1.2 GENERAL PLANT DESCRIPTION

1.2.1 PRINCIPAL DESIGN CRITERIA

The principal criteria for design, construction, and testing of the Susquehanna SES are summarized below. Specific criteria, codes and standards are addressed in Section 3.0.

1.2.1.1 General Design Criteria

The Susquehanna SES design conforms to the requirements given in 10CFR50, Appendix A. Specific compliance is discussed in Section 3.1.

1. The plant is designed, fabricated, and erected to produce electrical power in accordance with the codes, standards, and regulations as described in Section 3.1.

2. Safety related systems are designed to permit safe plant operation and to accommodate postulated accidents without endangering the health and safety of the public.

1.2.1.2 System Design Criteria

1.2.1.2.1 Nuclear System Criteria

1. The fuel cladding is designed to retain integrity as a radioactive material barrier for the design power range. The fuel cladding is designed to accommodate, without loss of integrity, the pressures generated by the fission gases released from the fuel material throughout the design life of the fuel.

2. The fuel cladding, in conjunction with other plant systems, is designed to retain integrity throughout any abnormal operational transient.

3. Those portions of the nuclear system which form part of the reactor coolant pressure boundary are designed to retain integrity as a radioactive material barrier during normal operation following abnormal operational transients and accidents.

4. Heat removal systems are provided in sufficient capacity and operational adequacy to remove heat generated in the reactor core for the full range of normal operational conditions from shutdown to design power, and for any abnormal operational transient.

5. Heat removal systems are provided to remove decay heat generated in the core under circumstances wherein the normal operational heat removal systems become inoperative. The capacity of such systems is adequate to prevent fuel clad damage. The reactor is capable of being shut down automatically in sufficient time to permit decay heat removal systems to become effective following loss of operation of normal heat removal systems.

6. The reactor core and reactivity control systems are designed so that control rod action is capable of bringing the core subcritical and maintaining it so, even with the rod of highest reactivity worth fully withdrawn and unavailable for insertion.

7. The nuclear system is designed so there is no tendency for divergent oscillation of any operating characteristics, considering the interaction of the nuclear system with other appropriate plant systems.

8. The reactor core is designed so that its nuclear characteristics do not contribute to a divergent power transient.

1.2.1.2.2 Safety Related Systems Criteria

1.2.1.2.2.1 General

1. Safety systems act in response to abnormal operational transients so that fuel cladding retains its integrity as a radioactive material barrier.

2. Safety systems and engineered safety features act to ensure that no damage to the nuclear system process barrier results from internal pressures caused by abnormal operational transients or accidents.

3. Where positive, precise actions are required in immediate response to accidents, such actions are automatic and require no decision or manipulation of controls by operations personnel.

4. Essential safety actions are carried out by equipment of sufficient redundance and independence that no single failure of active components can prevent the required actions. For systems or components to which IEEE 279-1971 is applicable, single failures of passive electrical components are considered as well as single failures of active components in recognition of the higher anticipated failure rates of passive electrical components relative to passive mechanical components.

5. Features of the station that are essential to the mitigation of accident consequences are designed, fabricated, and erected to quality standards which reflect the importance of the safety function to be performed.

6. The design of safety systems and engineered safety features includes allowances for environmental phenomena at the site.

7. Provision is made for control of active components of safety systems and engineered safety features from the control room.

8. Safety systems and engineered safety features are designed to permit demonstration of their functional performance requirements.

1.2.1.2.2.2 Containment and Isolation Criteria

1. A primary containment is provided to completely enclose the reactor vessel. It is designed to act as a radioactive material barrier during accidents that release radioactive material into the primary containment. It is possible to test the primary containment integrity and leak tightness at periodic intervals.

2. A secondary containment that completely encloses both primary containment and fuel storage areas is provided and is designed to act as a radioactive material barrier.

3. The primary and secondary containments, in conjunction with other engineered safety features, act to prevent radioactive material released from the containment volumes from exceeding the guideline values of applicable regulations.

4. Provisions are made for the removal of energy from within the primary containment as necessary to maintain the integrity of the containment system following accidents that release energy to the primary containment.

5. Piping that penetrates the primary containment structure and could serve as a path for the uncontrolled release of radioactive material to the environs is automatically isolated whenever there is a threat of uncontrolled radioactive material being released. Such isolation is effected in time to prevent radiological effects from exceeding the values of applicable regulations.

1.2.1.2.2.3 Emergency Core Cooling System (ECCS) Criteria

1. ECCS systems are provided to limit fuel cladding temperature to temperatures below the onset of fragmentation (2200°F) in the event of a loss of coolant accident.

2. The ECCS provides for continuity of core cooling over the complete range of postulated break sizes in the nuclear system process barrier.

3. The ECCS is diverse, reliable, and redundant.

4. Operation of the ECCS is initiated automatically when required regardless of the availability of off-site power supplies and the normal generating system of the plant.

1.2.1.2.3 Process Control Systems Criteria

1.2.1.2.3.1 Nuclear System Process Control Criteria

1. Control equipment is provided to allow the reactor to respond to limited load changes, major load changes and abnormal operational transients.

2. It is possible to control the reactor power level manually.

3. Control of the nuclear system is possible from a single location.

4. Nuclear system process controls are arranged to allow the operator to rapidly assess the condition of the nuclear system and to locate process system malfunctions.

5. Interlocks, or other automatic equipment, are provided as a backup to procedural controls to avoid conditions requiring the actuation of nuclear safety systems or engineered safety features.

6. If the control room is inaccessible, it is possible to bring the reactor from power range operation to a hot shutdown condition by manipulation of controls and equipment which are available outside of the control room. Furthermore, station design does not preclude the ability, in this event, to bring the reactor to a cold shutdown condition from the hot shutdown condition.

1.2.1.2.3.2 Power Conversion Systems Process Control Criteria

1. Controls are provided to maintain temperature and pressure to below design limitations. This system will result in a stable operation and response for all allowable variations.

2. Controls are designed to provide indication of system trouble.

3. Control of the power conversion system is possible from a single location.

4. Controls are provided to ensure adequate cooling of power conversion system equipment.

5. Controls are provided to ensure adequate condensate purity.

6. Controls are provided to regulate the supply of water so that adequate reactor vessel water level is maintained.

1.2.1.2.3.3 Electrical Power System Process Control Criteria

1. Controls are provided to ensure that sufficient electrical power is provided for startup, normal operation, prompt shutdown and continued maintenance of the station in a safe condition.

2. Control of the electrical power system is possible from a single location.

1.2.1.2.4 Electrical Power System Criteria

1. The station electrical power systems are designed to deliver the electrical power generated.

2. Sufficient normal auxiliary sources of electrical power are provided to attain prompt shutdown and continued maintenance of the station in a safe condition. The capacity of the power sources is adequate to accomplish all required engineered safety features under postulated design basis accident conditions.

3. Standby electrical power sources are provided to allow prompt reactor shutdown and removal of decay heat under circumstances where preferred power is not available. They provide sufficient power to all engineered safety features requiring electrical power.

1.2.1.2.5 Fuel Handling and Storage Facilities

1. Fuel handling and storage facilities are located in the reactor building and are designed to preclude criticality and to maintain adequate shielding and cooling for spent fuel.

   Additional spent fuel storage facilities are provided at the Independent Spent Fuel Storage Installation (ISFSI) located north of the Low Level Radwaste Holding Facility (LLRWHF). The ISFSI is described in detail in Section 11.7. Handling of spent fuel stored at the ISFSI is in the Reactor Building and is designed to preclude criticality and to maintain adequate shielding and cooling for spent fuel.

1.2.1.2.6 Auxiliary Systems Criteria

1. Multiple independent station auxiliary systems are provided for the purpose of cooling and servicing the station, the reactor and the station containment systems under various normal and abnormal conditions.

2. Backup reactor shutdown capability is provided independent of normal reactivity control provisions. This backup system has the capability to shut down the reactor from any operating condition, and subsequently to maintain the shutdown condition.

1.2.1.2.7 Power Conversion Systems Criteria

Components of the power conversion systems are designed to fulfill the following basic objectives:

a) Generate electricity with the turbine generator from steam produced in the reactor, condense the exhaust steam in the condenser and return the condensed water to the reactor as heated feedwater with most of the non-condensable gases and impurities removed.

b) Ensure that any fission products or radioactivity associated with the steam and condensate during normal operation are safely contained inside the system or are released under controlled conditions in accordance with waste disposal procedures.

1.2.1.2.8 Radioactive Waste Disposal Criteria

1. Gaseous, liquid, and solid waste disposal facilities are designed so that the discharge of radioactive effluents, storage, and off-site shipment of radioactive material are made in accordance with applicable regulations.

2. These facilities include means for informing station operating personnel whenever operational limits on the release of radioactive material are exceeded.

3. A separate facility for interim on-site storage of low level radioactive waste material as of April 30, 1988 was included under the 10CFR Part 50 facility operating licenses.

1.2.1.2.9 Shielding and Access Control Criteria

1. Radiation shielding is provided and access control patterns are established to allow the operating staff to control radiation doses within the limits of applicable regulations in any mode of normal station operation. The design and establishment of the above include conditions that deal with fission product release from failed fuel elements and contamination of station areas from system leakage.

2. The control room is shielded against radiation and has suitable environmental control so that occupancy under design basis accident conditions is possible.

1.2.2 PLANT DESCRIPTION

1.2.2.1 Site Characteristics

1.2.2.1.1 Location and Size

In terms of its relationship to metropolitan areas, the plant site lies approximately 20 miles southwest of Wilkes-Barre, approximately 50 miles northwest of Allentown, and approximately 70 miles northeast of Harrisburg. The plant is a two unit, Boiling Water Reactor. Each unit has a nominal rating of 1300 MWe. It is located on a 1,574 acre property owned by PPL in Salem Township, Luzerne County, Pennsylvania, along the west bank of the Susquehanna River approximately 5 miles northeast of the borough of Berwick, Pennsylvania. In addition, 717 acres of the site are located on the east side of the river in Conyngham and Hollenback Township. Total site acreage is approximately 2,355 acres. There are no structures or facilities on the east side of the river with the exception of transmission lines and facilities. A map of the site area including major structures and facilities is provided as Figure 2.1-12. PPL owns the entire 1800 foot plant exclusion area (except for Township Route T-419) and has complete authority to regulate any and all access and activity within that area.

The property in Salem Township is open deciduous woodland, interspersed with grassland and orchards and is bounded on its eastern flank by the Susquehanna River, which has a low water elevation of 484 feet MSL in this vicinity. Much of the northern property boundary runs along the slopes of an east-west trending ridge rising to a maximum elevation of 1060 feet MSL. This ridge abuts a rolling plateau to the south which in turn falls gradually in an easterly direction toward the floodplain of the Susquehanna River. The plant site is located on this plateau at an approximate grade of 675 feet MSL. Rainfall runoff leads into two main valleys that form intermittent waterways draining to the Susquehanna River, east of the property.

Also, in Salem Township a portion of the long abandoned North Branch Canal runs north-south across the floodplain between the Susquehanna River and U.S. Route 11. Within the property limits, the northern portion of the canal is generally dry and overgrown with trees and shrubs whereas the southern portion contains stagnant water. A permanent 30 acre body of water named Lake "Took-a-While" is located just west of the canal. An approximate 400 acre recreation area has been developed on the floodplain.

Acreage in Conyngham and Hollenback Townships of Luzerne County on the east side of the Susquehanna River is open to the public for hunting, fishing and hiking. One of the trails leads to the scenic view from Council Cup, a 700-foot high bluff overlooking the Susquehanna River valley.

A multiple-use land management program coordinated through a 10-year forest stewartship plan is aimed at providing a mix of woodlands, farming, recreation, a wildlife habitat, timber production, and historical protection.

1.2.2.1.2 Road and Rail Access

US Route 11 runs north-south through the property along the western edge of the floodplain. In this vicinity, the highway has a pavement width of 36 feet and is of bitumen-topped concrete slab construction. Township road T419, which follows the toe of the east-west trending ridge described above, leads off US 11 to traverse the property and link with Township road T438 which passes through the western portion of the property. Both of these Township roads are

paved in this vicinity. A railroad line on the floodplain parallels US 11 in traversing the property. The North Shore Railroad Co. operates this line owned by The Commonwealth of Pennsylvania. The railroad is a single track, non-electrified line of standard gage. The nearest railroad station is at Berwick, 5 miles to the south. Access to the various facilities is provided as follows:

a) a MAIN and a SECONDARY ACCESS ROAD leading from US 11 at separate locations to serve the main power block and surrounding structures

b) an ACCESS RAILROAD SPUR leading from the Conrail (Erie-Lackawanna) Railroad to serve the main power block and cooling tower areas

c) PLANT ROADS providing access to all structures as well as connecting with the Main and Secondary access roads

d) a RIVER FACILITIES ACCESS ROAD leading from US 11 to the intake structure on the river bank

e) a PERIMETER PATROL ROAD paralleling and within the plant security fence

1.2.2.1.3 Description of Plant Environs

1.2.2.1.3.1 Geology and Soils

The property is underlain by a series of tightly folded strata of Paleozoic age, trending generally northeast-southwest. Pleistocene glacial outwash deposits mantle much of the area, particularly in topographic depressions. Underlying these glacial deposits are strata of Devonian and older ages. The plant site area is underlain by a series of siltstone, sandstone and shale beds of the Hamilton and Susquehanna groups of Devonian age.

Soils in the area are derived from parent material of glacial origin. These soils are acidic, well drained and generally not well suited for agricultural purposes.

1.2.2.1.3.2 Groundwater

The two principal aquifer systems in the region are the unconsolidated glacial and alluvial deposits and the underlying bedrock formations. The glacial deposits consist of drift, till, and outwash materials and vary in permeability from very low to high. The thickness of these deposits is highly variable in the site vicinity, ranging from 1 or 2 feet to over 100 feet. Wells penetrating the bedrock produce water from Onondaga limestone and strata of the Hamilton siltstone group.

The groundwater table in the area is a subdued replica of the surface topography. At the site, the water table is found generally within 35 feet of the ground surface, usually just below the bedrock surface but sometimes within the overburden soils.

Groundwater is the primary source of water supply in the region. The plant potable water supply is obtained from groundwater. There will be no impact to surrounding wells (groundwater level) due to Plant usage, as documented in Dames & Moore report titled "Environmental Feasibility for Groundwater Supply at SSES" dated 9/24/86.

1.2.2.1.3.3 Hydrology

The Susquehanna River Basin comprises an area of about 27,500 square miles in the states of New York, Pennsylvania and Maryland. The plant site is located on the west side of the Main Branch of the Susquehanna River, approximately 42 miles upstream from the confluence of the Main and West Branches at Sunbury, Pennsylvania. The Main Branch has its source at Otsego Lake about 35 miles southeast of Utica, New York. From Otsego Lake, the river flows generally southwest. The Lackawanna River joins it near Pittston, Pennsylvania. From there it flows past the site.

The extreme and average daily flows recorded at the gaging station at Wilkes-Barre, about 20 miles upstream from the site, are:

	Flow *(cfs)*	*(gpm)*	*Date*
Minimum	528	2.38×10^5	September 27,1964
Average	13,000	5.85×10^6	70 years of record
Maximum	345,000	1.55×10^8	June 24, 1972 (Hurricane Agnes)

The River's path is controlled by the geologic structure in the site area, following the regional folding and jointing pattern. Just north of the site area, the river cuts across the regional fold axis along a major joint set before swinging west-southwest and paralleling the regional fold axis.

The river gradient is approximately 1.5 feet per mile near the site. The river has long pools, short riffles and shallow bedrock flats. Scouring during spring floods has removed most silt deposits except in quiet pools. The river formerly had large deposits of coal silt, most of which have been removed by dredging. Below Shickshinny, a large pool extends downstream to a point near Mocanaqua; below this point the water is shallower and faster. The bed is rock and gravel, and the river is interspersed with islands. Both upstream and downstream portions of these islands have clean gravel bars.

1.2.2.1.3.4 Meteorology

A modified continental-type climate prevails in the general area of the site. Normally, the frost-free season extends from late April to mid October. Minimum temperatures during December, January, and February usually are below freezing, but rarely dip below 0°F. Maximum temperatures above 100°F have seldom been recorded. The yearly relative humidity averages about 70 percent. Mean annual snowfall in the region is about 52 inches. In winter the area has about 40 percent of sunshine; the summer percentage is about 60 percent. Heavy thunderstorms have occasionally caused damage over limited areas and tornado-force winds have been reported.

The annual precipitation is about 38 inches. July is normally the wettest month, with an average rainfall of about 5 inches, and February is the driest, with about 2 inches.

The dominant wind is from the West-Southwest Sector.

1.2.2.1.3.5 Seismicity

Of the very few earthquakes which have occurred in Pennsylvania during historical times, most have been in the southeastern part of the State.

Only minor damage has ever been recorded from earth movement in Pennsylvania, with the exception of the two disturbances at Wilkes-Barre in 1954. It is doubtful whether the latter were the direct result of an earthquake. Since the affected area was only 2000 square feet and no record was made of the disturbances at any of the nearest seismic stations, it is likely they were associated with mining activities and the readjustment of alluvial deposits.

Because of the small correlation between seismic activity and known faults or tectonics, the area can be said to constitute an inactive seismotectonic province.

1.2.2.2 General Arrangement of Structures and Equipment

The principal structures and facilities located at the Plant Site are shown on Figure 2.1-11.

The equipment arrangements for these structures are shown in Dwgs. M-220, Sh. 1, M-221, Sh. 1, M-222, Sh. 1, M-223, Sh. 1, M-224, Sh. 1, M-225, Sh. 1, M-226, Sh. 1, M-227, Sh. 1, M-230, Sh. 1, M-231, Sh. 1, M-232, Sh. 1, M-233, Sh. 1, M-234, Sh. 1, M-235, Sh. 1, M-236, Sh. 1, M-237, Sh. 1, M-240, Sh. 1, M-241, Sh. 1, M-242, Sh. 1, M-243, Sh. 1, M-244, Sh. 1, M-245, Sh. 1, M-246, Sh. 1, M-247, Sh. 1, M-248, Sh. 1, M-249, Sh. 1, M-250, Sh. 1, M-251, Sh. 1, M-252, Sh. 1, M-253, Sh. 1, M-254, Sh. 1, M-255, Sh. 1, M-256, Sh. 1, M-257, Sh. 1, M-258, Sh. 1, M-259, Sh. 1, M-260, Sh. 1, M-261, Sh. 1, M-270, Sh. 1, M-271, Sh. 1, M-272, Sh. 1, M-273, Sh. 1, M-274, Sh. 1, M-276, Sh. 1, M-280, Sh. 1, M-281, Sh. 1, M-282, Sh. 1, M-284, Sh. 1, M-5200, Sh. 1, and M-5200, Sh. 2.

1.2.2.3 Nuclear System

The nuclear system includes a single cycle, forced circulation, General Electric Boiling Water Reactor producing steam for direct use in the steam turbine. Heat balances showing the major parameters of the nuclear system for the rated power condition, at rated core flow and at 108 Mlb/hr increased core flow, are shown in Figures 1.2-49 and 1.2-49-1 for Unit 1 and Figures 1.2-49-2 and 1.2-49-3 for Unit 2. The reactor heat balances differ slightly from the turbine heat balances (Figures 10.1-1a and 10.1-1b). The reactor heat balances are based on the measured moisture fraction exiting the reactor, and the moisture fraction exiting the MISV's is determined by the expected pressure drop between the reactor vessel steam dome and the MSIV exit. The turbine heat balance is based on turbine inlet design conditions, which allow for a slightly greater moisture fraction in the steam.

1.2.2.3.1 Reactor Core and Control Rods

The fuel for the reactor core consists of depleted, natural, and/or slightly enriched uranium dioxide pellets contained in sealed Zircaloy-2 tubes. These fuel rods are assembled into individual fuel assemblies. The number of fuel assemblies in the complete core is 764. Gross control of the core is achieved by movable, bottom entry control rods. The control rods are of

cruciform shape and are distributed evenly throughout the core. The control rods are positioned by individual control rod drives.

Each fuel assembly has several fuel rods with gadolinia ($Gd_2 O_3$) mixed in solid solution with the UO_2. The gadolinia is burnable poison which diminishes the reactivity of the fresh fuel. It is depleted as the fuel reaches the end of its first cycle.

A conservative limit of plastic strain is the design criterion used for fuel rod cladding failure. The peak linear heat generation for steady-state operation is well below the fuel damage limit even late in life. Experience has shown that the control rods are not susceptible to distortion and have an average life expectancy many times the residence time of a fuel loading.

1.2.2.3.2 Reactor Vessel and Internals

The reactor vessel contains the core and supporting structure; the steam separators and dryers; the jet pumps; the control rod guide tubes; distribution lines for the feedwater, core spray, and standby liquid control; the incore instrumentation; and other components. The main connections to the vessel include the steam lines, the coolant recirculation lines, the feedwater lines, the control rod drive housings, and the ECCS lines.

The reactor vessel is designed and fabricated in accordance with applicable codes for a pressure of 1250 psig. The nominal operating pressure is 1050 psia in the steam space above the separators. The vessel is fabricated of carbon steel and is clad internally with stainless steel (except for the top head which is not clad).

The core is cooled by reactor water that enters the lower portion of the core and boils as it flows upward around the fuel rods. The chemistry of reactor water is controlled to minimize corrosion of the fuel cladding, reactor vessel internals and reactor coolant system pressure boundary and to control the transport and deposition of corrosion product activity. The steam leaving the core is dried by steam separators and dryers, located in the upper portion of the reactor vessel. The steam is then directed to the turbine through four main steam line(s). Each steam line is provided with two isolation valves in series, one on each side of the primary containment barrier.

1.2.2.3.3 Reactor Recirculation System

The Reactor Recirculation System pumps reactor coolant through the core to remove the energy generated in the fuel. This is accomplished by two recirculation loops external to the reactor vessel but inside the primary containment. Each loop has one motor-driven recirculation pump. Recirculation pump speed can be varied to allow some control of reactor power level through the effects of coolant flow rate on moderator void content.

1.2.2.3.4 Residual Heat Removal System

The Residual Heat Removal System (RHRS) consists of pumps, heat exchangers and piping that fulfill the following functions:

a. Removal of decay heat during and after plant shutdown.

b. Rapid injection of water into the reactor vessel following a loss of coolant accident, at a rate sufficient to reflood the core maintain fuel cladding below the limits contained in 10 CFR 50.46. This is discussed in Subsection 1.2.2.4.

c. Removal of heat from the primary containment following a loss-of-coolant accident (LOCA) to limit the increase in primary containment pressure. This is accomplished by cooling and recirculating the water inside the primary containment. The redundancy of the equipment provided for the containment is further extended by a separate part of the RHRS which sprays cooling water into the drywell. This latter capability is discussed in Subsection 1.2.2.4.12.

d. Provide for cooling of the spent fuel pool(s) following a seismic event which results in a loss of normal spent fuel pool cooling, in conjunction with normal shutdown of both units.

1.2.2.3.5 Reactor Water Cleanup System (RWCU)

A Reactor Water Cleanup System, which includes filter demineralizers, is provided to clean up the reactor cooling water, to reduce the amounts of activated corrosion products in the water, and to remove reactor coolant from the nuclear system under controlled conditions.

1.2.2.4 Safety Related Systems

Safety related systems provide actions necessary to assure safe shutdown, to protect the integrity of radioactive material barriers, and/or to prevent the release of radioactive material in excess allowable dose limits. These systems may be components, groups of components, systems, or groups of systems. Engineered Safety Feature (ESF) systems are included in this category. ESF systems have a sole function of mitigating the consequences of design basis accidents.

1.2.2.4.1 Reactor Protection System

The Reactor Protection System initiates a rapid, automatic shutdown (scram) of the reactor. This action is taken in time to prevent excessive fuel cladding temperatures and any nuclear system process barrier damage following abnormal operational transients. The Reactor Protection System overrides all operator actions and process controls.

1.2.2.4.2 Neutron-Monitoring System

Not all of the Neutron Monitoring System qualifies as a nuclear safety system; only those portions that provide high neutron flux signals and neutron flux oscillation signals to the Reactor Protection System are safety related. The intermediate range monitors (IRM), oscillation power range monitors (OPRM), and average power range monitors (APRM), which monitor neutron flux via in-core detectors, signal the Reactor Protection System in time to prevent excessive fuel clad temperatures as a result of abnormal operational transients.

1.2.2.4.3 Control Rod Drive System

When a scram is initiated by the Reactor Protection System, the Control Rod Drive System inserts the negative reactivity necessary to shut down the reactor. Each control rod is controlled individually by a hydraulic control unit. When a scram signal is received, high pressure water from an accumulator for each rod forces each control rod rapidly into the core.

1.2.2.4.4 Nuclear System Pressure Relief System

A Pressure Relief System, consisting of safety-relief valves mounted on the main steam lines, prevents excessive pressure inside the nuclear system following either abnormal operational transients or accidents.

1.2.2.4.5 Reactor Core Isolation Cooling System

The Reactor Core Isolation Cooling System (RCIC) provides makeup water to the reactor vessel whenever the vessel is isolated from the main condenser and feed water system. The RCICS uses a steam driven turbine-pump unit and operates automatically, in time and with sufficient coolant flow, to maintain adequate reactor vessel water level.

1.2.2.4.6 Primary Containment

A pressure-suppression primary containment houses the reactor vessel, the reactor coolant recirculating loops, and other branch connections of the reactor primary system. The pressure suppression system consists of a drywell, a pressure-suppression chamber storing a large volume of water, a connecting vent system between the drywell and the water pool, isolation valves, containment cooling systems, and other service equipment. In the event of a process system piping failure within the drywell, reactor water and steam would be released into the drywell air space. The resulting increased drywell pressure would then force a mixture of air, steam, and water through the vents into the pool of water stored in the suppression chamber. The steam would condense rapidly in the suppression pool, resulting in a rapid pressure reduction in the drywell. Air transferred to the suppression chamber pressurizes the suppression chamber and is subsequently vented to the drywell to equalize the pressure between the two chambers. Cooling systems remove heat from the reactor core, the drywell, and from the water in the suppression chamber, thus providing continuous cooling of the primary containment under accident conditions. Appropriate isolation valves are actuated during this period to ensure containment of radioactive materials within the primary containment.

1.2.2.4.7 Primary Containment and Reactor Vessel Isolation Control System

The Primary Containment and Reactor Vessel Isolation Control System automatically initiates closure of isolation valves to close off all process lines which are potential leakage paths for radioactive material to the environs. This action is taken upon indication of a potential breach in the nuclear system process barrier.

1.2.2.4.8 Secondary Containment

Any leakage from the primary containment system is to the secondary containment system. This system includes the Standby Gas Treatment System and the Reactor Building Recirculation System. The secondary containment system is designed to minimize the release at ground level of airborne radioactive materials, and to provide for the controlled, filtered release of the reactor building atmosphere at roof level under accident conditions.

1.2.2.4.9 Main Steam Line Isolation Valves

Although process lines which penetrate the primary containment and offer a potential release path for radioactive material are provided with redundant isolation capabilities, the main steam lines, because of their large size and large mass flow rates, are given special isolation consideration. Two automatic isolation valves, each powered by both air pressure and spring force, are provided in each main steam line. These valves fulfill the following objectives:

a. To prevent excessive damage to the fuel barrier by limiting the loss of reactor coolant from the reactor vessel resulting either from a major leak from the steam piping outside the primary containment or from a malfunction of the pressure control system resulting in excessive steam flow from the reactor vessel.

b. To limit the release of radioactive materials, by closing the nuclear system process barrier, in case of a gross release of radioactive materials from the fuel to the reactor coolant and steam.

c. To limit the release of radioactive materials, by closing the primary containment barrier, in case of a major leak from the nuclear system inside the primary containment.

In addition the main steamline isolation valve leakage Isolated Condenser Treatment Method (Section 6.7) is provided to process the fission products after a LOCA. By directing the leakage from the closed main steamline isolation valves through the main steam drain line to the condenser, this leakage is processed prior to release to the atmosphere.

1.2.2.4.10 Main Steam Line Flow Restrictors

A venturi-type flow restrictor is installed in each steam line close to the reactor vessel. These devices limit the loss of coolant from the reactor vessel and prevent uncovering of the core before the main steam line isolation valves are closed in case of a main steam line break.

1.2.2.4.11 Emergency Core Cooling Systems (ECCS)

Four Core Standby Cooling Systems are provided to prevent excessive fuel clad temperatures if a breach in the nuclear system process barrier results in a loss of reactor coolant. The four Core Standby Cooling Systems are:

1. High Pressure Coolant Injection System (HPCIS)

   The HPCIS provides and maintains an adequate coolant inventory inside the reactor vessel to prevent excessive fuel clad temperatures as a result of postulated small breaks in the Reactor Coolant Pressure Boundary (RCPB). A high pressure system is needed for such breaks because the reactor vessel depressurizes slowly, preventing low pressure systems from injecting coolant. The HPCIS includes a turbine driven pump powered by reactor steam. The system is designed to accomplish its function on a short-term basis without reliance on plant auxiliary power supplies other than the dc power supply.

2. Automatic Depressurization System (ADS)

The ADS acts to rapidly reduce reactor vessel pressure in a LOCA situation in which the HPCIS fails to automatically maintain reactor vessel water level. The depressurization provided enables the low pressure standby cooling systems to deliver cooling water to the reactor vessel. The ADS uses some of the safety-relief valves which are part of the Nuclear System Pressure Relief System. The automatic safety-relief valves are arranged to open when a break in the nuclear system process barrier has occurred and the HPCIS is not delivering sufficient cooling water to the reactor vessel to maintain the water level above a preselected value. The ADS will not be activated unless either the Core Spray or the Low Pressure Coolant Injection System Pumps are operating.

3. Core Spray System

The Core Spray System consists of two independent pump loops that deliver cooling water to spray spargers over the core. The system is actuated by conditions indicating that breach exists in the nuclear system process barrier, but water is delivered to the core only after reactor vessel pressure is reduced. This system provides the capability to cool the fuel by spraying water onto the core. Either core spray loop, together with another ECCS system, is capable of preventing excessive fuel clad temperatures following a LOCA.

4. Low Pressure Coolant Injection (LPCI)

Low Pressure Coolant Injection (LPCI) is an operating mode of the Residual Heat Removal System (RHRS) and is an engineered safety feature. LPCI uses the pump loops of the RHRS to inject cooling water at low pressure into a reactor recirculation loop. LPCI is actuated by conditions indicating a breach in the nuclear system process barrier, but water is delivered to the core only after reactor vessel pressure is reduced. LPCI operation, together with the core shroud and jet pump arrangement, provides the capability of core reflooding following a LOCA in time to prevent excessive fuel clad temperatures.

1.2.2.4.12 Residual Heat Removal System (Containment Cooling)

The Residual Heat Removal System (RHRS) for containment cooling is placed in operation to limit the temperature of the water in the suppression pool following a design basis LOCA. In the containment cooling mode of operation, the RHRS pumps take suction from the suppression pool and deliver the water through the RHRS heat exchangers, where cooling takes place by transferring heat to the RHR service water. The fluid is then discharged back to the suppression pool.

As an alternative, RHRSW can be aligned to an RHR heat exchanger when RHR is aligned in the LPCI operating mode to support long term containment cooling.

Another portion of the RHRS is provided to spray water into the primary containment as a means of reducing containment pressure following a LOCA. This capability is in excess of the required energy removal capability and can be placed into service at the discretion of the operator.

1.2.2.4.13 Control Rod Velocity Limiter

A control rod velocity limiter is a part of each control rod and limits the velocity at which a control rod can fall out of the core should it become detached from its control rod drive. The rate of reactivity insertion resulting from a rod drop accident is limited by this feature. The limiters contain no moving parts.

1.2.2.4.14 Control Rod Drive Housing Supports

Control rod drive housing supports are located underneath the reactor vessel near the control rod housings. The supports limit the travel of a control rod in the event that a control rod housing is ruptured. The supports prevent a nuclear excursion as a result of a housing failure, thus protecting the fuel barrier.

1.2.2.4.15 Reactor Building Recirculation and Standby Gas Treatment Systems

The Reactor Building Recirculation System and the Standby Gas Treatment System (SGTS) are both a part of the secondary containment. The recirculation system has the capability of recirculating the reactor building air volume prior to its discharge via the SGTS, following a LOCA. Under normal wind conditions, the SGTS has the capability of maintaining a negative pressure within the reactor building with respect to the outside atmosphere. The air moving through the SGTS is filtered and discharged through the turbine building exhaust vent.

1.2.2.4.16 Standby ac Power Supply

The Standby ac Power Supply System consists of four diesel-generator sets. The diesel-generators are sized so that three diesels can supply all the necessary power requirements for one unit in the design basis accident condition, plus the necessary required loads to effect the safe shutdown of the second unit. The diesel generators are specified to start up and attain rated voltage and frequency within 10 seconds. Four independent 4 kV engineered safety feature switchgear assemblies are provided for each reactor unit. Each diesel-generator feeds an independent 4 kV bus for each reactor unit.

Additionally, a spare diesel generator is provided which can be manually realigned as a replacement for any one of the other four diesel generators. This spare diesel generator has the emergency loading capability of any of the other four diesel generators.

Each diesel-generator starts automatically upon loss of off-site power or detection of a nuclear accident. The necessary engineered safety feature system loads are applied in a preset time sequence. Each generator operates independently and without paralleling during a loss of off-site power or LOCA signal.

1.2.2.4.17 dc Power Supply

Each reactor unit is provided with four independent 125 V and two independent 250 V dc systems. Each dc system is supplied from a separate battery bank and battery charger. The 125 V dc systems are provided to supply station dc control power and dc power to four diesel generators and their associated switchgears. The 250 V dc systems are provided to supply power required for the larger loads such as dc motor driven pumps and valves.

Additionally, a separate 125V dc system is provided for the spare diesel generator. This separate 125V dc system is provided to supply dc control power and dc power to the spare diesel generator auxiliaries and its associated switchgear.

The 125 and 250-V dc Systems are designed to supply power adequate to satisfy the engineered safety feature load requirements of the unit with the postulated loss of off-site power and any concurrent single failure in the dc system.

1.2.2.4.18 Residual Heat Removal Service Water System

A Residual Heat Removal Service Water System is provided to remove the heat rejected by the Residual Heat Removal System during shutdown operation and accident conditions.

1.2.2.4.19 Emergency Service Water System

The Emergency Service Water System supplies water to cool the standby diesel-generators and the ECCS and Engineered Safety Features equipment rooms, and other essential heat loads.

1.2.2.4.20 Main Steam Line Radiation Monitoring System

The Main Steam Line Radiation Monitoring System consists of four gamma radiation monitors located external to the main steam lines just outside of the primary containment. The monitors are designed to detect a gross release of fission products from the fuel. Upon detection of high radiation, an alarm signal is initiated. A trip signal to the Mechanical Vacuum Pump (MVP) and its suction valves is generated by the monitors upon detection of a high high radiation signal.

1.2.2.4.21 Reactor Building Ventilation Radiation Monitoring System

The Reactor Building Ventilation Radiation Monitoring System consists of a number of radiation monitors arranged to monitor the activity level of the ventilation exhaust from the reactor building. Upon detection of high radiation, the reactor building is automatically isolated and the Standby Gas Treatment System is started.

1.2.2.4.22 Nuclear Leak Detection System

The Nuclear Leak Detection System consists of temperature, pressure, flow, and fission-product sensors with associated instrumentation and alarms. This system detects and annunciates leakage in the following systems:

1) Main steam lines

2) Reactor water cleanup (RWCU) system

3) Residual heat removal (RHR) system

4) Reactor core isolation cooling (RCIC) system

5) High pressure coolant injection (HPCI) system

6) Instrument lines

Small leaks generally are detected by temperature and pressure changes, fillup rate of drain sumps, and fission product concentration inside the primary containment. Large leaks are also detected by changes in reactor water level and changes in flow rates in process lines.

1.2.2.5 Instrumentation and Control

1.2.2.5.1 Nuclear System Process Control and Instrumentation

1.2.2.5.1.1 Reactor Manual Control System

The Reactor Manual Control System provides the means by which control rods are manipulated from the control room for gross power control. The system controls valves in the Control Rod Drive Hydraulic System. Only one control rod can be manipulated at a time. The Reactor Manual Control System includes the controls that restrict control rod movement (rod block) under certain conditions as a backup to procedural controls.

1.2.2.5.1.2 Recirculation Flow Control System

The Recirculation Flow Control System controls the speed of the reactor recirculation pumps. Adjusting the pump speed changes the coolant flow rate through the core. This effects changes in core power level. The system is arranged to automatically adjust reactor power output to the load demand by adjusting the frequency of the electrical power supply for the reactor recirculation pumps.

1.2.2.5.1.3 Neutron Monitoring System

The Neutron Monitoring System is a system of in-core neutron detectors and out-of-core electronic monitoring equipment. The system provides indication of neutron flux, which can be correlated to thermal power level for the entire range of flux conditions that may exist in the core. The system also provides detection of neutron flux oscillations, which may indicate thermal-hydraulic instability. The source range monitors (SRM) and the intermediate range monitors (IRM) provide flux level indications during reactor startup and low power operation. The local power range monitors (LPRM), oscillation power range monitors (OPRM) and average power range monitors (APRM) allow assessment of local and overall flux conditions during power range operation. The average power range monitors also provide post-accident neutron flux information. A rod block monitor (RBM) is provided to prevent rod withdrawal when reactor power should not be increased at the existing reactor coolant flow rate. The Traversing In-core Probe System (TIPS) provides a means to calibrate the individual LPRM's.

1.2.2.5.1.4 Refueling Interlocks

A system of interlocks, restricting the movements of refueling equipment and control rods when the reactor is in the refuel mode, is provided to prevent an inadvertent criticality during refueling operations. The interlocks back up procedural controls that have the same objective. The interlocks affect the refueling bridge, the refueling bridge hoists, the fuel grapple and control rods.

1.2.2.5.1.5 Reactor Vessel Instrumentation

In addition to instrumentation provided for the nuclear safety systems and engineered safety features, instrumentation is provided to monitor and transmit information that can be used to

assess conditions existing inside the reactor vessel and the physical condition of the vessel itself. The instrumentation provided monitors reactor vessel pressure, water level, temperature, internal differential pressures and coolant flow rates, and top head flange leakage.

1.2.2.5.1.6 Process Computer System

An on-line process computer is provided to monitor and log process variables and to make certain analytical computations. In conjunction with approved operating procedure, the rod worth minimizer function prevents improper rod withdrawal under low power conditions. The effect of the rod worth minimizer function is to limit the reactivity worth of the control rods by enforcing adherence to the preplanned rod pattern.

1.2.2.5.1.7 Remote Shutdown System

A Remote Shutdown Panel and associated procedures are provided for each unit so that the plant can be maintained in a safe shutdown condition in the event that the main control room becomes uninhabitable.

1.2.2.5.2 Power Conversion Systems Process Control and Instrumentation

1.2.2.5.2.1 Pressure Regulator and Turbine Control

The pressure regulator maintains control of turbine control valves; it regulates pressure at the turbine inlet and, therefore, the pressure of the entire nuclear system. Pressure regulation is coordinated with the turbine speed system and the load control system so that rapid control valve closure can be initiated when necessary to provide turbine overspeed protection for large load rejection.

1.2.2.5.2.2 Feedwater System Control

A three-element control system regulates the Feedwater System so that proper water level is maintained in the reactor vessel. Signals used by the control system are main steam flow rate, reactor vessel water level, and feedwater flow rate. The feedwater control signal is used to control the speed of the steam turbine-driven feedwater pumps.

1.2.2.5.2.3 Electrical Power System Control

Controls for the electrical power system are located in the control room to permit safe startup, operation, and shutdown of the plant.

1.2.2.6 Electrical Systems

1.2.2.6.1 Transmission and Generation Systems

Redundant sources of off-site power are provided to each unit by separate transmission lines to ensure that no single failure of any active component can prevent a safe and orderly shutdown. The two independent off-site sources provide auxiliary power for startup and for operating the engineered safety feature systems.

The main generator for each unit is an 1800-rpm, three-phase, 60-cycle synchronous machine rated at 1354 MVA. Each generator is connected directly to the turbine shaft and is equipped with an excitation system coupled directly to the generator shaft.

Power from the generators is stepped up from 24 kV to 230 kV on Unit No. 1 and from 24 kV to 500 kV on Unit No. 2 by the unit main transformers and supplied by overhead lines to the 230 kV and 500 kV switchyards, respectively.

1.2.2.6.2 Electric Power Distribution Systems

The electric power distribution system includes Class 1E and non-Class 1E ac and dc power systems. The class 1E power system supplies all safety related equipment and some non-class 1E loads while the non-Class 1E system supplies the balance of plant equipment.

The Class 1E ac system for each unit consists of four independent load groups. Two independent off-site power systems provide the normal electric power to these groups. Each load group includes 4.16 kV switchgear, 480 V load centers, motor control centers and 120 V control and instrument power panel. The vital ac instrumentation and control power supply systems include battery systems, static inverters. Voltages listed are nominal values, and all electrical equipment essential to safety is designed to accept a range of ±10 percent in voltage.

Four independent diesel generators are shared between the two units. Additionally, a spare diesel generator is provided which can be manually realigned as a replacement for any one of the other four diesel generators. This spare diesel generator has the emergency loading capability of any of the other four diesel generators. Each diesel generator is provided as a standby source of emergency power for one of the four Class 1E ac load groups in each unit. Assuming the total loss of off-site power and failure of one diesel generator, the remaining diesel generators have sufficient capacity to operate all the equipment necessary to prevent undue risk to public health and safety in the event of a design basis accident on one unit and a forced shutdown of the second unit.

The non-Class 1E ac system includes 13.8 kV switchgear, 4.16 kV switchgear, 480 V load centers and motor control centers.

Four independent Class 1E 125V dc batteries and two independent Class 1E 250V dc batteries and associated battery chargers provide direct current power for the Class 1E dc loads of each unit. Power for non-Class 1E dc loads is supplied from the Class 1E 125 and 250 V batteries. An additional circuit breaker is provided for each non-class 1E load connected to the class 1E system for redundant fault protection.

Additionally, a separate 125V dc system is provided for the spare diesel generator. This separate 125V dc system is provided to supply dc control power and dc power to spare diesel generator auxiliaries and its associated switchgear.

These systems are discussed in Chapter 8.

1.2.2.7 Fuel Handling and Storage Systems

1.2.2.7.1 New and Spent Fuel Storage Security-Related Information. Withheld Under 10 CFR 2.390

[redacted]

[redacted]

1.2.2.7.2 Fuel Pool Cooling and Cleanup System

A Fuel Pool Cooling and Cleanup System is provided to remove decay heat from spent fuel stored in the fuel pool and to maintain specified water temperature, purity, clarity, and level.

1.2.2.7.3 Fuel Handling Equipment

The major fuel servicing and handling equipment includes the reactor building cranes, the refueling service platform, fuel and control rod servicing tools, fuel sipping and inspection devices, and other auxiliary servicing tools.

1.2.2.8 Cooling Water and Auxiliary Systems

1.2.2.8.1 Service Water System

The Service Water System is designed to

a) Furnish cooling water to various heat exchangers located in the several plant buildings

b) Furnish water for diluting the oxidizing and non-oxidizing biocides and for injecting them into the circulating water systems. This is an intermittent service.

The system consists of three 50 percent capacity pumps with associated piping and valves. The cooling water supply to the pumps is taken from the cooling tower basin while the water being returned from the system is discharged into the cooling tower that acts as the heat sink. Equipment that requires service water and is common to Units 1 and 2 is provided with inter-ties to both service water systems so that either can provide the water.

1.2.2.8.2 Residual Heat Removal Service Water System (RHRSWS)

The objective of the RHRSWS is to provide a reliable supply of cooling water for heat removal from the Residual Heat Removal System under post-accident conditions and supply a source of water if post-accident flooding of the core or primary containment is required.

The system consists of two independent loops per unit, each of 100 percent capacity, and each loop consisting of two pumps, valves, piping and controls. Each loop uses the common spray pond with its spray distribution network as a heat sink.

During operation the pumps take water from the spray pond and circulate it through the tube side of the RHR heat exchangers. The warm water is returned to the spray pond through a network of spray nozzles that produce the cooling effect by causing an enthalpy gradient as a result of the convective heat transfer and partial evaporative cooling of the spray droplets. A radiation monitor is provided to check the radioactivity of the service water leaving each RHR heat exchanger. In the event of a high activity level (a tube leak in the RHR heat exchanger), an alarm will sound and the operator will make the decision to isolate the heat exchanger and minimize the volume of contaminated water that flows to the spray pond.

1.2.2.8.3 Emergency Service Water System (ESWS)

The objective of the ESWS is to supply cooling water to the RHR pumps and associated room coolers during normal and emergency conditions, as necessary, to safely shutdown the reactor or support normal and emergency conditions, as necessary, to safely shutdown the reactor or support "hot standby" conditions and in addition supply cooling water to the Diesel-Generator Units. The ESWS provides a reliable supply of cooling water to emergency equipment under a loss of off-site power condition or LOCA. The system consists of two independent loops supplying both units supplying both units (denoted "A" and "B") each of 100 percent capacity and containing two pumps, valves, piping and controls. Each loop uses the spray pond with its spray distribution system (common to both the Emergency Service Water and RHR Service Water Systems) as a heat sink. The ESWS is designed with sufficient redundancy so that no single active or passive system component failure can prevent it from achieving its safety objective. During operation, the ESWS pumps take water from the spray pond and circulate it through the various heat exchangers in the system. The warm water is returned to the spray pond through either a network of spray nozzles or directly through piping that bypass the spray arrays. The spray nozzles produce the cooling effect by causing an enthalpy gradient as a result of the convective heat transfer and partial evaporative cooling of the spray droplets.

1.2.2.8.4 Reactor Building Closed Cooling Water System

The Reactor Building Closed Cooling Water System is designed to accomplish the following objectives:

a) Provide cooling water to auxiliary plant equipment associated with the nuclear system and located in the reactor and radwaste buildings.

b) Provide cooling water to reactor building chilled water system in the event of unavailability of the chillers or loss of off-site power.

The Reactor Building Closed Cooling Water System consists of two 100 percent capacity pumps, two 100 percent capacity heat exchangers, a head tank, chemical addition tank, associated piping, valves and controls. The reactor building cooling water system is a closed loop cooling water system using inhibited demineralized water. The systems for Units 1 and 2 are separate from each other.

During normal plant operation one pump and heat exchanger will be in service, transferring heat to the service water system, with the other pump on automatic standby. Upon complete loss of off-site power, without occurrence of DBA, both cooling water pumps will start automatically when the buses are re-energized by Diesel Generators. The reactor building closed cooling water heat exchangers can be transferred from service water to emergency service water and one pump can be taken out of service, both by remote manual switching.

1.2.2.8.5 Turbine Building Closed Cooling Water System (TBCCWS)

The TBCCWS is designed to provide cooling water to the auxiliary plant equipment associated with the nuclear and power conversion systems in the Turbine Building. The TBCCWS consists of two 100 percent capacity pumps, two 100 percent capacity heat exchangers, a head tank, chemical addition tank, associated piping and valves. The Turbine Building Cooling water System is a closed loop cooling water system using inhibited demineralized water. The systems for Units 1 and 2 are separate from each other. During normal plant operation, the turbine building closed cooling water heat exchanger transfers heat from the Turbine Building Closed Cooling Water System to the Service Water system. After a loss of off-site power, the pumps start automatically the pumps start automatically and the turbine building closed cooling water heat exchangers can be transferred by remote switching to Emergency Service Water System. The heat load during this period will be rejected to the emergency service water. One turbine building closed cooling water pump and heat exchanger will be normally in service and the other pump will be on automatic standby.

1.2.2.8.6 Standby Liquid Control System

Although not intended to provide rapid reactor shutdown, the Standby Liquid Control System provides a redundant, independent, and alternative method to the control rods to bring the reactor subcritical and to maintain it subcritical as the reactor cools. The system makes possible an orderly and safe shutdown if not enough control rods can be inserted into the reactor core to accomplish normal shutdown. The system is sized to counteract the positive reactivity effect from rated power to the cold shutdown condition.

The system will also be used to buffer suppression pool pH to prevent iodine re-evolution following a postulated design basis loss of coolant accident.

1.2.2.8.7 Fire Protection System

A Fire Protection System supplies fire fighting water to points throughout the plant. In addition, automatic Halon and carbon dioxide protection systems and portable fire extinguishers are also provided.

1.2.2.8.8 Plant Heating, Ventilating, and Air-Conditioning Systems

The Plant Heating, Ventilating, and Air-Conditioning Systems supply and circulate filtered fresh air for personnel comfort and equipment cooling.

1.2.2.8.9 Compressed Air System

The Compressed Air Systems (e.g., instrument air, service air and containment instrument air) supply air of suitable quality and pressure for various plant operations.

1.2.2.8.10 Makeup Water Treatment System

A Makeup Water Treatment System furnishes a supply of treated water suitable for use as makeup for the plant.

1.2.2.8.11 Domestic and Sanitary Water Systems

A water system for drinking and sanitary uses is provided for the plant.

1.2.2.8.12 Plant Equipment and Floor Drainage Systems

The Plant Equipment and Floor Drainage System handles both radioactive and non-radioactive drains. Drains which may contain radioactive materials are pumped to the radwaste system for cleanup, reuse, or discharge. Non-radioactive drains are discharged to the environs.

1.2.2.8.13 Process Sampling System

The Process Sampling System is provided to monitor the operation of plant equipment and to provide information needed to make operational decisions.

1.2.2.8.14 Plant Communication System

The Plant Communication System provides communication between various plant buildings and locations.

1.2.2.8.15 Process Valve Stem Leakoff System

The Process valve stem leak-off collection system is designed to reduce and control leakage to the atmosphere from valves greater than 2 1/2 in. that are used in the turbine building in systems containing radioactive steam or water and not connected to the main condenser.

Valves in the turbine building were originally provided with valve stem packing leakoff connections. Research and testing has shown that improved packing provides an effective seal to prevent leakage into the Turbine Building. As a result, these leakoff connections are in the process of being removed and package configurations changed, as appropriate, to conform with the new requirements. As part of this effort, leakoff isolation valves and piping will be removed (or abandoned in place) and the leakoff collection header piping will be removed or abandoned in place.

1.2.2.8.16 Diesel Auxiliary Systems

Diesel auxiliary systems are those systems which directly support operation of the emergency diesel generators. The following are diesel auxiliary systems:

a) Diesel Generator Fuel Oil Storage and Transfer System

b) Diesel Generator Cooling Water System

c) Diesel Generator Starting System

d) Diesel Generator Lubrication System

e) Diesel Generator Combustion Air Intake and Exhaust System

1.2.2.8.17 Auxiliary Steam System

The auxiliary steam system consists of two electrode steam boilers and auxiliary equipment. The system is designed to provide flexibility for accommodating varying steam demands during all operating modes.

1.2.2.9 Power Conversion System

1.2.2.9.1 Turbine-Generator

The turbine-generator consists of the turbine, generator, exciter, controls, and required subsystems designed for a nominal plant rating output of 1300 MWe for both Unit 1 and Unit 2.

Each turbine is an 1800 rpm, tandem-compound, six-flow, non-reheat unit with an electrohydraulic control system. The main turbine comprises one double-flow high pressure turbine and three double-flow low pressure turbines. Exhaust steam from the high pressure turbine passes through moisture separators before entering the three low pressure turbines

The generator is a direct-driven, three-phase, 60 Hz, 24,000 V, 1800 rpm, conductor-cooled, synchronous generator rated on the basis of guaranteed best turbine efficiency MW rating at 0.935 power factor, 75 psig hydrogen pressure. The generator-exciter system is shaft-driven, complete with static type voltage regulator and associated switchgear. The following are the turbine generator auxiliary systems:

a) Generator Hydrogen System
b) Generator Seal Oil System
c) Turbine Lube Oil System
d) Steam Seal System
e) Gland Exhaust System
f) Generator Stator Cooling System

1.2.2.9.2 Main Steam System

The main steam system delivers steam from the nuclear boiler system via four 24 in. OD steam lines to the turbine-generator. This system also supplies steam to the steam jet air ejectors, the reactor feed pump turbines, the main condenser hotwell at startup and low loads, and the steam seal evaporator.

1.2.2.9.3 Main Condenser

The main condenser is a triple pass, triple-pressure, deaerating type with a reheating-deaerating hotwell and divided water boxes. The condenser consists of three sections, and each section is located below one of three low-pressure turbines. The condensers are supported on the turbine foundation mat, with rubber expansion joints provided between each turbine exhaust opening and the steam inlet connections in the condenser shells.

During normal operation, steam from the low pressure turbine is exhausted directly downward into the condenser shells through exhaust openings in the bottom of the turbine casings and is condensed. The condenser also serves as a heat sink for several other flows, such as exhaust steam from feed condenser drain, gland seal condenser drain, feedwater heater shell operating vents, and condensate pump suction vents.

During abnormal conditions the condenser is designed to receive (not simultaneously) turbine bypass steam, feedwater heater high level dump(s), and relief valve discharge (from crossover steam lines, feedwater heater shells, steam seal regulator, and various steam supply lines).

Other flows occur periodically; they originate from condensate pump and reactor feed pump startup vents, reactor feed pump and condensate pump minimum recirculation flows, feedwater line startup flushing, turbine equipment clean drains, low point drains, deaerating steam, makeup, condensate, etc.

1.2.2.9.4 Main Condenser Gas Removal System

The main condenser Gas Removal System removes the non-condensable gases from the main condenser and exhausts them to the Off-Gas System. One steam jet air ejector (100 percent capacity), is provided for the removal of air and radiolysis gases during normal operation. One motor-driven mechanical vacuum pump is to establish or maintain vacuum during startup and shutdown.

1.2.2.9.5 Steam Seal System

The steam seal system provides clean, non-radioactive steam to the seals of the turbine valve packings and the turbine shaft packings. The sealing steam is supplied by the seal steam evaporator. The auxiliary boiler provides an auxiliary steam supply for startup and when the seal steam evaporator is not operating.

1.2.2.9.6 Steam Bypass and Pressure Control System

The turbine steam bypass and pressure control system control the reactor pressure for the following operating modes:

a) During reactor heatup to rated pressure.

b) While the turbine is being brought up to speed and synchronized.

c) During transient power operation when the reactor steam generation exceeds the turbine steam requirements.

d) When cooling down the reactor.

1.2.2.9.7 Circulating Water System

The Circulating Water system is a closed loop system designed to circulate the flow of water required to remove the heat load from the main condenser and auxiliary heat exchanger equipment and discharge it to the atmosphere through a natural draft cooling tower.

1.2.2.9.8 Condensate Cleanup System

The function of the Condensate Cleanup System is to maintain the required purity of the feedwater flowing to the reactor.

The system consists of full flow deep bed demineralizers using ion exchange resins which remove dissolved and a portion of the suspended solids from the feedwater to maintain the purity necessary for the reactor. The demineralizers will also remove some of the radioactive material produced by corrosion as well as fission product carryover from the reactor. The radioactivity from these sources does not have a significant effect on the resins.

1.2.2.9.9 Condensate and Feedwater System

The Condensate and Feedwater System is designed to deliver the required feedwater flow to the reactor vessels during stable and transient operating conditions throughout the entire operating range from startup to full load to shutdown. The system operates using four condensate pumps to pump deaerated condensate from the hotwell of the main condenser through the steam jet air ejector condenser, the gland steam condenser, the condensate filters, and thence to the condensate demineralizer. The demineralized feedwater then flows through three parallel strings of feedwater heaters, each string consisting of five heaters, to the suction of three reactor feed pumps which deliver the feedwater to the reactor.

1.2.2.9.10 Condensate and Refueling Water Storage and Transfer System

The function of the Condensate and Refueling Water Storage and Transfer System is to store condensate to be used as follows:

a) Supply water for the RCIC and HPCI systems.

b) Maintain the required condensate level in the hotwell either by receiving excess condensate rejected from the main condensate system or by supplying condensate to the main condensate system to makeup for a deficiency.

c) Fill up the reactor well of either reactor during refueling and receive this water back for storage after it has been cleaned up by the demineralizer.

d) Provide condensate where required for miscellaneous equipment in the radwaste building and both reactor buildings.

The makeup to condensate storage tanks and the refueling storage tank is provided by the demineralized water storage tank.

1.2.2.10 Radioactive Waste Systems

The Radioactive Waste Systems are designed to confine the release of plant produced radioactive material to well within the limits specified in 10CFR20. Various methods are used to achieve this end, e.g. collection, filtration, holdup for decay, dilution and concentration.

1.2.2.10.1 Liquid Radwaste System

The Liquid Radwaste System collects, treats, stores, and disposes of all radioactive liquid wastes. These wastes are collected in sumps and drain tanks at various locations throughout the plant and then transferred to the appropriate collection tanks in the radwaste building prior to treatment, storage and disposal. Processed liquid wastes are returned to the Condensate System, packaged for offsite shipment, or discharged from the plant.

Equipment is selected, arranged, and shielded to permit operation, inspection, and maintenance within radiation allowances for personnel exposure. For example, tanks and processing equipment which will contain significant radiation sources are shielded and sumps, pumps, instruments, and valves are located in controlled access rooms or spaces. Processing equipment is selected and designed to require a minimum of maintenance.

Valving redundancy, instrumentation for detection, alarms of abnormal conditions, and procedural controls protect against the accidental discharge of liquid radioactive waste.

1.2.2.10.2 Solid Radwaste System

Solid wastes originating from nuclear system equipment are stored for radioactive decay in the fuel storage pool and prepared for reprocessing or off-site storage in approved shipping containers. Examples of these wastes are spent control rods, and in-core ion chambers.

Process solid wastes as applicable are collected, dewatered, solidified, packaged, and stored in shielded compartments prior to off-site shipment. Examples of these solid wastes are filter residue, spent resins, paper, air filters, rags, and used clothing.

If off-site shipment of solidified liners or dry active waste is not practicable, these items may be temporarily stored at the Low Level Radioactive Waste Holding Facility, as described in Section 11.6, provided they are packaged for off-site disposal.

1.2.2.10.3 Gaseous Radwaste System

Radioactive gaseous wastes are discharged to the reactor building vent via the Gaseous Radwaste System. This system provides hydrogen-oxygen recombination, filtration, and holdup of the off-gases to ensure a low rate of release from the reactor building vent.

The off-gases from the main condenser are the greatest source of gaseous radioactive waste. The treatment of these gases reduces the released activity to below permissible levels.

1.2.2.11 Radiation Monitoring and Control

1.2.2.11.1 Process Radiation Monitoring

Radiation monitors are provided on various lines to monitor for radioactive materials released to the environs via process liquids and gases or for detection of process system malfunctions. These monitors annunciate alarms and/or provide signals to initiate isolation and corrective actions.

1.2.2.11.2 Area Radiation Monitors

Radiation monitors are provided to monitor for abnormal radiation at various locations in the reactor building, turbine building, and radwaste building. These monitors annunciate alarms when abnormal radiation levels are detected.

1.2.2.11.3 Site Environs Radiation Monitors

Radiation monitors are provided outside the plant buildings to monitor radiation levels. These data are used for determining the contribution of plant operations to on-site and off-site radiation levels.

1.2.2.11.4 Liquid Radwaste System Control

Liquid wastes to be discharged are handled on a batch basis with protection against accidental discharge provided by procedural controls. Instrumentation, with alarms, to detect abnormal concentration of the radwastes, is provided.

1.2.2.11.5 Solid Radwaste Control

The Solid Radwaste System collects, treats, and prepares solid radioactive wastes for off-site shipment. Wastes are handled on a batch basis. Radiation levels of the various batches are determined by the operator.

1.2.2.11.6 Gaseous Radwaste System Control

The Gaseous Radwaste System is continuously monitored by the turbine building vent radiation monitor and the off-gas pre-treatment radiation monitor. A high level signal will annunciate alarms.

1.2.2.12 Shielding

Shielding is provided throughout the plant, as required, to reduce radiation levels to operating personnel and to the general public within the applicable limits set forth in 10CFR20 and 10CFR50. It is also designed to protect certain plant components from radiation exposure resulting in unacceptable alterations of material properties or activation.

1.2.2.13 Steam Dryer Storage

A separate Steam Dryer Storage Facility(SDSF) is provided within the plant protected area for the storage, shielding, and radioactive decay of replaced Reactor steam dryers. The steam dryers are cut in half and packaged into steel containers for storage in the SDSF. They are not considered as radioactive waste but are treated as irradiated plant equipment. The SDSF is a reinforced concrete vault with removable roof slab access only, meeting 10CFR20 dose limits.

1.2.2.14 FLEX Equipment Storage Building

A separate FLEX Equipment Storage Building is provided within the plant protected area for the storage of portable equipment needed to respond to a Beyond Design Basis External Event (BDBEE). B.5.b equipment (i.e., pumper truck, etc.) is also stored in this building. This is strictly an emergency equipment storage facility (no personnel occupancy amenities) constructed to meet all plant extreme environmental conditions (i.e., seismic, tornado, missile).

THIS FIGURE HAS BEEN
REPLACED BY DWG.
M-220, Sh. 1

FSAR REV. 65

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Figure 1.2-1 replaced by dwg.
M-220, Sh. 1

FIGURE 1.2-1, Rev. 57

AutoCAD Figure 1_2_1.doc

THIS FIGURE HAS BEEN
REPLACED BY DWG.
M-221, Sh. 1

FSAR REV. 65

SUSQUEHANNA STEAM ELECTRIC STATION
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Figure 1.2-2 replaced by dwg.
M-221, Sh. 1

FIGURE 1.2-2, Rev. 56

AutoCAD Figure 1_2_2.doc

THIS FIGURE HAS BEEN
REPLACED BY DWG.
M-222, Sh. 1

FSAR REV. 65

SUSQUEHANNA STEAM ELECTRIC STATION
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Figure 1.2-3 replaced by dwg.
M-222, Sh. 1

FIGURE 1.2-3, Rev. 48

AutoCAD Figure 1_2_3.doc

THIS FIGURE HAS BEEN
REPLACED BY DWG.
M-223, Sh. 1

FSAR REV. 65

SUSQUEHANNA STEAM ELECTRIC STATION UNITS 1 & 2 FINAL SAFETY ANALYSIS REPORT
Figure 1.2-4 replaced by dwg. M-223, Sh. 1 FIGURE 1.2-4, Rev. 48

AutoCAD Figure 1_2_4.doc

THIS FIGURE HAS BEEN
REPLACED BY DWG.
M-224, Sh. 1

FSAR REV. 65

SUSQUEHANNA STEAM ELECTRIC STATION UNITS 1 & 2 FINAL SAFETY ANALYSIS REPORT
Figure 1.2-5 replaced by dwg. M-224, Sh. 1 FIGURE 1.2-5, Rev. 48

AutoCAD Figure 1_2_5.doc

THIS FIGURE HAS BEEN
REPLACED BY DWG.
M-225, Sh. 1

FSAR REV. 65

SUSQUEHANNA STEAM ELECTRIC STATION UNITS 1 & 2 FINAL SAFETY ANALYSIS REPORT
Figure 1.2-6 replaced by dwg. M-225, Sh. 1 FIGURE 1.2-6, Rev. 48

AutoCAD Figure 1_2_6.doc

THIS FIGURE HAS BEEN
REPLACED BY DWG.
M-226, Sh. 1

FSAR REV. 65

SUSQUEHANNA STEAM ELECTRIC STATION
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Figure 1.2-7 replaced by dwg.
M-226, Sh. 1

FIGURE 1.2-7, Rev. 48

AutoCAD Figure 1_2_7.doc

THIS FIGURE HAS BEEN
REPLACED BY DWG.
M-227, Sh. 1

FSAR REV. 65

SUSQUEHANNA STEAM ELECTRIC STATION
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Figure 1.2-8 replaced by dwg.
M-227, Sh. 1

FIGURE 1.2-8, Rev. 48

AutoCAD Figure 1_2_8.doc

THIS FIGURE HAS BEEN
REPLACED BY DWG.
M-230, Sh. 1

FSAR REV. 65

SUSQUEHANNA STEAM ELECTRIC STATION
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Figure 1.2-9 replaced by dwg.
M-230, Sh. 1

FIGURE 1.2-9, Rev. 57

AutoCAD Figure 1_2_9.doc

THIS FIGURE HAS BEEN
REPLACED BY DWG.
M-231, Sh. 1

FSAR REV. 65

SUSQUEHANNA STEAM ELECTRIC STATION UNITS 1 & 2 FINAL SAFETY ANALYSIS REPORT
Figure 1.2-10 replaced by dwg. M-231, Sh. 1 FIGURE 1.2-10, Rev. 49

AutoCAD Figure 1_2_10.doc

THIS FIGURE HAS BEEN
REPLACED BY DWG.
M-232, Sh. 1

FSAR REV. 65

SUSQUEHANNA STEAM ELECTRIC STATION
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Figure 1.2-11 replaced by dwg.
M-232, Sh. 1

FIGURE 1.2-11, Rev. 48

AutoCAD Figure 1_2_11.doc

THIS FIGURE HAS BEEN
REPLACED BY DWG.
M-233, Sh. 1

FSAR REV. 65

SUSQUEHANNA STEAM ELECTRIC STATION
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Figure 1.2-12 replaced by dwg.
M-233, Sh. 1

FIGURE 1.2-12, Rev. 49

AutoCAD Figure 1_2_12.doc

THIS FIGURE HAS BEEN
REPLACED BY DWG.
M-234, Sh. 1

FSAR REV. 65

SUSQUEHANNA STEAM ELECTRIC STATION UNITS 1 & 2 FINAL SAFETY ANALYSIS REPORT
Figure 1.2-13 replaced by dwg. M-234, Sh. 1 FIGURE 1.2-13, Rev. 48

AutoCAD Figure 1_2_13.doc

THiS FIGURE HAS BEEN
REPLACED BY DWG.
M-235, Sh. 1

FSAR REV. 65

SUSQUEHANNA STEAM ELECTRIC STATION
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Figure 1.2-14 replaced by dwg.
M-235, Sh. 1

FIGURE 1.2-14, Rev. 48

AutoCAD Figure 1_2_14.doc

THiS FIGURE HAS BEEN
REPLACED BY DWG.
M-236, Sh. 1

FSAR REV. 65

SUSQUEHANNA STEAM ELECTRIC STATION UNITS 1 & 2 FINAL SAFETY ANALYSIS REPORT
Figure 1.2-15 replaced by dwg. M-236, Sh. 1 FIGURE 1.2-15, Rev. 48

AutoCAD Figure 1_2_15.doc

THIS FIGURE HAS BEEN
REPLACED BY DWG.
M-237, Sh. 1

FSAR REV. 65

SUSQUEHANNA STEAM ELECTRIC STATION
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Figure 1.2-16 replaced by dwg.
M-237, Sh. 1

FIGURE 1.2-16, Rev. 48

AutoCAD Figure 1_2_16.doc

THIS FIGURE HAS BEEN
REPLACED BY DWG.
M-240, Sh. 1

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SUSQUEHANNA STEAM ELECTRIC STATION
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Figure 1.2-17 replaced by dwg.
M-240, Sh. 1

FIGURE 1.2-17, Rev. 48

AutoCAD Figure 1_2_17.doc

THIS FIGURE HAS BEEN
REPLACED BY DWG.
M-241, Sh. 1

FSAR REV. 65

SUSQUEHANNA STEAM ELECTRIC STATION UNITS 1 & 2 FINAL SAFETY ANALYSIS REPORT
Figure 1.2-18 replaced by dwg. M-241, Sh. 1 FIGURE 1.2-18, Rev. 55

AutoCAD Figure 1_2_18.doc

THIS FIGURE HAS BEEN
REPLACED BY DWG.
M-242, Sh. 1

FSAR REV. 65

SUSQUEHANNA STEAM ELECTRIC STATION UNITS 1 & 2 FINAL SAFETY ANALYSIS REPORT
Figure 1.2-19 replaced by dwg. M-242, Sh. 1 FIGURE 1.2-19, Rev. 48

AutoCAD Figure 1_2_19.doc

THIS FIGURE HAS BEEN
REPLACED BY DWG.
M-243, Sh. 1

FSAR REV. 65

SUSQUEHANNA STEAM ELECTRIC STATION UNITS 1 & 2 FINAL SAFETY ANALYSIS REPORT
Figure 1.2-20 replaced by dwg. M-243, Sh. 1 FIGURE 1.2-20, Rev. 55

AutoCAD Figure 1_2_20.doc

THIS FIGURE HAS BEEN
REPLACED BY DWG.
M-244, Sh. 1

FSAR REV. 65

SUSQUEHANNA STEAM ELECTRIC STATION
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Figure 1.2-21 replaced by dwg.
M-244, Sh. 1

FIGURE 1.2-21, Rev. 56

AutoCAD Figure 1_2_21.doc

THIS FIGURE HAS BEEN
REPLACED BY DWG.
M-245, Sh. 1

FSAR REV. 65

SUSQUEHANNA STEAM ELECTRIC STATION UNITS 1 & 2 FINAL SAFETY ANALYSIS REPORT
Figure 1.2-22 replaced by dwg. M-245, Sh. 1 FIGURE 1.2-22, Rev. 48

AutoCAD Figure 1_2_22.doc

THIS FIGURE HAS BEEN
REPLACED BY DWG.
M-246, Sh. 1

FSAR REV. 65

SUSQUEHANNA STEAM ELECTRIC STATION
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Figure 1.2-23 replaced by dwg.
M-246, Sh. 1

FIGURE 1.2-23, Rev. 48

AutoCAD Figure 1_2_23.doc

THIS FIGURE HAS BEEN
REPLACED BY DWG.
M-247, Sh. 1

FSAR REV. 65

SUSQUEHANNA STEAM ELECTRIC STATION
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Figure 1.2-24 replaced by dwg.
M-247, Sh. 1

FIGURE 1.2-24, Rev. 48

AutoCAD Figure 1_2_24.doc

THIS FIGURE HAS BEEN
REPLACED BY DWG.
M-248, Sh. 1

FSAR REV. 65

SUSQUEHANNA STEAM ELECTRIC STATION UNITS 1 & 2 FINAL SAFETY ANALYSIS REPORT
Figure 1.2-25 replaced by dwg. M-248, Sh. 1 FIGURE 1.2-25, Rev. 48

AutoCAD Figure 1_2_25.doc

THIS FIGURE HAS BEEN
REPLACED BY DWG.
M-249, Sh. 1

FSAR REV. 65

SUSQUEHANNA STEAM ELECTRIC STATION
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Figure 1.2-26 replaced by dwg.
M-249, Sh. 1

FIGURE 1.2-26, Rev. 48

AutoCAD Figure 1_2_26.doc

THIS FIGURE HAS BEEN
REPLACED BY DWG.
M-250, Sh. 1

FSAR REV. 65

SUSQUEHANNA STEAM ELECTRIC STATION
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Figure 1.2-27 replaced by dwg.
M-250, Sh. 1

FIGURE 1.2-27, Rev. 48

AutoCAD Figure 1_2_27.doc

THIS FIGURE HAS BEEN
REPLACED BY DWG.
M-251, Sh. 1

FSAR REV. 65

SUSQUEHANNA STEAM ELECTRIC STATION UNITS 1 & 2 FINAL SAFETY ANALYSIS REPORT
Figure 1.2-28 replaced by dwg. M-251, Sh. 1 FIGURE 1.2-28, Rev. 56

AutoCAD Figure 1_2_28.doc

THIS FIGURE HAS BEEN
REPLACED BY DWG.
M-252, Sh. 1

FSAR REV. 65

SUSQUEHANNA STEAM ELECTRIC STATION
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Figure 1.2-29 replaced by dwg.
M-252, Sh. 1

FIGURE 1.2-29, Rev. 48

AutoCAD Figure 1_2_29.doc

THIS FIGURE HAS BEEN
REPLACED BY DWG.
M-253, Sh. 1

FSAR REV. 65

SUSQUEHANNA STEAM ELECTRIC STATION
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Figure 1.2-30 replaced by dwg.
M-253, Sh. 1

FIGURE 1.2-30, Rev. 55

AutoCAD Figure 1_2_30.doc

THIS FIGURE HAS BEEN
REPLACED BY DWG.
M-254, Sh. 1

FSAR REV. 65

SUSQUEHANNA STEAM ELECTRIC STATION
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Figure 1.2-31 replaced by dwg.
M-254, Sh. 1

FIGURE 1.2-31, Rev. 55

AutoCAD Figure 1_2_31.doc

THIS FIGURE HAS BEEN
REPLACED BY DWG.
M-255, Sh. 1

FSAR REV. 65

SUSQUEHANNA STEAM ELECTRIC STATION
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Figure 1.2-32 replaced by dwg.
M-255, Sh. 1

FIGURE 1.2-32, Rev. 48

AutoCAD Figure 1_2_32.doc

THIS FIGURE HAS BEEN
REPLACED BY DWG.
M-256, Sh. 1

FSAR REV. 65

SUSQUEHANNA STEAM ELECTRIC STATION
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Figure 1.2-33 replaced by dwg.
M-256, Sh. 1

FIGURE 1.2-33, Rev. 48

AutoCAD Figure 1_2_33.doc

THIS FIGURE HAS BEEN
REPLACED BY DWG.
M-257, Sh. 1

FSAR REV. 65

SUSQUEHANNA STEAM ELECTRIC STATION
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Figure 1.2-34 replaced by dwg.
M-257, Sh. 1

FIGURE 1.2-34, Rev. 48

AutoCAD Figure 1_2_34.doc

THIS FIGURE HAS BEEN
REPLACED BY DWG.
M-258, Sh. 1

FSAR REV. 65

SUSQUEHANNA STEAM ELECTRIC STATION
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Figure 1.2-35 replaced by dwg.
M-258, Sh. 1

FIGURE 1.2-35, Rev. 48

AutoCAD Figure 1_2_35.doc

THIS FIGURE HAS BEEN
REPLACED BY DWG.
M-259, Sh. 1

FSAR REV. 65

SUSQUEHANNA STEAM ELECTRIC STATION UNITS 1 & 2 FINAL SAFETY ANALYSIS REPORT
Figure 1.2-36 replaced by dwg. M-259, Sh. 1 FIGURE 1.2-36, Rev. 48

AutoCAD Figure 1_2_36.doc

THIS FIGURE HAS BEEN
REPLACED BY DWG.
M-260, Sh. 1

FSAR REV. 65

SUSQUEHANNA STEAM ELECTRIC STATION UNITS 1 & 2 FINAL SAFETY ANALYSIS REPORT
Figure 1.2-37 replaced by dwg. M-260, Sh. 1 FIGURE 1.2-37, Rev. 48

AutoCAD Figure 1_2_37.doc

THIS FIGURE HAS BEEN
REPLACED BY DWG.
M-261, Sh. 1

FSAR REV. 65

SUSQUEHANNA STEAM ELECTRIC STATION UNITS 1 & 2 FINAL SAFETY ANALYSIS REPORT
Figure 1.2-38 replaced by dwg. M-261, Sh. 1 FIGURE 1.2-38, Rev. 48

AutoCAD Figure 1_2_38.doc

THIS FIGURE HAS BEEN
REPLACED BY DWG.
M-270, Sh. 1

FSAR REV. 65

SUSQUEHANNA STEAM ELECTRIC STATION UNITS 1 & 2 FINAL SAFETY ANALYSIS REPORT
Figure 1.2-39 replaced by dwg. M-270, Sh. 1 FIGURE 1.2-39, Rev. 56

AutoCAD Figure 1_2_39.doc

THIS FIGURE HAS BEEN
REPLACED BY DWG.
M-271, Sh. 1

FSAR REV. 65

SUSQUEHANNA STEAM ELECTRIC STATION
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Figure 1.2-40 replaced by dwg.
M-271, Sh. 1

FIGURE 1.2-40, Rev. 48

AutoCAD Figure 1_2_40.doc

THIS FIGURE HAS BEEN
REPLACED BY DWG.
M-272, Sh. 1

FSAR REV. 65

SUSQUEHANNA STEAM ELECTRIC STATION UNITS 1 & 2 FINAL SAFETY ANALYSIS REPORT
Figure 1.2-41 replaced by dwg. M-272, Sh. 1 FIGURE 1.2-41, Rev. 48

AutoCAD Figure 1_2_41.doc

THIS FIGURE HAS BEEN
REPLACED BY DWG.
M-273, Sh. 1

FSAR REV. 65

SUSQUEHANNA STEAM ELECTRIC STATION UNITS 1 & 2 FINAL SAFETY ANALYSIS REPORT
Figure 1.2-42 replaced by dwg. M-273, Sh. 1 FIGURE 1.2-42, Rev. 48

AutoCAD Figure 1_2_42.doc

THIS FIGURE HAS BEEN
REPLACED BY DWG.
M-274, Sh. 1

FSAR REV. 65

SUSQUEHANNA STEAM ELECTRIC STATION
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Figure 1.2-43 replaced by dwg.
M-274, Sh. 1

FIGURE 1.2-43, Rev. 48

AutoCAD Figure 1_2_43.doc

THIS FIGURE HAS BEEN
REPLACED BY DWG.
M-276, Sh. 1

FSAR REV. 65

SUSQUEHANNA STEAM ELECTRIC STATION
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Figure 1.2-44 replaced by dwg.
M-276, Sh. 1

FIGURE 1.2-44, Rev. 48

THIS FIGURE HAS BEEN
REPLACED BY DWG.
M-280, Sh. 1

FSAR REV. 65

SUSQUEHANNA STEAM ELECTRIC STATION
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Figure 1.2-45 replaced by dwg.
M-280, Sh. 1

FIGURE 1.2-45, Rev. 48

AutoCAD Figure 1_2_45.doc

THIS FIGURE HAS BEEN
REPLACED BY DWG.
M-281, Sh. 1

FSAR REV. 65

SUSQUEHANNA STEAM ELECTRIC STATION
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Figure 1.2-46 replaced by dwg.
M-281, Sh. 1

FIGURE 1.2-46, Rev. 48

AutoCAD Figure 1_2_46.doc

THIS FIGURE HAS BEEN
REPLACED BY DWG.
M-282, Sh. 1

FSAR REV. 65

SUSQUEHANNA STEAM ELECTRIC STATION
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Figure 1.2-47 replaced by dwg.
M-282, Sh. 1

FIGURE 1.2-47, Rev. 48

AutoCAD Figure 1_2_47.doc

THIS FIGURE HAS BEEN
REPLACED BY DWG.
M-284, Sh. 1

FSAR REV. 65

SUSQUEHANNA STEAM ELECTRIC STATION
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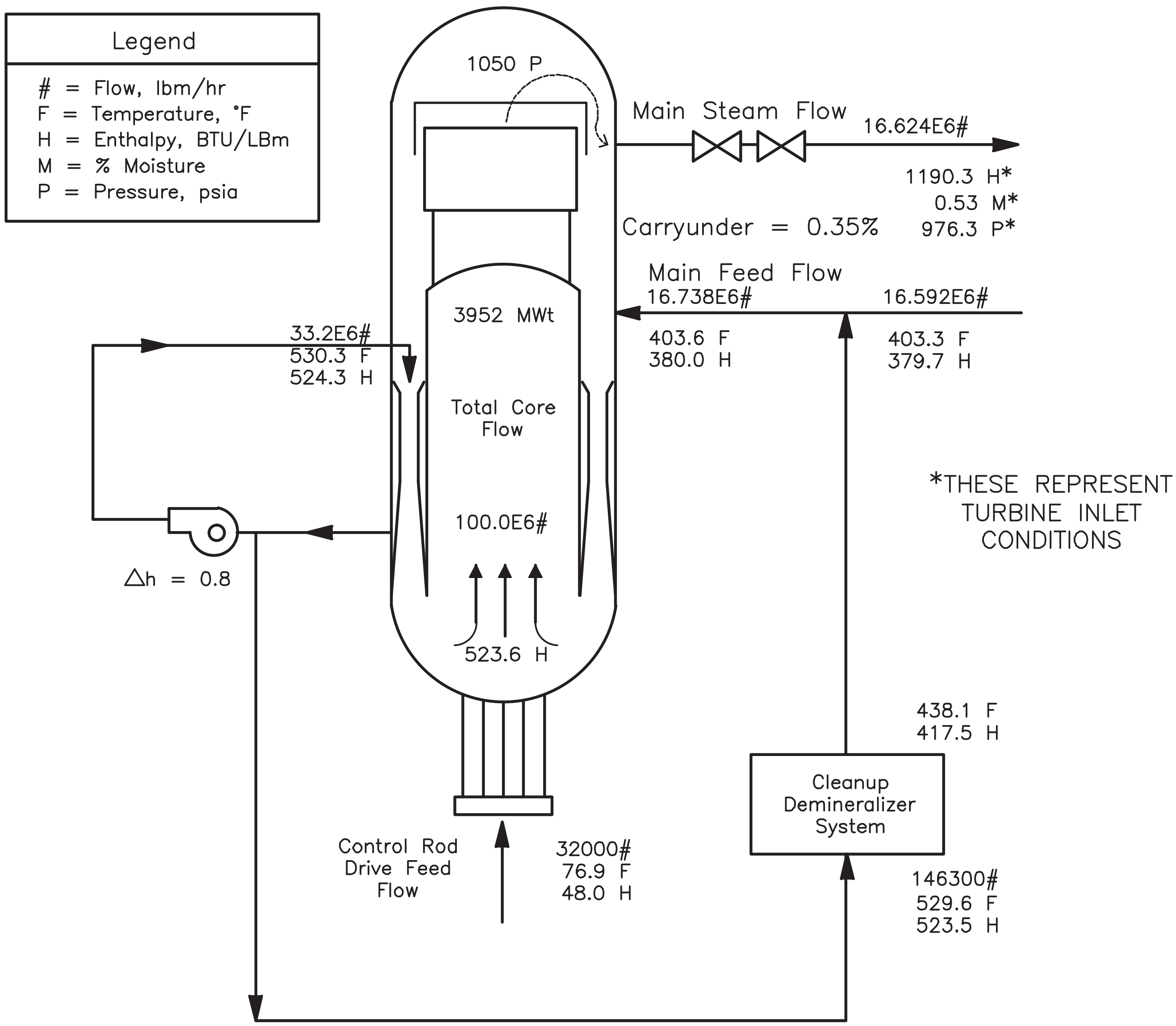
Figure 1.2-48 replaced by dwg.
M-284, Sh. 1

FIGURE 1.2-48, Rev. 48

AutoCAD Figure 1_2_48.doc

SSES — Reactor Heat Balance



Core Thermal Power	3952.0 MWt
Pump Heating	8.1
Cleanup Losses	−4.5
Other System Losses	−2.8
Turbine Cycle Use	3952.8 MWt

FSAR REV.67

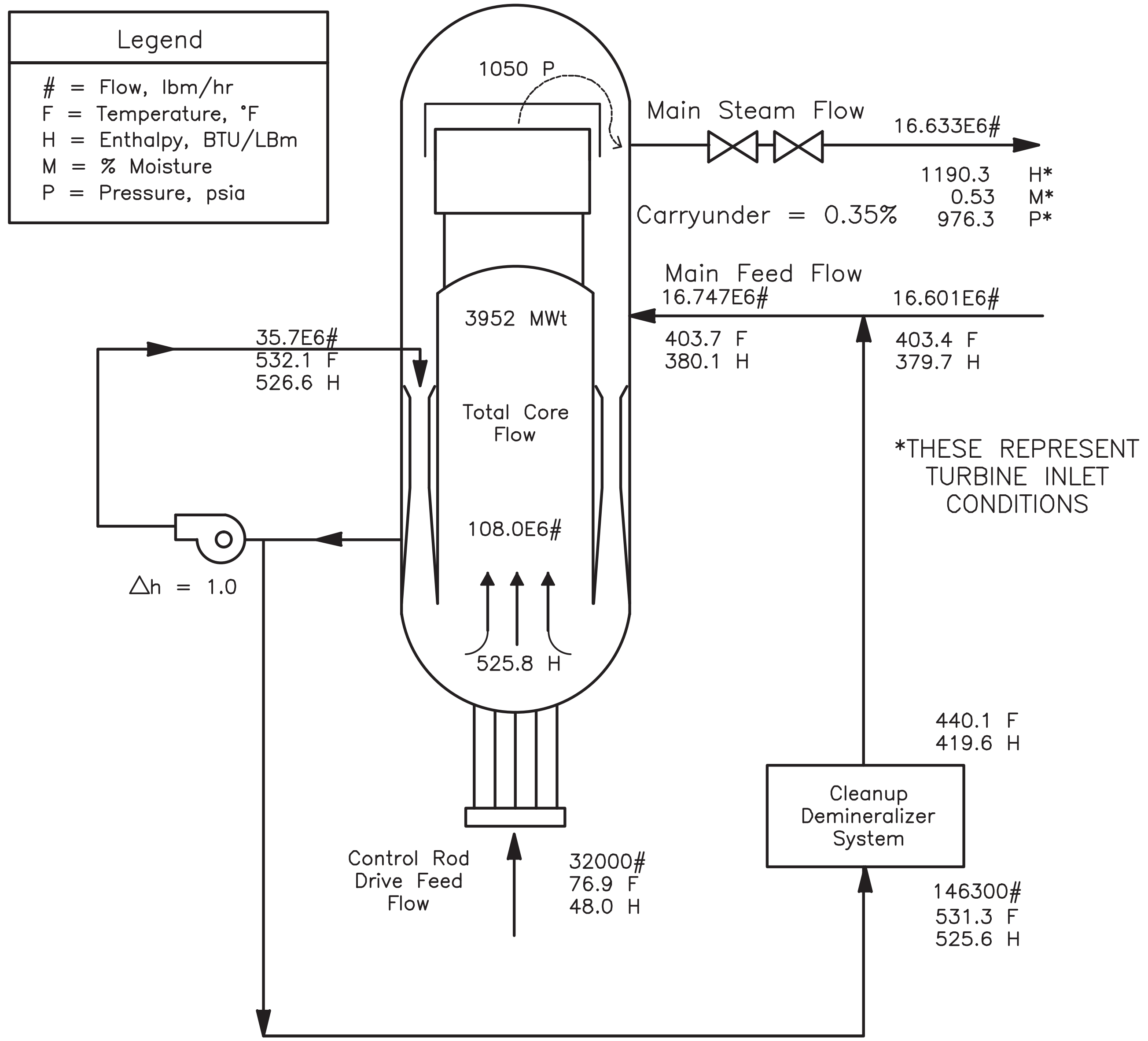
SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

UNIT 1
HEAT BALANCE AT RATED POWER
WITH 100 X 10^6LBm/hr CORE FLOW

FIGURE 1.2-49, Rev 60

AutoCAD: Figure Fsar 1_2_49.dwg

SSES — Reactor Heat Balance



Core Thermal Power	3952.0 MWt
Pump Heating	10.0
Cleanup Losses	−4.5
Other System Losses	−2.8
Turbine Cycle Use	3954.7 MWt

FSAR REV.67

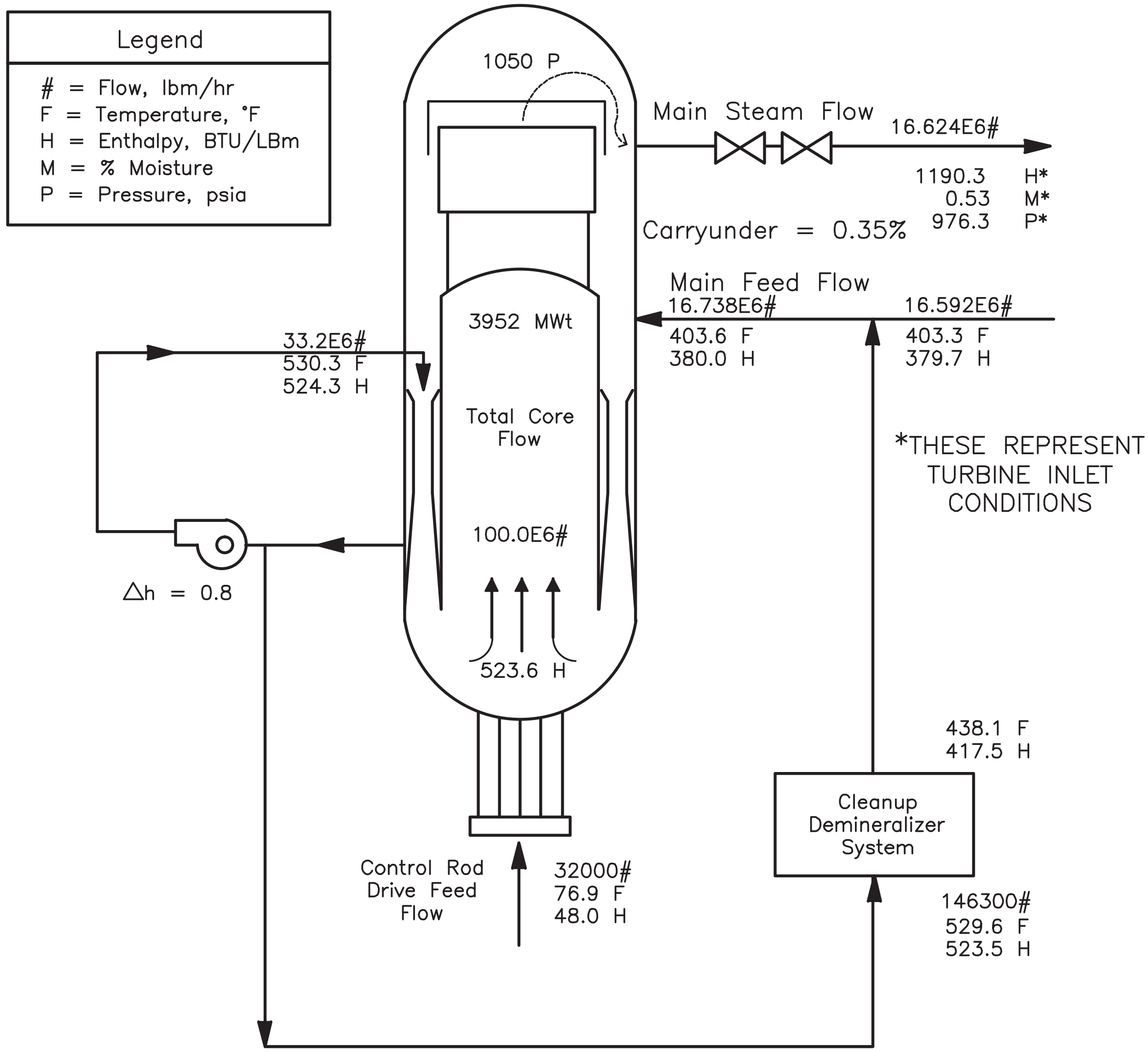
SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

UNIT 1
HEAT BALANCE AT RATED POWER
WITH 108 X 10^6 LBm/hr INCREASED
CORE FLOW

FIGURE 1.2-49-1, Rev 60

AutoCAD: Figure Fsar 1_2_49_1.dwg

SSES — Reactor Heat Balance



Core Thermal Power	3952.0 MWt
Pump Heating	8.1
Cleanup Losses	−4.5
Other System Losses	−2.8
Turbine Cycle Use	3952.8 MWt

FSAR REV.67

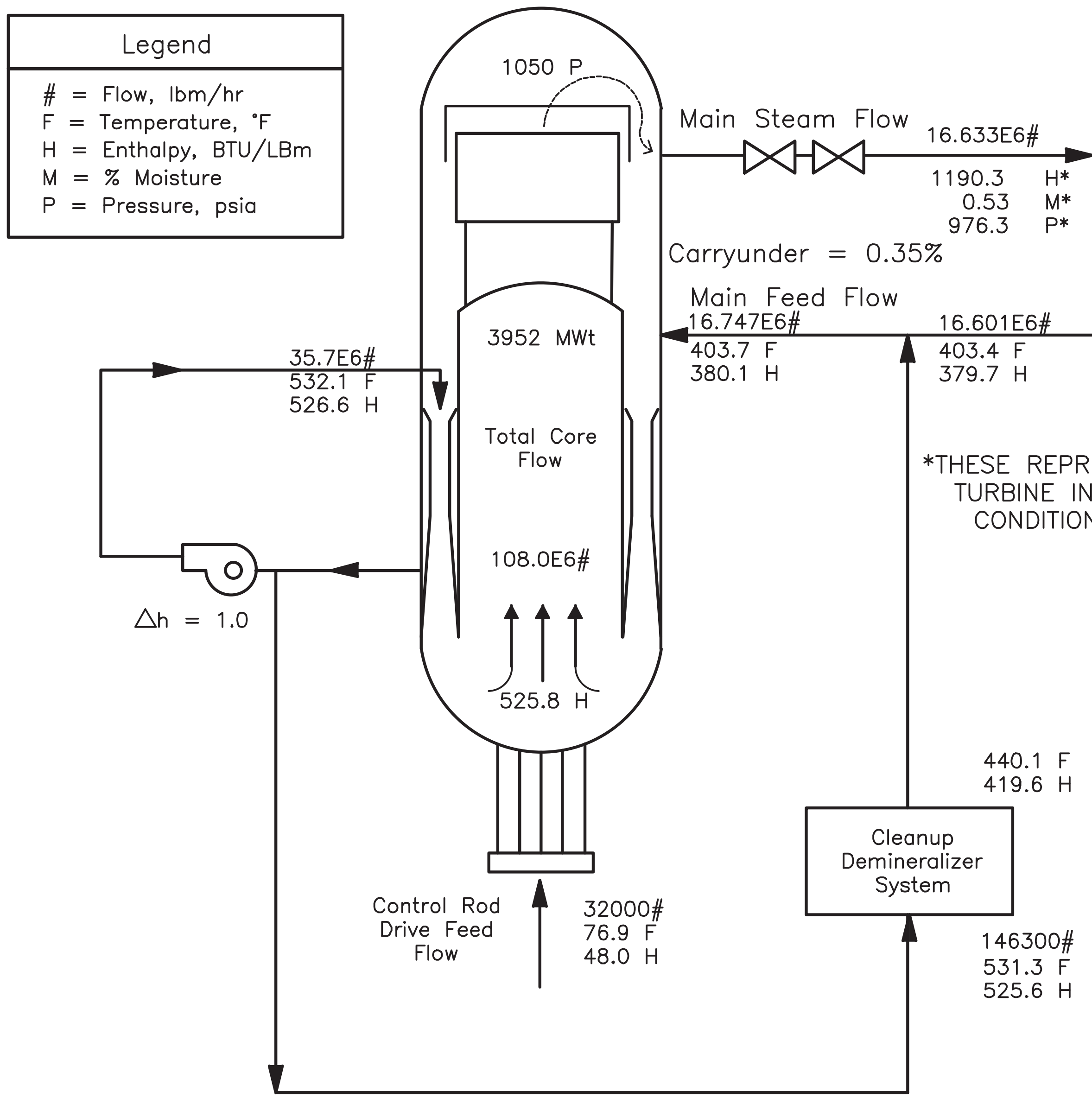
SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

UNIT2
HEAT BALANCE AT RATED POWER
WITH 100 X 10^6LBm/hr CORE FLOW

FIGURE 1.2-49-2, Rev 3

AutoCAD: Figure Fsar 1_2_49_2.dwg

SSES — Reactor Heat Balance



Core Thermal Power	3952.0 MWt
Pump Heating	10.0
Cleanup Losses	−4.5
Other System Losses	−2.8
Turbine Cycle Use	3954.7 MWt

FSAR REV.67

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

UNIT 2
HEAT BALANCE AT RATED POWER
WITH 108 X 10^6 LBm/hr INCREASED
CORE FLOW

FIGURE 1.2-49-3, Rev 3

AutoCAD: Figure Fsar 1_2_49_3.dwg

THIS FIGURE HAS BEEN
REPLACED BY DWG.
M-5200, Sh. 1

FSAR REV. 65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

Figure 1.2-50 replaced by dwg.
M-5200, Sh. 1

FIGURE 1.2-50, Rev. 52

AutoCAD Figure 1_2_50.doc

THIS FIGURE HAS BEEN
REPLACED BY DWG.
M-5200, Sh. 2

FSAR REV. 65

SUSQUEHANNA STEAM ELECTRIC STATION UNITS 1 & 2 FINAL SAFETY ANALYSIS REPORT
Figure 1.2-51 replaced by dwg. M-5200, Sh. 2 FIGURE 1.2-51, Rev. 52

AutoCAD Figure 1_2_51.doc