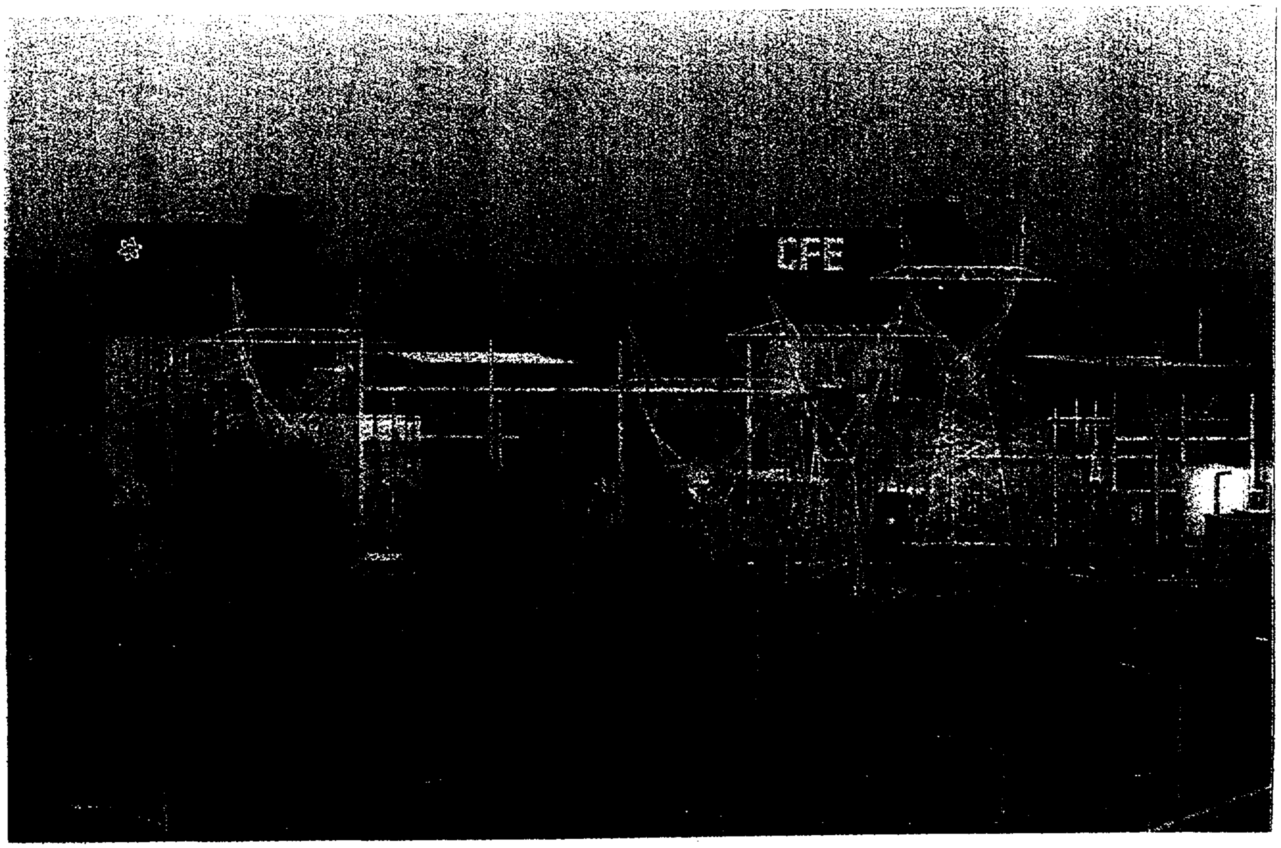
GERENCIA DE CENTRALES NUCLEOELECTRICAS



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COMISION FEDERAL DE ELECTRICIDAD

LAGUNA VERDE, VER.

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Contents

1. PLANT PRESENTATION

1.1 Overall Site and Plant Description.

1.1.1 Overall Site

Laguna Verde Units 1 and 2 are located on the Gulf of Mexico in the municipality of Alto Lucero in the state of Veracruz, 69 km NNW of the city of Veracruz. 60 km ENE of the city of Jalapa, the state capital, and 260 km ENE of Mexico City. The site shown in Figure 7.1 is bounded on the east by the Gulf of Mexico and on the west by Federal Highway No. 180. The site includes Laguna Verde on the North, and Laguna Salada on the south and has a total area of 3,700,000 sq. m.

The plant site grade level is 10.908 m above Mean Sea Level (MSL). A belt of sand dunes runs from north to south and vegetation consists of small birches, shrubs, and grass around the lagoons and along the coast line.

1.1.2 Type of Nuclear Steam Supply.

Both units have a boiling water reactor nuclear steam supply system as designed and supplied by the General Electric Company and designated as BWR 5.

1.1.3 Type of Containment.

The primary containment is part of the overall containment system which provides the capability to reliably limit the release of radioactive materials to the environs subsequent to the occurrence of the postulated Loss-of-Coolant Accident (LOCA) so that offsite doses are below the "reference values" stated in Title 10 of the United States Code of Federal Regulations, Part 100 (10 CFR 100). The design employs the drywell pressure-suppression features of the BWR/Mark II containment concept. The containment provides dual barriers consisting of the primary containment and the secondary containment. The primary containment is a steel lined reinforced concrete structure of the over-and-under configuration. The secondary containment is the Reactor Building which encloses the Reactor Pressure Vessel (RPV) and its primary containment.

1.1.4 Power Levels.

Each reactor has a rated power level of 1931 MWT and a design power level of 2015 MWT. The plant utilizes a single cycle forced circulation Boiling Water Reactor (BWR) provided by General Electric. The heat balance for rated power is shown in Figure 7.2. The units are designed to operate at a gross electrical output power of approximately 675 MWE and a net electrical output power of approximately 650 MWE.

1.1.5 Turbine Generator.

The turbine-generator was provided by Mitsubishi Heavy Industries, the turbine is an 1800 rpm, tandem compound (4-flow machine with 44" last row blades) reheat unit equipped with an electrohydraulic control system for normal operation. The rating of the turbine-generator is approximately 674,480 kw at the generator terminals when the turbine exhaust pressure is 2.0 in Hg Abs.

The generator is a direct-driven, three phase, 60Hz, 22,000 volts, 1,800 rpm, hydrogen inner-cooled, synchronous machine rated at 750,000 kVA with a 0.9 power factor and a short circuit ratio of 0.58 to the design capacity.

1.2 Plant History and Current Operating Status.

1.2.1 Plant History

The principal events in Laguna Verde history were as follows:

1976 Construction Unin 1 is started with the foundations of Reactor Building.

1976/84 Erection of main structures is carried out. Reactor vessel, piping and electro-mechanical equipment are installed.

1984/87 Generic tests and "cold" preoperational tests are performed.

1988 On October 21st, after having fulfilled all the regulatory requirements, the CNSNS authorizes the Initial Fuel Load, and Startup Tests are initiated.

1990	The Startup Test Program was finished. Counting with the CNSNS certification, the Secretaría de Energía, Minas e Industria Paraestatal (Energy, Mines and Government Industry Secretariat) granted to CFE the Unit 1 License for Commercial Operation (July 29th).

It should be mentioned that during the statup tests phase, which lasted about 21.5 months, a significative number of modifications and maintenance activities were performed, which later contributed to a good unit operation. Also, during this phase there were 5 planned scrams and 25 unplanned scrams.

UNIT 2

1977	Construction Unin 2 is started with the foundations of Reactor Building.
1977/82	Little advances in construction mainly due to budgetary problems.
1982/84	Construction work is suspended.
1985	Construction is reinitiated.
1987	Electromechanical installation work is reinitiated.
1994	On August 25th after having fulfilled all the regulatory requirements the CNSNS authorizes the initial Fuel Load, and Startup Test are initiated.
1995	The Startup Test Program was finished. Counting with the CNSNS certification, the Secretaría de Energía, (Energy Secretariat) granted to CFE the Unit 2 License for Commercial Operation (April 10th).

It should be mentioned that the Startup Tests phase length (7.5 months) represents an international achievement for plants of this type. There were 3 planned scrams and 4 unplanned scrams, which represents a good performance during this phase (see figure 7.3).

1.3 CFE Organization

Comisión Federal de Electricidad (CFE) is a mexican government owned utility, with 58 years of experience in the design, construction and operation of power generating plants in Mexico. It has designed. built and operates diverse types of plants such as thermoelectric, combined cycle, gas turbine, internal combustion. and dual fuel power plants; as well as nuclear, geothermal, hydroelectric. and wind turbine power plants. It has 32,000 MW of installed capacity. and generates more than 141,000 GW/hour annually. Utility information is shown in the pamphlets annex to this report. "Comisión Federal de Electricidad: Power for México and the World", and "Comisión Federal de Electricidad - International Division".

1.4 Plant Organization

The responsibility for the appropriate management of the plant is with the CFE General Director, who delegates the responsability in the appointed Nuclear Director. The safe operations of the plant is under responsibility of the Plant Manager who manages the staff (see figures 7.7 and 7.8).

In the Chapter "Management. Organization and Administration" it is presented the appropriate description.

1.5 Plant Layout

The arrangement of major plant structures and buildings is shown in fig. 7.4.

2. DESIGN INFORMATION

2.1 General Arrangement of Structures and Equipment

The plant site principal structures are as follows (see fig. 7.4):

a) Reactor Building (2):
This is the building that houses the major portion of the Nuclear Steam Supply System (NSSS), the primary containment (drywell and suppression pool), the spent fuel pool, the new fuel storage vault, the refueling equipment and the emergency core cooling systems.

b) Turbine Building (2):
This is the building that houses the power conversion equipment.

c) Control Building (2):
This is the building that houses the control room and essential switchgear, and health physics area.

d) Diesel Generator Building (2):
This is the building that houses the standby diesel generators, oil day tanks, and associated controls and instrumentation.

e) Oil Storage Vault (2):
This is the structure that houses the oil storage tanks for the diesel generators.

f) Radwaste Building (1):
This is the building that houses the liquid and solid radwaste processing systems and components of the off-gas treatment system for both Units 1 and 2. It also houses the filter demineralizers and condensate polishers, the spent fuel pool cleanup system and the Reactor Water Cleanup System (RWCU) filter demineralizer for Unit 1.

g) Purification Building (1):
This is the building that houses the condensate polishers, spent fuel pool cleanup system and RWCU, for Unit 2, and some portions of the radwaste liquid processing systems.

h) Circulating Water Structure (1):
This is the structure housing the circulating water pumps and Turbine Building service water pumps for both Units 1 and 2.

i) Nuclear Service Water Structure (1):
This is the structure housing the redundant nuclear service water pumps for both Units 1 and 2.

j) Fire Pump House (1):
This is the building housing the fire pumps for the site fire protection system.

k) Water Treament Building and Machine Shop (1):
This is the facility providing demineralized water for both Units 1 and 2, and serving both Units for maintenance works.

l) Sodium Hypochlorite Generating Facility (1):
This facility functions to provide low level chlorination of the intake water for both Units 1 and 2.

m) Condensate Storage Tanks (4):
There are two tanks per unit to store condensate.

n) Meteorological Tower (1):
This tower provides continuous meteorological data for both Units 1 and 2.

o) Switchyard (1):
Encloses the 230 kv system and the 400 kv system.

p) Central Alarm Station (CAS) Building (1):
This building is for site security.

q) Guard House (1):
The Guard House is for site access.

r) Administration Building:
This building is serving the site.

s) Onsite Radwaste Storage Facility.
This facility serves both Units 1 and 2.

t) SIIP Building (1):
This is the building housing the computer room for both Units 1 and 2 Process Computer System, or SIIP (Sistema Integral de Información de Proceso).

2.2 Major Process and Safety Systems

A brief description of the principal systems of the plant, grouped in Nuclear Systems; Nuclear Safety Systems and Engineered Safety Features; Power Conversion Systems; Electrical Systems and Instrumentation and Control; Fuel Handling and Storage Systems; Cooling Water and Auxiliary Systems; Radioactive Waste Systems; and Radiation Monitoring and Control is presented.

2.2.1 Nuclear System

The nuclear system includes a direct cycle, forced circulation, boiling water reactor that produces steam for direct use in the steam turbine. A heat balance showing the major parameters of the nuclear system for the rated power conditions is shown in Figure 7.2.

2.2.1.1 Reactor Core and Control Rods

Fuel for the reactor core consists of slightly enriched uranium dioxide pellets sealed in Zircalloy-2 tubes. These tubes (or fuel rods) are assembled intoindividual fuel assemblies. Gross control of the core is achieved by movable, bottom-entry control rods. The control rods are cruciform in shape and are dispersed throughout the lattice of fuel assemblies. The control rods are positioned by individual control rod drive-hydraulic control units.

2.2.1.2 Reactor Vessel Internals

The reactor vessel contains the core and supporting structures; the steam separators and dryers; the jet pumps; the control rod guide tubes; the distribution lines for the feedwater and core sprays; the incore instrumentation; and other components. The main connections to the vessel include steam lines, coolant recirculation lines, feedwater lines, control rod drive and incore nuclear instrument housings, core spray lines, residual heat removal lines, core differential pressure line, jet pump pressure sensing lines, water level instrumentation.

The reactor vessel is designed and fabricated in accordance with applicable codes for a pressure of 1250 psig. The nominal operating pressure in the steam space above the separators is 1020 psia. The vessel is fabricated of low alloy steel and is clad internally with stainless steel (except for the top head, nozzles, and nozzle weld zones which are unclad).

2.2.1.3 Reactor Recirculation System

The reactor recirculation system consists of two recirculation pump loops external to the reactor vessel. These loops provide the piping path for the driving flow of water to the reactor vessel jet pumps.

Each external loops contains one high capacity motor-driven recirculation pump, two motor operated maintenance valves and one hydraulically operated flow control valve. The variable position hydraulic flow control valve operates in conjunction with a low frequency motor-generator set to control reactor power level through the effects of coolant flow rate on moderator void content.

2.2.1.4 Residual Heat Removal System

The Residual Heat Removal (RHR) System is a system of pumps, heat exchangers, and piping that fulfills the following functions:

1) It removes decay and sensible heat during and after plant shutdown.

2) It injects water into the reactor vessel, following a loss-of-coolant accident, to reflood the core independent of other core cooling systems.

3) It also removes heat from the primary containment, following a loss-of-coolant accident, to limit the increase in primary containment pressure. This is accomplished by cooling and recirculating the suppression pool water (containment cooling) and by spraying the drywell and suppression pool air spaces (containment spray) with suppression pool water.

2.2.1.5 Reactor Water Cleanup System

The reactor water cleanup system recirculates a portion of reactor coolant through a filter-demineralizer to remove particulate and dissolved impurities from the reactor coolant. It also removes excess coolant from the reactor system under controlled conditions.

2.2.1.6 Nuclear Leak Detection System

The nuclear leak detection and monitoring 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: Main Steam Lines, Reactor Water Cleanup System, Residual Heat Removal (RHR) System, Reactor Core Isolation Cooling (RCIC) System, Feedwater system, ECCS Systems, and Miscellaneous Systems.

2.2.2 Nuclear Safety Systems and Engineered Safety Features

2.2.2.1 Reactor Protection System

The Reactor Protection System (RPS) initiates a rapid, automatic shutdown (scram) of the reactor. It acts on time to prevent fuel cladding damage and any nuclear system process barrier damage following abnormal operational transients. The reactor protection system overrides all operator actions and process controls and is based on a fail-safe design philosophy that allows appropriate protective action even if a single failure occurs.

2.2.2.2 Neutron Monitoring System

Those portions of the neutron monitoring system that are part of the reactor protection system qualify as a nuclear safety system. The Intermediate Range Monitors (IRM) in U1 or the Source Range Neutron Monitoring (SRNM) in U2, and the Average Power Range Monitors (APRM) in U1 or Power Range Neutron Monitoring (PRNM) in U2, which monitor neutron flux via incore detectors, provide scram logic inputs to the reactor protection system to prevent excessive fuel clad damage as a result of overpower transients.

2.2.2.3 Control Rod Drive System

Each control rod is controlled individually by a hydraulic control unit. When a scram signal is received, high pressure water stored in an accumulator in the hydraulic control unit or reactor pressure forces its control rod into the core. This system also positions the control rods for a gross control of the core.

2.2.2.4 Control Rod Velocity Limiter

A control rod velocity limiter is attached to each control rod to limit the velocity at which a control rod can fall out of the core should it become detached from its control rod drive. This action limits the rate of reactivity insertion resulting from a rod drop accident. The limiters contain no moving parts.

2.2.2.5 Nuclear System Pressure Relief System

A pressure relief system consisting of safety/relief valves mounted on the main steam lines is provided to prevent excessive pressure buildup inside the nuclear system for operational transients or accidents.

2.2.2.6 Reactor Core Isolation Cooling System

The Reactor Core Isolation Cooling System (RCIC) provides makeup water to the reactor vessel when the vessel is isolated. The RCIC system uses a steam-driven turbine-pump unit and operates automatically in time and with sufficient coolant flow to maintain adequate water level in the reactor vessel for some events.

2.2.2.7 Emergency Core Cooling Systems (see fig. 7.5)

Four emergency core cooling systems are provided to maintain fuel cladding below the temperature limit of 10 CFR 50.46 in the event of a breach in the reactor coolant pressure boundary that results in a loss of reactor coolant. The systems are: High Pressure Core Spray (HPCS) System, Automatic Depressurization System (ADS), Low Pressure Core Spray (LPCS) System, and Low Pressure Coolant Injection (LPCI) System, an operating mode of the residual heat removal system.

2.2.2.7.1 High Pressure Core Spray

The HPCS system provides and maintains an adequate coolant inventory inside the reactor vessel to limit fuel cladding temperatures in the event of breaks in the reactor coolant pressure boundary. The system is initiated by either high pressure in the drywell or low water level in the vessel. It operates independently of all other systems over the entire range of pressure differences from greater than normal operating pressure to zero. The HPCS cooling decreases vessel pressure to enable the low pressure cooling systems to function. The HPCS system pump motor is powered by a diesel generator if auxiliary power is not available, and the system may also be used as a backup for the RCIC system.

2.2.2.7.2 Automatic Depressurization System (ADS)

The ADS uses five of the safety/relief valves that are part of the nuclear pressure relief system. The automatic safety/relief valves are arranged to open on conditions indicating both that a break in the reactor coolant pressure boundary has ocurred and that the HPCS system is not delivering sufficient cooling water to the reactor vessel to maintain the water level above a preselected value.

The ADS is not activated unless either the LPCS or LPCI pumps are operating. This is to ensure that adequate coolant will be available to maintain reactor water level after the depressurization.

2.2.2.7.3 Low Pressure Core Spray System

The LPCS system consists of one independent pump and the valves and piping to deliver cooling water to a spray sparger over the core. The system is actuated by conditions indicating that a breach exists in the reactor coolant pressure boundary 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 into each fuel channel. The LPCS loop functioning in conjunction with the ADS or HPCS can provide sufficient fuel cladding cooling following a loss-of-coolant accident.

2.2.2.7.4 Low Pressure Coolant Injection System

Low pressure coolant injection is an operating mode of the Residual Heat Removal (RHR) System, but is discussed here because the LPCI mode acts as an engineered safety feature in conjunction with the other emergency core cooling systems. LPCI uses the pump loops of the RHR to inject cooling water into the pressure vessel. LPCI is actuated by conditions indicating a breach in the reactor coolant pressure boundary, but water is delivered to the core only after reactor vessel pressure is reduced.

2.2.2.8 Primary Containment

The primary containment is part of an overall containment system which has the capability of reliably limiting the release of radioactive materials to the environs subsequent to the occurrence of the postulated design basis loss-of-coolant accident such that offsite doses will be within the reference values stated in 10CFR100. Its design employs an over-and-under steel lined reinforced concrete structure which serves as a housing for the reactor pressure vessel, the reactor coolant recirculating loops, main steam header and other branch connections od the reactor primary coolant system. The pressure suppression system consists of a drywell over a pressure suppression chamber, or wetwell, containing a large volume of water, a submerged vent system between the two compartments, relief valves, isolation valves and a containment cooling system. In the event of a failure of the reactor coolant piping boundary within the drywell, reactor water and steam would be released into the drywell air space. The ensuing increase in drywell pressure then forces a mixture of air and steam through the vents into the suppression pool, where the steam is condensed, thus providing a rapid means of pressure reduction in the drywell. Air which is transferred to the suppression chamber during this process, pressurizes the suppression chamber and is ultimately vented back to the drywell via vacuum breakers mounted on the downcomers. In order to avoid any negative pressure inside containment from spray actuation or other sources vacuum breakers are installed between the Reactor Building and the wetwell air space.

2.2.2.9 Containment and Reactor Vessel Isolation Control System

The 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 breach in the reactor coolant pressure boundary.

2.2.2.10 Main Steam Line Isolation Valves

Although all pipelines that both penetrate the 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. Automatic isolation valves are provided in each main steam line. Each is powered by both air pressure and spring force to close actuator. In addition a third isolation valve (shutoff valve) which is remote manually actuated is provided on each main steam line.

2.2.2.11 Main Steam Line Flow Restrictors

A venturi-type flow restrictor is installed in each steam line. These devices limit the loss of coolant from the reactor vessel before the main steam line isolation valves are losed in case of a main steam line break outside the containment.

2.2.2.12 Main Steam Line Radiation Monitoring System

The main steam line radiation monitoring system consists of four gamma radiation monitors located externally to the main steam lines just outside the containment. The monitors are designed to detect a gross release of fission products from the fuel. On detection of high radiation, the trip signals generated by the monitors are used by the reactor protection system to initiate a reactor scram and to close the main steam line isolation valves.

2.2.2.13.Residual Heat Removal System (Containment Cooling)

The containment cooling subsystem is placed in operation to limit the temperature of the water in the suppression pool and of the atmospheres in the drywell and suppression chamber following a design basis loss-of-coolant accident, to control the pool temperature during normal operation of the safety/relief valves and the RCIC system, and to reduce the pool temperature following an isolation transient. In the containment cooling mode of operation, the RHR main system pumps take suction from the suppression pool and pump the water through the RHR heat exchangers where cooling takes place by transferring heat to the nuclear service water. The fluid is then discharged back to the suppression pool, to the drywell spray header, to the suppression chamber spray header, or to the reactor pressure vessel.

2.2.2.14 Ventilation Exhaust Radiation Monitoring System

The ventilation radiation monitoring system consists of a number of radiation monitors arranged to monitor the activity level of the air exhaust from the Reactor Building, Turbine Building, Radwaste Building and Purification Building.

2.2.2.15 Standby Gas Treatment System

The Standby Gas Treatment System (SGTS) consists of two identical process trains. Each train includes in the direction of flow a moisture separator, an electric heating coil, a bank of medium efficiency prefilters, a bank of High Efficiency Particulate Air (HEPA) prefilters, a carbon-adsorber, a bank of after-HEPA filters and a fan with backdraft check valve. Either process train may be considered as an installed spare with the other train capable of passing the required amount of air to provide a minimum of one complete change of Reactor Building air in a 24-hour period. The system maintains a 0.25 in. water gauge negative pressure in the Reactor Building.

2.2.2.16 Reactor Building

The Reactor Building completely surrounds the primary containment. This provides secondary containment when the primary containment is closed and in service; primary containment during those periods when the primary containment is open, such as during refueling. The Reactor Building also houses refueling and reactor servicing equipment, new and spent fuel storage facilities. and other reactor safety and auxiliary systems.

2.2.2.17 Standby AC Power Supply System

The standby AC power supply system consists of three diesel-generator sets with associated switchgear, distribution equipment and auxiliary components.

Two diesel-generator sets are associated with redundant (Division 1 and 2) separate electrical divisions. Each diesel-generator set powers one independent division with a capacity sufficient to achieve shutdown during complete loss of offsite power even assuming a LOCA concurrent with the loss of offsite power. Since the plant load distribution is such that redundant auxiliary systems required for shutdowns are separated by divisions; safe shutdown can be achieved with the operation of only one diesel-generator set.

Another independent diesel-generator set provides AC power exclusively to the HPCS system (Division 3) in the event of Loss of Offsite Power.

2.2.2.18 DC Power Supply System

The DC power supply system consists of station batteries, battery chargers, distribution equipment and related auxiliary components.
The DC power supply system furnishes power at three voltage levels: 250 volts, 125 volts, and 24 volts. The 250v DC and 125v DC subsystems supply power to both class 1E and non-class 1E loads, while the 24v DC subsystem supplies power for the start-up range and power-range neutron monitoring systems. The ampere-hour capacitiy of each battery is capable of supplying all essential loads for a minimum of 2 hours in the event of loss of DC output from the associated battery charger.

2.2.2.19 Standby Liquid Control System

Although not intended to provide prompt reactor shutdown as the control rods are, the standby liquid control system provides a redundant, independent, and alternate way to bring the nuclear fission reaction to subcriticality and to maintain subcriticality as the reactor cools. The system makes possible an orderly and safe shutdown in the event that not enough control rods can be inserted into the reactor core to accomplish shutdown in the normal manner. The system is sized to counteract the positive reactivity effect from rated power to the cold shutdown condition.

2.2.2.20 Safe Shutdown from Outside the Control Room

In the event that the control room becomes inaccessible, the reactor can be brought from power range operation to cold shutdown conditions by the use of the controls provided on the remote shutdown panel that is located outside the control room.

2.2.3 Power Conversion System

2.2.3.1 Turbine-Generator

The turbine is an 1800 rpm, tandem compound (4-flow machine with 44" last row blades) reheat unit equipped with an electrohydraulic control system for normal operation. The rating of the turbine generator is approximately 674,480 kw at the generator terminals when the turbine exhaust pressure is 2.0 in Hg Abs.

The generator is a direct-driven, three-phase, 60 Hz, 22,000 volts, 1,800 rpm, hydrogen inner cooled, synchronous machine rated at 750,000 kVA with a 0.9 power factor and a short circuit ratio of 0.58 to the design capacity.

2.2.3.2 Main Condenser

The main condenser is of the divided water box dual pressure type and is designed with a condensing capacity of 4,255 x 10E6 Btu/hr. This is sufficient to allow rated turbine output when the circulating water temperature at the condenser inlet is 78.7 F or below and to maintain exhaust pressure of 1.91 in Hg at the LP turbine and 2.51 in Hg at the HP turbine. This is the optimized design condition. The main condenser is designed to receive up to 27.5 percent of the rated full load main steam flow from the turbine by-pass system and to provide a hotwell effluent oxygen content of 5 ppm or less over a load range of 25 to 100 percent. Leakage from steam lines valves 2" and larger in the Turbine Building is also routed to the main condenser.

2.2.3.3 Main Condenser Evacuation System

The main condenser evacuation system is designed to remove non- condensable gases from the condenser, including air inleakage and radiolytic dissociation products originating in the reactor, continuously exhausting them to the gaseous radwaste processing system during operation.

2.2.3.4 Steam Bypass System and Pressure Control System

A turbine bypass system in provided which passes steam directly to the main condenser under the control of the pressure regulator. Steam is bypassed to the condenser whenever the reactor steaming rate exceeds the load permitted to pass to the turbine generator except under low vacuum conditions when bypass to the condenser will be prevented. The capacity of the turbine bypass system is 27.5 percent of the turbine rated steam flow.

2.2.3.5 Circulating Water System

The circulating water system is an open system and provides a continuous supply of cooling water to the condenser. Four 25 percent capacity circulating water pumps located in the circulating water intake structure take water from the Gulf of Mexico, pass it through the condenser and discharge it back to the Gulf of Mexico via Laguna Salada. The combination of a channel in Laguna Salada with a breakwater around the intake structure ensures that the discharged water wil have a minimal effect on the temperature of the sea water at the intake structure.

2.2.3.6 Condensate Storage Facilities and Condensate Supply System

Two 300,000 gallon condensate storage tanks are provided per unit. Each is designed to ensure that at least 100,000 gallons of condensate will always be available for the RCIC or HPCS system requirements. The condensate supply system is not required to perform a safety function but does provide a source of condensate for testing of pumps during normal operations, for suppression pool makeup and for RCIC system operation in the hot standby or shutdown modes. One

100,000 gallon demineralized water storage tank is also provided for the plant for makeup to the condensate storage system of both units.

2.2.3.7 Condensate and Feedwater System

The condensate and feedwater system pumps water from the condenser hotwell to the reactor pressure vessel. Condensate is pumped by means of two of three 50 percent capacity motor driven condensate pumps through the steam jet air ejector intercondenser, the gland seal steam condenser and the full flow condensate filter demineralizer system. The effluent from the filter demineralizer is then pumped by the condensate booster pumps through the five low-pressure heaters. Three 50 percent nominal capacity motor driven booster pumps are provided. The last low-pressure heater discharges to the suction of the two 50 percent capacity steam turbine driven reactor feedwater pumps. These pumps serve to pump the condensate through the sixth stage of feedwater heating and into the reactor vessel. Feedwater flow is controlled by varying the speed of the steam driven turbine.

2.2.4 Electrical Systems, Instrumentation and Control

2.2.4.1 Electrical Power Systems (see figure 7.6)

The station consists of two duplicate main generator units designated as Unit 1 and Unit 2. Each main generator is connected to a main power transformer through a low-voltage main generator circuit breaker built into the isolated phase bus duct. The main power transformer steps up the output of the 22kV generator to a nominal 400 kV transmission system voltage.

Station auxiliary power is provided by a 22kV/4.36kV three winding normal auxiliary transformers tapped off the 22kV generator isolated phase bus duct. This transformer is used for normal startup power with the generator breaker open, normal auxiliary power with the generator breaker closed and normal shut-down power with the generator breaker opened. This transformer is sized to provide power for the total plant auxiliary load.

The main transformer output is connected to a 400kV switchyard and consists of circuit breakers, disconnect switches, buses as well as associated equipment arranged in a breaker and a half configuration.

The 400kV switchyard is connected by a 300MVA auto-transformer to a 230kV switchyard on site. The 230 kV switchyard equipment is similar to that in the 400 kV switchyard. It is also arranged in the breaker and a half configuration. The 400/230 kV auto-transformer is provided with a 34.5kV tertiary winding of ample capacity to furnish power to both Units 1 and 2 through a three winding 34.5kV/4.36kV standby transformer, one per unit; thus, ensuring the station with an offsite power supply from either the 230kV or 400kV systems.

In addition, a two winding 230kV/34.5kV transformer is connected to the 230kV system. It can provide the necessary power to two 34.5kV/4.36kV two winding backup transformers, one per unit; sized to provide power for the plant safe shutdown loads.

2.2.4.2 Instrumentation and Control

2.2.4.2.1 The reactor manual control system

Provides the means by which control rods are positioned from the control room for power control. The system operates valves in each control rod drive hydraulic control unit to change control rod position. Only one control rod can be manipulated at a time.

2.2.4.2.2 Recirculation Flow Control System

During normal power operation a variable position discharge valve is used to control flow. Adjusting this valve changes the coolant flow rate through the core. It also changes the core power level. The system automatically adjusts the reactor power output to the load demand. For startup and shutdown flow changes at lower power, the pump speed is changed by reducing the frequency of the electrical power supply.

2.2.4.2.3 Neutron Monitoring System

The neutron monitoring system is a system of incore 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 can exist in the core.

2.2.4.2.4 Refueling Interlocks

A system of interlocks that restricts movement of refueling equipment and control rods when the reactor is in the refueling and startup modes is provided to prevent an inadvertent criticality during refueling operation. The interlocks back up procedural controls that have the same objective. The interlocks affect the refueling platform, refueling platform hoists, fuel grapple and control rods.

2.2.4.2.5 Reactor Vessel Instrumentation

In addition to instrumentation 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. This instrumentation monitors reactor vessel pressure, water level, coolant temperature, reactor core differential pressure, coolant flow rates, and reactor vessel head inner seal ring leakage.

2.2.4.2.6 Pressure Regulator and Turbine-Generator Control

The pressure regulator maintains control of the turbine control and turbine bypass valves to allow proper generator and reactor response while maintaining the nuclear system pressure essentially constant.

The turbine generator speed load controls can initiate rapid closure of turbine control valves (rapid opening of the turbine bypass valves) to prevent turbine overspeed on loss of the generator electric load.

2.2.4.2.7 The feedwater control system

Automatically controls the flow of feedwater into the reactor pressure vessel to maintain the water within the vessel at predetermined levels. A conventional three element control system is used to accomplish this function.

2.2.5 Fuel Handling and Storage Systems

2.2.5.1 New and Spent Fuel Storage

New and spent fuel storage racks are designed to prevent inadvertent criticality and load buckling. Sufficient coolant and shielding are maintained to prevent overheating as well as excessive personnel exposure, respectively. The design of the fuel storage areas provides for corrosion resistance, adherence to Seismic Category I requirements, and prevention of Keff from reaching 0.95 under dry or flooded conditions.

2.2.5.2 Fuel Handling System

The fuel handling equipment includes a fuel inspection stand, fuel preparation machine, a 125 ton crane, a refueling platform, jib cranes, and other related tools for fuel and reactor servicing. All equipment conforms to applicable codes and standards.

2.2.6 Cooling Water and Auxiliary Systems

2.2.6.1 Nuclear Closed Cooling Water System

The nuclear closed cooling water system consists of four pumps per unit, heat exchangers, controls and instrumentation in order to provide adequate cooling for the various equipment located in the Reactor Building and Radwaste Building.

2.2.6.1.1 Fuel Pool Cooling and Cleanup System

The fuel pool and cleanup subsystem provides for the removal of decay heat from stored spent fuel and maintain specified water temperature, purity, clarity, and level.

2.2.6.1.2 Nuclear Service Water System

Nuclear service water is supplied by nuclear service water pumps located in the nuclear service water intake structure. Four nuclear service water pumps are provided per unit and take suction from the Gulf of Mexico. This system removes heat from the Nuclear Closed Cooling Water System as well as from other equipment and discharges in the Gulf of Mexico, via Laguna Salada.

2.2.6.1.3 Ultimate Heat Sink

The Gulf of Mexico serves as the ultimate heat sink for the Laguna Verde Nuclear Power Station.

2.2.6.1.4 Condensate Storage Transfer System

The condensate storage facility provides a source of water for testing HPCS and RCIC pumps and for main condenser makeup during operation. The 300,000 gallon condensate storage tanks are interconnected to simultaneously supply condensate to the main condenser, to the CRD pumps, to the RCIC and HPCS systems, and to the condenser filter demineralizer back wash pumps. In addition, the condensate supply pumps deliver condensate to miscellaneous services in the Reactor and Radwaste Buildings.

2.2.6.1.5 Demineralized Water Makeup System

The demineralized water makeup system is comprised of the plant makeup water treatment system and the water demineralization system.

2.2.6.1.6 Potable and Sanitary Waste Water Systems

The plant potable water system provides water for drinking and sanitary purposes. Potable water is supplied by the plant makeup water treatment system from three deep wells located near Laguna Verde.

2.2.6.1.7 Plant Chilled Water System

Separate water chillers are provided for air conditioning service in the Control Building, for chilled water service to the fan coil units inside primary containment as well as for chilled water service to certain nonsafety related fan coil units in the Reactor Building.

2.2.6.1.8 Process Sampling System

The process sampling system is furnished to provide process information that is required to monitor plant and equipment performance and changes to operating parameters. Representative liquid and gas samples are taken automatically and/or manually during normal plant operation for laboratory or online analyses.

2.2.6.1.9 Plant Equipment and Floor Drainage

Plant equipment and floor drainages systems handle both radioactive and nonradioactive wastes. Drainage systems that carry radioactive waste are isolated from those that do not transport radioactive waste.

2.2.6.1.10 Service and Instrument Air System

Three air compresors are provided to supply service and instrument air for the plant. The plant air compressors discharge to the after-coolers, moisture separators, through to air receivers and then to supply headers throughout the plant.

2.2.6.1.11 Normal Auxiliary AC Power

The plant normal auxiliary AC power system consists of a full capacity normal auxiliary transformer connected directly to the main generator bus capable of supplying all auxiliary power requirements during normal plant operations. In addition, one full capacity standby and one reduced capacity back-up auxiliary transformer are also provided.

2.2.6.1.12 Diesel Generator Fuel-Oil Storage and Transfer System

The fuel oil storage and transfer system consists of separate and independent diesel oil supply subsystems serving each of two emergency diesel generators and the HPCS diesel generator. Each full capacity subsystem consists of a fuel storage tank, two full capacity transfer pumps, a day tank, interconnecting piping, strainers, valves, associated instrumentation and controls.

2.2.6.1.13 Auxiliary Steam System

The auxiliary steam system normally operates only when the clean steam evaporators are inoperative during the plant startup and shutdown.

2.2.6.1.14 Ventilating and Air Conditioning Systems

The ventilation and air conditioning (VAC) systems are designed to remove heat released by equipment to ensure meeting acceptable maximum operating conditions and to maintain proper air quality for personnel comfort in those areas that are continually occupied. In addition. the main control room. critical switchgear area and cable spreading room VAC systems, the nuclear service water pump room heat removal system, the Reactor Building ESF pump rooms and critical electric equipment area cooling systems and the ventilation system for the standby diesel generators, are designed to operate under all plant conditions. The primary containment drywell cooling system is designed to operate during all normal and transient conditions. with the exception of the post LOCA conditions. However, air mixing fans located in the drywell are designed to continue operation in the post-LOCA environment.

2.2.6.1.15 Lighting Systems

The plant lighting systems are normal AC lighting, emergency AC lighting, DC lighting and battery powered emergency lighting.

2.2.6.1.16 Fire Protection System

The fire protection system is designed to provide for the detection and extiguishment of fires. Automatic detection and suppression systems backed up by manual fire hose stations provide plant capability to mitigate the consequences of fires.

In addition, each unit is provided with a unique and separate Safe Shutdown Earthquake Fire Protection System (SSEFP) that serves safety related equipment as well as areas in the Reactor, Control and Diesel Generator Buildings. The SSEFP can deliver a minimum of 75 GPM to two hose stations and is designed to function during and after the safe shutdown earthquake.

2.2.6.1.17 Communications Systems

The plant communications systems are designed to provide reliable communication inside and outside the plant. It also provides communications from the plant to local fire and law enforcement authorities as well as to emergency medical facilities. The system utilizes a page party/public address system with five party channels and building wide alarm system, the public telephone system and a radio communication system.

2.2.7 Radioactive Waste Systems

2.2.7.1 Gaseous Radwaste System

The purpose of the gaseous radwaste system is to process and control the release of gaseous radioactive wastes to the site environs so that the total radiation exposure to persons outside the controlled area does not exceed the maximum limits of the applicable regulations 10 CFR 20 and 10 CFR 50 Appendix I, during normal operations.

2.2.7.2 Liquid Radwaste System

This system collects, treats, stores, and disposes of all radioactive liquid wastes. These wastes are accumulated directly in radwaste tanks or in sumps at various locations throughout the plant for subsequent transfer to collection tanks in the radwaste facility. Wastes are processed on a batch basis with each batch being processed by such method or methods appropriate for the quality and quantity of materials determined to be present. Processed liquid wastes may be returned to the condensate system or discharged to the circulating water discharge line to Laguna Salada. The liquid wastes in the discharge piping are diluted with circulating water to achieve a concentration at the site boundary which is below the limits of 10 CFR 20.

2.2.7.3 Solid Radwaste Systems

Solid radioactive wastes are collected, processed, mixed with a binder, if appropriate and packaged for storage and ultimate burial. These wastes are generally stored on the site until the short half-lived isotopes have decayed. Wet solid wastes are collected, dewatered, mixed with a binder and packaged in steel containers. Examples of these wastes are filter residues, concentrated wastes, and spent resins. Dry solid wastes such as paper, air filters, rags, and used clothing are compressed and packaged in steel containers.

2.2.8 Radiation Monitoring and Control

2.2.8.1 Process Radiation Monitoring

Process radiation monitoring systems are provided to monitor as well as control radioactivity in process streams. These systems also monitor and control effluent streams and in all cases, activate appropriate alarms and controls. The following major process monitoring systems are provided:

- Main steam line radiation monitoring system,
- Air ejector and off-gas radiation monitoring system,
- Liquid radwaste effluent radiation monitoring system,
- Service water discharge radiation monitoring system,
- Reactor Building ventilation exhaust radiation monitoring system,
- Reactor Building nuclear closed cooling water radiation monitoring system,
- Turbine Building ventilation exhaust radiation monitoring system,
- Radwaste Building ventilation exhaust radiation monitoring system,
- Purification Building ventilation exhaust radiation monitoring system.

2.2.8.2 Area Radiation Monitoring

Radiation monitoring devices are located in key areas throughout the CLV buildings to provide alerts for incipient high radiation airborne releases or high radiation liquid spill conditions. They also ensure that plant personnel are not inadvertently exposed to high radiation dosages.

2.2.8.3 Site Environs Radiation Monitoring

A comprehensive radiation surveillance program has been in progress since the beginning of plant construction to measure baseline radiation levels in the environs sorrounding the plant site. The program measures radiation doses or radioisotope levels in at least eight different media Atmospheric dose is monitored by means of Thermoluminiscent Dosimeters (TLD) while airborne particulates are measured by filtration of a known volume of air and analysis of the filtered material.

2.3 Key Design Parameters

2.3.1 Containment

It is a reinforced concrete structure with a steel lining, consisting of a drywell and a suppression chamber communicated through 68 venting tubes:

DRYWELL

Minimum Air Volume:	4360.9 m3 (145000 ft3)
Maximum Internal Pressure:	3.16 Kg/cm2 (45 psig.)
Maximum External Pressure:	0.141 Kg/cm2 (2 psig.)
Differential Pressure DW-WW:	1.757 Kg/cm2 (25 psid)
Maximum Internal Temperature:	171 °C (340 °F)

SUPPRESSION CHAMBER

Air Region Volume:	3171.49 m3 (112000 ft3)
Water Region Volume:	3208.3 m3 (113300 ft3)
Maximum Internal Pressure:	3.16 Kg/cm2 (45 psig.)
Maximum External Pressure:	0.141 Kg/cm2 (2 psig.)

Maximum System Temperature:	140 °C (285 °F)
Differential Pressure WW-DW:	0.176 Kg/cm2 (2.5 psid)

2.3.2 REACTOR

Type:	Boiling Water (BWR)
Fuel:	Enriched UO2
Uranium Total Weight:	81.285 tons.
Cruciform Control Rods:	109 steel rods filled with boron carbide.
Number of fuel bundles:	444 bundles with 62 fuel rods and 2 water rods each.
Reactor Nominal Pressure:	70.69 Kg/cm2
Reactor Thermal Power:	1931 MWt
Steam Flow:	3759 Kton/hr
Recirculation Pumps:	2
Recirculation Flow:	9600 ton/hr
Internal Recir. Jet Pumps:	20

2.3.3 TURBINE

Type:	4-flow impulse-reaction
High Pressure Turbine:	1 turbine
Speed:	1800 rpm
Inlet Steam Temperature:	283 C

Inlet Steam Pressure:	68.2 Kg/cm2
Outlet Steam Pressure:	13.7 Kg/cm2
Number of extractions:	4
Low Pressure Turbines:	2 turbines
Inlet Steam Pressure:	13.3 Kg/cm2
Outlet Steam Pressure:	710 mmHg
Inlet Steam Temperature:	267 C
Number of extractions	10

2.3.4 GENERATOR

Type:	Closed, hydrogen inner cooled
Maximum Capacity:	675 MWe
Frequency:	60 Hz
Voltage:	22 kV
Speed:	1800 rpm
Current:	19683 A
Power Factor:	0.9

2.3.5 FUEL CHARACTERISTICS

Fuel Type:	U-1	U-2
Fuel Type:	GE9B	GE6/GE9B
Fuel Bundle Geometry:	8X8	8X8

Bundle Fuel Rods Number:	62	62
Water Rods:	2	2
Peak Technical Specification:		
Linear heat generation rate:	47.2 kw/m (14.4 kw/ft)	43.9 kw/m *GE6 (13.4 kw/ft) 47.2 kw/m *GE9B (14.4 kw/ft)
Minimum Critical Power Ratio:	1.08	1.08

3. EXTERNAL ORGANIZATIONS

3.1 Nuclear Division of CFE (Comisión Federal de Electricidad = Federal Commission of Electricity).

CFE is a federal and public services institution dedicated to the generation and distribution of electricity to all citizens within the nation.

CFE Organization is headed by the CFE General Director who reports to a Government Council of which the Executive President is the Secretary of Energy.

CFE is the owner of Laguna Verde Nuclear Power Plant (CLV) and is held responsible for its design, engineering, construction, operation and maintenance. Therefore, CFE is equally responsible for overall management of CLV throughout the operative life of the plant.

Fig. 7.7 shows CFE's Nuclear Division structure that has total responsibility of activities performed at Laguna Verde.

The CFE General Director holds overall responsibility before CNSNS (National Commission of Nuclear Safety and Saveguards) to ensure that the management, design, engineering, construction, operation and maintenance of Laguna Verde units 1 & 2 are carried out in such manner that they propose no undue risk to general public health and safety. The General director will also delegate this responsibility to the Nuclear Director.

3.1.1 Nuclear Director

The Nuclear Director is held wholly responsible for establishing the policies and practices for the management and maintenance of LV units 1 & 2 and for insuring that all activities are carried out within the frame of quality in order to comply with stipulations set forth by operations licensing and applicable requirements.

He is held responsible for establishing an effective interface between Plant Manager (responsible for adequate plant operations) and the support groups (engineering, construction, administration, licensing, quality assurance, emergency plan) whose joint efforts guarantee correct operation and maintenace of both units.

3.1.2 Plant Organization

The Plant organization and main functions are illustrated in Fig. 7.8 and described in detailed in section entitled "Management Organization and Administration" of this manual. This organization is directly involved with the operation and maintenance of both CLV units.

3.1.3 Engineering Group

This group will serve as a technical support group during units 1 & 2 operation. The engineering group is headed by the engineering manager who reports directly to the Nuclear Director.

They provide all technical support necessary that might fall outside the plant's normal scope. The a-fore- mentioned group is comprised by personnel with the knowledge and experience necessary to oversee problems as to provide areas with support in the following:

a) Engineering of materials, metallurgy, mechanics, electricity, thermohydraulics, structures, and I&C.
b) Plant chemistry
c) Radiation protection
d) Fuel and outage
e) Maintenance
f) Licensing
g) Fuel handling
h) Environmental monitoring

Experience in the above mentioned disciplines is achieved by CFE qualified personnel assigned to this group and contractor consultants. The Nuclear Director can and will appeal to other CFE departments for procurement of technical support, if necessary.

3.1.4 Construction Group

These was personnel in charge of CLV units 1 & 2 construction, during start-up (commissioning) this group provided the necessary support to adequacy and modifications of plant installations. At present, they are in the process of concluding.

3.1.5 Administrative Group

In charge of providing all administrative support to plant development by means of purchasing, procurement, contracts, human resources, etc.

3.1.6 Licensing & Nuclear Safety Group is responsible for:

Keeping up-to-date before CNSNS, safety reports and any other documents required to obtain and maintain validity of the operations license of plant units.

Reporting to CNSNS the changes carried out on CLV which are unrevised non safety matters and to negotiate CNSNS authorization of changes, tests or experiments that are unrevised safety matters, in accordance with 10CFR50.59 or those involving technical specifications changes.

Advising operation organization on Licensing basics and criteria included in manuals, specifications and procedures.

3.1.7 Quality Assurance Group

The Quality Assurance Group is responsible for verifying through meetings, surveillances and/or audits that CLV various participant organizations perform their activities in compliance with established controls and requirements setforth in the operations Q.A. Plan as in guidelines, standards, codes, licenses and approved procedures which apply.

3.1.8 Emergency Preparedness Group

This group activities are presented in more detail in section entitled "Emergency Planning and Preparedness" of this manual.

3.2 Industry Organizations

National and foreign organizations that at certain time provide support to CVL normal and/or infrequent activities, can be divided into: national institutes of research and application and foreign organizations.

3.2.1 National Institutes

Among those associations which provide support to plant activities we could mention:

a) **Instituto de Investigaciones Electricas IIE (Electrical Research Institute)**; by means of training, design and simulator maintenance services, rotating machinery analysis, etc.

b) **Instituto Nacional de Investigaciones Nucleares ININ (National Institute of Nuclear Research)**; provides training sservices, radiation protection technicians in normal & outages, they also participate jointly with CFE and General Electric in CLV fuel fabrication (pilot plan), etc.

c) **Laboratorio de Equipos, Pruebas y Materiales LAPEM (Tests, equipment and material laboratories)**By means of Non Destructive Testing (PND) with In Service Inspection or Out of Service Inspection.

d) **Instituto Mexicano del Petroleo IMP (Mexican Petroleum Institute)**They have participated with CLV in the performance of Fire Protection Systems audits and provided consultant services in systems stress analysis methodology.

3.2.2 Foreign Organizations

Among foreign organizations we count with the support of goods and services procurement associations with which CFE has agreements or contracts of colaboration to guarantee the necessary support.

a) General Electric Co. (GE)

Provider of the CLV nuclear island as well as the main safety analysis reports to insure plant's adequate design. It also provides the appropriate engineering support in case of required modifications to original design calculations.

b) Raytheon Enginering & Constructors, Ebasco Division

Architec-Engineer of records for Unit 1, providing continuos engineering & technical support as required by CFE.

c) Mitsubishi Heavy Industries

Main turbine generator suplier, supling technical support & services for maintenence and upgrades.

d) Boiling Water Reactor Owners Group (BWROG)

CLV is a member of this association in charge of analysis & proposals to solutions of common BWRs problems.

e) Institute of Nuclear Power Operations (INPO)

CLV is a member of INPO and there is a continuos exchange of information for the benefit of operational experience.

f) World Association of Nuclear Operations (WANO)

CLV is also a member and holds exchange of information in accordance with WANO standards.

g) Electrical Power Research Institute (EPRI)

CLV is suscribed to this institue in some activities; among of which, the most outstanding: the VIP (Vessel Internals Program).

h) Southwest Research Institute (SWRI)

CLV holds an agreement with this research instiute for the development of welds inspection program, as requested by operations tech specs.

i) Nuclear Operation Maintenance Information Services (NOMIS)

CLV has a 1 year contract with NOMIS for advisory services to plant members to be able to make/receive direct technical advise from plant to plant without intervention from any agency. In addition, they facilitate an urgent spare parts procurement.

3.3 Regulatority Authorities

The Nuclear Regulatory Body of Mexico is the National Comission of Nuclear Safety and Safeguards (CNSNS), it was created by the Mexican law (reglamentary law of Art. 27 of the Constitution) in January 26, 1979. It's subjected to the Secretariat of Energy, and is integrated by an Advisory Board and a General Director.

There are representatives of the following secretariats on the board Foreign Relations; National Defense; Marine; Agriculture, Live Stock and Rural Development; Communications and Transportations; Environment, Natural Resources and Fishing; Labor and Social Services.

The Secretary of Energy attends the meetings as chairman of the board.

CNSNS was created in order to fulfill the need of having an independent authority for the regulation of peaceful utilization of nuclear energy.

The CNSNS program comprises the following activities:

a) Evaluation of safety reports and granting of licenses for nuclear facilities and installations.

b)- Activities related to the compliance of radiological safety regulations in handling and usage of radioactive materials.

c) Activities related to the physical security of nuclear facilities as well as the application of safeguards in handling and utilization of nuclear materials.

d) Activities related to the compliance of radiological safety regulations in mining and processing of transuranium.

The program includes other support functions such as maintain an adequate systems. relations with regulatory agencies in other countries and with the IAEA. as well as training programs, in the aforementioned areas the CNSNS is involved in.

Establishing the rules and regulations of the law in the involved aspects concerning nuclear safety, and surveillance compliance.

- Defining the working procedures for carrying out the regulatory work.
- Establishing training programs to maintain the staff level in specialized areas.
- Acquiring the necessary equipment for performing the involved activities.

In order to define the organization and its necessary resources. the IAEA's recomendations for the structure of a regulatory authority were taken into consideration.

The CNSNS, (see organization chart) is headed by a General Director, under the Secretary of Energy (Chairman of the Board).

The chairman of the board, delegates to the General Director the functions conferred by law, as well as sufficient authority to carry out the assigned responsabilities.

The General Director is entrusted with planning, programming, organizing and directing the CNSNS staff's activities.

The advisory board, with it's interministerial composition, allows an adequate interaction of the CNSNS with the various secretariats represented on the board and receives the necessary support and cooperation when several government agencies have jurisdiction on matters where the regulatory authority should act.

The CNSNS as the regulatory body in the nuclear industry, should:

a) Establish regulatory guides or requirements so that in the development of the nuclear industry the safety and health of the population are protected.

b) Perform the surveillance within the country, for the compliance of legal orders and international treaties signed by Mexico on the subjects of nuclear safety, physical security, radiological safety and safeguards.

c) Review and authorize, when applicable, the basis for design, construction operation and modification activities of nuclear facilities.

d) Establish and maintain a national system of accounting and control of nuclear and radioactive materials.

e) Establish norms for nuclear, physical and radiological safety in order to assure the safe operation of nuclear facilities in the country.

f) Establish radiological protection norms for handling, transportation, possession, and use for radioactive material in the country.

g) Grant permisions, via licenses for importation and exportation of radioactive and nuclear materials and supervise compliance with the aforementioned norms.

h) The activities of CNSNS, in order to comply with the functions conferred by the law, can be divided in two groups:

i) Evaluation of design and construction conditions and authorization of operation of nuclear facilities, as well as supervision and surveillance during their operation, in order to assure that the applicable regulation are complied with and that the public safety and health will not be affected during the life of the plant.

j) Evaluation, and eventual authorization, of facilities to use radioactive material for industrial, medical and research uses, and to assure the strict compliance of applicable norms and adequate radiological protection of the working personnel.

k) The CNSNS has a staff of specialists, highly qualified in several engineering areas. It has technical support from the IAEA, whose colaboration started with Laguna Verde Project's inception.

l) In addition the CNSNS, when necessary, has contracted consultants, either from Mexico of from foreing countries, to complement it's technical capabilities.

m) Other valuable technical supports have come from the United States Nuclear Regulatory Commission and the Junta de Energía Nuclear de España, with whom there is close cooperation and exchange of information agreements.

Another aspect that deserves CNSNS attention is related to aproximatly 1200 users of radioactive materials, spread across the country, to whom the Commission is responsable for issuing licenses for the acquisition and utilization of radioactive material.

Furthermore, the CNSNS takes care that the applicable regulation are adequately observed. This function, due to the high and increasing number of users, demands necessary personnel since all the national territory is covered. Fortunately, the CNSNS receives due support from the higher levels of the Mexican Government to carry out it's functions in a proper way.

3.4 Other Agencies

In addition to the reports submitted to CNSNS in compliance with technical specifications, Laguna Verde submits monthly reports to the Comision Nacional del Agua (National Water Comission), and to the Procuraduría Federal de Protección al Ambiente (Environment Protection Federal Agency). These reports have information about residual discharges and main condenset cooling discharges to the Gulf of México.

3.5 Inspection Agencies (Technical Experts)

In addition to the a-fore-said agencies, at CLV we receive the visits from different groups of experts who conduct activities verification and/or revision and compliance with international standards, and to verify maintenance is performed according to adequate safety and functional standards; among which the following could be mentioned:

a) International Atomic Energy Agency (IAEA) experts who have performed safeguards inspections of fuel control, various OSART missions, and technical visits on specific topics.

b) INPO inspections conducted to verify the plant functioning, planning and control.-

c) Peer Reviews conducted by WANO experts focussed in operations performance.

d) Insurance Company inspections (ASEMEX), to verify compliance with safety regulations as per business policies.

3.6 Main suppliers and subcontractors

According to CFE's Nuclear Division policies, main contractors must comply with their Quality Program veifications. This task is developed and carried out by the Quality Assurance department. The list below shows the plants qualified vendors, of which among the main ones are:

General Electric Co.	CLV's nuclear island, technical support in safety matters, plant modifications, etc)
Mitsubishi Heavy Ind.	Turbogenerator, technical support on BOP
ABB-Power Distribution	Distribution panels & spare parts
EBASCO Serv.Co.(Raytheon)	Architect-Engineer
Southwest Research Institute	In Service Inspection
Dumez Copisa	Engineering & design services
ASTECA, S.A. de C.V.	Tech. support in engineering, insp. and audits
Barlett Nuclear Inc.	Health Phisics & Descontamination Serv.
CIMEX	Analysis and chemical and metalographic tests.

5. MAIN CONCERNS AND PROBLEMS

5.1 Reactor Recirculation Loops Vibration

During U-1 operation cycles, there have been some problems with the isolation discharge valve of the "A" loop, and in the vent and drain valves welds of this same line. This is probably due to high frequency vibrations. It is suspected that the loop geometry is causing flow turbulences which in turn cause these vibrations.

A task group has been formed to address this problem. This group is composed of plant personnel (Operations, Maintenance, System Engineering & Reactor Engineering) personnel of CFE Nuclear Division, and nuclear island supplier (GE) personnel.

This Task Group recommended actions to identify the root cause of the problem and define the pertinent corrective actions, as follows:

- Data collection of the recirculation piping behavior during unit startup (mainly high frequency data) through a data aquisition system installed in both recirculation loops.
- Complete analysis of collected data through specific models, and identification of the root cause.
- Depending on the results of the root cause analysis, development of appropriate corrective actions plan.

5.2 Collective Radiation Exposure and Radiation Level in the Drywell.

One of the CLV main concerns is to reduce the radiation exposure levels reached in the past operation cycles, which have increased in the last U-1 refueling outage.

To reach this objective a special committee was formed with the task ofestablishing a "Dose Minimization Program". In this program, specific concepts have been established in which the committee is currently working. To have a better control of the committee activities, five sections have been formed as follows:

- Radiation Sources Reduction

- Minimization of Leakages
- Human Factors
- Decontamination/Crud Removal
- Miscellaneous

A big concern of this are the drywell radiation levels. Laguna Verde Units 1 and 2 are of BWR 5 GE's design, using two loops and internal jet pumps to recirculate the reactor coolant. These loops, including pumps and valves, are a strong radiation source in the drywell. The fact that the material of many components has a "normal" Cobalt content increases the radiation problem.

In this radiation area CLV experienced technical problems with the recirculation loops, specially with the discharge isolation valve. Their repair and additional maintenance have been an additional source of collective dose.

5.3 Plant Modifications

Laguna Verde Units 1 and 2 have experienced a significative number of modifications, mainly originated from three sources:

Plant Staff (mainly system engineering) propose, in accordance with their duty, modifications to improve safety and efficiency of the system and tore duce radiation doses.

On the utility level (Plant Manager and Nuclear Division Director) decisions were made to implement some modifications to improve the efficiency and reliability of the plant. (e.g. turbine modifications, process computer modification, laundry installation, etc.)

Mandatory changes caused by technical necessities or requirements by regulatory authorities (e.g. off-gas system modifications, loose parts monitoring system installation, etc.).

The following table shows the number of modifications implemented during refueling outages for both Units:

U-1

REFUELING OUTAGE	1st	2nd	3rd	4th	5th
NUMBER OF MODIFICATIONS	78	70	78	84	94

U-2

REFUELING OUTAGE	1st
NUMBER OF MODIFICATIONS	78

6. GOOD PRACTICES/ PERFORMANCE

6.1 Intergranular Stress Corrosion Cracking Prevention

To prevent or mitigate the IGSCC on RPV internals and reactor coolant pressure boundary (RCPB) piping, CFE has taken different actions to improve the component corrosion resistant properties applied to both units of CLV, as follows:

a. Design modifications and welding process control U-1 / U-2.

- Recirculation Bypass minimum flow has been deleted.
- Control Rod Drive Return line has been deleted.
- Core spray Lines HPCS to vessel nozzle N-16 and LPCS to nozzle N-5 are both especified as carbon steel SA/A-106 G B rather than austenitic SS, removing the potential for IGSCC.
- RRC Jet Pump Riser and 10" Inlet Nozzles have been replaced with type 316L SS.
- RRC loops A and B, and attached piping are fabricated of type 304 material and solution heat treated after fabrication.

- Two austenitic SS welding procedures were used for fabrication and installation: manual Shielded Metal Arc Welding (SMAW) and manual Gas Tungsten Arc Welding (GTAW) with consumable insert or filler metal added, both of these processes had heat input controls.

b. Preventive actions taken to reduce igscc for rpv

U-1

- RPV austenitic stainless steel safe-ends replacement in N2, N4, N5 & N16 and N6 nozzles.
- Induction Heating Stress Improvement (IHSI) was performed for recirculation piping system welds.
- Inconel 82 was applied on the Ring to Access Hole Cover 180° weld.
- Design change to N2, N5, N6 and N16 thermal sleeves to avoid potential crevices.

U-2

- RPV austenitic stainless steel safe-end replacement in N2, N4, N5 & N16 and N6 nozzles.
- Induction Heating Stress Improvement (IHSI) was performed for recirculation piping system welds.
- Corrosion Resistant Cladding (CRC) was performed on suction spool riser close out welds, loop "A".
- RPV nozzles modification (Thermal Sleeve) weld overlay clad N1, N2, N4, N5, N16, N6 and N9.
- Water Level Nozzles RPV interior weld overlay clad modification N12, N13 and N14.
- Inconel 82 overlay clad was applied on Jet Pump Diffuser to Shroud Support Baffle Plate weld.
- Inconel 82 overlay clad was applied on Shroud to Shroud Support weld.

- Inconel 82 overlay clad was applied on Incore Housing to Bottom Head weld.
- Inconel 82 overlay clad was applied on Control Rod Drive Housing to Stub-Tube welds.
- Inconel 82 overlay clad was applied on the Ring to Access Hole Cover 180° weld.
- Surface condition upgrade of the Shroud, Shroud Head & Steam Dryer machined and ground surfaces.
- Design change to N2, N5, N6 and N16 thermal sleeves to avoid potential crevices.

c. Residual stress improvement U-1 / U-2

- The recirculation lines in loop "A" and "B" (49 welds) for U-1, (28 welds) for U-2 and RHR Let Down Line (15 welds), have been treated with the Induction Heat Stress Improvement (IHSI) technique, to provide compresion stresses on the inner surface of the weld (root) and Heat Affected Zone (HAZ) of piping.
- 304 SS Type was the material used in the fabrication of the spools for recirculation loops and have been Solution Annealed after manufacture. this meets the recomendations of NUREG-0313 Rev.1 addressing the IGSCC prevention.

d. High quality reactor coolant inventory chemistry control

- Control of oxygen content of 0.2 ppm at rated operating conditions.
- Control of water conductivity with an average of during the first five years of U-1 Operation
- No Reactor Coolant additives are used in the BWR'S primary process loop.
- Deaeration using the condenser and steam jet air ejectors and purification using condensate demineralizers for the steam/feedwater cycle and the Reactor Water Clean-up system.

e. Inspection

- The CLV has the IVVI long term plan to perform the inspection of RPV internals, in accordance with the ASME section XI Code 1989 Edition, GE Engineering recomendations and NUREG'S, SIL'S, IE Bulletins that have been issued to notify the BWR'S the actions to detect the IGSCC indications using the appropriate NDE methods, equipments and techniques.

- We are also following the guidance of Article IWG proposed by ASME Section XI Committee.

- The ISI/IVVI program is in the reviewing process in accordance with the susceptibility analysis results recently performed by CFE for U-1, and will be under continuous conformance with the EPRI/VIP Inspection and Integrity Assessment Component Management Prioritization.

f. Evaluation

- The CLV has been using the manufacturer acceptance criteria, but is reviewing the IWG/ASME acceptance criteria, which will be provided through the EPRI/BWR-VIP group in the near future for the main internal RPV components and CLV is in progress to be a member of the BWR VIP for RPV Reactor Internals Management.

g. Mitigation

- Hydrogen Water Chemistry and Analysis of the dose rate impact for Hydrogen injection is in process to schedule their implementation as soon as posible on U-1 and planning for U-2.

- Feedwater Filters installation on G-16 system is in process, to improve water coolant quality.

6.2 SAFETY RELIEF VALVES TEST STAND

A utility-developed test stand allows testing of main steam safety relief valves on site, is cost effective, and reduces valve-out-of-service time. The design specifications were developed by Laguna Verde personnel and the test stand was designed and constructed by a local vendor. The test stand contains a small boiler that allows the safety relief valves

(SRV's) to be tested and calibrated using steam as the pressure source. This eliminates potential problems with using conversion tables for testing with an alternative pressure medium.

Use of the test stand also avoids turnaround times associated with off-site calibrations. Station personnel are able to obtain more accurate test results with the test stand than previously obtained using a vendor.

6.3 EMERGENCY PREPAREDNESS

a. Emergency Classification Flow Diagram

The process used to classify emergencies at Laguna Verde NPP helps in the prompt recognition of emergency action levels. Contributing to the efficient process is a large color-coded flow diagram that is prominently located in each control room and technical support center. Categories include considerations for radiation, loss of reactor coolant system or containment barriers, loss of safe shutdown systems or instrumentation, and other miscellaneous events or natural phenomena. The logic used in classification of events follows the guidance in the National Environmental Studies Project (NESP) developed by NUMARC. This guidance was used in conjunction with Laguna Verde original classification method, based on NUREG 0654 methodology, to formulate the emergency classification flow diagram.

The layout of the flow diagram and the incorporation of key words that are consistently used to describe event trigger points provide an effective tool for use by emergency response personnel. Additionally, a detailed basis document explaining specific classification criteria is readily available for reference. The organization of this document is such that questions concerning a classification may be researched in a timely manner.

b. Emergency Operating Procedure Flow Diagram

References to specific entry points in the emergency classification flow diagram are incorporated into the emergency operating procedure flow diagrams to enhance the recognition of plant conditions that could escalate the emergency. Each reference is clearly recognizable using a multicolor designation. The classification reference relates to a specific location on the emergency classification flow diagram.

The flow diagrams are displayed in each control room and in the technical support center. The flow diagrams are color coded and are designed in an organized fashion that promotes efficient recognition of emergency conditions.

c. Multiple Event Classification

The guidance for classification of emergency events and activation of the emergency response organization (ERO) during situations when multiple events at different nuclear plant units occur simultaneously is well organized. This guidance consists of a one pagle table that uses different combinations of scenarios to illustrate the following:

* the proper classification to declare for the site

* the unit shift supervisor that initially is in charge of the emergency(s)

* the number of personnel in the emergency response organization (ERO) to activate

* the correct unit (Unit 1 or Unit 2) emergency response facilities to activate.

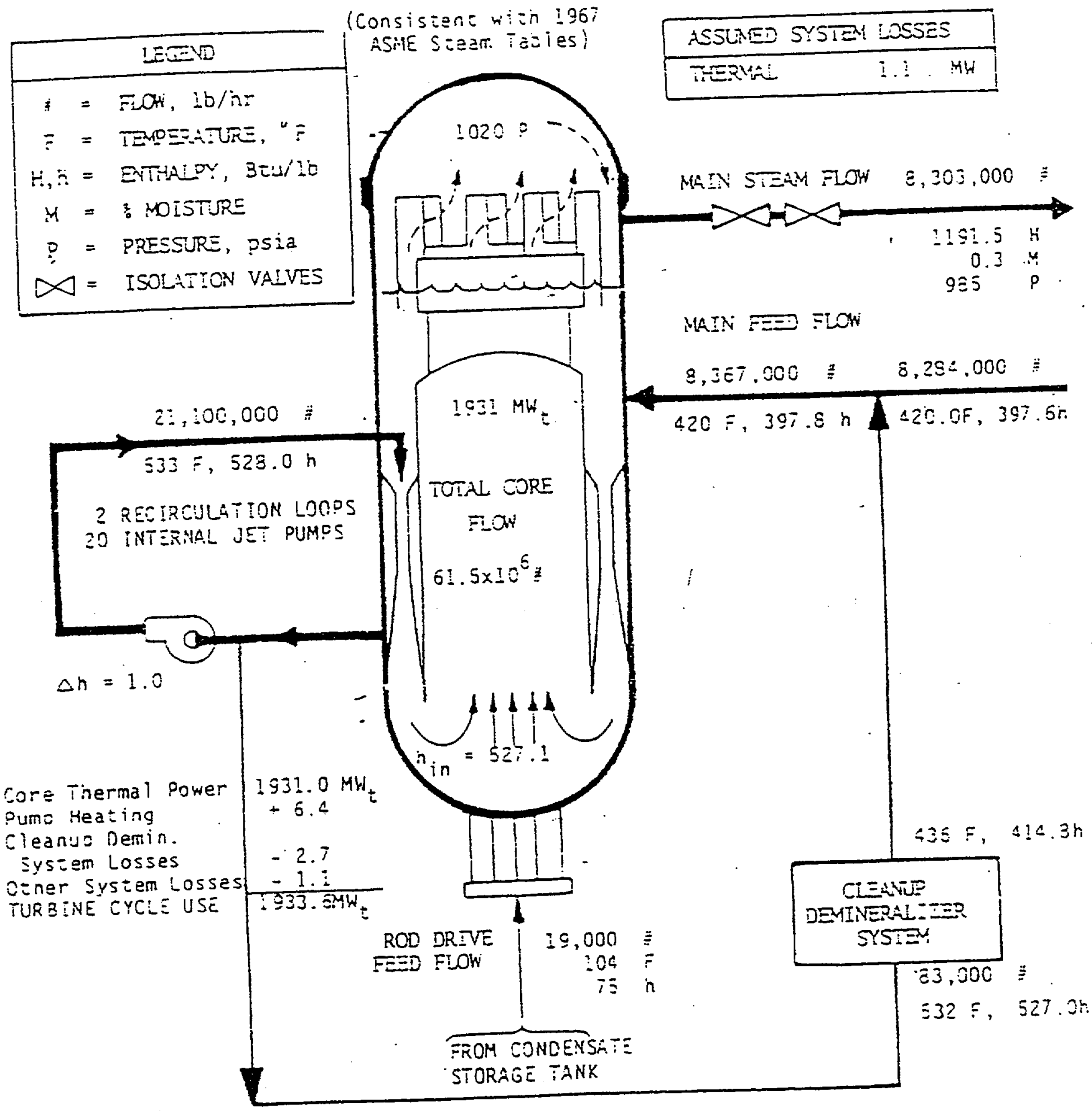


FIGURE 7.2. HEAT BALANCE AT RATED POWER

FIGURE 7.3. START-UP TESTING CHRONOLOGY (Comparison between U-1 and U-2).

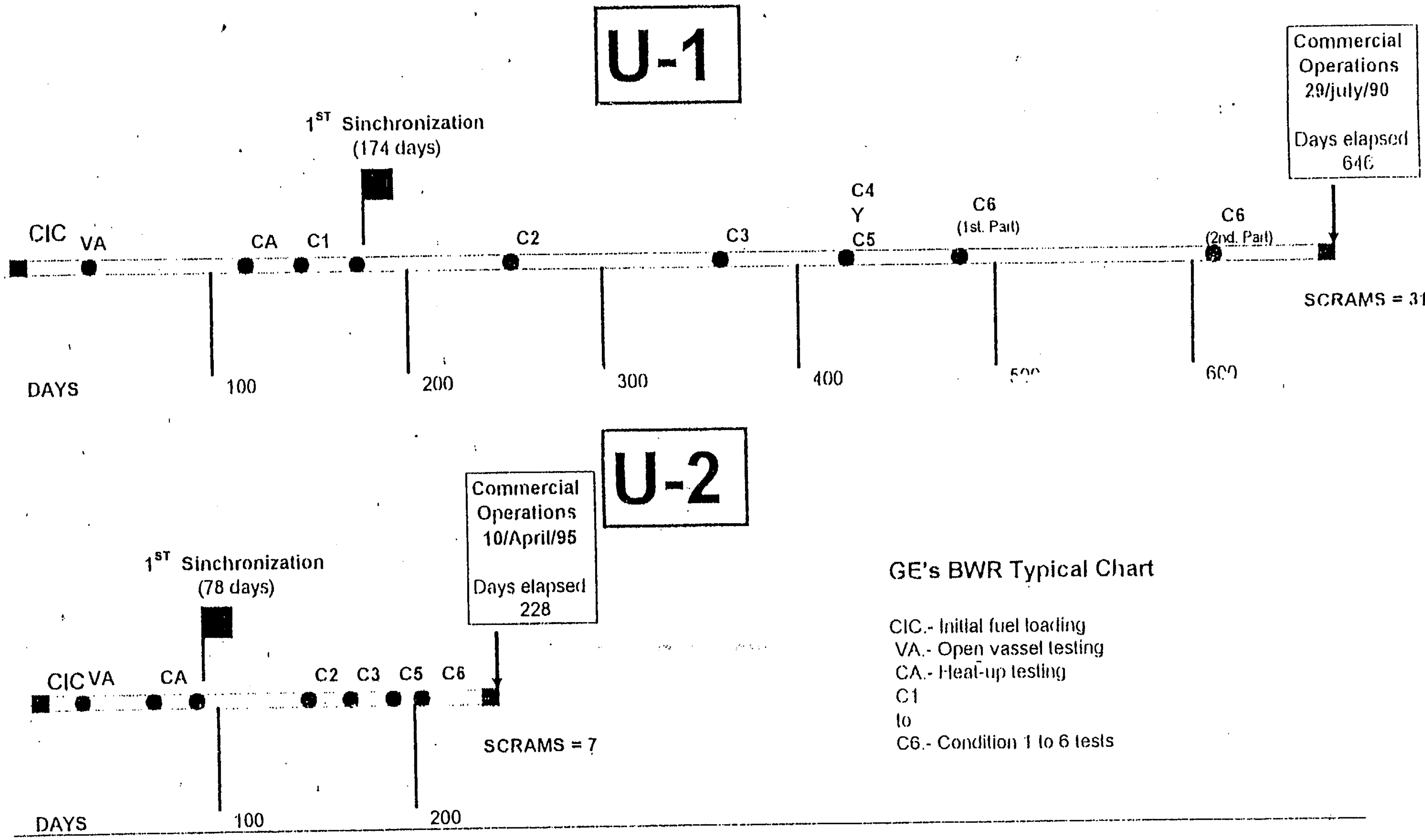
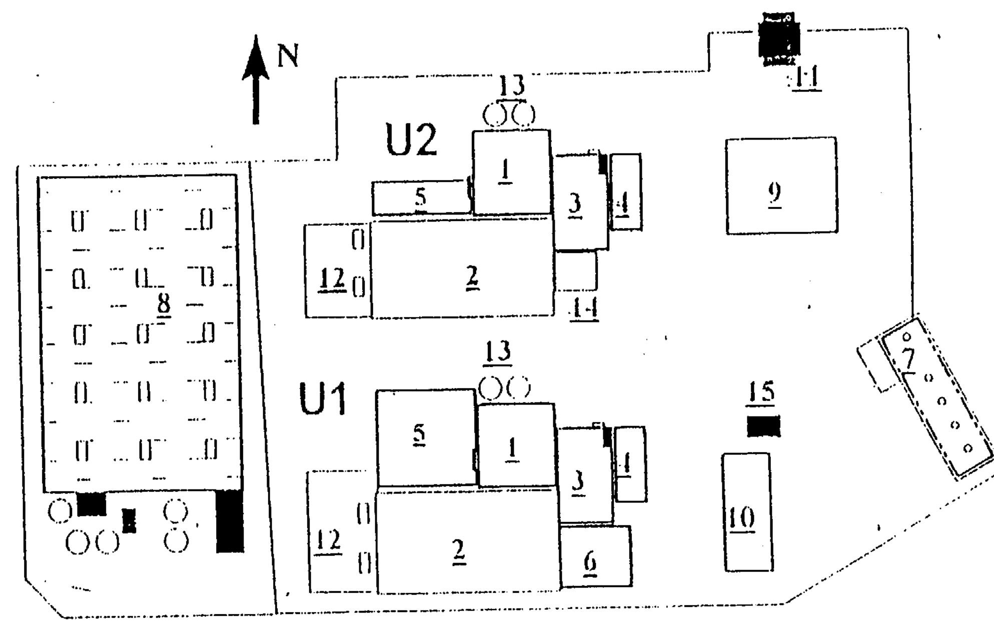


FIGURE 7.4. PLANT LAYOUT



1.- Reactor Building
2.- Turbogenerator Building
3.- Control Building
4.- Diesel Generators Building
5.- Radwaste Building
6.- Water Treatment & Shop Building
7.- Water Intake Building
8.- Switchyard
9.- Administrative Building
10.- Spare Parts Warehouse
11.- Guard House
12.- Main Transformers Yard
13.- Condensate Storage Tanks
14.- SIIP Building
15.- Alarm Central Station

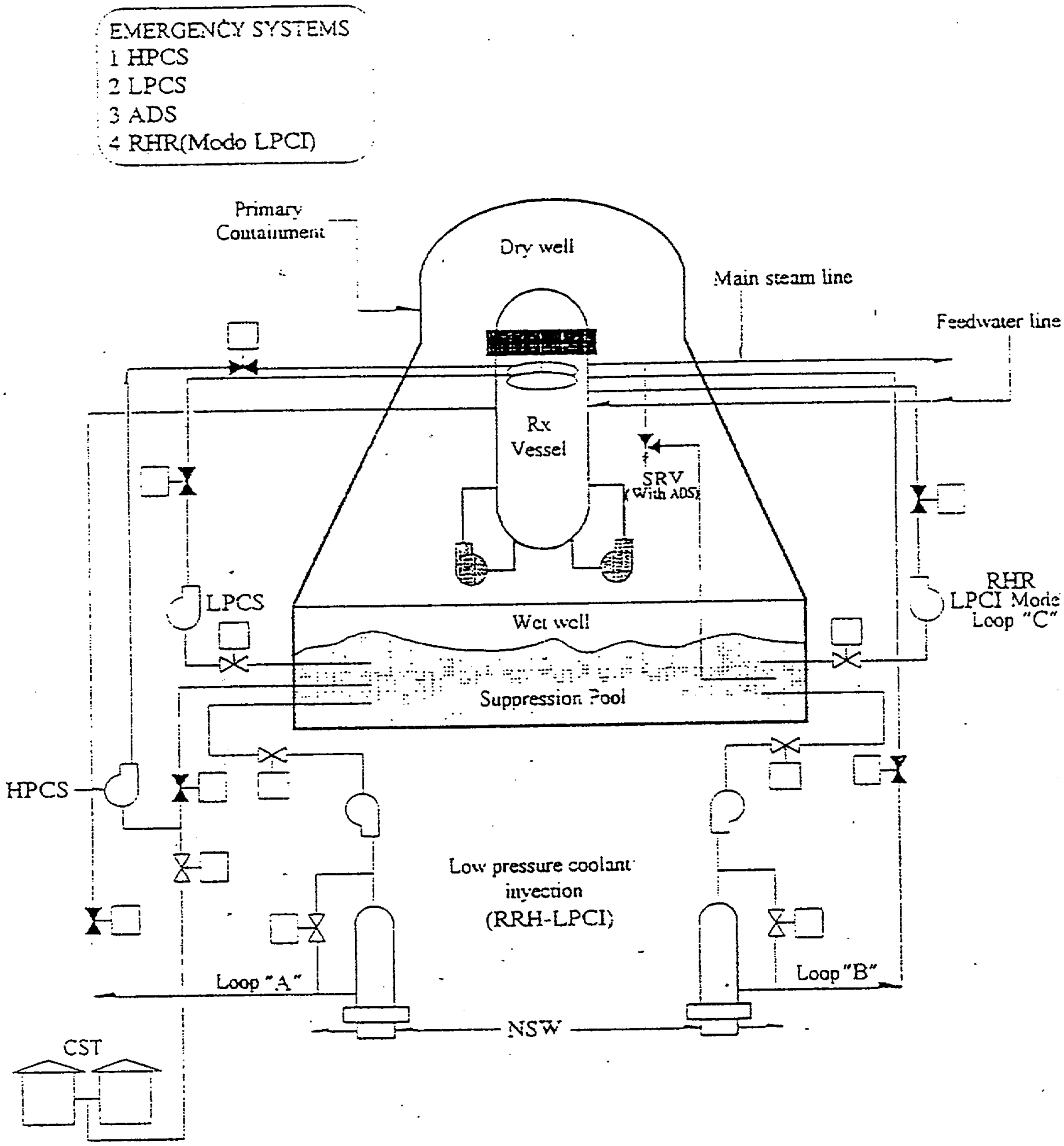


Figure 7.5. EMERGENCY CORE COOLING SYSTEMS

FIGURE 7.6. PLANT AUXILIARY POWER DISTRIBUTION DIAGRAM

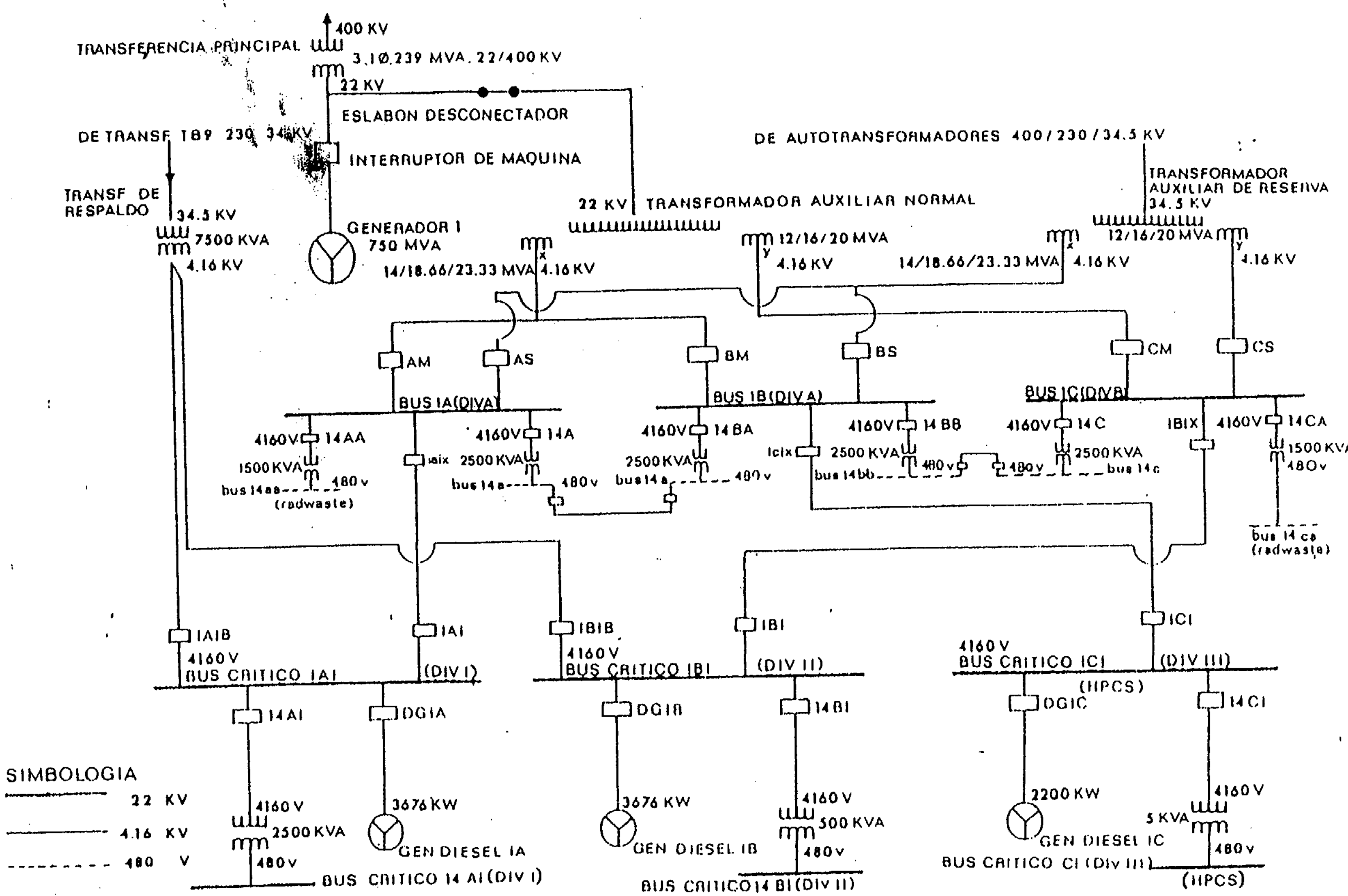


FIGURE 7.7. CFE ORGANIZATIONS AND NUCLEAR DIRECTION

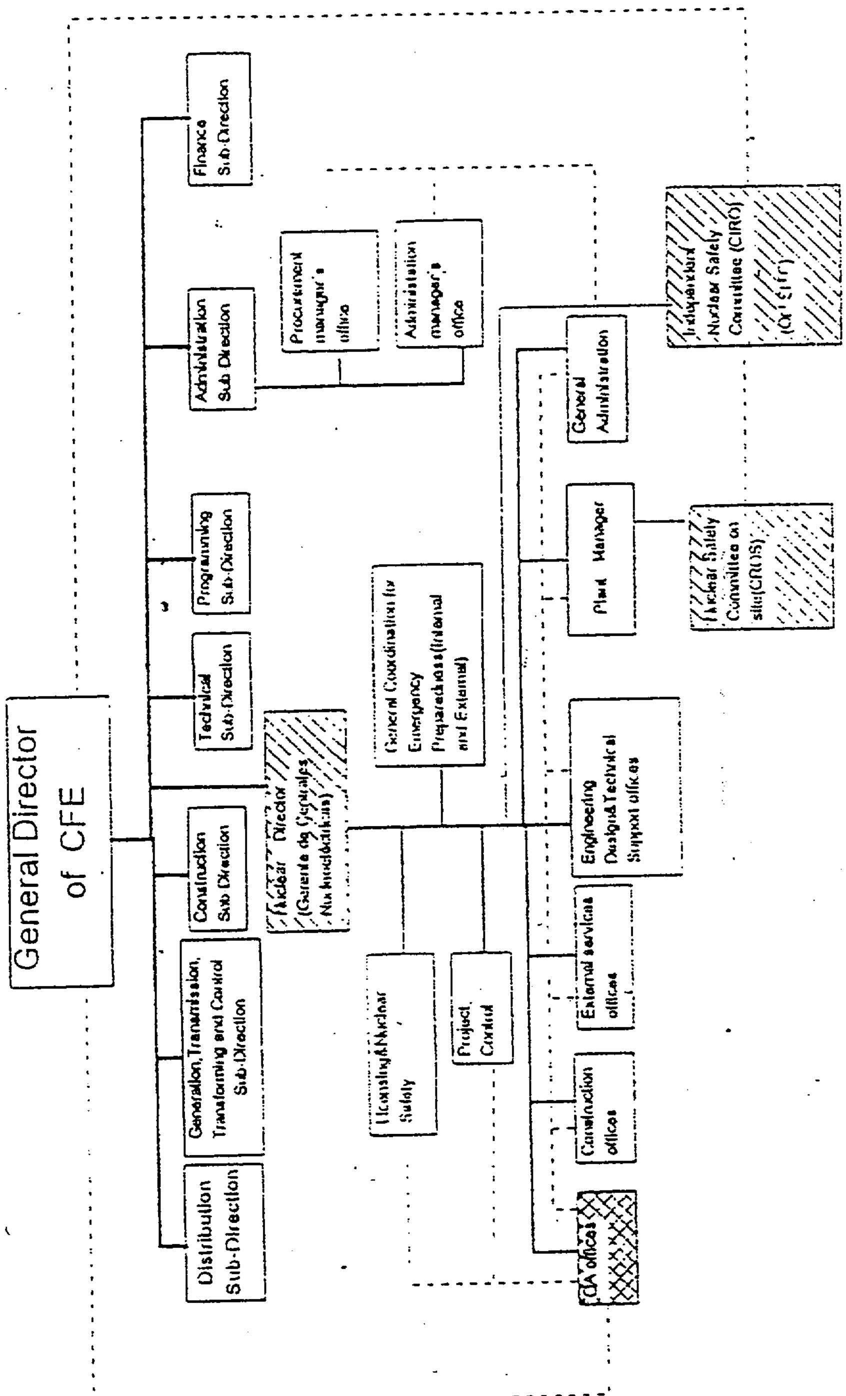


FIGURE 7.8. OPERATIONS ORGANIZATION CHART