J-2	From 66'-6" to 82'-6"	2'-6"	Yes
Column Line 3 wall	From J-1 to J-2	From 100'-0" to 135'-3"	2'-6"	Yes

Table 3.3-1 (cont.)
Definition of Wall Thicknesses for Nuclear Island Buildings and Annex Building(1)

Wall or Section Description	Column Lines	Floor Elevation or Elevation Range	Concrete Thickness(2)(3)	Applicable Radiation Shielding Wall (Yes/No)
Column Line 3 wall	From J-2 to K-2	From 66'-6" to 135'-3"	2'-6"	Yes
Column Line 3 wall	From K-2 to L-2	From 66'-6" to 94'-3"	2'-6"	Yes
Column Line 4 wall	From I to J-1	From 66'-6" to 153'-0"	2'-6"	Yes
Column Line 4 wall	From J-1 to J-2	From 66'-6" to 92'-6"	2'-6"	Yes
Column Line 4 wall	From J-1 to J-2	From 100'-0" to 135'-3"	2'-6"	Yes
Column Line 4 wall	From J-2 to K-2	From 66'-6" to 135'-3"	2'-6"	Yes
Column Line 4 wall	From I to intersection with shield building wall	From 135'-3" to 180'-0"	2'-0"	Yes
Column Line 5 wall	From I to J-1	From 66'-6" to 160'-6"	2'-0"	Yes
Column Line 7.1 wall	From I to J-1	From 66'-6" to 82'-6"	2'-0"	Yes
Column Line 7.2 wall	From I to J-1	From 66'-6" to 100'-0"	2'-0"	Yes
Column Line 7.3 wall	From I to K	From 66'-6" to 100'-0"	3'-0"	Yes
Column Line 7.3 wall	From I to K	From 100'-0" to 160'-6"	2'-0"	No
Column Line 11 wall	From I to Q	From 66'-6" to 100'-0"	3'-0"	No
Column Line 11 wall	From I to Q	From 100'-0" to 117'-6"	2'-0"	Yes
Column Line 11 wall	From I to L	From 117'-6" to 153'-0"	2'-0"	Yes
Column Line 11 wall	From L to M	From 117'-6" to 135'-3"	4'-0"	Yes
Column Line 11 wall	From M to P	From 117'-6" to 135'-3"	2'-0"	Yes
Column Line 11 wall	From P to Q	From 117'-6" to 135'-3"	4'-0"	Yes
Column Line 11 wall	From L to Q	From 135'-3" to 153'-0"	2'-0"	Yes
Column Line I wall	From 1 to 11	From 66'-6" to 100'-0"	3'-0"	No
Column Line I wall	From 1 to 4	From 100'-0" to 180'-0"	2'-0"	Yes
Column Line I wall	From 4 to 7.3	From 100'-0" to 160'-6"	2'-0"	No

Table 3.3-1 (cont.)
Definition of Wall Thicknesses for Nuclear Island Buildings and Annex Building[1]

Wall or Section Description	Column Lines	Floor Elevation or Elevation Range	Concrete Thickness[2][3]	Applicable Radiation Shielding Wall (Yes/No)
Column Line I wall	From 7.3 to 11	From 100'-0" to 153'-0"	2'-0"	No
Column Line J-1 wall	From 1 to 2	From 82'-6" to 100'-0"	2'-0"	Yes
Column Line J-1 wall	From 2 to 4	From 66'-6" to 135'-3"	2'-6"	Yes
Column Line J-1 wall	From 2 to 4	From 135'-3" to 153'-0"	2'-0"	Yes
Column Line J-1 wall	From 4 to 5	From 66'-6" to 107'-2"	2'-0"	Yes
Column Line J-2 wall	From 2 to 4	From 66'-6" to 135'-3"	2'-6"	Yes
Column Line J-2 wall	From 4 to intersection with shield building wall	From 82'-6" to 107'-2"	2'-0"	Yes
Column Line K-2 wall	From 2 to 4	From 66'-6" to 135'-3"	4'-9"	Yes
Column Line L-2 wall	From 2 to 4	From 66'-6" to 135'-3"	4'-0"	Yes
Column Line N wall	From 1 to 2	From 66'-6" to 119'-9"	3'-0"	No
Column Line N wall	From 1 to 2	From 119'-9" to 135'-3"	3'-0"	Yes
Column Line N wall	From 2 to 4	From 66'-6" to 98'-0"	3'-0"	No
Column Line N wall	From 2 to 4	From 98'-0" to 135'-3"	5'-6"	Yes
Column Line N wall	From 1 to 4	From 135'-3" to 180'-0"	2'-0"	Yes
Column Line J wall	From 7.3 to 11	From 66'-6" to 117'-6"	2'-0"	No
Column Line K wall	From 7.3 to 11	From 60'-6" to 135'-3"	2'-0"	Yes
Column Line L wall	From shield building wall to 11	From 60'-6" to 153'-0"	2'-0"	Yes
Column Line M wall	From shield building wall to 11	From 66'-6" to 153'-0"	2'-0"	Yes
Column Line P wall	From shield building wall to 11	From 66'-6" to 153'-0"	2'-0"	Yes
Column Line Q wall	From shield building wall to 11	From 66'-6" to 100'-0"	3'-0"	No
Column Line Q wall	From shield building wall to 11	From 100'-0" to 153'-0"	2'-0"	Yes
Labyrinth Wall between Col. Line 3 and 4 and J-1 to J-2	Not Applicable	From 82'-6" to 100'-0"	2'-0"	Yes
N-S Shield Wall (low wall)	Between K-2 and L-2 extending from column line 1 north	From 100'-0" to 107'-2"	2'-6"	Yes
N-S Shield Wall	Between K-2 and L-2 extending from column line 1 north	From 100'-0" to 125'-0"	2'-3"	Yes

Table 3.3-1 (cont.)
Definition of Wall Thicknesses for Nuclear Island Buildings and Annex Building[1]

Wall or Section Description	Column Lines	Floor Elevation or Elevation Range	Concrete Thickness[2][3]	Applicable Radiation Shielding Wall (Yes/No)
E-W Shield Wall	Between 1 and 2 extending from column line N east	From 100'-0" to 125'-0"	2'-9"	Yes
Column Line 9.2 wall	From I to J and K to L	From 117'-6" to 135'-3"	2'-0"	Yes
Labyrinth Wall between Column Line 7.3 and 9.2 and J to K	Corner wall	From 117'-6" to 135'-3"	2'-0"	Yes
Auxiliary Area Basemat	From 1-11 and I-Q, excluding shield building	From 60'-6" to 66'-6"	6'-0"	No
Nuclear Island Basemat	Below shield building	From 60'-6" to containment vessel or 82'-6"	6'-0" to 22'-0" (varies)	No
Floor	From 1 to 2 and I to N	82'-6"	2'-0"	Yes
Floor	From 2 to 5 and J-1 to J-2	82'-6"	0'-9"	Yes
Pipe Chase Floor	From 2 to 5 and J-1 to J-2	92'-6"	2'-0"	Yes
Floor	From 2 to 3 and J-2 to K-2	92'-6"	3'-0"	Yes
Floor	From 3 to 4 and J-2 to K-2	92'-6"	2'-0"	Yes
Floor	From 4 to 7.3 and I to J-1	82'-6"	2'-0"	Yes
Floor	From 1 to 2 and I to N	100'-0"	3'-0"	Yes
Floor	From 2 to 4 and K-2 to L-2	94'-3"	4'-9"	Yes
Floor	From I to J-2 and 4 to intersecting vertical wall before column line 5	107'-2"	2'-0"	Yes
Floor	From I to shield building wall and from intersecting vertical wall before column line 5 to column line 5	107'-2"	0'-9"	Yes
Floor	From 5 to 7.3 and I to shield building wall	100'-0"	2'-0"	Yes
Floor	From K to L and shield building wall to column line 10	100'-0"	0'-9"	Yes
Floor	From 1 to 1.6 and L-2 to N	125'-0"	3'-0"	Yes
Floor	From 1.6 to 2 and L-2 to N	117'-6"	2'-0"	Yes
Main Control Room Floor	From 9.2 to 11 and I to L	117'-6"	2'-0"	Yes
Floor	Bounded by shield bldg, 7.3, J, 9.2 and L	117'-6"	2'-0"	Yes
Floor	From 9.2 to 11 and L to Q	117'-6"	2'-0"	Yes

Table 3.3-1 (cont.)
Definition of Wall Thicknesses for Nuclear Island Buildings and Annex Building(1)

Wall or Section Description	Column Lines	Floor Elevation or Elevation Range	Concrete Thickness(2)(3)	Applicable Radiation Shielding Wall (Yes/No)
Floor	From 3 to 4 and J-2 to K-2	117'-6"	2'-0"	Yes
Floor	From 2 to 4 and I to J-1	153'-0"	1'-3"	Yes
Floor	From 1 to 4 and I to N	180'-0"	1'-3"	Yes
Floor	From 4 to short of column line 5 and from I to intersection with shield building wall	135'-5"	0'-9"	Yes
Floor	From short of column line 5 to column line 5 and from I to intersection with shield building wall	133'-0"	0'-9"	Yes
Floor	From 5 to 7.3 and from I to intersection with shield building wall	135'-3"	0'-9"	Yes
Annex Building				
Column line 2 wall	From E to H	From 107'-2" to 135'-3"	19 3/4"	Yes
Column line 4 wall	From E to H	From 107'-2" to 162'-6" & 166'-0"	2'-0"	Yes
N-S Shield Wall between E and F	From 2 to 4	From 107'-2" to 135'-3"	1'-0"	Yes
Column line 4.1 wall	From E to H	From 107'-2" to 135'-3"	2'-0"	Yes
E-W Labyrinth Wall between column line 7.1 and 7.8 and G to H	Not Applicable	From 100'-0" to 112'-0"	2'-0"	
N-S Labyrinth Wall between column line 7.8 and 9 and G to H	Not Applicable	From 100'-0" to 112'-0"	2'-0"	
E-W Labyrinth Wall between column line 7.1 and 7.8 and G to H	Not Applicable	From 100'-0" to 112'-0"	2'-0"	Yes
N-S Labyrinth Wall between column line 7.8 and 9 and G to H	Not Applicable	From 100'-0" to 112'-0"	2'-0"	Yes
N-S Shield Wall on Column line. F	From 4.1 North	From 100'-0" to 117'-6"	1'-0"	Yes
Column Line 9 wall	From E to connecting wall between G and H	From 107'-2" to 117'-6"	2'-0"	Yes
Column Line E wall	From 9 to 13	From 100'-0" to 135'-3"	2'-0"	Yes
Column Line 13 wall	From E to I.1	From 100'-0" to 135'-3"	2'-0"	Yes
Column Line I.1 wall	From 11.09 to 13	From 100'-0" to 135'-3"	2'-0"	Yes
Corridor Wall between G and H	From 9 to 13	From 100'-0" to 135'-3"	1'-6"	Yes

Table 3.3-1 (cont.)
Definition of Wall Thicknesses for Nuclear Island Buildings and Annex Building[1]

Wall or Section Description	Column Lines	Floor Elevation or Elevation Range	Concrete Thickness[2][3]	Applicable Radiation Shielding Wall (Yes/No)
Column Line 9 wall	From I to H	From 117'-6" to 158'-0"	2'-0"	Yes
Floor	2 to 4 from shield wall between E and F to column line H	135'-3"	0'-6"	Yes
Floor	From 4 to 4.1 and E to H	135'-3"	1'-0"	Yes
Floor	From 9 to 13 and E to I.1	117'-6"	0'-6"	Yes
Floor	From 9 to 13 and E to I.1	135'-3"	0'-8"	Yes
Containment Filtration Rm A (North Wall)	Between column line E to H	From 135'-3" to 158'-0"	1'-0"	Yes
Containment Filtration Rm A (East wall)	Between column line E to F	From 135'-3" to 158'-0"	1'-0"	Yes
Containment Filtration Rm A (West wall)	Between column line G to H	From 135'-3" to 158'-0"	1'-0"	Yes
Containment Filtration Rm A (Floor)	Between column line E to H	135'-3"	1'-0"	Yes
Containment Filtration Rm B (Floor)	Between column line E to H	146'-3"	0'-6"	Yes
Containment Filtration Rm B (West wall)	Between column line G to H	From 146'-3" to 158'-0"	1'-0"	Yes
North wall (Room 50351)	N/A	100'-0" to top of wall	1'-4"	Yes
East Wall (Room 50351)	DR from 2R past 3R	100'-0" to top of wall	1'-4"	Yes
West wall (Room 50351)	DR from 2R past 3R	100'-0" to top of wall	1'-4"	Yes
East wall (Room 50352)	FR from 1R to 2R	100'-0" to top of wall	2'-0"	Yes
South wall (Room 50352)	1R from FR to DR	100'-0" to top of wall	2'-0"	Yes
West Wall (Room 50352)	DR from 1R to 2R	100'-0" to top of wall	2'-0"	Yes

Table 3.3-2
Nuclear Island Building Room Boundaries
Required to Have Flood Barrier Floors and Walls

Boundary/ Maximum Flood Level (inches)	Between Room Number to Room Number	
	Room with Postulated Flooding Source	Adjacent Room
Floor/36	12306	12211
Floor/3	12303	12203/12207
Floor/3	12313	12203/12207
Floor/1	12300	12201/12202/12207 12203/12204/12205
Floor/3	12312	12212
Wall/36	12306	12305
Floor/1	12401	12301/12302/12303 12312/12313
Wall/1	12401	12411/12412
Floor/36	12404	12304
Floor/4	12405	12305
Floor/36	12406	12306
Wall/36	12404	12401
Wall/1	12421	12452
Floor/3	12501	12401/12411/12412
Floor/3	12555	12421/12423/12422
Wall/36	12156/12158	12111/12112

Table 3.3-3 Class 1E Divisions in Nuclear Island Fire Areas				
	Class 1E Divisions			
Fire Area Number	A	C	B	D
1200 AF 01	Yes	Yes	-	-
1200 AF 03	-	-	Yes	Yes
1201 AF 02	-	-	Yes	-
1201 AF 03	-	-	-	Yes
1201 AF 04	-	-	Yes	Yes
1201 AF 05	-	-	Yes	Yes
1201 AF 06	-	-	Yes	Yes
1202 AF 03	-	Yes	-	-
1202 AF 04	Yes	-	-	-
1204 AF 01	Yes	-	-	-
1220 AF 01	-	-	Yes	Yes
1220 AF 02	-	-	-	Yes
1230 AF 01	Yes	Yes	-	-
1230 AF 02	-	-	Yes	Yes
1240 AF 01	Yes	Yes	-	-
1242 AF 02	Yes		-	

Note: Dash (-) indicates not applicable.

Table 3.3-4
Nuclear Island Rooms with Postulated High Energy Line Breaks/Essential Targets/Pipe Whip Restraints and Related Hazard Source

Room Number	Room Description	Essential Target Description	Hazard Source
11201	Steam Generator Compartment-01	Automatic depressurization system (ADS) Stage 4 valves (RCS-V004A, RCS-V004C, RCS-V014A, and RCS-V014C	1) Reactor Coolant System (RCS)-Pressurizer Spray Line, 4" L110A: Terminal End Break at RCS Cold Leg 1A 2) RCS-Pressurizer Spray Line, 4" L106: Terminal End Break at RCS Cold Leg 1B
11209	Pipe Chase to CVS Equipment Room	CVS makeup, CVS letdown, CVS hydrogen supply, and SGS steam generator blowdown piping	1) Steam Generator System (SGS)-Blowdown Line, 4" L009A: Terminal End Break at Containment Penetration P27 2) SGS-Blowdown Line, 4" L009B: Terminal End Break at Containment Penetration P28 3) CVS-Makeup Line, 3" L056: Terminal End Break at In-Line Anchor
11303	Lower Pressurizer Compartment	SGS steam generator blowdown and steam generator drain piping. RCS pressurizer pressure and level instrumentation; pressurizer support steel	1) RCS-CVS Purification Line, 3" L112: Intermediate Break at Outlet to Valve CVS-V082
11400	Maintenance Floor Mezzanine	Steam generator supports	1) SGS-Startup Feedwater Line, 6" L005B: Terminal End Break at Containment Penetration P45
11401	Steam Generator 01 Compartment	ADS Stage 4 valves (RCS-V004A, RCS-V004C, RCS-V014A, and RCS-V014C)	1) RCS Pressurizer Spray Line, 4" L106: Terminal End Break at In-Line Anchor
11403	Pressurizer Spray Valve Room	ADS Stage 4 valves (RCS-V004A, RCS-V004C, RCS-V014A, and RCS-V014C)	1) RCS Pressurizer Spray Line, 4" L213: Intermediate Break at 4x2 Tee Connection to Auxiliary Spray Line 2) RCS CVS Letdown Line, 3" L111: Intermediate Break at Inlet to Valve CVS-V001

Table 3.3-4 (cont.)
Nuclear Island Rooms with Postulated High Energy Line Breaks/Essential Targets/Pipe Whip Restraints and Related Hazard Source

Room Number	Room Description	Essential Target Description	Hazard Source
11503	Upper Pressurizer Compartment	ADS Stage 1, 2, and 3 valves, lower tier platform support steel	1) RCS-Pressurizer Spray Line, 4" L215: Terminal End Break at Pressurizer Nozzle
11601	Steam Generator-01 Feed Water Nozzle Area	RCS head vent piping SGS level instrumentation piping	1) SGS-Startup Feedwater Line, 6" L005A: Terminal End Break at Steam Generator Loop 1 Nozzle 2) SGS-Main Feedwater Line, 16" L003A: Terminal End Break at Steam Generator Loop 1 Nozzle
11602	Steam Generator-02 Feedwater Nozzle Area	SGS level instrumentation piping	1) SGS-Main Feedwater line, 16" L003B: Terminal End Break at Steam Generator Loop 2 Nozzle
11603	Lower ADS Valve Area	ADS Stage 2 and 3 valves (RCS-V002B, RCS-V003B, RCS-V012B, and RCS-V013B) Raceways and cable for Divisions A/C and B/D	1) RCS-Automatic Depressurization System Stage 1 Line, 4" L010B: Terminal End Break at Inlet to Valve RCS V011B
11703	Upper ADS Valve Area	ADS Stage 2 and 3 valves (RCS-V002A, RCS-V003A, RCS-V012A, and RCS-V013A) Raceways and cables for Division A/C	1) RCS-Automatic Depressurization System Stage 1 Line, 4" L010A: Terminal End Break at Inlet to Valve RCS V011A
12244	Lower Annulus Valve Area	CVS Makeup valve – CVS-V090	1) CVS-Makeup Line, 3" L131: Terminal End at In-Line Anchor

Table 3.3-5
Key Dimensions of Nuclear Island Building Features

Key Dimension	Reference Dimension (Figure 3.3-14)	Nominal Dimension	Tolerance
Distance between Outside Surface of walls at Column Line I & N when Measured at Column Line 1	X1	91 ft-0 in	+3 ft -1 ft
Distance from Outside Surface of wall at Column Line 1 to Column Line 7 when Measured at Column Line I	X2	138 ft-0 in	+3 ft -1 ft
Distance from Outside Surface of wall at Column Line 11 to Column Line 7 when Measured at Column Line I	X3	118 ft-0 in	+3 ft -1 ft
Distance between Outside Surface of walls at Column Line I & Q when Measured at Column Line 11	X4	117 ft-6 in	+3 ft -1 ft
Distance from Outside Surface of wall at Column Line Q to Column Line N when Measured at Column Line 11	X5	29 ft-0 in	+3 ft -1 ft
Distance between Outside Surface of shield building wall to shield building centerline when Measured on West Edge of Shield Building	X6	72 ft-6 in	+3 ft -1 ft
Distance between shield building centerline to Reactor Vessel centerline when Measured along Column Line N in North-South Direction	X7	7 ft-6 in	± 3 in
Distance from Bottom of Containment Sump to Top Surface of Embedded Containment Shell	-	2 ft-8 in	± 3 in
Distance from top of Basemat to Design Plant Grade	-	33 ft-6 in	± 1 ft
Distance of Design Plant Grade (Floor elevation 100'-0") relative to Site Grade	-	0 ft	± 3 ft-6 in
Distance from Design Plant Grade to Top Surface of Shield Building Roof	-	208 ft-6 in	± 1 ft

Table 3.3-6
Inspections, Tests, Analyses, and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. The physical arrangement of the nuclear island structures and the annex building is as described in the Design Description of this Section 3.3 and Figures 3.3-1 through 3.3-14. The physical arrangement of the radwaste building, the turbine building, and the diesel generator building is as described in the Design Description of this Section 3.3.	An inspection of the nuclear island structures, the annex building, the radwaste building, the turbine building, and the diesel generator building will be performed.	The as-built nuclear island structures, the annex building, the radwaste building, the turbine building, and the diesel generator building conform with the physical arrangement as described in the Design Description of this Section 3.3 and Figures 3.3-1 through 3.3-14.
2.a) The nuclear island structures, including the critical sections listed in Table 3.3-7, are seismic Category I and are designed and constructed to withstand design basis loads as specified in the Design Description, without loss of structural integrity and the safety-related functions.	i) An inspection of the nuclear island structures will be performed. Deviations from the design due to as-built conditions will be analyzed for the design basis loads.	i) A report exists which reconciles deviations during construction and concludes that the as-built nuclear island structures, including the critical sections, conform to the approved design and will withstand the design basis loads specified in the Design Description without loss of structural integrity or the safety-related functions.
	ii) An inspection of the as-built concrete thickness will be performed.	ii) A report exists that concludes that the as-built concrete thicknesses conform with the building sections defined on Table 3.3-1.
2.b) Site grade level is located relative to floor elevation 100'-0" per Table 3.3-5.	Inspection of the as-built site grade will be conducted.	Site grade is consistent with design plant grade within the dimension defined on Table 3.3-5.
2.c) The containment and its penetrations are designed and constructed to ASME Code Section III, Class MC.[1]	See Tier 1 Material, Subsection 2.2.1, Containment System.	See Tier 1 Material, Subsection 2.2.1, Containment System.

1. Containment isolation devices are addressed in subsection 2.2.1, Containment System.

Table 3.3-6 (cont.) Inspections, Tests, Analyses, and Acceptance Criteria		
Design Commitment	**Inspections, Tests, Analyses**	**Acceptance Criteria**
2.d) The containment and its penetrations retain their pressure boundary integrity associated with the design pressure.	See Tier 1 Material, Subsection 2.2.1, Containment System.	See Tier 1 Material, Subsection 2.2.1, Containment System.
2.e) The containment and its penetrations maintain the containment leakage rate less than the maximum allowable leakage rate associated with the peak containment pressure for the design basis accident.	See Tier 1 Material, Subsection 2.2.1, Containment System.	See Tier 1 Material, Subsection 2.2.1, Containment System.
2.f) The key dimensions of nuclear island structures are defined on Table 3.3-5.	An inspection will be performed of the as-built configuration of the nuclear island structures.	A report exists and concludes that the key dimensions of the as-built nuclear island structures are consistent with the dimensions defined on Table 3.3-5.
2.g) The containment vessel greater than 7 feet above the operating deck provides a heat transfer surface. A free volume exists inside the containment shell above the operating deck.	The maximum containment vessel inside height from the operating deck is measured and the inner radius below the spring line is measured at two orthogonal radial directions at one elevation.	The containment vessel maximum inside height from the operating deck is 121'-1" (with tolerance of +12", -6"), and the inside diameter is 130 feet nominal (with tolerance of +12", -6").
3. Walls and floors of the nuclear island structures as defined on Table 3.3-1 except for designed openings or penetrations provide shielding during normal operations.	Inspection of the as-built nuclear island structures wall and floor thicknesses will be performed.	A report exists and concludes that the shield walls and floors of the nuclear island structures as defined on Table 3.3-1 except for designed openings or penetrations are consistent with the concrete wall thicknesses provided in Table 3.3-1.

Table 3.3-6 (cont.) Inspections, Tests, Analyses, and Acceptance Criteria		
Design Commitment	**Inspections, Tests, Analyses**	**Acceptance Criteria**
4. Walls and floors of the annex building as defined on Table 3.3-1 except for designed openings or penetrations provide shielding during normal operations.	Inspection of the as-built annex building wall and floor thicknesses will be performed.	A report exists and concludes that the shield walls and floors of the annex building as defined on Table 3.3-1 except for designed openings or penetrations are consistent with the minimum concrete wall thicknesses provided in Table 3.3-1.
5.a) Exterior walls and the basemat of the nuclear island have a water barrier up to site grade.	An inspection of the as-built exterior walls and the basemat of the nuclear island up to floor elevation 100'-0", for application of water barrier will be performed during construction before the walls are poured.	A report exists that confirms that a water barrier exists on the nuclear island exterior walls up to site grade.
5.b) The boundaries between rooms identified in Table 3.3-2 of the auxiliary building are designed to prevent flooding of rooms that contain safety-related equipment.	An inspection of the auxiliary building rooms will be performed.	A report exists that confirms floors and walls as identified on Table 3.3-2 have provisions to prevent flooding between rooms up to the maximum flood levels for each room defined in Table 3.3-2.
5.c) The boundaries between the following rooms, which contain safety-related equipment – PXS valve/accumulator room A (11205), PXS valve/accumulator room B (11207), and CVS room (11209) – are designed to prevent flooding between these rooms.	An inspection of the boundaries between the following rooms which contain safety-related equipment – PXS Valve/ Accumulator Room A (11205), PXS Valve/Accumulator Room B (11207), and CVS Room (11209) – will be performed.	A report exists that confirms that provisions to prevent flooding of other rooms to a maximum floor level of 108 feet are provided.
6.a) The available room volumes of the radiologically controlled area of the auxiliary building between floor elevations 66'-6" and 82'-6" exceed the volume of the liquid radwaste storage tanks (WLS-MT-05A, MT-05B, MT-06A, MT-06B, MT-07A, MT-07B, MT-07C, MT-11).	An inspection will be performed of the as-built radiologically controlled area of the auxiliary building between floor elevations 66'-6" and 82'-6" to define volume.	A report exists and concludes that the as-built available room volumes of the radiologically controlled area of the auxiliary building between floor elevations 66'-6" and 82'-6" exceed the volume of the liquid radwaste storage tanks (WLS-MT-05A, MT-05B, MT-06A, MT-06B, MT-07A, MT-07B, MT-07C, MT-11).

Table 3.3-6 (cont.) Inspections, Tests, Analyses, and Acceptance Criteria		
Design Commitment	**Inspections, Tests, Analyses**	**Acceptance Criteria**
6.b) The radwaste building package waste storage room has a volume greater than or equal to 1293 cubic feet.	An inspection of the radwaste building packaged waste storage room (50352) is performed.	The volume of the radwaste building packaged waste storage room (50352) is greater than or equal to 1293 cubic feet.
7.a) Class 1E electrical cables, fiber optic cables associated with only one division, and raceways are identified according to applicable color-coded Class 1E divisions.	Inspections of the as-built Class 1E cables and raceways will be conducted.	Class 1E electrical cables, fiber optic cables associated with only one division, and raceways are identified by the appropriate color code.
7.b) Class 1E divisional electrical cables and fiber optic cables associated with only one division are routed in their respective divisional raceways.	Inspections of the as-built Class 1E divisional cables and raceways will be conducted.	Class 1E electrical cables and fiber optic cables associated with only one division are routed in raceways assigned to the same division. There are no other safety division electrical cables in a raceway assigned to a different division.
7.c) Separation is maintained between Class 1E divisions in accordance with the fire areas as identified in Table 3.3-3.	i) Inspections of the as-built Class 1E division electrical cables, fiber optic cables associated with only one division, and raceways located in the fire areas identified in Table 3.3-3 will be conducted. ii) Inspections of the as-built fire barriers between the fire areas identified in Table 3.3-3 will be conducted.	i) Results of the inspection will confirm that the separation between Class 1E divisions is consistent with Table 3.3-3. ii) Results of the inspection will confirm that fire barriers exist between Class 1E divisions consistent with the fire areas identified in Table 3.3-3.

Table 3.3-6 (cont.) Inspections, Tests, Analyses, and Acceptance Criteria		
Design Commitment	**Inspections, Tests, Analyses**	**Acceptance Criteria**
7.d) Physical separation is maintained between Class 1E divisions and between Class 1E divisions and non-Class 1E cables.	Inspections of the as-built Class 1E raceways will be performed to confirm that the separation between Class 1E raceways of different divisions and between Class 1E raceways and non-Class 1E raceways is consistent with the following: – Within the main control room and remote shutdown area, the minimum vertical separation is 3 inches and the minimum horizontal separation is 1 inch. – Within other plant areas (limited hazard areas), the minimum separation is defined by one of the following: 1) The minimum vertical separation is 5 feet and the minimum horizontal separation is 3 feet. 2) The minimum vertical separation is 12 inches and the minimum horizontal separation is 6 inches for raceways containing only instrumentation and control and low-voltage power cables <2/0 AWG. 3) For configurations that involve exclusively limited energy content cables (instrumentation and control), the minimum vertical separation is 3 inches and the minimum horizontal separation is 1 inch.	Results of the inspection will confirm that the separation between Class 1E raceways of different divisions and between Class 1E raceways and non-Class 1E raceways is consistent with the followings: – Within the main control room and remote shutdown area, the vertical separation is 3 inches or more and the horizontal separation is 1 inch or more. – Within other plant areas (limited hazard areas), the separation meets one of the following: 1) The vertical separation is 5 feet or more and the horizontal separation is 3 feet or more except. 2) The minimum vertical separation is 12 inches and the minimum horizontal separation is 6 inches for raceways containing only instrumentation and control and low-voltage power cables <2/0 AWG. 3) For configurations that involve exclusively limited energy content cables (instrumentation and control), the minimum vertical separation is 3 inches and the minimum horizontal separation is 1 inch.

Table 3.3-6 (cont.) Inspections, Tests, Analyses, and Acceptance Criteria		
Design Commitment	**Inspections, Tests, Analyses**	**Acceptance Criteria**
	4) For configurations involving an enclosed raceway and an open raceway, the minimum vertical separation is 1 inch if the enclosed raceway is below the open raceway. 5) For configuration involving enclosed raceways, the minimum separation is 1 inch in both horizontal and vertical directions.	4) For configurations that involve an enclosed raceway and an open raceway, the minimum vertical separation is 1 inch if the enclosed raceway is below the raceway. 5) For configurations that involve enclosed raceways, the minimum vertical and horizontal separation is 1 inch.
	– Where minimum separation distances are not maintained, the circuits are run in enclosed raceways or barriers are provided.	– Where minimum separation distances are not met, the circuits are run in enclosed raceways or barriers are provided.
	– Separation distances less than those specified above and not run in enclosed raceways or provided with barriers are based on analysis.	– A report exists and concludes that separation distances less than those specified above and not provided with enclosed raceways or barriers have been analyzed.
	– Non-Class 1E wiring that is not separated from Class 1E or associated wiring by the minimum separation distance or by a barrier or analyzed is considered as associated circuits and subject to Class 1E requirements.	– Non-Class 1E wiring that is not separated from Class 1E or associated wiring by the minimum separation distance or by a barrier or analyzed is treated as Class 1E wiring.
7.e) Class 1E fiber optic cables which interconnect two divisions are routed and separated such that the Protection and Safety Monitoring System voting logic is not defeated by the loss of any single raceway or fire area.	Inspections of the as-built Class 1E fiber optic cables will be conducted.	Class 1E fiber optic cables which interconnect two divisions are routed and separated such that the Protection and Safety Monitoring System voting logic is not defeated by the loss of any single raceway or fire area.

Table 3.3-6 (cont.)
Inspections, Tests, Analyses, and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
8. Equipment labeled as essential targets in Table 3.3-4 and located in rooms identified in Table 3.3-4 are protected from the dynamic effects of postulated pipe breaks.	An inspection will be performed of the as-built high energy pipe break pipe whip restraints features for systems located in rooms identified in Table 3.3-4.	An as-built Pipe Rupture Hazard Analysis Report exists and concludes that equipment labeled as essential targets in Table 3.3-4 and located in rooms identified in Table 3.3-4 can withstand the effects of postulated pipe rupture without loss of required safety function.
9. The reactor cavity sump has a minimum concrete thickness as shown in Table 3.3-5 between the bottom of the sump and the steel containment.	An inspection of the as-built containment building internal structures will be performed.	A report exists and concludes that the reactor cavity sump has a minimum concrete thickness as shown on Table 3.3-5 between the bottom of the sump and the steel containment.
10. The shield building roof and PCS storage tank support and retain the PCS water sources. The PCS storage tank has a stainless steel liner which provides a barrier on the inside surfaces of the tank. Leak chase channels are provided on the tank boundary liner welds.	i) A test will be performed to measure the leakage from the PCS storage tank based on measuring the water flow out of the leak chase collection system. ii) An inspection of the PCS storage tank exterior tank boundary and shield building tension ring will be performed before and after filling of the PCS storage tank to the overflow level. The vertical elevation of the shield building roof will be measured at a location at the outer radius of the roof (tension ring) and at a location on the same azimuth at the outer radius of the PCS water storage tank before and after filling the PCS storage tank.	i) A report exists and concludes that total water flow from the leak chase collection system does not exceed 10 gal/hr. ii) A report exists and concludes that there is no visible water leakage from the PCS storage tank and that inspection and measurement of the structure before and after filling of the tank shows structural behavior under normal loads to be acceptable.

Table 3.3-6 (cont.)
Inspections, Tests, Analyses, and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
11. The construction approach for soft soil sites having unconsolidated deposits with shear wave velocities in the range from 1,000 to 2,000 feet per second includes two limits: i) Concrete will not be placed in the shield building or containment internal structure above floor elevation 84′-0″ before the walls of the auxiliary building are completed to floor elevation 82′-6″, and ii) Concrete will not be placed in the auxiliary building above floor elevation 117′-6″ before the shield building is completed to floor elevation 82′-6″.	Construction records for the nuclear island structures will be reviewed.	A review of the construction records concludes that: i) The walls of the auxiliary building were completed to floor elevation 82′-6″ prior to placement of concrete in the shield building above floor elevation 84′-0″ or containment internal structures above floor elevation 84′-0″. ii) The concrete was not placed in the auxiliary building above floor elevation 117′-6″ before the shield building was completed to floor elevation 82′-6″.
12. The extended turbine generator axis intersects the shield building.	An inspection of the as-built turbine generator will be performed.	The extended axis of the turbine generator intersects the shield building.
13. Separation is provided between the structural elements of the turbine, annex and radwaste buildings and the nuclear island structure. This separation permits horizontal motion of the buildings in the safe shutdown earthquake without impact between structural elements of the buildings.	An inspection of the separation of the nuclear island from the annex, radwaste and turbine building structures will be performed. The inspection will verify the specified horizontal clearance between structural elements of the adjacent buildings, consisting of the reinforced concrete walls and slabs, structural steel columns and floor beams.	The minimum horizontal clearance above floor elevation 100′-0″ between the structural elements of the annex and radwaste buildings and the nuclear island is 4 inches. The minimum horizontal clearance above floor elevation 100′-0″ between the structural elements of the turbine building and the nuclear island is 12 inches.
14. Protected Area/Vital Area walls that are accessible and unmonitored are security hardened.	An inspection of the as-built Protected Area/Vital Area walls that are accessible and unmonitored will be performed.	The as-built inspection report exists and concludes that the Protected Area/Vital Area walls that are accessible and unmonitored meet the requirements of being security hardened.

Table 3.3-7 Nuclear Island Critical Structural Sections
Containment Internal Structures South west wall of the refueling cavity South wall of the west steam generator cavity North east wall of the in-containment refueling water storage tank In-containment refueling water storage tank steel wall Column supporting the operating floor
Auxiliary and Shield Building South wall of auxiliary building (column line 1), elevation 66'-6" to elevation 180'-0" Interior wall of auxiliary building (column line 7.3), elevation 66'-6" to elevation 160'-6" West wall of main control room in auxiliary building (column line L), elevation 117'-6" to elevation 153'-0" North wall of MSIV east compartment (column line 11 between lines P and Q), elevation 117'-6" to elevation 153'-0" Shield building cylinder, elevation 160'-6" to elevation 200'-0" Roof slab at elevation 180'-0" adjacent to shield building cylinder Floor slab on metal decking at elevation 135'-3" 2'-0" slab in auxiliary building (tagging room ceiling) at elevation 135'-3" Finned floor in the main control room at elevation 135'-3" Shield building roof, exterior wall of the PCS water storage tank Shield building roof, tension ring and columns between air inlets, elevation 241'-0" to elevation 250'-0" Divider wall between the spent fuel pool and the fuel transfer canal
Nuclear Island Basemat Below Auxiliary Building Bay between reference column lines 9.1 and 11, and K and L Bay between reference column lines 1 and 2 and K-2 and N

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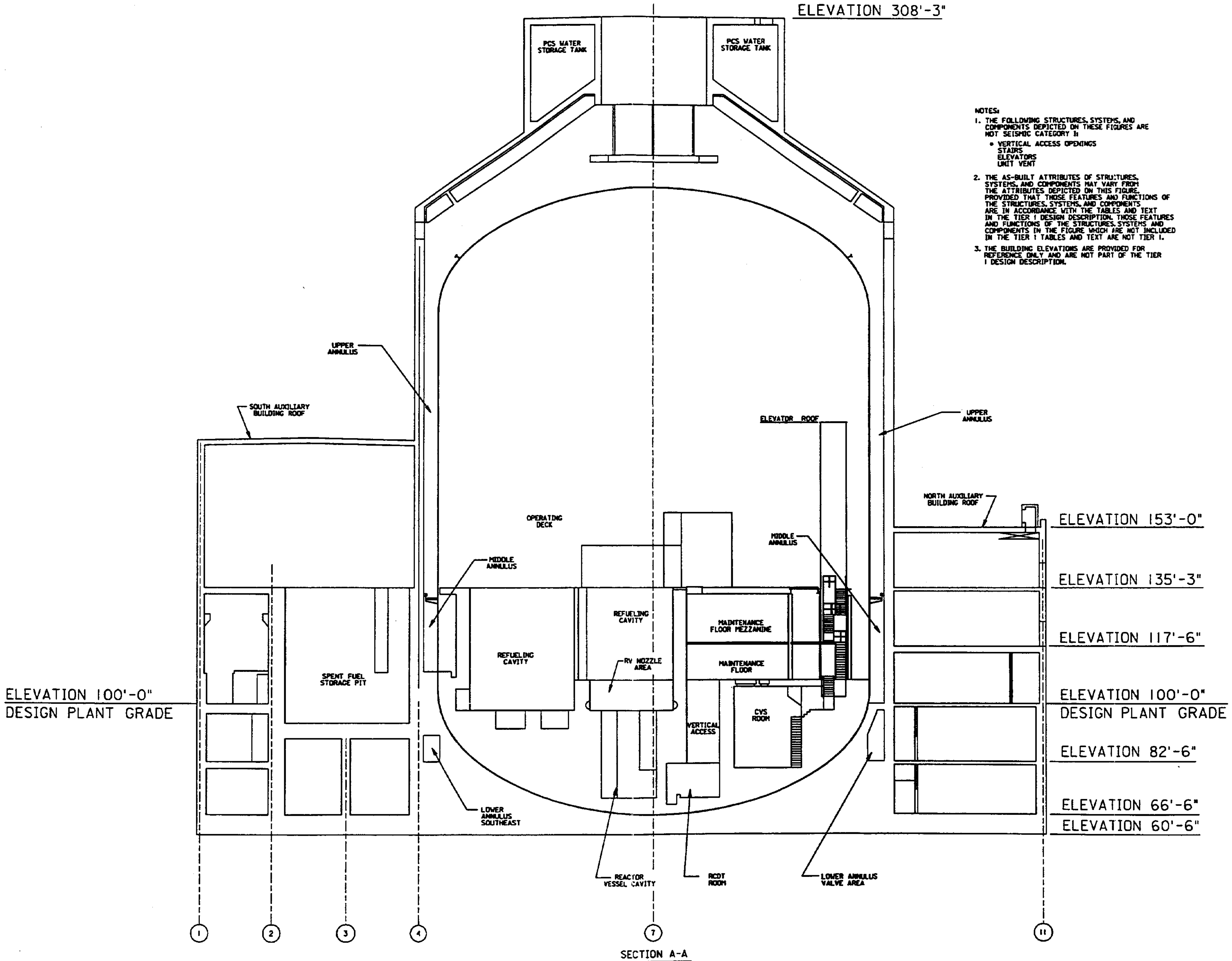


Figure 3.3-1
Nuclear Island Section A-A

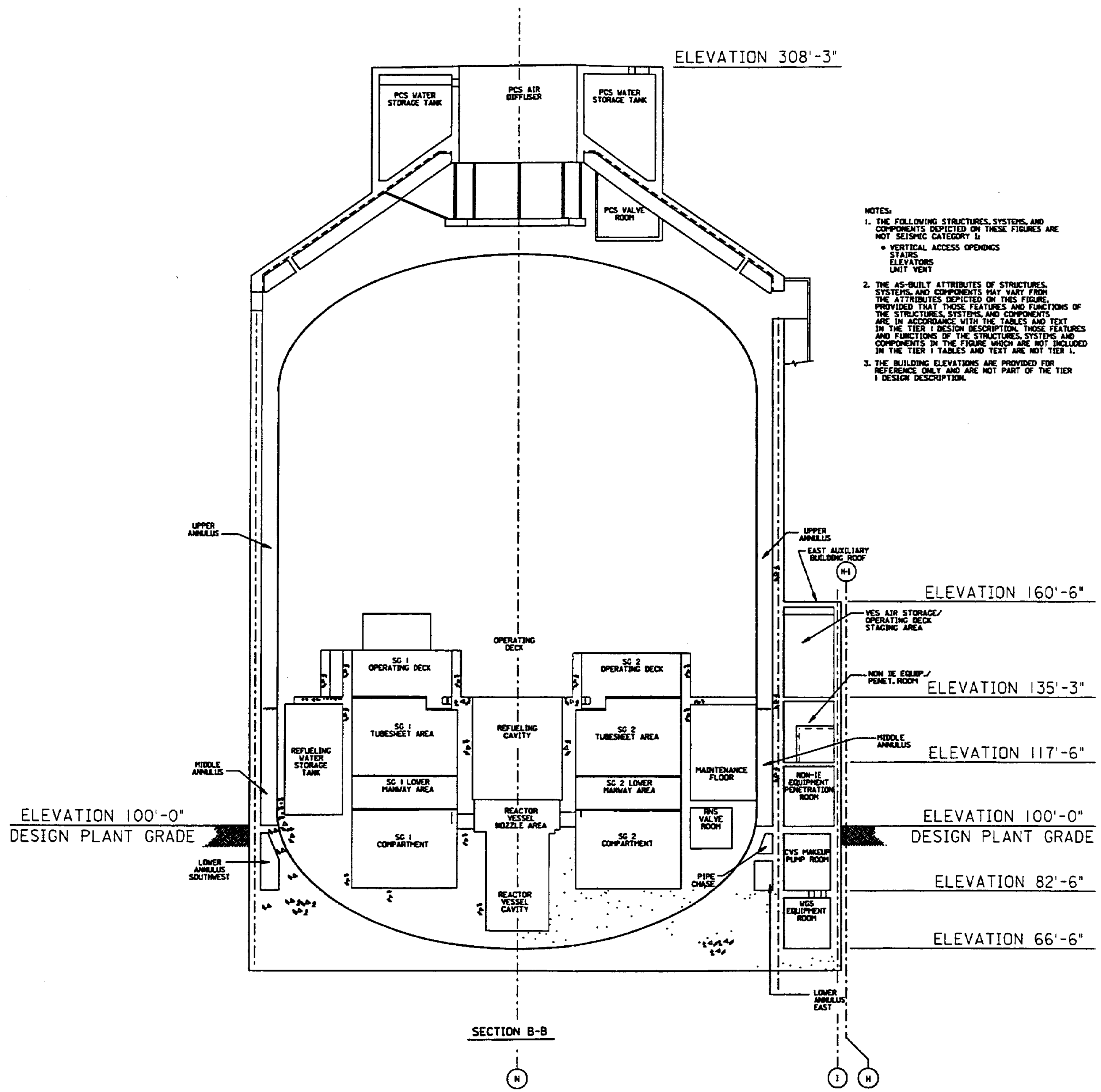
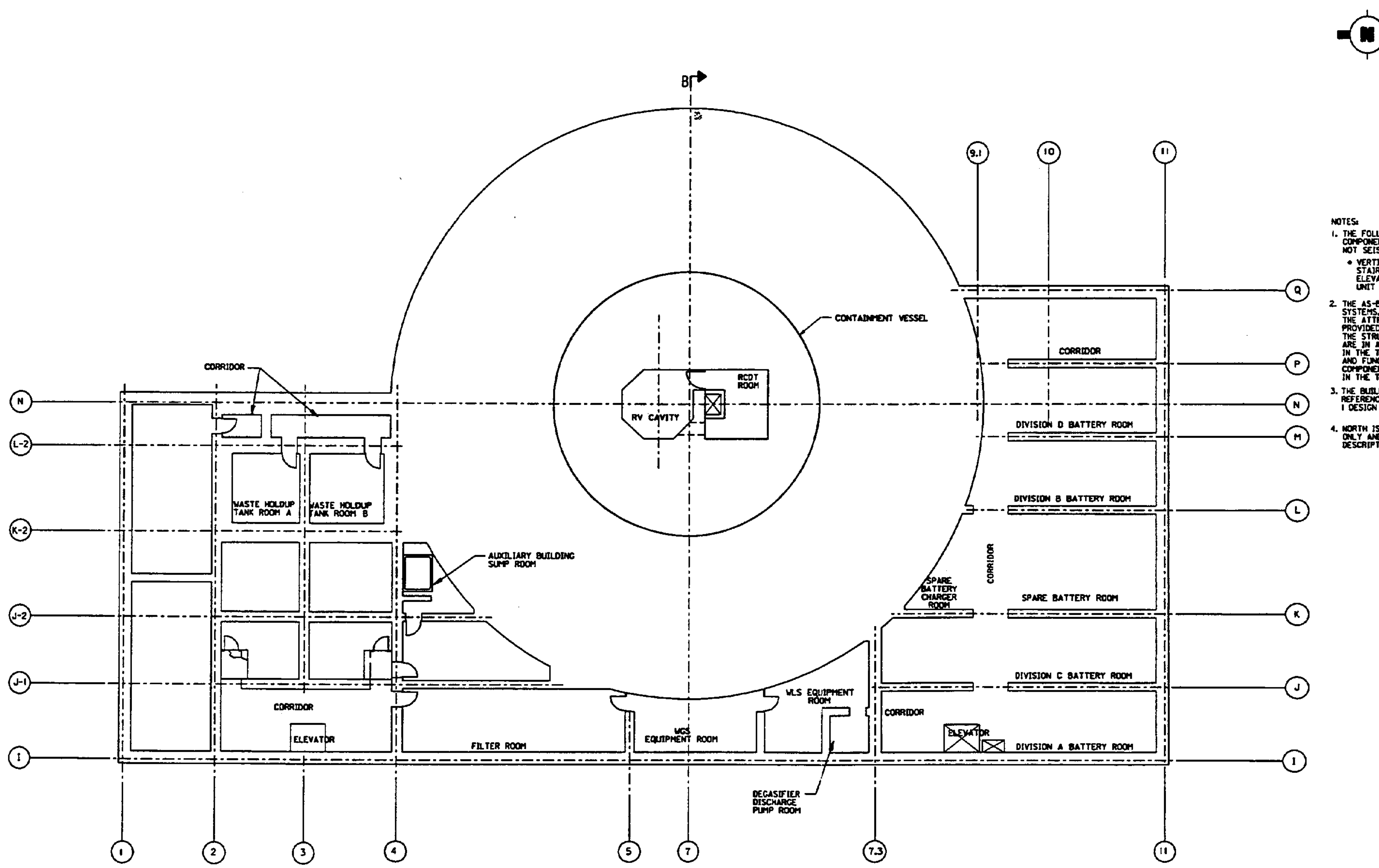


Figure 3.3-2
Nuclear Island Section B-B



NOTES:

1. THE FOLLOWING STRUCTURES, SYSTEMS, AND COMPONENTS DEPICTED ON THESE FIGURES ARE NOT SEISMIC CATEGORY I:
   - VERTICAL ACCESS OPENINGS
     STAIRS
     ELEVATORS
     UNIT VENT
2. THE AS-BUILT ATTRIBUTES OF STRUCTURES, SYSTEMS, AND COMPONENTS MAY VARY FROM THE ATTRIBUTES DEPICTED ON THIS FIGURE, PROVIDED THAT THOSE FEATURES AND FUNCTIONS OF THE STRUCTURES, SYSTEMS, AND COMPONENTS ARE IN ACCORDANCE WITH THE TABLES AND TEXT IN THE TIER 1 DESIGN DESCRIPTION. THOSE FEATURES AND FUNCTIONS OF THE STRUCTURES, SYSTEMS AND COMPONENTS IN THE FIGURE WHICH ARE NOT INCLUDED IN THE TIER 1 TABLES AND TEXT ARE NOT TIER 1.
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4. NORTH IS PROVIDED AS A REFERENCE DIRECTION ONLY AND IS NOT PART OF THE TIER 1 DESIGN DESCRIPTION.

Figure 3.3-3
Nuclear Island Plan View at Elevation 66'-6"

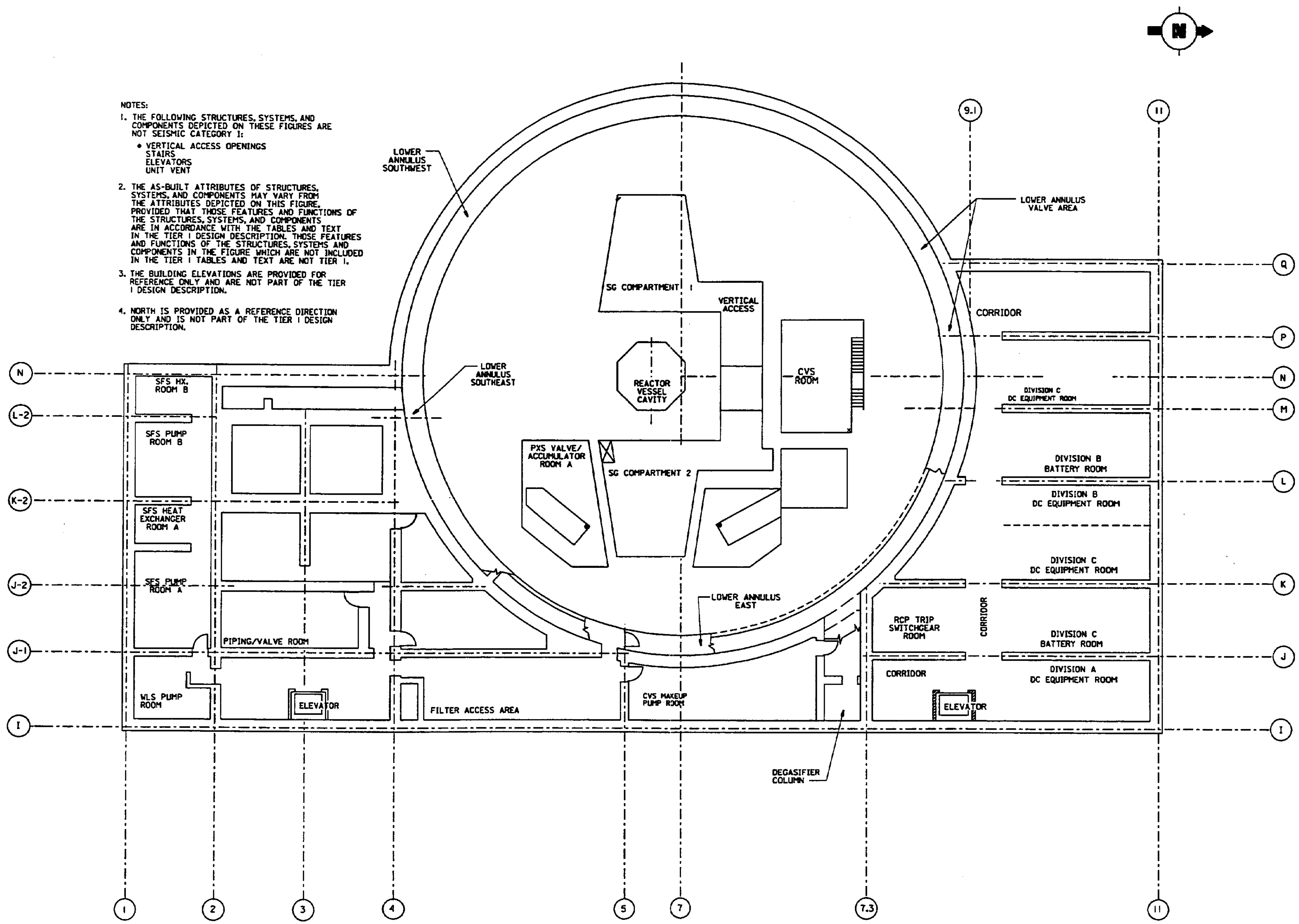
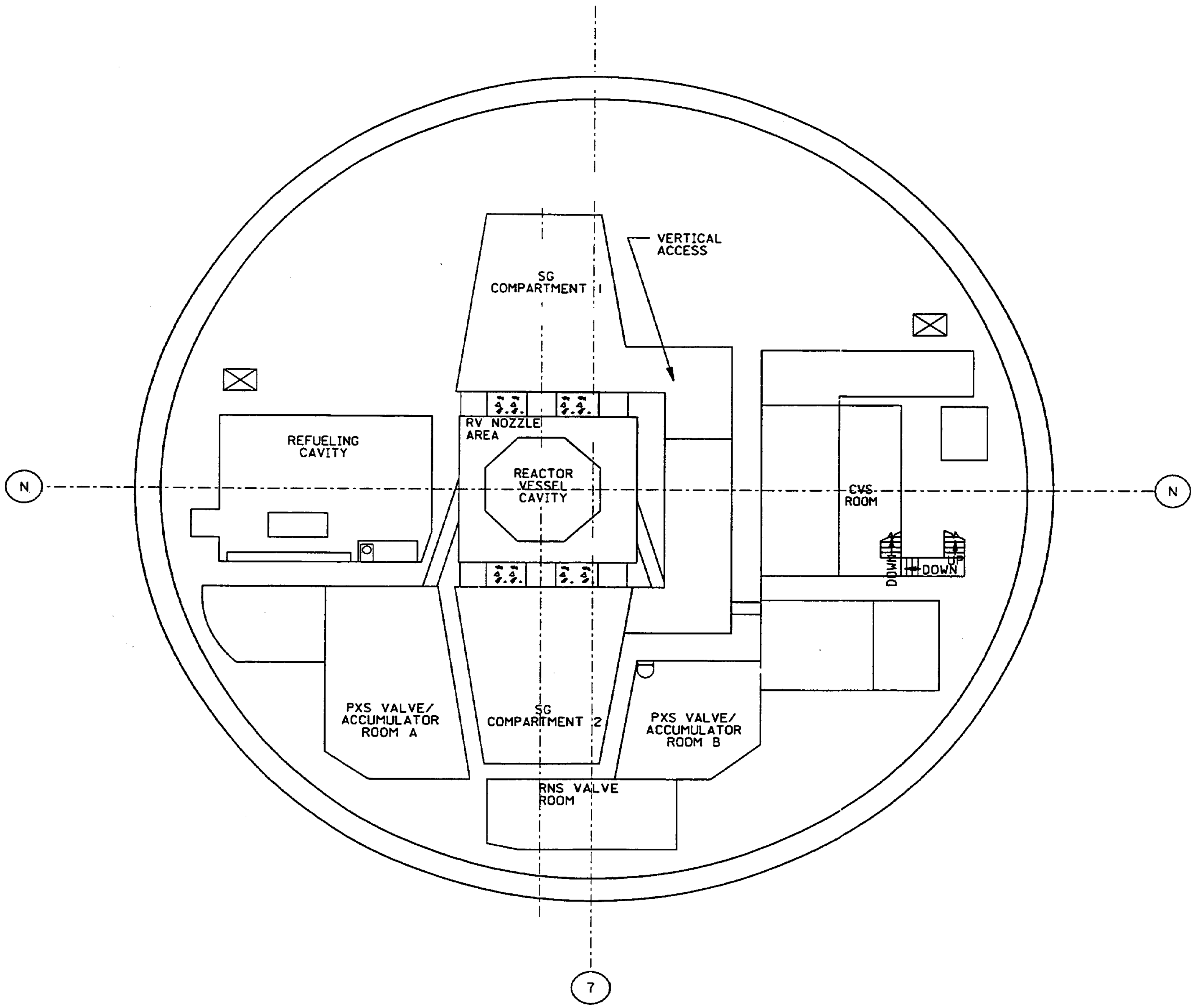


Figure 3.3-4
Nuclear Island Plan View at Elevation 82'-6"



NOTES:

1. THE FOLLOWING STRUCTURES, SYSTEMS, AND COMPONENTS DEPICTED ON THESE FIGURES ARE NOT SEISMIC CATEGORY I:
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     ELEVATORS
     UNIT VENT
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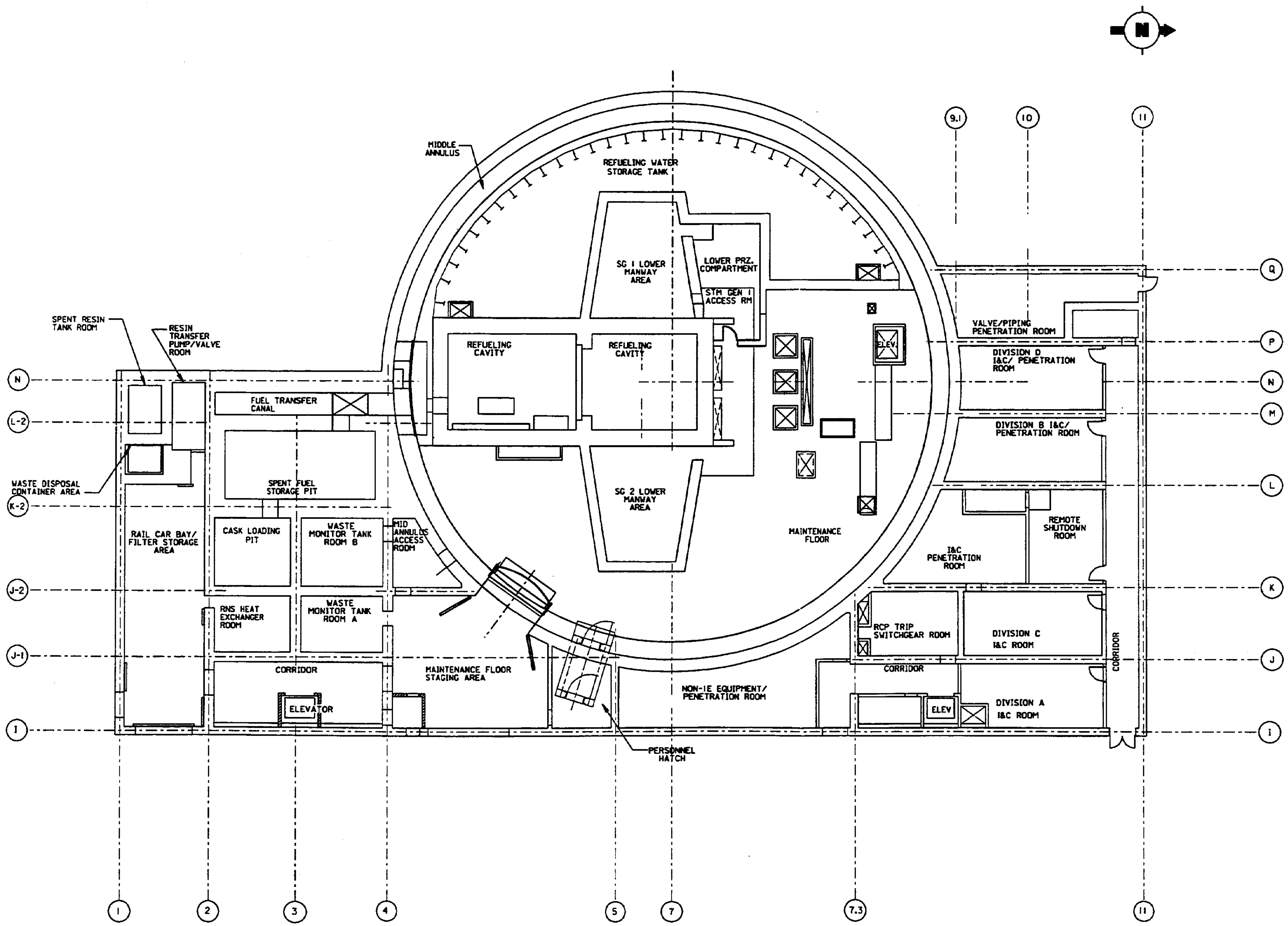
Figure 3.3-5
Nuclear Island Plan View at Elevation 96'-6"



NOTES:

1. THE FOLLOWING STRUCTURES, SYSTEMS, AND COMPONENTS DEPICTED ON THESE FIGURES ARE NOT SEISMIC CATEGORY I:
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     STAIRS
     ELEVATORS
     UNIT VENT
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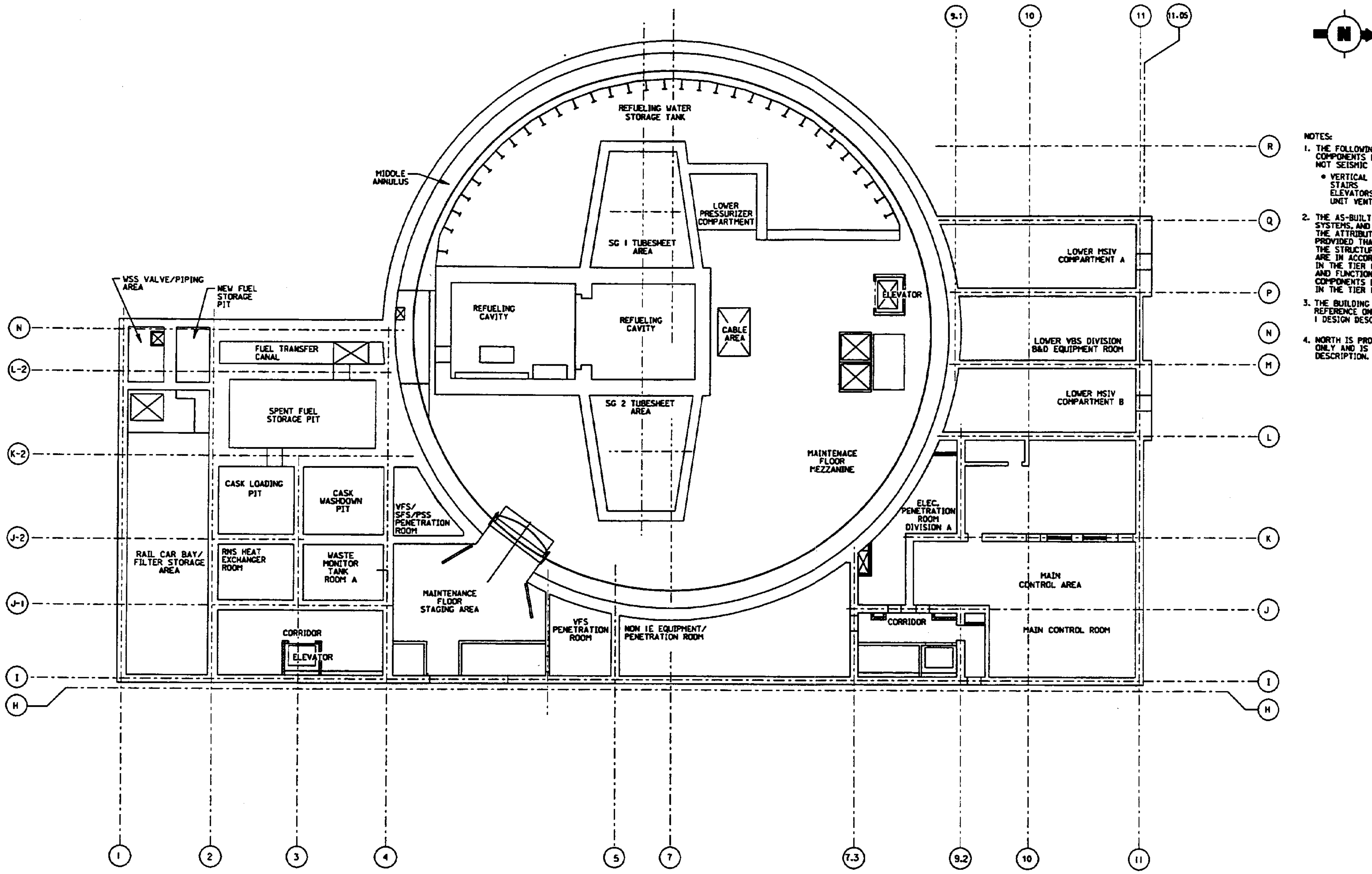
Figure 3.3-6
Nuclear Island Plan View at Elevation 100'-0"



NOTES:

1. THE FOLLOWING STRUCTURES, SYSTEMS, AND COMPONENTS DEPICTED ON THESE FIGURES ARE NOT SEISMIC CATEGORY I:
   - VERTICAL ACCESS OPENINGS
     STAIRS
     ELEVATORS
     UNIT VENT
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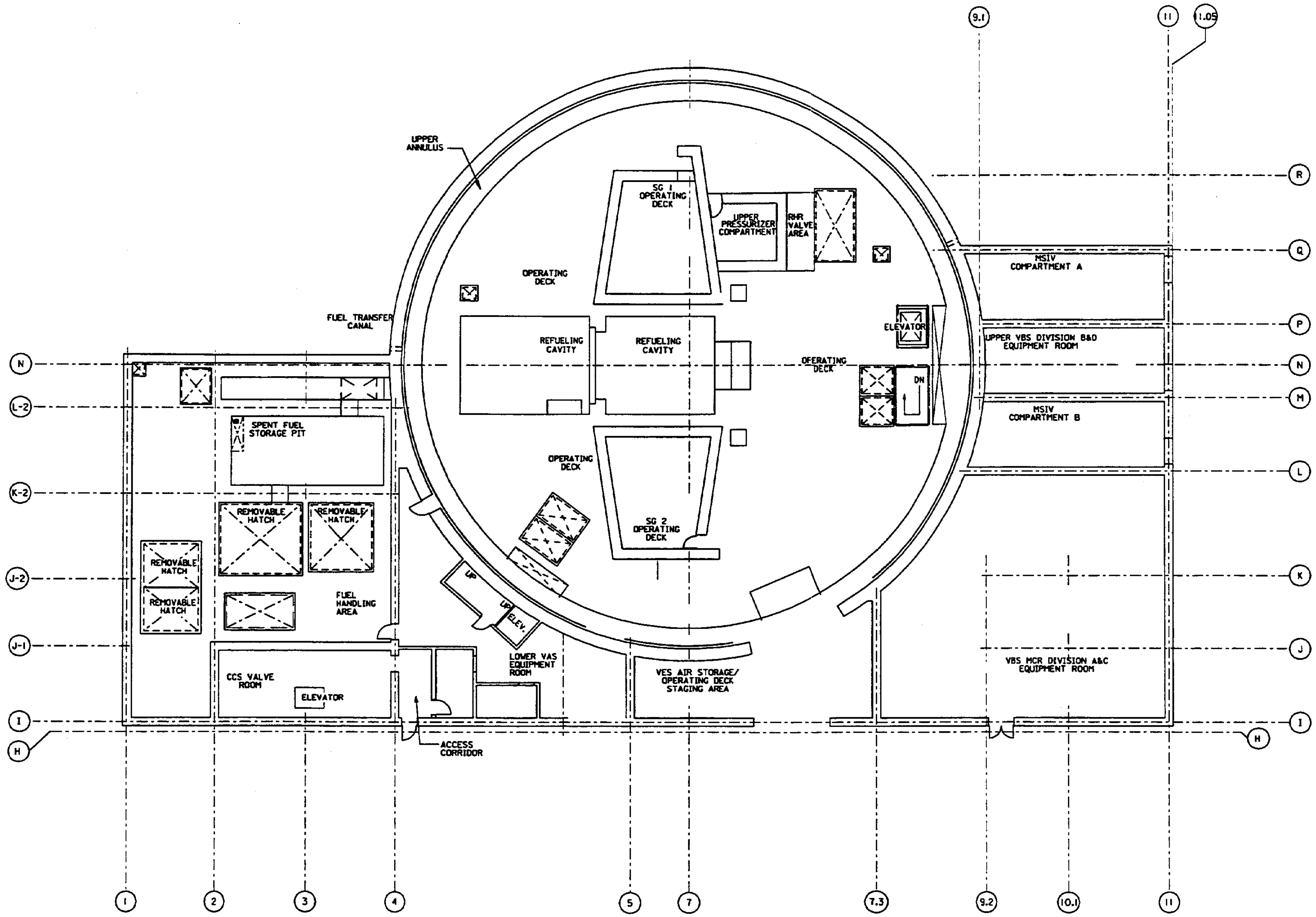
Figure 3.3-7
Nuclear Island Plan View at Elevation 117'-6"



NOTES:

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     STAIRS
     ELEVATORS
     UNIT VENT
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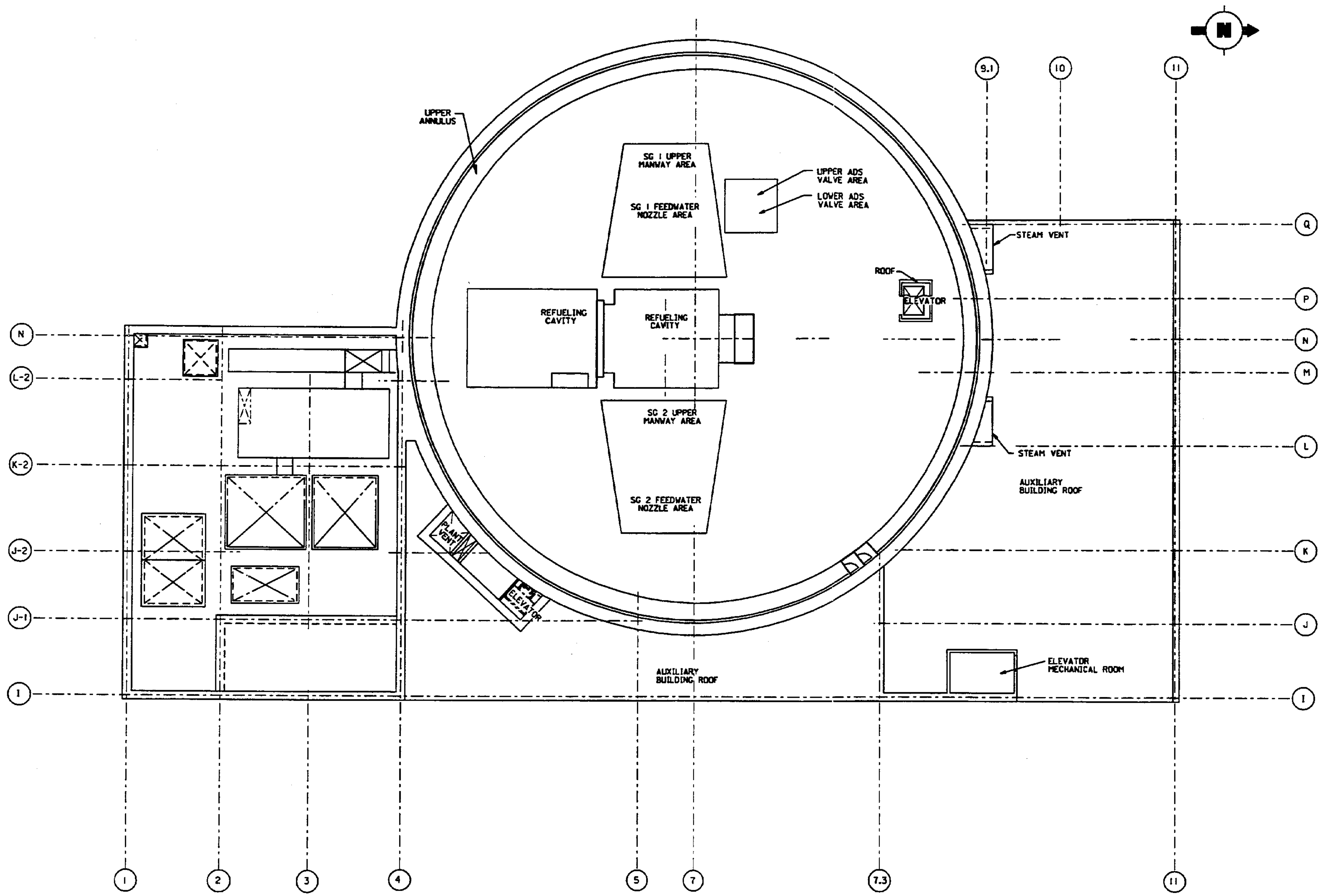
Figure 3.3-8
Nuclear Island Plan View at Elevation 135'-3"



NOTES:

1. THE FOLLOWING STRUCTURES, SYSTEMS, AND COMPONENTS DEPICTED ON THESE FIGURES ARE NOT SEISMIC CATEGORY I:
   - VERTICAL ACCESS OPENINGS
     STAIRS
     ELEVATORS
     UNIT VENT
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Figure 3.3-9
Nuclear Island Plan View at Elevation 153'-03" and 160'-6"

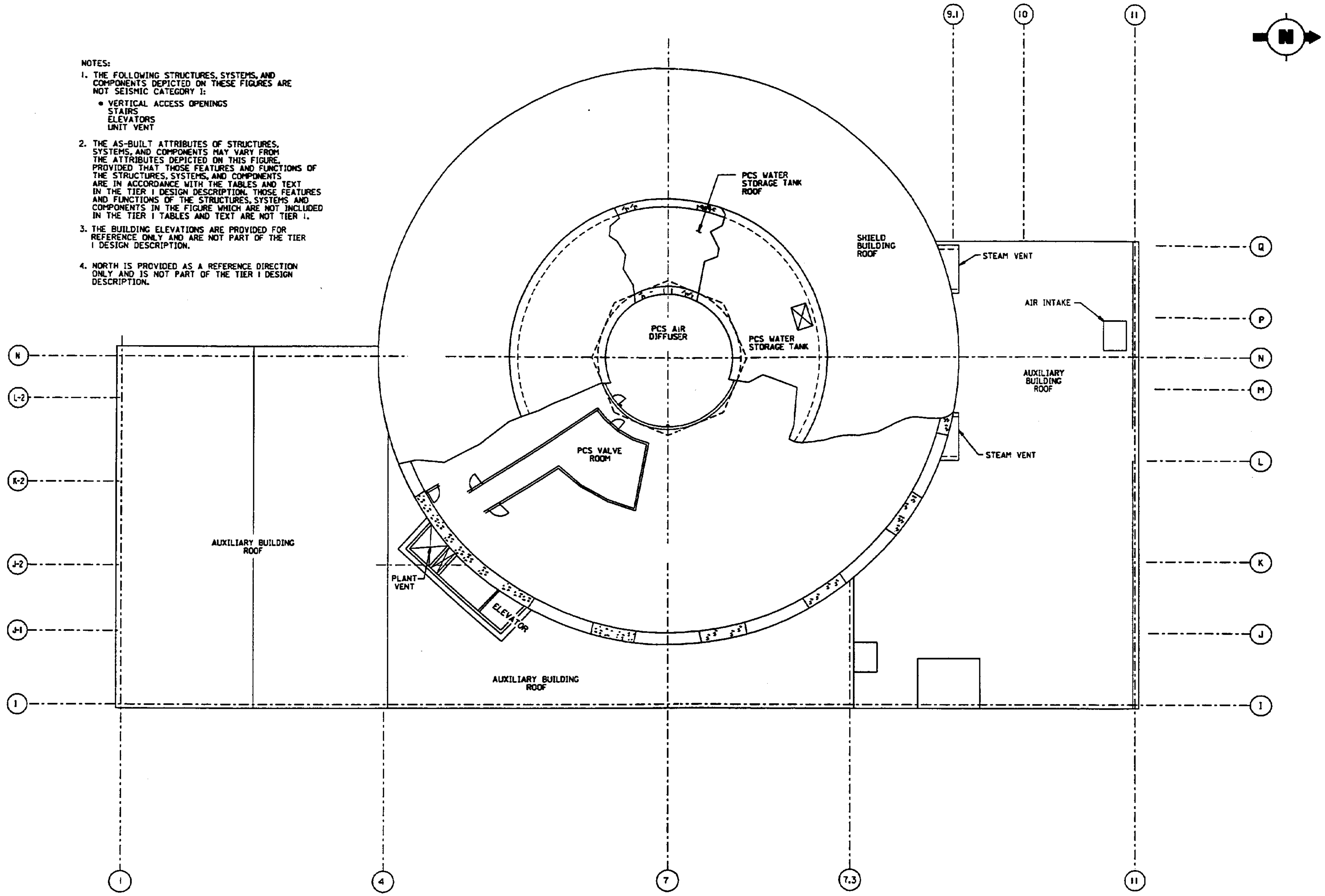
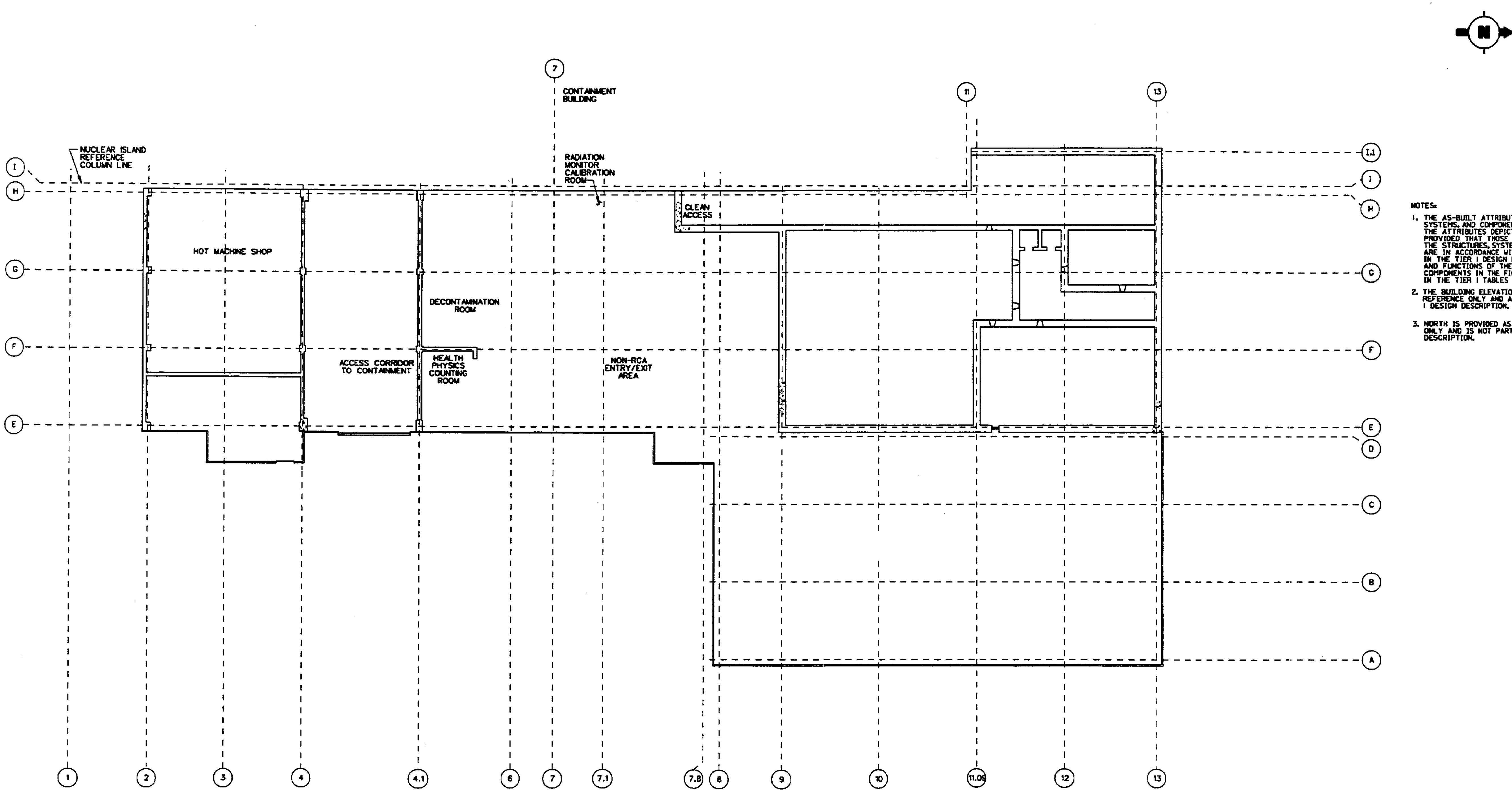


Figure 3.3-10
Nuclear Island Plan View at Shield Building Roof



NOTES:

1. THE AS-BUILT ATTRIBUTES OF STRUCTURES, SYSTEMS, AND COMPONENTS MAY VARY FROM THE ATTRIBUTES DEPICTED ON THIS FIGURE, PROVIDED THAT THOSE FEATURES AND FUNCTIONS OF THE STRUCTURES, SYSTEMS, AND COMPONENTS ARE IN ACCORDANCE WITH THE TABLES AND TEXT IN THE TIER 1 DESIGN DESCRIPTION. THOSE FEATURES AND FUNCTIONS OF THE STRUCTURES, SYSTEMS AND COMPONENTS IN THE FIGURE WHICH ARE NOT INCLUDED IN THE TIER 1 TABLES AND TEXT ARE NOT TIER 1.

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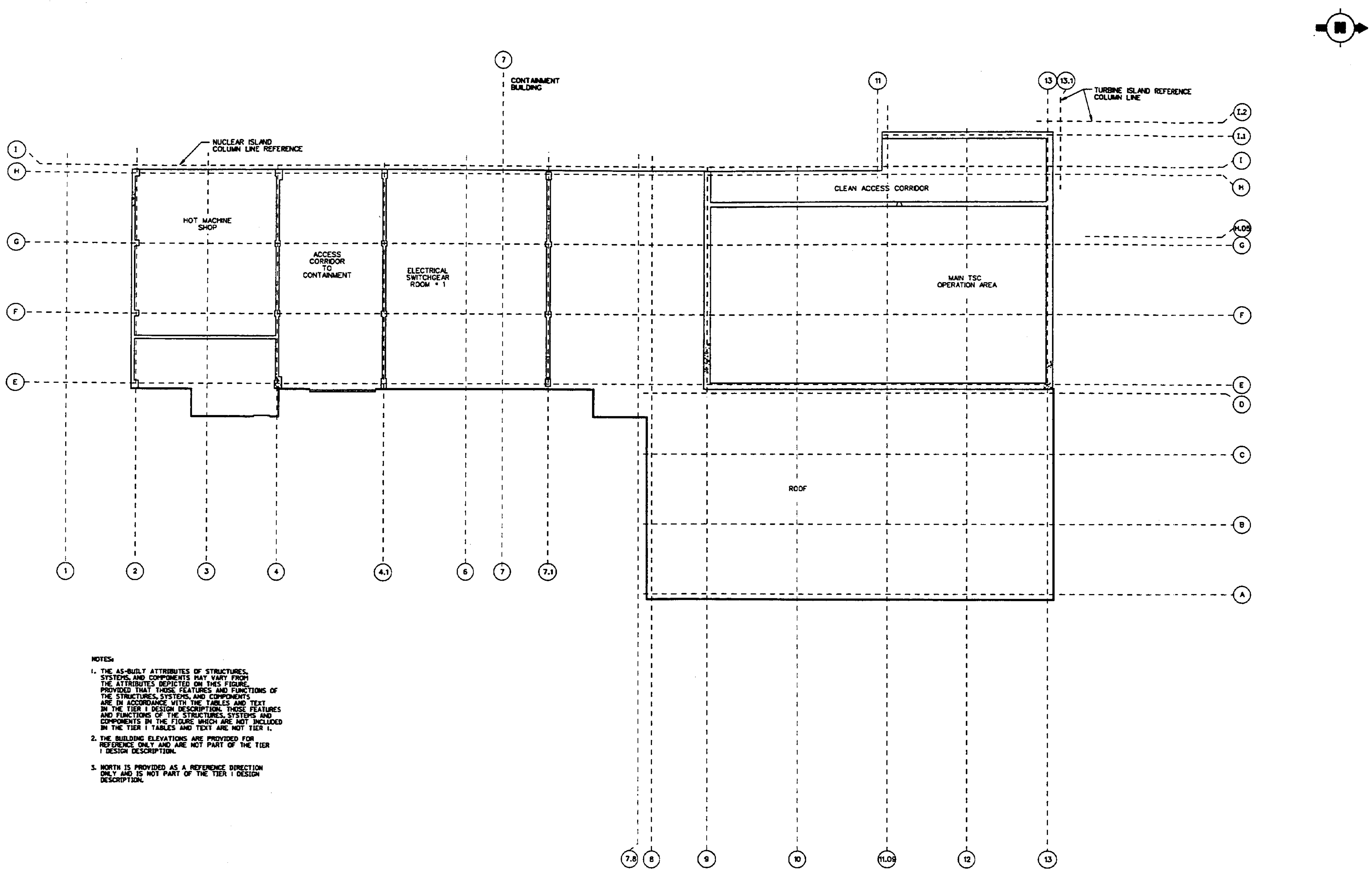
Figure 3.3-11
Annex Building Plan View at Elevation 100'-0"



NOTES:

1. THE AS-BUILT ATTRIBUTES OF STRUCTURES, SYSTEMS, AND COMPONENTS MAY VARY FROM THE ATTRIBUTES DEPICTED ON THIS FIGURE, PROVIDED THAT THOSE FEATURES AND FUNCTIONS OF THE STRUCTURES, SYSTEMS, AND COMPONENTS ARE IN ACCORDANCE WITH THE TABLES AND TEXT IN THE TIER 1 DESIGN DESCRIPTION. THOSE FEATURES AND FUNCTIONS OF THE STRUCTURES, SYSTEMS AND COMPONENTS IN THE FIGURE WHICH ARE NOT INCLUDED IN THE TIER 1 TABLES AND TEXT ARE NOT TIER 1.
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Figure 3.3-12
Annex Building Plan View at Elevation 117'-6"

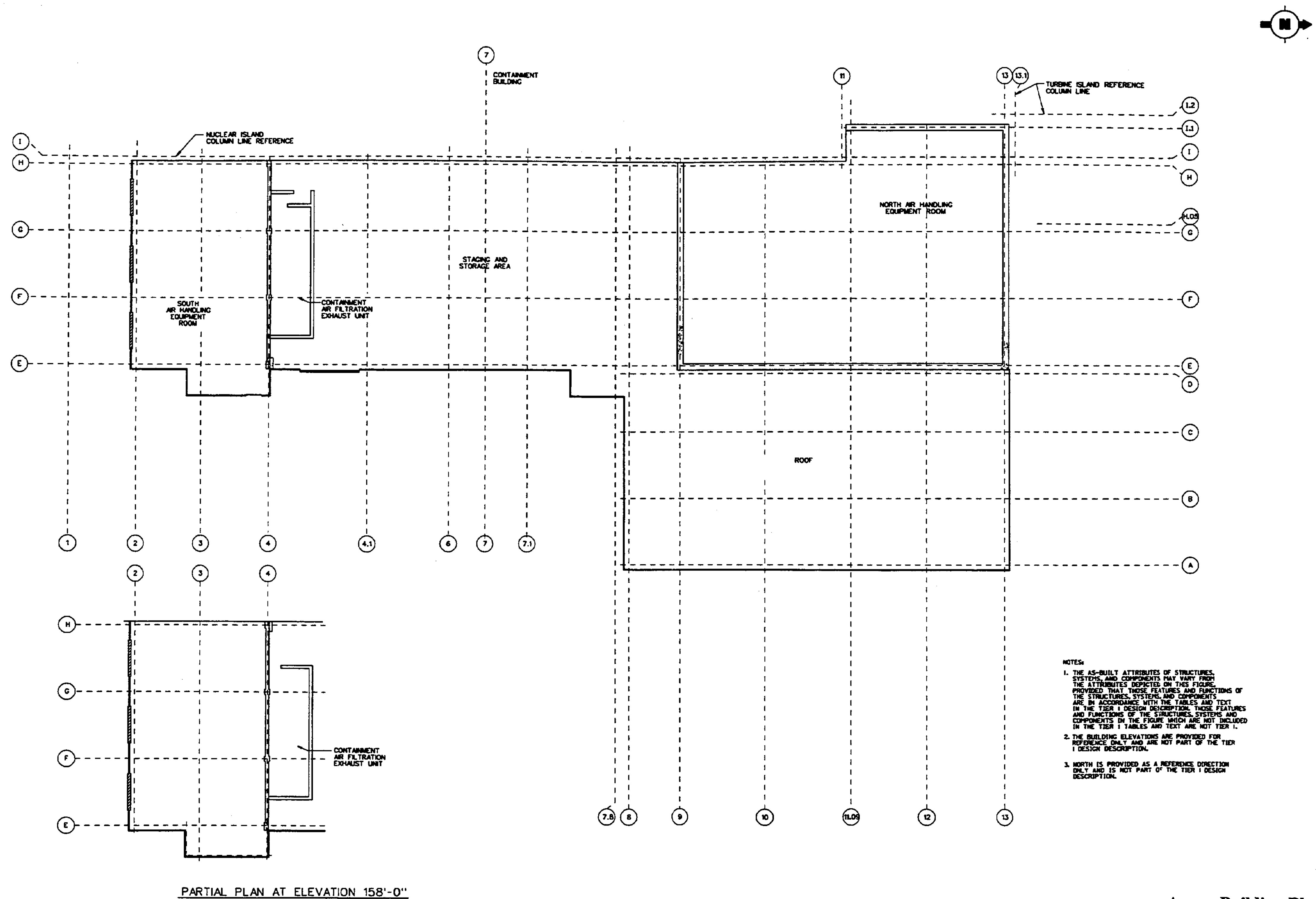

Figure 3.3-13
Annex Building Plan View at Elevation 135'-3"

NOTES:

1. THE AS-BUILT ATTRIBUTES OF STRUCTURES, SYSTEMS, AND COMPONENTS MAY VARY FROM THE ATTRIBUTES DEPICTED ON THIS FIGURE, PROVIDED THAT THOSE FEATURES AND FUNCTIONS OF THE STRUCTURES, SYSTEMS, AND COMPONENTS ARE IN ACCORDANCE WITH THE TABLES AND TEXT IN THE TIER 1 DESIGN DESCRIPTION. THOSE FEATURES AND FUNCTIONS OF THE STRUCTURES, SYSTEMS AND COMPONENTS IN THE FIGURE WHICH ARE NOT INCLUDED IN THE TIER 1 TABLES AND TEXT ARE NOT TIER 1.
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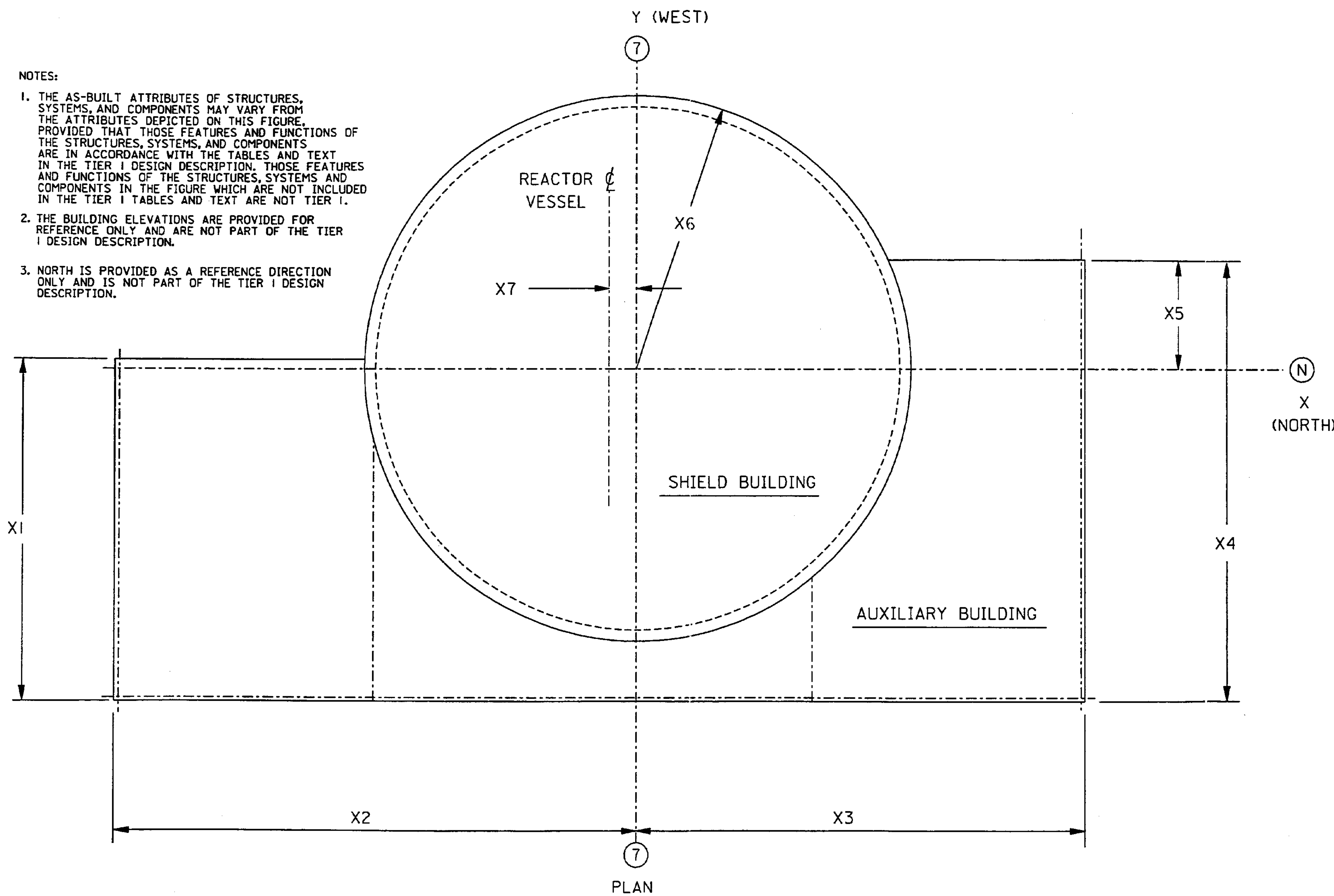

Figure 3.3-14
Nuclear Island Structures Dimensions at Elevation 66'-6"

3.4 Initial Test Program

Design Description

This section represents a commitment that combined license applicants referencing the AP600 certified design will implement an initial test program.

An initial test program is performed during the initial startup of each AP600 plant. The initial test program consists of a series of tests categorized as construction and installation, preoperational (prior to fuel load), and startup (during and after fuel load). All ITAAC will be completed prior to fuel load; therefore, no ITAAC are performed during the startup test phase of the initial test program.

Construction and installation tests are performed to verify the adequacy of construction, installation, and preliminary operation of components and systems. Various electrical and mechanical tests are performed including cleaning and flushing, hydrostatic testing, electrical checks, operability checks, and instrumentation calibration. The completion of the construction and installation test program demonstrates that the system is ready for preoperational testing.

Preoperational tests are performed for each system after construction and installation tests, but prior to initial fuel loading to demonstrate that equipment and systems perform in accordance with design criteria so that initial fuel loading, initial criticality, and subsequent power operation can be safely undertaken. Preoperational tests include, as appropriate, logic and interlock tests, control and instrumentation functional tests, component functional tests, operational and performance tests, and expansion, vibration, and dynamic effects tests.

Startup tests begin with the initial fuel loading and are performed to demonstrate the capability of individual systems, as well as the integrated plant, to meet performance requirements. Startup testing is conducted in four categories: tests related to initial fuel loading, tests performed after initial fuel loading but prior to initial criticality, tests related to initial criticality and those performed at low power (less than 5 percent), and tests performed at power levels greater than 5 percent (ascension to power tests). Startup tests include a controlled fuel load, reactor core and component performance tests, initial criticality, control and protection system operational tests, and plant system performance tests.

Preoperational and startup tests are performed using test specifications and test procedures. The test procedures delineate the test methods to be used in the conduct of the Initial Test Program and the applicable acceptance criteria against which performance is evaluated. Test specifications and procedures are developed and reviewed by qualified personnel. Copies of the test specifications and test procedures for preoperational tests are available to NRC personnel prior to the scheduled performance of these tests. Copies of the test specifications and test procedures for startup tests are provided to NRC inspection personnel prior to the scheduled fuel loading date. Administrative procedures are used to control the conduct of the test program; the review, evaluation and approval of test results; and test record retention.

3.5 Radiation Monitoring

Design Description

Radiation monitoring is provided for those plant areas where there is a significant potential for airborne contamination, for those process and effluent streams where contamination is possible, and in accessible areas to provide indication of unusual radiological events as identified in Tables 3.5-1, 3.5-2, 3.5-3, 3.5-4, and 3.5-5. The radiation monitoring component locations are as shown in Table 3.5-7.

1. The seismic Category I equipment identified in Table 3.5-1 can withstand seismic design basis loads without loss of safety function.

2. The Class 1E equipment identified in Table 3.5-1 as being qualified for a harsh environment can withstand the environmental conditions that would exist before, during, and following a design basis accident without loss of safety function for the time required to perform the safety function.

3. Separation is provided between system Class 1E divisions, and between Class 1E divisions and non-Class 1E cable.

4. Safety-related displays identified in Table 3.5-1 can be retrieved in the main control room (MCR).

5. The process radiation monitors listed in Table 3.5-2 are provided.

6. The effluent radiation monitors listed in Table 3.5-3 are provided.

7. The airborne radiation monitors listed in Table 3.5-4 are provided.

8. The area radiation monitors listed in Table 3.5-5 are provided.

Inspections, Tests, Analyses, and Acceptance Criteria

Table 3.5-6 specifies the inspections, tests, analyses, and associated acceptance criteria for radiation monitoring.

Table 3.5-1					
Equipment Name	**Tag No.**	**Seismic Cat. I**	**Class 1E**	**Qual. for Harsh Envir.**	**Safety-Related Display**
Containment High Range Monitor	PXS-RE160	Yes	Yes	Yes	Yes
Containment High Range Monitor	PXS-RE161	Yes	Yes	Yes	Yes
Containment High Range Monitor	PXS-RE162	Yes	Yes	Yes	Yes
Containment High Range Monitor	PXS-RE163	Yes	Yes	Yes	Yes
MCR Radiation Monitoring Package A(1)	VBS-RE01A	Yes	Yes	No	Yes
MCR Radiation Monitoring Package B(1)	VBS-RE01B	Yes	Yes	No	Yes
Containment Atmosphere Monitor (Gaseous)	PSS-RE026	Yes	No	No	No
Containment Atmosphere Monitor (N13)	PSS-RE027	Yes	No	No	No

Notes: (1) Each MCR Radiation Monitoring Package includes particulate, iodine and gaseous radiation monitors.

Table 3.5-2 Process Radiation Monitors	
Equipment List	**Equipment No.**
Steam Generator Blowdown	BDS-RE010
Steam Generator Blowdown	BDS-RE011
Component Cooling Water	CCS-RE001
Main Steam Line	SGS-RE026
Main Steam Line	SGS-RE027
Service Water Blowdown	SWS-RE008
Primary Sampling System Liquid Sample	PSS-RE050
Primary Sampling System Gaseous Sample	PSS-RE052
Containment Air Filtration Exhaust	VFS-RE001
Gaseous Radwaste Discharge	WGS-RE017

Table 3.5-3 Effluent Radiation Monitors	
Equipment List	**Equipment No.**
Plant Vent (Normal Range Particulate)	VFS-RE101
Plant Vent (Normal Range Iodine)	VFS-RE102
Plant Vent (Normal Range Radiogas)	VFS-RE103
Plant Vent (Mid Range Radiogas)	VFS-RE104A
Plant Vent (High Range Radiogas)	VFS-RE104B
Turbine Island Vent	TDS-RE001
Liquid Radwaste Discharge	WLS-RE229
Wastewater Discharge	WWS-RE021

Table 3.5-4 Airborne Radiation Monitors	
Equipment List	**Equipment No.**
Fuel Handling Area Exhaust Radiation Monitor	VAS-RE-001
Auxiliary Building Exhaust Radiation Monitor	VAS-RE-002
Annex Building Exhaust Radiation Monitor	VAS-RE003
Health Physics and Hot Machine Shop Exhaust Radiation Monitor	VHS-RE001
Radwaste Building Exhaust Radiation Monitor	VRS-RE023

Table 3.5-5 Area Radiation Monitors	
Primary Sampling Room Area Monitor	RMS-RE008
Technical Support Center Area Monitor	RMS-RE016
Main Control Room Area Monitor	RMS-RE010

Table 3.5-6 Inspections, Tests, Analyses, and Acceptance Criteria		
Design Commitment	**Inspections, Tests, Analyses**	**Acceptance Criteria**
1. The seismic Category I equipment identified in Table 3.5-1 can withstand seismic design basis loads without loss of safety function.	i) Inspection will be performed to verify that the seismic Category I equipment identified in Table 3.5-1 is located on the Nuclear Island.	i) The seismic Category I equipment identified in Table 3.5-1 is located on the Nuclear Island.
	ii) Type tests, analyses, or a combination of type tests and analyses of seismic Category I equipment will be performed.	ii) A report exists and concludes that the seismic Category I equipment can withstand seismic design basis loads without loss of safety function.
	iii) Inspection will be performed for the existence of a report verifying that the as-installed equipment including anchorage is seismically bounded by the tested or analyzed conditions.	iii) A report exists and concludes that the as-installed equipment including anchorage is seismically bounded by the tested or analyzed conditions.
2. The Class 1E equipment identified in Table 3.5-1 as being qualified for a harsh environment can withstand the environmental conditions that would exist before, during, and following a design basis accident without loss of safety function for the time required to perform the safety function.	Type tests, analyses, or a combination of type tests and analyses will be performed on Class 1E equipment located in a harsh environment.	A report exists and concludes that Class 1E equipment identified in Table 3.5-1 as being located in a harsh environment can withstand the environmental conditions that would exist before, during, and following a design basis accident without loss of safety function for the time required to perform the safety function.
3. Separation is provided between system Class 1E divisions, and between Class 1E divisions and non-Class 1E cable.	See Tier 1 Material, Section 3.3, Nuclear Island Buildings.	See Tier 1 Material, Section 3.3, Nuclear Island Buildings.
4. Safety-related displays identified in Table 3.5-1 can be retrieved in the MCR.	Inspection will be performed for retrievability of the displays in the MCR.	Safety-related displays identified in Table 3.5-1 can be retrieved in the MCR.
5. The process radiation monitors listed in Table 3.5-2 are provided.	Inspection for the existence of the monitors will be performed.	Each of the monitors listed in Table 3.5-2 exists.

Table 3.5-6 (cont.) Inspections, Tests, Analyses, and Acceptance Criteria		
Design Commitment	**Inspections, Tests, Analyses**	**Acceptance Criteria**
6. The effluent radiation monitors listed in Table 3.5-3 are provided.	Inspection for the existence of the monitors will be performed.	Each of the monitors listed in Table 3.5-3 exists.
7. The airborne radiation monitors listed in Table 3.5-4 are provided.	Inspection for the existence of the monitors will be performed.	Each of the monitors listed in Table 3.5-4 exists.
8. The area radiation monitors listed in Table 3.5-5 are provided.	Inspection for the existence of the monitors will be performed.	Each of the monitors listed in Table 3.5-5 exists.

Table 3.5-7		
Component Name	**Tag No.**	**Component Location**
Containment High Range Radiation Monitor	PXS-RE160	Containment
Containment High Range Radiation Monitor	PXS-RE161	Containment
Containment High Range Radiation Monitor	PXS-RE162	Containment
Containment High Range Radiation Monitor	PXS-RE163	Containment
MCR Radiation Monitoring Package A	VBS-RE01A	Auxiliary Building
MCR Radiation Monitoring Package B	VBS-RE01B	Auxiliary Building
Containment Atmosphere Radiation Monitor (Gaseous)	PSS-RE026	Auxiliary Building
Containment Atmosphere Radiation Monitor (N13)	PSS-RE027	Auxiliary Building
Steam Generator Blowdown Radiation Monitor	BDS-RE010	Turbine Building
Steam Generator Blowdown Radiation Monitor	BDS-RE011	Turbine Building
Component Cooling Water Radiation Monitor	CCS-RE001	Turbine Building
Main Steam Line Radiation Monitor	SGS-RE026	Auxiliary Building
Main Steam Line Radiation Monitor	SGS-RE027	Auxiliary Building
Service Water Blowdown Radiation Monitor	SWS-RE008	Turbine Building
Primary Sampling System Liquid Sample Radiation Monitor	PSS-RE050	Auxiliary Building
Primary Sampling System Gaseous Sample Radiation Monitor	PSS-RE052	Auxiliary Building
Containment Air Filtration Exhaust Radiation Monitor	VFS-RE001	Annex Building
Gaseous Radwaste Discharge Radiation Monitor	WGS-RE017	Auxiliary Building
Plant Vent (Normal Range Particulate) Radiation Monitor	VFS-RE101	Plant Vent
Plant Vent (Normal Range Iodine) Radiation Monitor	VFS-RE102	Plant Vent

Table 3.5-7 (cont.)		
Component Name	**Tag No.**	**Component Location**
Plant Vent (Normal Range Radiogas) Radiation Monitor	VFS-RE103	Plant Vent
Plant Vent (Mid Range Radiogas) Radiation Monitor	VFS-RE104A	Plant Vent
Plant Vent (High Range Radiogas) Radiation Monitor	VFS-RE104B	Plant Vent
Turbine Island Vent Radiation Monitor	TDS-RE001	Turbine Building
Liquid Radwaste Discharge Monitor	WLS-RE229	Auxiliary Building
Wastewater Discharge Radiation Monitor	WWS-RE021	Turbine Building
Fuel Handling Area Exhaust Radiation Monitor	VAS-RE-001	Auxiliary Building
Auxiliary Building Exhaust Radiation Monitor	VAS-RE-002	Auxiliary Building
Annex Building Exhaust Radiation Monitor	VAS-RE003	Auxiliary Building
Health Physics and Hot Machine Shop Exhaust Radiation Monitor	VHS-RE001	Annex Building
Radwaste Building Exhaust Radiation Monitor	VRS-RE023	Radwaste Building
Primary Sampling Room Area Radiation Monitor	RMS-RE008	Auxiliary Building
Technical Support Center Area Radiation Monitor	RMS-RE016	Annex Building
Main Control Room Area Radiation Monitor	RMS-RE010	Auxiliary Building

3.6 Reactor Coolant Pressure Boundary Leak Detection

Design Description

The reactor coolant pressure boundary leakage detection monitoring provides a means of detecting and quantifying the reactor coolant leakage. To detect unidentified leakage inside containment, the following diverse methods are provided to quantify and assist in locating the leakage:

- Containment Sump Level
- Reactor Coolant System Inventory Balance
- Containment Atmosphere Radiation

Leakage detection monitoring is accomplished using instrumentation and other components of several systems.

1. The diverse leak detection methods provide the nonsafety-related function of detecting small leaks when RCS leakage indicates possible reactor coolant pressure boundary degradation.

Inspection, Tests, Analyses, and Acceptance Criteria

Table 3.6-1 specifies the inspections, tests, analyses, and associated acceptance criteria for the leak detection equipment.

Table 3.6-1
Inspections, Tests, Analyses, and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. The diverse leak detection methods provide the nonsafety-related function of detecting small leaks when RCS leakage indicates possible reactor coolant pressure boundary degradation.	See Tier 1 Material sections: i) Subsection 2.3.10 for the containment sump level measuring instruments WLS-034 and WLS-035 ii) Section 3.5 for the containment atmosphere N^{13}/F^{18} radioactivity monitor PSS-RE027 iii) Subsection 2.1.2 for the pressurizer level measuring instruments RCS-195A, RCS-195B, RCS-195C, and RCS-195D iv) Subsection 2.1.2 for the RCS hot and cold leg temperature instruments RCS-121A, RCS-121B, RCS-121C, RCS-121D, RCS-122A, RCS-122B, RCS-122C, RCS-122D, RCS-131A, RCS-131B, RCS-131C, RCS-131D, RCS-132A, RCS-132B, RCS-132C, RCS-132D v) Subsection 2.1.2 for the RCS pressure instruments RCS-140A, RCS-140B, RCS-140C, RCS-140D vi) Subsection 2.3.2 for the letdown and makeup flow instruments CVS-001 and CVS-025 vii) Subsection 2.3.10 for the reactor coolant drain tank level instrument WLS-002	See Tier 1 Material sections: i) Subsection 2.3.10 for the containment sump level measuring instruments WLS-034 and WLS-035 ii) Section 3.5 for the containment atmosphere N^{13}/N^{18} radioactivity monitor PSS-RE027 iii) Subsection 2.1.2 for the pressurizer level measuring instruments RCS-195A, RCS-195B, RCS-195C, and RCS-195D iv) Subsection 2.1.2 for the RCS hot and cold leg temperature instruments RCS-121A, RCS-121B, RCS-121C, RCS-121D, RCS-122A, RCS-122B, RCS-122C, RCS-122D, RCS-131A, RCS-131B, RCS-131C, RCS-131D, RCS-132A, RCS-132B, RCS-132C, RCS-132D v) Subsection 2.1.2 for the RCS pressure instruments RCS-140A, RCS-140B, RCS-140C, RCS-140D vi) Subsection 2.3.2 for the letdown and makeup flow instruments CVS-001 and CVS-025 vii) Subsection 2.3.10 for the reactor coolant drain tank level instrument WLS-002

3.7 Design Reliability Assurance Program

The Design Reliability Assurance Program (D-RAP) is a program that will be performed during the detailed design and equipment specification phase prior to initial fuel load. The D-RAP evaluates and sets priorities for the structures, systems, and components (SSCs) in the design, based on their degree of risk significance. The risk-significant components are listed in Table 3.7-1.

The objective of the D-RAP program is to provide reasonable assurance that risk significant SSC's (Table 3.7-1) are designed such that: (1) assumptions from the risk analysis are utilized, (2) SSC's (Table 3.7-1) when challenged, function in accordance with the assumed reliability, (3) SSC's (Table 3.7-1) whose failure results in a reactor trip, function in accordance with the assumed reliability, (4) maintenance actions to achieve the assumed reliability are identified.

1. The D-RAP provides reasonable assurance that the design of risk-significant SSCs is consistent with their risk analysis assumptions.

Inspections, Tests, Analyses, and Acceptance Criteria

Table 3.7-3 specifies the inspections, tests, analyses, and associated acceptance criteria for the D-RAP.

Table 3.7-1
Risk-Significant Components

Equipment Name	Tag No.
PMS Actuation Software (used to provide automatic control functions listed in Tables 2.5.2-2 and 2.5.2-3)	-
PMS Actuation Hardware (used to provide automatic control functions listed in Tables 2.5.2-2 and 2.5.2-3)	-
MCR 1E Displays	OCS-JC-010, OCS-JC-011
MCR 1E System Level Controls	OCS-JC-010, OCS-JC-011
Reactor Trip Switch Gear	PMS-JP-RTS A01/2 PMS-JP-RTS B01/2 PMS-JP-RTS C01/2 PMS-JP-RTS D01/2
Reactor Coolant Pump Circuit Breakers	ECS-ES-51, -52, -53, -54 ECS-ES-61, -62, -63, -64
Annex Building UPS Distribution Panels (provide power to DAS)	EDS1-EA-14 EDS2-EA-14
PLS Actuation Software and Hardware (used to provide automatic control functions listed in Table 3.7-2)	-
DAS Actuation Hardware (used to provide automatic and manual actuation)	DAS-JD-001 DAS-JD-002 OCS-JC-020
Containment Isolation Valves Controlled by DAS	Refer to Table 2.2.1-1

Note: Dash (-) indicates not applicable.

Table 3.7-1 (cont.) Risk-Significant Components	
Equipment Name	**Tag No.**
Control Rod MG Set Field Breakers	PLS-MG-01A, PLS-MG-01B
Makeup Pumps	CVS-MP-01A, -01B
RNS Pumps	RNS-MP-01A, -01B
Startup Feedwater Pumps	FWS-MP-03A, -03B
SFS Pumps	SFS-MP-01A, -01B
CCS Pumps	CCS-MP-01A, -01B
Service Water Pumps	SWS-MP-01A, -01B
PCCWST Recirculation Pumps	PCS-MP-01A, -01B
PCCWST Drain Isolation Valves	PCS-PL-V001A/B
Standby Diesel Generators	ZOS-MG-02A, -02B
Ancillary Diesel Generators	ECS-MG-01, -02
MCR Ancillary Fans	VBS-MA-10A, -10B
I&C Room B/C Ancillary Fans	VBS-MA-11, -12
Hydrogen Ignitors	VLS-EH-1 through -60
Containment Vessel	CNS-MV-50
Pressurizer Safety Valves	RCS-PL-V005A RCS-PL-V005B
First-Stage ADS MOV	RCS-PL-V001A RCS-PL-V001B RCS-PL-V011A RCS-PL-V011B
Second-Stage ADS MOV	RCS-PL-V002A RCS-PL-V002B RCS-PL-V012A RCS-PL-V012B
Third-Stage ADS MOV	RCS-PL-V003A RCS-PL-V003B RCS-PL-V013A RCS-PL-V013B

Table 3.7-1 (cont.)
Risk-Significant Components

Equipment Name	Tag No.
Fourth-Stage ADS Squib Valves	RCS-PL-V004A RCS-PL-V004B RCS-PL-V004C RCS-PL-V004D
RCS Hot Leg Level Sensors	RCS-160A RCS-160B
Pressurizer Pressure Sensors	RCS-191A RCS-191B RCS-191C RCS-191D
Pressurizer Level Sensors	RCS-195A RCS-195B RCS-195C RCS-195D
Main Steam Line Isolation Valves	SGS-PL-V040A SGS-PL-V040B
Steam Generator Narrow-Range Level Sensors	SGS-001 SGS-002 SGS-003 SGS-004 SGS-005 SGS-006 SGS-007 SGS-008
Steam Generator Wide-Range Level Sensors	SGS-011 SGS-012 SGS-013 SGS-014 SGS-015 SGS-016 SGS-017 SGS-018

Table 3.7-1 (cont.) Risk-Significant Components	
Equipment Name	**Tag No.**
Steam Line Pressure Sensors	SGS-030 SGS-031 SGS-032 SGS-033 SGS-034 SGS-035 SGS-036 SGS-037
Main Steam Safety Valves	SGS-PL-V030A SGS-PL-V030B SGS-PL-V031A SGS-PL-V031B SGS-PL-V032A SGS-PL-V032B
IRWST Screens	PXS-MY-Y01A PXS-MY-Y01B
Containment Recirculation Screens	PXS-MY-Y02A PXS-MY-Y02B
CMT Discharge Isolation Valves	PXS-PL-V014A PXS-PL-V014B PXS-PL-V015A PXS-PL-V015B
CMT Discharge Check Valves	PXS-PL-V016A PXS-PL-V016B PXS-PL-V017A PXS-PL-V017B
IRWST Gutter Bypass Isolation Valves	PXS-PL-V130A PXS-PL-V130B
Accumulator Discharge Check Valves	PXS-PL-V028A PXS-PL-V028B PXS-PL-V029A PXS-PL-V029B
PRHR HX Control Valves	PXS-PL-V108A PXS-PL-V108B
Containment Recirculation Isolation Motor-operated Valves	PXS-PL-V117A PXS-PL-V117B

Table 3.7-1 (cont.)
Risk-Significant Components

Equipment Name	Tag No.
Containment Recirculation Squib Valves	PXS-PL-V118A PXS-PL-V118B PXS-PL-V120A PXS-PL-V120B
IRWST Injection Check Valves	PXS-PL-V122A PXS-PL-V122B
IRWST Injection Squib Valves	PXS-PL-V123A PXS-PL-V123B PXS-PL-V124A PXS-PL-V124B PXS-PL-V125A PXS-PL-V125B
CMT Level Sensors	PXS-011A PXS-011B PXS-011C PXS-011D PXS-012A PXS-012B PXS-012C PXS-012D PXS-013A PXS-013B PXS-013C PXS-013D PXS-014A PXS-014B PXS-014C PXS-014D
IRWST Level Sensors	PXS-045 PXS-046 PXS-047 PXS-048
125 Vdc 24-Hour Battery	IDSA-DB-1A IDSA-DB-1B IDSB-DB-1A IDSB-DB-1B IDSC-DB-1A IDSC-DB-1B IDSD-DB-1A IDSD-DB-1B

Table 3.7-1 (cont.) Risk-Significant Components	
Equipment Name	**Tag No.**
125Vdc Distribution Panels	IDSA-DD-1 IDSB-DD-1 IDSC-DD-1 IDSD-DD-1 IDSA-EA-1 IDSA-EA-2 IDSB-EA-1 IDSB-EA-2 IDSB-EA-3 IDSC-EA-1 IDSC-EA-2 IDSC-EA-3 IDSD-EA-1 IDSD-EA-2
Fused Transfer Switch Box	IDSA-DF-1 IDSB-DF-1 IDSB-DF-2 IDSC-DF-1 IDSC-DF-2 IDSD-DF-1
125 Vdc MCC	IDSA-DK-1 IDSB-DK-1 IDSC-DK-1 IDSD-DK-1

Table 3.7-2 PLS D-RAP Automatic Control Functions
CVS Reactor Makeup
RNS Reactor Injection from IRWST
Startup Feedwater from CST
Spent Fuel Cooling
Component Cooling of RNS and SFS Heat Exchangers
Service Water Cooling of CCS Heat Exchangers
Standby Diesel Generators
Hydrogen Ignitors

Table 3.7-3
Inspections, Tests, Analyses and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. The D-RAP provides reasonable assurance that the design of risk-significant SSCs is consistent with their risk analysis assumptions.	Inspection will be performed for the existence of a report which establishes the estimated reliability of as-built risk-significant SSCs.	A report exists and concludes that the estimated reliability of each as-built component identified in Table 3.7-1 is at least equal to the assumed reliability and that industry experience including operations, maintenance, and monitoring activities were assessed in estimating the reliability of these SSCs.