INTRODUCTION AND GENERAL DESCRIPTION OF PLANT

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1.0 INTRODUCTION AND GENERAL DESCRIPTION OF PLANT

1.1 INTRODUCTION

This Updated Final Safety Analysis Report (UFSAR) is submitted in accordance with the requirements of 10 CFR 50.71(e). It is based on the original FSAR, including 14 amendments, which was submitted in support of an application by Florida Power & Light Company for a license to operate a nuclear power unit designated as St. Lucie Unit 2. The unit is located on Hutchinson Island in St. Lucie County about halfway between the cities of Fort Pierce and Stuart on the east coast of Florida.

This submittal contains updated information which is accurate for the period up to six months prior to the most recent revision of this document. The updated material is of the same level of detail presented in the original FSAR. It includes changes necessary to reflect information and analysis submitted to the NRC or prepared pursuant to Commission requirements, and it includes changes describing physical modifications to the plant.

Generally, the information provided in the original FSAR where no update is required is retained for historical purposes.

The original Nuclear Steam Supply System (NSSS) is a pressurized water reactor system designed by Combustion Engineering Incorporated. The containment structure is comprised of a steel containment vessel designed by Chicago Bridge & Iron Company, and is surrounded by a reinforced concrete Shield Building designed by Ebasco Services Incorporated.

The initial rating of the NSSS thermal power level was 2570 Mwt (including a 10 Mwt net heat addition from reactor coolant pumps). 2560 Mwt was the projected initial operating power of the core and the power at which the thermal and hydraulic aspects of the core had been analyzed. The corresponding net electrical output for the rated power level was 802 Mwe. Subsequent to the Cycle 2 reload, St. Lucie Unit 2 requested and was granted a stretch power rating of 2700 Mwt. This corresponds to a net electrical output of 830 Mwe. The UFSAR has been modified, where necessary to reflect the revisions brought about by this increased power level. The design thermal power level is 2700 Mwt, the maximum expected output of the Nuclear Steam Supply System. This is the basis for the design of the balance of plant and related facilities, including the major systems and components, the Engineered Safety Features and for site radiological release calculations.

Prior to the Cycle reload, St. Lucie Unit 2 requested an extended power rating of 3020 Mwt, comprised of an approximate 10% Extended Power Uprate (EPU) and a 1.7% Measurement Uncertainty Recapture (MUR). This represents an approximate 11.85% increase from the stretch power rating of 2700 Mwt. The UFSAR has been modified, where necessary, to reflect revisions brought about by this increased power level.

Both original steam generators (OSGs) were removed and replacement steam generators (RSGs) designed and manufactured by AREVA were installed. The effect of the RSG installation on the information provided in the UFSAR is specifically noted in the affected sections.

1.2 GENERAL PLANT DESCRIPTION

1.2.1 PRINCIPAL SITE CHARACTERISTICS

The site for St. Lucie Units 1 and 2 consists of approximately 1,132 acres. The unimproved area of the site is generally flat, covered with water and has a dense vegetation characteristic of Florida coastal mangrove swamps. At the ocean shore the land rises slightly in a dune or ridge to approximately 15 ft, above mean low water.

The island and the adjoining mainland are sparsely populated. The southern most boundary of the nearest population center is the City of Fort Pierce which is 4.1 miles from the site. The City of Fort Pierce has an estimated population of 33,083 people as of a 1978 estimate. The minimum site exclusion radius is 5,100 ft. Site characteristics are given in Chapter 2.

1.2.2 PRINCIPAL DESIGN CRITERIA

Principal structures, system and equipment which may serve either to prevent accidents or to mitigate their consequences are designed and erected in accordance with applicable codes to withstand the most severe earthquakes, flooding conditions, windstorms, temperature and other deleterious natural phenomena which could be reasonably assumed to occur at the site during the lifetime of the plant. Principal structures, systems and equipment are sized for the design power level of the nuclear steam supply system output.

Redundancy is provided in the reactor protective and engineered safety feature systems so that no single failure of any active component of the systems can prevent action necessary to avoid an unsafe condition. The plant is designed to facilitate inspection and testing of systems and components whose reliability are important to plant shutdown and to the protection of the public and plant personnel.

Provisions are made to minimize the probability and effect of fires and explosions, in accordance with 10 CFR 50.48(c), NFPA 805.

Systems and components which are significant from the standpoint of nuclear safety are designed, fabricated and erected to quality standards commensurate with the safety function to be performed.

Section 3.1 addresses the implementation of the NRC General Design Criteria for Nuclear Power Plants, 10 CFR Part 50, Appendix A. Chapter 17 describes the quality assurance program for the design and operation of St. Lucie Unit 2.

1.2.2.1 Reactor

The reactor is of the pressurized water-type, designed to provide heat to steam generators which, in turn, provide steam to drive a turbine generator. The full power core thermal output is 3020 megawatts.

The reactor core is fueled with UO_2 and UO_2-Gd_2 O_3 and/or UO_2 Er_2O_3 pellets enclosed in zircaloy tubes pressurized with helium and fitted with welded end plugs.

The tubes are fabricated into assemblies in which end fittings prevent axial motion and spacer grids prevent lateral motion of the tubes. Beginning with Region N, the fuel incorporates the GUARDIAN™ fuel assembly design to screen and entrap debris. The GUARDIAN™ design employs a redesigned bottom spacer grid that provides positive axial restraint to the rods and added screening features. Region N also includes the addition of “backup arches” adjacent to all cantilevered springs in the interior of the upper H1D-1L spacer grid or top Inconel grid (beginning with Region U). The backup arch limits the possible compression of the grid spring, and thereby better maintains the proper geometry between the grid support features and the fuel rod during fabrication and operation. This same feature was present in peripheral locations in each Zircaloy spacer grid for all previous St. Lucie 2 fuel batches. In these locations, the backup arches protect the grid springs that may be subject to compression during fuel handling, when peripheral fuel rods can be pressed inward as bowed fuel assemblies are slid past one another in the core. In the new upper grid design, the arches will be present at all 440 interior spring locations in the grid. The backup arches will thus limit compression of grid springs in all interior locations during fuel rod loading. The control element assemblies (CEAs) consist of inconel clad boron carbide absorber rods which are guided by zircaloy tubes located within the fuel assembly. The core consists of 217 fuel assemblies.

Beginning with Cycle 23, the feed fuel is of the AREVA CE-16 HTP™ fuel design.The AREVA CE-16 HTP™ fuel consists of a 16x16 assembly configuration with M5® clad fuel rods, Zircaloy-4 MONOBLOC™ Corner Guide tubes, an Alloy 718 HMP™ spacer at the lowermost axial elevation, Zircaloy-4 HTP™ spacers in all other axial elevations, a FUELGUARD™ lower tie plate, and the AREVA reconstitutable upper tie plate.

The M5® Zirconium alloy has been consistently shown to provide superior corrosion resistance and growth performance.The robust FUELGUARD™ lower tie plate provides highly effective debris resistance with good flow characteristics and an acceptable pressure drop.The HTP™ spacer grid design has shown improved protection against fuel rod fretting damage and increased structural integrity of fuel assemblies. The bottom HMP™ spacer grid design made from Alloy 718 reduces cell relaxation during irradiation to prevent fuel rod movement.The MONOBLOC™ corner guide tubes increases the wall thickness in the bottom approximately 14 inches of the guide tube, slightly increasing fuel assembly stiffness.

The AREVA CE-16 HTP™ fuel has been approved by the USNRC for implementation at St. Lucie Unit 2 as described in the Safety Evaluation (SE) in Reference 4.

Minimum departure from nucleate boiling ratio (DNBR) during normal operation and anticipated operational occurrences is not less than 1.28 (cycle 1 was 1.19) using the CE-1 correlation. The maximum center line temperature of the fuel, evaluated at the design overpower condition, is below that value which could lead to fuel rod failure. The melting points of the UO_2 and UO_2-Gd_2O_3 and/or UO_2-Er_2O_3 are not reached during routine operation and anticipated operational occurrences. For the AREVA CE16 HTP™ fuel, Minimum DNBR during normal operation and anticipated operational occurrences is not less than the correlation safety limit using the HTP correlation.

The combined response of the fuel temperature coefficient, the moderator temperature coefficient, the moderator void coefficient and the moderator pressure coefficient to an increase in reactor thermal power is a decrease in reactivity. In addition, the reactor power transient remains bounded and damped in response to any expected changes in any operating variable.

Control element assemblies (CEAs) are capable of holding the core sub-critical at hot zero power conditions with margin following a trip even with the most reactive CEA stuck in the fully withdrawn position.

Fuel rod clad is designed to maintain cladding integrity throughout fuel life. Fission gas release within the rods and other factors affecting design life are considered for the maximum expected exposures.

The reactor and control systems are designed so that any xenon transients are adequately damped.

The reactor in conjunction with the Reactor Protective System is designed to accommodate safely and without fuel damage, the anticipated operational occurrences.

The reactor vessel and its closure head are fabricated from manganese molybdenum nickel steel internally clad with austenitic stainless steel. The vessel and its internals are designed so that the integrated neutron flux does not exceed 4.9 x 10^{19} n/cm^2 (E > 1 Mev) over the 60 year design life of the vessel.

Power excursions which could result from any credible reactivity addition do not cause damage, either by deformation or rupture of the reactor vessel and do not impair operation of the Engineered Safety Features.

The internal structures include the core support barrel, the lower support structure, the core shroud, the hold-down ring and the upper guide structure assembly. The core support barrel is a right circular cylinder supported from a ring flange from a ledge on the reactor vessel. The flange carries the entire weight of the core. The lower support structure transmits the weight of the core to the core support barrel by means of vertical columns and a beam structure. The core shroud surrounds the core and limits the amount of coolant bypass flow. The upper guide structure provides a flow shroud for the CEAs and prevents upward motion of the fuel assemblies during pressure transients. Lateral motion limiters or snubbers are provided at the lower end of the core support barrel assembly. The hold-down ring acts as a shim and is set between the reactor vessel head and the upper guide structure to resist axial upward movement.

Further details concerning the reactor are given in Chapters 3 and 4.

1.2.2.2 Reactor Coolant and Auxiliary Systems

The Reactor Coolant System is arranged as two closed loops connected in parallel to the reactor vessel. Each loop consists of one 42 in. ID outlet (hot) pipe, one steam generator, two 30 in. ID inlet (cold) pipes and two reactor coolant pumps. An electrically heated pressurizer is connected to the hot leg of one of the loops and a safety injection line is connected to each of the four cold legs.

The Reactor Coolant System operates at a nominal pressure of 2235 psig. The reactor coolant enters near the top of the reactor vessel, and flows downward between the reactor vessel shell and the core support barrel into the lower plenum. It then flows upward through the core, leaves the reactor vessel, and flows through the tube side of the two vertical U-tube steam generators where heat is transferred to the secondary system. Reactor coolant pumps return the reactor coolant to the reactor vessel.

The two steam generators are vertical shell and U-tube units. The steam generated in the shell side of the steam generator flows upward through moisture separators and scrubber plate dryers which reduce the moisture content to less than 0.2 percent. All surfaces in contact with the reactor coolant are either stainless steel or NiCrFe alloy in order to minimize corrosion.

The reactor coolant is circulated by four electric motor driven single suction vertical centrifugal pumps. The pump shafts are sealed by mechanical seals. Each pump motor is equipped with an antireverse mechanism to prevent reverse rotation.

Components of the Reactor Coolant System are designed and operated so that no stresses are imposed on the structural materials that result in loss of function. The necessary consideration has been given to the ductile characteristics of the materials at low temperatures.

The Reactor Coolant System is designed and constructed to maintain its integrity throughout the plant life. Appropriate means of test and inspection are provided.

See Chapter 5 for further information.

1.2.2.3 Engineered Safety Features

The plant design incorporates redundant Engineered Safety Features. These systems in conjunction with the containment system, ensure that the offsite radiological consequences following any LOCA up to and including a double ended break of the largest reactor coolant pipe do not exceed the guidelines established for design basis accidents. The systems also ensure that the guidelines of 10 CFR 50 Appendix K, "Acceptance Criteria for Emergency Core Cooling Systems" are satisfied, based upon analytical methods, assumptions and procedures accepted by the NRC. The Engineered Safety Features include: (a) independent redundant systems (Containment Cooling System and Containment Spray System) to remove heat from and reduce the pressure in the containment vessel in order to maintain containment integrity, (b) a high and low pressure Safety Injection System to limit fuel and cladding damage to an amount which does not interfere with adequate emergency core cooling and to limit metal-water reactions to negligible amounts, (c) a Shield Building Ventilation System to reduce offsite consequences due to leakage from the containment vessel, (d) a containment isolation system to minimize post-LOCA radiological effects offsite, (e) a hydrogen control system to maintain safe post-LOCA hydrogen concentration within the containment, and (f) a control room habitability system.

EC291393

The Reactor Building, which is a dual containment design, is comprised of a steel containment vessel surrounded by an annular space and enclosed by a reinforced concrete Shield Building. The containment vessel is a low leakage steel shell which is designed to confine the radioactive material that could be released from a postulated design basis, Loss-of-Coolant Accident, (LOCA). It is a cylindrical vessel with hemispherical dome and ellipsoidal bottom. The Shield Building is a medium leakage concrete structure which surrounds the annulus and steel containment vessel. It protects the containment vessel from external missiles, and provides biological shielding and a means of collecting radioactive fission products that may leak from the containment following a major hypothetical accident (see Subsection 6.2.1 for details).

The containment in conjunction with either of the associated spray and cooling systems is designed to withstand the internal pressure and coincident temperature resulting from the energy release associated with the design basis accident. The containment is equipped with two 100 percent capacity heat removal systems, each comprised of one containment spray loop and two containment cooling units.

The Containment Spray System supplies borated water to cool and reduce pressure in the containment atmosphere. The pumps take suction initially from the refueling water tank. Long term cooling is based on suction from the containment sump through the recirculation lines.

The Containment Cooling System provides containment atmosphere mixing by recirculation. The cooling coils and fans of the Containment Cooling System are sized to provide adequate containment cooling at post-accident conditions of temperature, pressure and humidity (see Subsection 6.2.2 for details).

In the event of a LOCA, the Safety Injection System described in Section 6.3 injects borated water into the Reactor Coolant System. This provides cooling to limit core damage and fission product release, and assures adequate shutdown margin. The injection system also provides continuous long term post-accident cooling of the core by recirculation of borated water from the containment sump through the shutdown heat exchangers and back to the reactor core.

The Shield Building Ventilation System is provided to maintain a negative pressure in the annulus between the steel containment vessel and the concrete Shield Building following a LOCA. Two independent 100 percent capacity systems are provided. This system filters any radioactivity leakage from the containment vessel and therefore reduces the effects on the environment (see Subsection 6.2.3 for details). The SBVS is provided with carbon absorbers for iodine removal in the Shield Building. EC291393

A containment isolation system consisting of valves and associated actuators and controls is provided for each line penetrating the containment that must be closed to prevent a radioactivity release in the case of a loss-of-coolant accident (see Subsection 6.2.4 for details).

A hydrogen control system is provided which consists of a hydrogen sampling system. A hydrogen purge system is provided as a non-safety, diverse system (see Subsection 6.2.5 for details). EC295353

The control room habitability system is provided to limit control room doses from airborne activity to within GDC 19 limits (see Section 6.4 for details).

1.2.2.4 Protection, Control, Instrumentation and Electrical Systems

a) Reactor Protective System

The reactor parameters are maintained within the acceptable limits by the inherent characteristics of the reactor, by the Reactor Regulating System, by boron in the moderator and by the operating procedures. In addition in order to preclude unsafe conditions for plant equipment or personnel, the Reactor Protective System initiates reactor trip if a selected parameter reaches its preset limit. Four independent channels normally monitor each of the selected plant parameters. The Reactor Protective System logic initiates protective action whenever the signal of any two of three channels reaches the preset limit. A fourth channel is provided as a spare and allows bypassing of one channel while maintaining a two-out-of-three system. If any two channels receive coincident signals, the power supply to the magnetic jack control element drive mechanisms is interrupted releasing the control elements to drop into the core to shutdown the reactor. Redundancy is provided in the Reactor Protective System to assure that no single failure prevents protective action when it is required. The protective system is completely independent of and separate from the control system (see Section 7.2 for details).

b) Control System

The reactor is controlled by a combination of control element assemblies (CEAs) and dissolved boric acid in the reactor coolant. Boric acid is used for reactivity changes associated with large but gradual changes in water temperature, core xenon, fuel burnup and power levels. Additions of boric acid also provide an increased shutdown margin during the initial loading and subsequent refuelings. The boric acid solution is prepared and stored at a temperature sufficiently high to prevent precipitation. CEA movement provides changes in reactivity for shutdown or power changes. The CEAs are actuated by control drive mechanisms mounted on the reactor vessel head. The control drive mechanisms are designed to permit rapid insertion of the CEAs into the reactor core by gravity. CEA trip motion can be initiated manually or automatically.

The Reactor Regulating System (RRS) was designed to control reactivity to maintain the programmed reactor coolant temperature and power level which includes the capability to load follow. The RRS was designed to match the Nuclear Steam Supply System capability of following a ramp change from 15 percent to 100 percent power at a rate of five percent per minute and at greater rates over smaller load change increments up to a step change of 10 percent.

A RRS temperature controller is used to compare the existing average reactor coolant temperature with the value corresponding to the power called for by the temperature control program. If the temperature is different, the CEAs are manually adjusted to bring the two temperatures within the prescribed control band. Regulation of the reactor coolant temperature in accordance with this program maintains the secondary steam pressure within operating limits and matches reactor power to load demand.

The CEAs are moved through manual operation by the operator.

The pressure in the Reactor Coolant System is controlled by regulating the temperature of the coolant in the pressurizer, where steam and water are held in thermal equilibrium. Steam is formed by the pressurizer heaters or condensed by the pressurizer spray to reduce variations caused by expansion and contraction of the reactor coolant temperature changes. The pressure and water level control systems are described in Subsection 7.7.1.1.

Overpressure protection of the Reactor Coolant System is provided by power operated relief valves and spring loaded safety valves connected to the pressurizer. The discharge from the pressurizer safety and relief valves is released under water in the pressurizer quench tank, where it is condensed and cooled. In the event the discharged steam exceeds the capacity of the tank, the tank relieves to the containment atmosphere (see Subsections 5.2.2, 5.2.6, and 5.4.13 for details).

A Turbine Control System is provided to regulate steam flow to the turbine as a function of system load. In the event of turbine trip, bypass systems are provided to release steam to the condenser and to the atmosphere. These systems are designed to reduce the sensible heat in the Reactor Coolant System, maintain the steam generator pressure during hot standby, and meet the original design basis of 45 percent steam bypass capability to mitigate challenges to the pressurizer and steam generator safety valves (see Section 7.7).

A Steam Generator Water Level Control System regulates feedwater flow to the steam generator (see Subsection 7.7.1.1). An Auxiliary Feedwater System is provided to ensure flow to the steam generators during plant startup, plant shutdown, and in the event of a plant design basis accident.

c) Instrumentation System

The nuclear instrumentation includes excore and incore neutron flux detectors. Twelve channels of excore instrumentation monitor the neutron flux and provide reactor protection and control signals during start up and power operation. Four of the channels are wide range logarithmic safety channels to measure neutron flux from source range to 200 percent of full power. Another four channels are power range safety channels to measure neutron flux linearly from one percent to 200 percent of full power. The power range safety channels are used by the reactor protection system to determine the neutron flux power and axial offset, and by the high power bypass circuitry for the high rate-of-change of power trip (see Subsection 7.2.1.1). There are two linear power range channels utilized for control purposes and two channels for startup and extended shutdown (see Subsection 7.7.1.1.9).

The original feedwater flow and temperature instrumentation consisting of venturis, differential pressure indications and Resistance Temperature Detectors (RTDs) has been supplemented by the installation of a Cameron/Caldon Leading Edge Flow Meter (LEFM) Checkplus system. This change supports the MUR 1.7% increase in core thermal power. The original feedwater flow and temperature instrumentation was retained and is used for companson monitoring of the LEFM system and as a backup feedwater mass flow measurement when needed (see Subsection 7.7.4).

The incore instrumentation consists of self-powered rhodium neutron detectors and background detectors to provide information on neutron flux distribution.

The process instrumentation monitoring includes those critical channels which are used for protective action. Temperature, pressure, flow and liquid level monitoring is provided, as required, to keep the operating personnel informed of plant conditions and to provide information from which plant processes can be evaluated and/or regulated.

Instrument signals transmitted from the containment are electric. Instrument signal transmission for the remaining plant instruments is either electric or pneumatic (see Chapter 7 for details).

The plant gaseous and liquid effluents are monitored to assure that they are maintained within acceptable radioactivity limits. Activity levels are displayed and off- normal values are annunciated. Area monitoring stations measure radioactivity at selected locations in the plant for personnel protection. A complete description of the radiation instrumentation is contained in Section 11.5 and Subsection 12.3.4.

d) Electrical System

Redundant sources of offsite power are provided by four separate transmission lines.

The unit includes a 1,200 MVA, 0.9 power factor generator delivering power to a 230 kV switchyard through step-up power transformers. Auxiliary power is utilized at 6.74 kV (a 6.9 kV winding is provided for the start up transformers), 4.16 kV, 480V, and 120V ac; 125V dc systems are also provided. For emergency power, Engineered Safety Features control, and essential nuclear instrumentation, all voltages except 6.74 kV are provided.

The auxiliary load is normally supplied from two auxiliary transformers connected to the main generator bus. Start up power is supplied from two start up transformers connected to the 230 kV switchyard. Emergency power for the Engineered Safety Features is supplied by redundant diesel generator sets (see Chapter 8 for details).

1.2.2.5 Power Conversion System

The power conversion system removes heat energy from the reactor coolant in two U-tube steam generators, and converts the steam into electrical energy by means of a turbine-generator. The unusable heat in the steam cycle is transferred to the main condenser for rejection by the Circulating Water System. The resulting condensate is deaerated in the condenser, then heated through feedwater heaters and returned to the steam generators as feedwater.

The turbine generator is a Siemens Energy Inc. unit. It is an 1,800 rpm tandem-compound, four-flow exhaust unit. The feedwater pumps are electric motor driven. Each of two strings of feedwater heaters consists of four low pressure and one high pressure heaters.

The Auxiliary Feedwater System contains two electric motor driven pumps and one pump driven by a noncondensing steam turbine. This system provides a source of water inventory to the steam generators during plant startup and hot standby, and during plant cooldown provides heat removal to bring the Reactor Coolant System to the shutdown cooling system activation window. (See Chapter 10 for details.)

1.2.2.6 Fuel Handling and Storage Systems

The fuel handling systems provide for the safe handling of fuel assemblies and control element assemblies under all foreseeable conditions and for the required assembly, disassembly, and storage of the reactor vessel head and internals. These systems include a refueling machine located inside containment above the refueling cavity, the fuel transfer carriage, the upending machine, the fuel transfer tube, a spent fuel handling machine in the Fuel Handling Building, and various devices used for handling the reactor vessel head and internals (see Subsection 9.1.4 for details). Dry storage of spent fuel is provided as discussed in Section 1.2.2.9.

New fuel is stored dry in vertical racks in the Fuel Handling Building. The rack and fuel assembly spacing precludes criticality (see Subsection 9.1.1 for details).

The spent fuel pool is a reinforced concrete structure, stainless steel lined. Spent fuel assemblies are stored in vertical racks. Spacing between fuel assemblies is such that the effective neutron multiplication factor (k_{eff}) will remain less than 1.0 for non-accident conditions when no credit is taken for the boron in the pool water (see Subsections 9.1.2 and 9.1.3.3.2 for details). As discussed in Subsection 9.1.2.3, partial credit is taken for the negative reactivity of soluble boron in fuel pool water during certain postulated accidents.

Cooling and purification equipment is provided for the fuel pool water. This equipment may also be used for cleanup of refueling water after each fuel change in the reactor (see Subsection 9.1.3 for details).

1.2.2.7 Cooling Water and Other Auxiliary Systems

a) Chemical and Volume Control System

The purity level in the Reactor Coolant System is controlled by continuous purification of a bypass stream of reactor coolant. Water removed from the Reactor Coolant System is cooled in the regenerative heat exchanger. From there, the coolant flows to the letdown heat exchanger and then through a filter and a demineralizer where corrosion and fission products are removed. It is then sprayed into the volume control tank and returned to the regenerative heat exchanger by the charging pumps where it is heated prior to return to the Reactor Coolant System.

The Chemical and Volume Control System automatically adjusts the amount of reactor coolant in order to maintain a constant level in the pressurizer. This compensates for changes in specific volume due to coolant temperature changes and reactor coolant pump shaft controlled seal leakage (see Subsection 9.3.4 for details).

The Chemical and Volume Control System is capable of adding boric acid to the reactor coolant at a rate sufficient to maintain an adequate shutdown margin during Reactor Coolant System cooldown at the maximum design rate following a reactor trip. The system is independent of the CEA system.

b) Shutdown Cooling System

The Shutdown Cooling System is used to reduce the temperature of the reactor coolant at a controlled rate and to maintain the proper reactor coolant temperature during refueling.

The Shutdown Cooling System utilizes the low pressure safety injection pumps to circulate the reactor coolant through two shutdown heat exchangers, returning it to the Reactor Coolant System through the low pressure injection header (see Subsection 5.4.7).

The Component Cooling System serves as a heat sink for the shutdown heat exchangers.

c) Sampling System

Two sampling systems are provided; one for the reactor coolant and its auxiliary systems and one for the turbine steam and feedwater system. These systems are used for determining both chemical and radiochemical conditions of the various process fluids used in the plant (see Subsection 9.3.2).

d) Cooling Water Systems

The turbine generator condenser is cooled by the Circulating Water System which takes suction from and discharges to the Atlantic Ocean.

An Intake Cooling Water System provides seawater from the Circulating Water System intake structure and serves as a heat sink for the component cooling water heat exchangers, the Turbine Closed Cooling System heat exchangers and the blowdown system open cooling water heat exchangers.

The Component Cooling Water System, consisting of three pumps and two heat exchangers, removes heat from the various auxiliary systems. Corrosion inhibited demineralized water is circulated by the system through auxiliary components of the Nuclear Steam Supply System that require cooling water. During reactor shutdown, component cooling water is also circulated through the shutdown heat exchangers. The Component Cooling Water System provides an intermediate barrier between the Reactor Coolant System and the Intake Cooling Water System (see Subsection 9.2.2 for details).

The blowdown system closed cooling water heat exchangers remove heat from the steam generator blowdown. This heat is, in turn, removed by the intake cooling water by the open blowdown cooling water system heat exchangers.

The Turbine Closed Cooling Water System removes heat from the turbine generator oil cooler, hydrogen coolers, feed pump oil coolers, sample coolers, and other components by providing corrosion inhibited demineralized water to those components (see Section 9.2 for details).

e) Plant Ventilation Systems

Separate ventilation systems are provided for the containment vessel, the control room, the Reactor Auxiliary Building, the Fuel Handling Building, Turbine Building, CCW structure, intake structure, and the Diesel Generator Building. Two purge systems are provided for the containment atmosphere (see Section 9.4).

The annular space between the steel containment vessel and the concrete Shield Building is evacuated by the Shield Building Ventilation System utilizing charcoal filters for removal of radioactive iodine. This system is automatically put into operation upon receipt of a containment isolation actuation signal following a LOCA (see Subsection 6.2.3).

f) Plant Fire Protection System

The Fire Protection System, common to St. Lucie Units 1 and 2, supplies water to fire hydrants, deluge systems and hose racks in the various areas of the plant. Additional design features are provided throughout the plant to ensure conformance to 10 CFR 50 Appendix A, GDC 3 and 10 CFR 50.48(c), NFPA 805. (See Subsection 9.5.1 and the Fire Protection Design Basis Document (Reference 5).)

g) Compressed Air System

The Compressed Air System supplies properly conditioned compressed air required to operate pneumatic instruments and controls, operate containment isolation valves and perform normal plant maintenance. It consists of the Instrument Air System, which supplies the various air operated valves, pneumatic instruments and controls, and the Station Air System which supplies various outlets throughout the plant.

Multiple compressor units and a cross-connection are provided between the Instrument and Station Air Systems. In case of loss of instrument air, all safety related pneumatically operated devices in the plant are designed to fail in a position which would allow safe shutdown. Where safety class valves are required to operate, accumulators are provided (see Subsection 9.3.1).

h) Diesel Generator Fuel Oil Storage and Transfer System

The Diesel Generator Fuel Oil System is provided to transfer diesel fuel oil from the onsite storage tanks to the day tanks which supply the emergency diesel generator sets. Redundant subsystems are provided, capable of supplying sufficient fuel to their respective diesel generator sets,

1.2.2.8 Radioactive Waste Management System

The Waste Management System provides the means for controlled handling, storage and disposal of liquid, gaseous and solid wastes. The principal design criterion is that plant personnel and the general public are protected by ensuring that all normal operating releases of radioactive material are made as low as reasonably achievable in accordance with the provisions of 10 CFR 50, Appendix I.

Reactor coolant from the Chemical and Volume Control System and from the reactor drain tank is processed by the boron management subsystem as described in Section 11.2.2.1.

Miscellaneous liquid wastes from the Reactor Auxiliary Building are collected in the equipment and chemical drain tanks and subsequently processed by the liquid waste subsystem as described in Section 11.2.2.2.

Waste gases are handled by the Gaseous Radwaste Treatment System. In this system, waste gases may be compressed and stored in the gas decay tanks which have a 30 day storage capacity or the gaseous effluent may be directly released to the plant vent if its activity level is sufficiently low. After decay, the gas in the waste gas decay tanks is sampled to ensure radioactivity levels are within acceptable limits, and is then released to the plant vent at a controlled rate.

Spent ion exchange resins and filters can be temporarily stored in high intensity containers (HICs) within the low level waste storage facility and ultimately transported in a shielded container from the plant.

Low activity wastes such as contaminated laundry, rags and paper are compacted and containerized for removal from the plant (see Chapter 11 for details).

1.2.2.9 Independent Spent Fuel Storage Installation (ISFSI)

An Independent Spent Fuel Storage Installation (ISFSI) has been constructed on the St. Lucie site to provide Unit 1 and Unit 2 spent fuel storage capacity through the current end of extended plant lives and to provide the storage required to facilitate decommissioning of the plant. The ISFSI provides the capability to store St. Lucie spent nuclear fuel, high-level radioactive waste, and reactor-related Greater Than Class C (GTCC) waste into dry storage casks.

The ISFSI is licensed under the General License provided to power reactor licensees under 10 CFR 72.210. ISFSI information is provided in References 1, 2, and 3. Therefore, only brief descriptions of the ISFSI are provided herein.

ISFSI soil improvements and construction changes have been evaluated and do not adversely affect safe plant operation. The ISFSI storm water management system limits storm water runoff to pre-construction levels. Other design and environmental effects of the ISFSI have been evaluated to ensure there are no adverse effects on safe plant operation.

1.2.2.10 Low Level Waste Storage Facility (LLWSF)

Due to the uncertainty of availability of offsite disposal options, a Low Level Waste Storage Facility (LLWSF) has been constructed on the site to provide interim low level waste storage capability for both St. Lucie units 1 and 2. Conservatively, both units produce a combined 840 cu. ft. of Class B/C low level radioactive waste (LLW) per year. This amount would fill approximately seven (7) 8-120 High Integrity Containers (HICs) per year. The LLWSF is designed to safely store five (5) years of LLW (36 HICs) within an array of concrete shields inside the precast panel concrete building.

The storage of Low Level Waste is licensed under the General License provided to power reactor licensees under 10 CFR Part 50.

The construction/implementation of the LLWSF including associated soil improvements have been evaluated and do not adversely affect safe plant operation. The existing storm water management system has the capacity to meet Florida Department of Environmental Protection requirements. Other design and environmental effects of the LLWSF have been evaluated to ensure there are no adverse effects on safe plant operation.

1.2.3 MAJOR STRUCTURES AND EQUIPMENT ARRANGEMENT

Refer to the Site Plan, Figure 1.2-1, and the Enlarged Plot Plan, Figure 1.2-2, for the site general layout including the ISFSI site. The plant structures arrangement plans and sections are shown on Figures 1.2-3 through 1.2-22.

The Turbine Building is oriented parallel to State Road A1A and the shoreline of the Atlantic Ocean, with the Reactor Building located on the east, or seaward, side of the Turbine Building. The Reactor Auxiliary Building is located perpendicular to and east of the Turbine Building, oriented in an east-west direction. The Fuel Handling Building is located east of the Reactor Building and the Reactor Auxiliary Building, oriented in a north-south direction.

The Reactor Containment Building encloses the steel containment structure, which houses the Nuclear Steam Supply System consisting of the reactor, steam generators, reactor coolant pumps, pressurizer, and other reactor auxiliaries. The containment structure is served by a polar bridge crane.

The Reactor Auxiliary Building houses the waste management facilities, Engineered Safety Features, heating and ventilating system components, electrical equipment, laboratories, offices, laundry and control room.

The Fuel Handling Building contains the spent fuel pool and new fuel storage facilities, as well as the cooling equipment for the fuel pool. The fuel is transferred from the Reactor Building to the Fuel Handling Building through the fuel transfer tube.

The Turbine Building houses the turbine generator, condensers, feedwater heaters, condensate and feedwater pumps, turbine auxiliaries and electrical switchgear assemblies and other electrical distribution systems which are non-Class 1E.

1.2.4 SHARED SYSTEMS AND INTERCONNECTIONS BETWEEN UNIT 1 AND UNIT 2

Normal plant shutdown requires the operation of several auxiliary systems, none of which are normally used by both units.

The following is a list of systems interconnected (one complete system on each unit which may, under certain conditions, be used by the other unit) between St. Lucie Units 1 and 2:

a) condensate storage tanks (AFW pump suction inter-tie),

b) diesel generator fuel oil storage and transfer system,

c) station blackout cross-tie,

d) liquid waste management system,

e) instrument air system,

f) station service air system, and

g) startup transformers.

A tie between the two units has been provided from the Unit 2 condensate storage tank to the Unit 1 auxiliary feedwater pump's suction for a backup tornado missile protected water supply. This cross-tie is normally isolated. The valve alignment assures that the minimum quantity of water required for safe shutdown is maintained at all times in both tanks.

The diesel generator fuel oil storage and transfer system has a seismic Category I interconnecting tie line between St. Lucie Units 1 and 2. Seismic Category I locked closed isolation valves assure that the tie line is opened only after administrative approval has been obtained.

In the event of a total loss of AC power, both onsite and offsite, (i.e., station blackout) power can be transferred from the non-blacked out unit's emergency diesel generator set via the station blackout tie to one of the blacked-out unit's redundant Class 1E electrical distribution trains. Plant procedures limit the amount of the power transferred so as not to affect the non-blacked out unit's safe shutdown equipment.

The liquid waste management system is interconnected at two non-seismic, non-safety locations by normally closed valves. One interconnection allows either unit to transfer liquid wastes to the other unit's holdup tanks. The other interconnection allows the transfer of liquid waste from the aerated waste storage tank of one unit to the other.

The instrument air system is interconnected but normally isolated between units via automatically controlled valves. As instrument air pressure is lost in one unit the isolation valves automatically open to allow compressed air be provided by the other unit.

The station service air system is interconnected between units, but is isolated via normally closed valves.

The startup transformers (1A-2A, 1B-2B) are provided with a manual switching arrangement which permits paralleling 4.16kV power to St. Lucie Units 1 and 2 (see Section 8.2.1.5 for additional discussion).

St. Lucie Units 1 and 2 are designed using the "slide along" concept. The following facilities, systems and components are shared (one system which may be used by either or both units) by both nuclear units:

a) ultimate heat sink,

b) steam generator blowdown treatment facility,

c) makeup demineralizer regeneration (water treatment facility),

d) domestic water and fire protection system,

e) switchyard, telemetering and load dispatch equipment,

f) seismic instrumentation,

g) site and offsite environmental monitors,

h) hypochlorite system,

i) turbine oil storage tank,

j) carbon dioxide, nitrogen and hydrogen systems,

k) auxiliary steam supply system,

l) safety assessment system, and

m) condensate polisher filter demineralizer system.

All facilities are constructed so that no failure can in any way preclude safe shutdown of the plant.

An accident or single failure in one unit does not affect safe shutdown of either unit. A failure in any of the share features may result in reduced load operation of either or both units, but the capability for safe shutdown is unaffected by such a failure.

The ISFSI (Section 1.2.2.9) is also shared by both units for dry storage of spent fuel.

The LLWSF (Section 1.2.2.10) is also shared by both units for the interim storage of low level waste prior to shipment off site.

1.2.5 SECURITY PLAN

As discussed in Section 13.7 of the Unit 1 UFSAR, a common site security plan is provided for St. Lucie Units 1 and 2.

1.2.6 EMERGENCY PLAN

As discussed in Section 13.3, a common site emergency plan is provided for St. Lucie Units 1 & 2.

1.2.7 SYMBOLS AND ABBREVIATIONS ON FIGURES

Definitions of symbols and abbreviations used throughout the chapters on fluid and electrical systems are shown in detail on Figures 1.2-23 and 1.2-24. The auxiliary pumps P&I diagram is shown on Figure 1.2-34.

1.2.8 REFERENCES FOR SECTION 1.2

1. Letter from M. Rahimi (NRC) to T. Neider (Transnuclear, Inc.), "Certificate of Compliance No. 1030 for the NUHOMS® HD System" dated January 10, 2007, including Safety Evaluation Report to Transnuclear, Inc. NUHOMS® HD Horizontal Modular Storage System for Irradiated Nuclear Fuel

2. Appendix A to Certificate of Compliance No. 1030: NUHOMS® HD System Generic Technical Specifications

3. Transnuclear NUHOMS® HD Horizontal Modular Storage System for Irradiated Nuclear Fuel Final Safety Analysis Report

4. ML16063A121, "St. Lucie Plant, Unit No. 2 - Issuance of Amendment Regarding Transitioning to AREVA Fuel (CAC No. MF5495)

5. DBD-FP-1, Fire Protection Design Basis Document

Refer to Drawing
2998-G-058

FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

SITE PLAN

FIGURE 1.2-1

Refer to Drawing
2998-G-059

FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

ENLARGED PLOT PLAN

FIGURE 1.2-2

Refer to Dwg.
2998-G-060

FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

GENERAL ARRANGEMENT
TURBINE BUILDING
GROUND FLOOR PLAN
FIGURE 1.2-3

Refer to Drawing
2998-G-061

FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

GENERAL ARRANGEMENT
TURBINE BUILDING
FIGURE 1.2-4

Refer to Dwg.
2998-G-062

FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

GENERAL ARRANGEMENT
TURBINE BUILDING
OPERATING FLOOR PLAN
FIGURE 1.2-5

Refer to Dwg.
2998-G-063

FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

GENERAL ARRANGEMENT
TURBINE BUILDING SECTIONS
SHEET 1
FIGURE 1.2-6

Refer to Dwg.
2998-G-064

FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

GENERAL ARRANGEMENT
TURBINE BUILDING SECTIONS
SHEET 2
FIGURE 1.2-7

Refer to Dwg.
2998-G-065

FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

GENERAL ARRANGEMENT
REACTOR BLDG FLOOR PLANS
SHEET 1
FIGURE 1.2-8

Refer to Dwg.
2998-G-066

FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

GENERAL ARRANGEMENT
REACTOR BLDG FLOOR PLANS
SHEET 2 AND MAIN STEAM TRESTLE
FIGURE 1.2-9

Refer to Dwg.
2998-G-067

FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

GENERAL ARRANGEMENT
REACTOR BUILDING SECTIONS
SHEET 1
FIGURE 1.2-10

Refer to Dwg.
2998-G-068

FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

GENERAL ARRANGEMENT
REACTOR BUILDING SECTIONS
SHEET 2
FIGURE 1.2-11

Refer to Dwg.
2998-G-069

FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

GENERAL ARRANGEMENT
REACTOR AUXILIARY BUILDING PLAN
SHEET 1
FIGURE 1.2-12

Refer to Dwg.
2998-G-070

FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

GENERAL ARRANGEMENT
REACTOR AUXILIARY BUILDING PLAN
SHEET 2
FIGURE 1.2-13

Refer to Dwg.
2998-G-071

FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

GENERAL ARRANGEMENT
REACTOR AUXILIARY BUILDING PLAN
SHEET 3
FIGURE 1.2-14

Refer to Dwg.
2998-G-072

FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

GENERAL ARRANGEMENT
REACTOR AUXILIARY BUILDING

FIGURE 1.2-15

Refer to Dwg.
2998-G-073

FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

GENERAL ARRANGEMENT
FUEL HANDLING BUILDING
PLANS
FIGURE 1.2-16

Refer to Dwg.
2998-G-074

FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

GENERAL ARRANGEMENT
FUEL HANDLING BUILDING
SECTIONS
FIGURE 1.2-17

Refer to Drawing
2998-G-075

FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

GENERAL ARRANGEMENT
REACTOR AUXILIARY

FIGURE 1.2-18

Refer to Dwg.
2998-G-076

FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

GENERAL ARRANGEMENT
REACTOR AUXILIARY BUILDING
MISCELLANEOUS PLANS AND
SECTIONS
FIGURE 1.2-19

Refer to Dwg.
2998-G-077 SH 1

FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

GENERAL ARRANGEMENT
COMPONENT COOLING WATER AREA
AND
DIESEL GENERATOR BUILDING
FIGURE 1.2-20

Refer to Drawing
2998-G-077 SH 2

FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

GENERAL ARRANGEMENT
COMPONENT COOLING AREA

FIGURE 1.2-21

Refer to Drawing
2998-G-077 SH 3

FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

GENERAL ARRANGEMENT
INTAKE STRUCTURE

FIGURE 1.2-22

Refer to Drawing
2998-G-078 SH 100

FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

FLOW DIAGRAM SYMBOLS

FIGURE 1.2-23

Refer to Dwg.
2998-B-276, Sheet 00-2

FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

CONTROL AND BLOCK
DIAGRAM

FIGURE 1.2-24

Figures 1.2-25 through 1.2-33
have been deleted

Refer to Dwg.
2998-G-078 SH 105A, B, C

Amendment No. 11, (5/97)

FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

FLOW DIAGRAM
AUXILIARY PUMPS

FIGURE 1.2-34

1.3 COMPARISONS

Comparisons contained herein were valid at the time the operating license for St. Lucie Unit 2 was issued, and are being retained in the Updated FSAR for document completeness and historical record. No present or future update of this section is required.

1.3.1 COMPARISONS WITH SIMILAR FACILITY DESIGNS

Table 1.3-I presents a summary of the characteristics of St. Lucie Unit 2 as originally licensed. The table presents comparative data for San Onofre Units 2 and 3; Arkansas Nuclear One, Unit 2; and St. Lucie Unit 1. Data was extracted from the applicable FSAR.

The San Onofre Units 2 and 3, and Arkansas Nuclear One, Unit 2 designs were selected for comparison because of the basic similarity of the reactor cores. Also they are well advanced in terms of licensing relative to St. Lucie Unit 2. St. Lucie Unit 1 was selected because it is an operating plant which is essentially the same design as St. Lucie Unit 2.

1.3.2 COMPARISON OF FINAL AND PRELIMINARY INFORMATION

1.3.2.1 General

This section contains a discussion of the significant changes that have been made in the St. Lucie Unit 2 design since submittal of the FSAR. Changes considered as significant would include changes in design bases or criteria for seismic Category I structures, and safety related systems or components, plant arrangement, mode of system operation, type of equipment, or gross changes in component or system capacity. In general, such changes further increase the safety margins and operating flexibility of St. Lucie Unit 2.

1.3.2.2 Fuel Load and Operation Dates

Fuel loading was scheduled to commence in October 1982 and 100 percent power operation was expected to be reached in April 1983. The operating license was actually issued in April 1983 and 100 percent power operation was achieved in July 1983.

1.3.2.3 Deletion of Chlorine Accident Detection System

A hypochlorite/biocide system has replaced the onsite use of bottled chlorine storage to control biological fouling in the Circulating Water System (refer to Subsection 10.4.5.4). As a result, the chlorine accident detection system is not required and has been eliminated. EC293500

1.3.2.4 New and Spent Fuel Storage Racks

The capacities of both the new fuel and spent fuel storage racks have increased as discussed in Subsections 9.1.1 and 9.1.2, respectively.

1.3.2.5 Construction Responsibility

Florida Power & Light Company (FP&L) has assumed responsibility for construction of St. Lucie Unit 2, with Ebasco Services Incorporated providing supervision and craft labor for performance of construction as directed or required by FP&L (refer to Section 1.4).

1.3.2.6 Pipe Rupture Criteria

Rupture restraint locations are selected on a "break anywhere" criteria based on Giambusso criteria which was accepted by the NRC review as delineated in the SER (November 1974). Rupture restraints are not provided where it was shown that the broken pipe does not cause unacceptable damage to essential systems. Rupture restraints are also not provided for system pressures under 275 psig, for slot breaks in lines less than four inches, and for systems only operating during accident and/or testing conditions.

In addition, a moderate energy piping analysis has been performed based on criteria as presented in Section 3.6.

The Shutdown Cooling System, which is used as high energy fluid system for only short operational periods and as moderate energy fluid system for the major operational periods, is classified and analyzed as a moderate energy system.

1.3.2.7 Clarification of Code Commitments

ACI-349 was not utilized as design criteria for St Lucie Unit 2 structures. For a clarification of the extent of use of ASME Code, Section III NF, refer to Subsections 3.8.3.2.1 and 3.9.3.4.

1.3.2.8 Containment Analysis

As discussed in Subsection 6.2.1.1, the computer code utilized to determine the containment pressure/temperature results from a loss-of-coolant-accident (LOCA) or main steam line break (MSLB) was CONTRANS (rather than CONTEMPT). In addition, the main feedwater and back-up isolation valves have changed to a 4.0 second closure time.

A spectrum of small break LOCAs are also analyzed.

1.3.2.9 NOT USED EC291393

1.3.2.10 Control Room Design and Analysis

The control room can support a 30 day occupancy throughout the duration of the accident without exceeding the guidelines of GDC 19. The control room is automatically isolated at the outset of the accident followed by the manual opening of an outside air intake, with filtration of the air through charcoal and HEPA filters.

The maximum temperature reached in the control room is based on having only one chiller of the Control Room Air Conditioning System available. Refer to Subsection 9.4.1 for further discussion.

1.3.2.11 Atmospheric Dump Valves and Main Feedwater Isolation Valves

In lieu of two 100 percent ac controlled atmospheric dump valves, four 50 percent capacity valves are provided, two on each main steam line, with ac controlled modulation and dc control for open/close operation (refer to Subsection 10.4.9).

The backup feedwater isolation valves have been relocated immediately upstream of the main feedwater isolation valves in place of the feedwater check valves (refer to Subsection 10.4.7) and are now classified as Quality Group B, seismic Category I.

1.3.2.12 Continuous Containment Purge/Hydrogen Purge System

A Continuous Containment Purge/Hydrogen System has been added, as described in Subsection 9.4.8. As a result, the Airborne Radioactivity Removal System and Containment Instrument Air Compressor inside the containment are no longer required and they have been eliminated.

1.3.2.13 Solid Waste Management System

As stated in Section 11.4, when solidification is performed, in lieu of a permanent system a portable solidification system provided by an outside contractor is utilized to prepare waste material for transportation to an offsite disposal facility.

1.3.2.14 Radiation Protection

The Radiation Monitoring System is a computer based digital system as described in Section 11.5 and Subsection 12.3.4.

In light of the ALARA concern, plant shielding has been improved where practicable, some of which was based on St Lucie Unit 1 experience. Some examples of improved shielding design are the shielding provided for the fuel transfer tube, and shielding for neutron streaming around the reactor vessel (refer to Subsection 12.3.1). Other changes such as a bottom-loaded filter system are provided to reduce doses to operating personnel.

1.3.2.15 Protection Logic

As described in Sections 7.2 and 7.3, the Reactor Protective System and engineered safety features system logic is designed to initiate protective action whenever the signal of any two of three channels reaches the preset limit. A fourth channel is provided as a spare and allows bypassing of one channel while maintaining a two-out-of-three system.

1.3.2.16 Meteorological Data Acquisition

New calculational techniques for updating the site meteorological data are used as detailed in Section 2.3.

1.3.2.17 Fire Safety Analysis

Design features which conform to 10 CFR 50 Appendix A, GDC 3 and 10 CFR 50.48(c), NFPA 805 are presented in the Fire Protection Design Basis Document. (Reference 1)

1.3.2.18 Auxiliary Feedwater System

The motor-operated valves required for the operation of the turbine-driven auxiliary feedwater pump are dc controlled (refer to Subsection 10.4.9).

1.3.2.19 Chapter 15 Accident Analysis

The chapter is structured around an event type/frequency matrix which categorizes the initiating events by type and expected frequency of occurrence. Only the limiting cases in each group have been quantitatively analyzed.

Incorporated into Chapter 15 is the Reload Safety Evaluation and Chapter 15 appendices.

1.3.3 REFERENCES FOR SECTION 1.3

1. DBD-FP-1, Fire Protection Design Basis Document.

TABLE 1.3-1

PLANT PARAMETER COMPARISON

Item	St. Lucie Unit 2 (Cycle 1)	Reference Section	San Onofre Units 2 and 3	ANO-2	St. Lucie Unit 1 (Cycle 1)
Hydraulic and Thermal Design Parameters					
Rated core heat output, MWt	2,560	4.4	3,390	2,815	2,560
Rated core heat output, Btu/hr	8,737 x 10^6	4.4	11,570 x 10^6	9,608 x 10^6	8,737 x 10^6
Heat generated in fuel, %	97.5	4.4	97.5	97.4	97.5
System pressure, nominal, psia	2,250	4.4	2,250	2,250	2,250
System pressure, minimum steady state,psia	2,200	4.4	2,200	2,200	2,200
Hot channel factors,					
Heat flux, F_q	2.57		2.35	2.35	2.85
DNB ratio at nominal conditions	2.64 (CE-1)	4.4	2.07 (CE-1)	2.26 (W-3)	2.30 (W-3)
Coolant flow					
Minimum allowable reactor flowrate, lb/hr	139.4 x 10^6	4.4	148 x 10^6	120.4 x 10^6	122 x 10^6
Effective flowrate for heat transfer, lb/hr	134.3 x 10^6	4.4	142.8 x 10^6	116.2 x 10^6	117.5 x 10^6
Effective flow area for heat transfer, ft^2	54.7	4.4	54.7	44.6	53.5
Average velocity along fuel rods, ft/sec	15.1	4.4	16.3	16.4	13.6
Average mass velocity, lb/hr-ft^2	2.45 x 10^6	4.4	2.61 x 10^6	2.60 x 10^6	2.20 x 10^6
Coolant temperatures, F					
Nominal inlet	548	4.4	553	553.5	538.9
Design inlet	550	4.4	556	556.5	544
Average rise in vessel	48	4.4	58	58.5	55
Average rise in core	50	4.4	60	60.5	56
Average in core	573	4.4	586	583.75	572
Average in vessel	572	4.4	582	582.75	571.5
Nominal outlet of hot channel	622	4.4	642	652	640

TABLE 1.3-1 (Cont'd)

Item	St. Lucie Unit 2 (Cycle 1)	Reference Section	San Onofre Units 2 and 3	ANO-2	St. Lucie Unit 1 (Cycle 1)
Hydraulic and Thermal Design Parameters (Cont'd)					
Heat transfer at 100% power					
Active heat transfer surface area, ft^2	56,315	4.4	62,000	51,000	48,400
Average heat flux, Btu/hr-ft^2	151,300	4.4	182,400	185,000	176,000
Maximum heat flux, Btu/hr-ft^2	388,800	4.4	428,000	433,800	501,300
Average thermal output, KW/ft (Fuel Rod Only)	4.43	4.4	5.34	5.41	5.94
Maximum thermal output, KW/ft (Fuel Rod Only)	11.4	4.4	12.5	12.7	17
Maximum clad surface temperature at nominal pressure, F	657.0	4.4	657.0	657	657
Fuel center temperature, F maximum at 100% power	2,986	4.4	3,180	3,420	3,890
Core Mechanical Design Parameters					
Fuel assemblies					
Design	CEA	4.2	CEA	CEA	CEA
Rod pitch, in.	0.506	4.2	0.5063	0.5063	0.58
Cross-section dimensions, in.	7.972 x 7.972	4.2	7.972 x 7.972	7.97 x 7.97	7.98 x 7.98
Fuel weight (as UO_2), lb_m	204.4×10^3	4.2	223.9×10^3	183,834	207,200
Total weight, lb_m	282.8×10^3	4.2	314,867	250,208	271,280
Number of grids per assembly	10	4.2	11	12	8
Fuel rods					
Number	49,580	4.2	49,580	40,644	36,896
Outside diameter, in.	0.382	4.2	0.382	0.382	0.44
Diametral gap, in.	0.007	4.2	0.007	0.007	0.0085
Clad thickness, in.	0.025	4.2	0.025	0.025	0.026
Clad material	Zircaloy-4	4.2	Zircaloy-4	Zircaloy	Zircaloy

TABLE 1.3-1 (Cont'd)

Item	St. Lucie Unit 2 (Cycle 1)	Reference Section	San Onofre Units 2 and 3	ANO-2	St. Lucie Unit 1 (Cycle 1)
Core Mechanical Design Parameters (Cont'd)					
Fuel pellets					
Material	UO_2 sintered	4.2	UO_2 sintered	UO_2 sintered	UO_2 sintered
Diameter, in.	0.325	4.2	0.325	0.325	0.3795
Length, in.	0.390	4.2	0.390	0.390	0.650
Control assemblies					
Neutron absorber	(See Table 4.2-1)	4.2	(See Table 4.2-1)	B_4C/Ag-In-Cd	B_4C/SS
Cladding material	Inconel 625	4.2	Inconel 625	NiCrFe alloy	NiCrFe alloy
Clad thickness	0.035	4.2	0.035	0.035	0.040
Number of assembly, full/part-length	83/0	4.2	83/8	73/8	73/8
Number of rods per assembly	4,5/5	4.2	4,5/5	5	5
Nuclear Design Data					
Structural characteristics					
Core diameter, in. (equivalent)	136	4.2	136	123	136
Core height, in. (active fuel)	136.7	4.2	150	150	136.7
H_20/UO_2 Unit Cell (cold), volume ratio	1.705	4.2	1.705	1.705	1.63
Number of fuel assemblies	217	4.2	217	177	217
UO_2 Rods per assembly, unshimmed/shimmed					
Batch A	236	4.3	236	236	176
Batch B	236/220	4.3	236/220	224	164
Batch C	236/224 or 220	4.3	236/224 or 220	224/234/233	176/164/164
Performance characteristics loading technique	3-batch mixed central zone	4.3	3-batch mixed central zone	3-batch mixed central zone	3-batch mixed central zone
Fuel discharge burnup, MWD/MTU					
Average first cycle	13,187	4.3	12,731	12,500	12,800

TABLE 1.3-1 (Cont'd)

Item	St. Lucie Unit 2 (Cycle 1)	Reference Section	San Onofre Units 2 and 3	ANO-2	St. Lucie Unit 1 (Cycle 1)
Nuclear Design Data (Cont'd)					
Feed enrichment, wt%					
Region 1	1.71	4.3	1.87	1.93	1.93
Region 2	2.28	4.3	2.38	2.27	2.33
Region 3	2.73	4.3	2.88	2.94	2.82
Control characteristics effective multiplication (beginning of life)					
Cold, no power, clean	1.170	4.3	1.170	1.195	1.170
Hot, no power, clean	1.119	4.3	1.125	1.139	1.134
Hot, full power, Xe equilibrium	1.070	4.3	1.067	1.082	1.078
Control Assemblies					
Total rod worth (hot), %	11.16 (EOC)	4.3	11.35	12.3	11.0
Boron concentrations for criticality:					
Zero power no rods inserted, clean, ppm Cold/Hot	901/809	4.3	899/832	1011/1001	945/935
At power with no rods inserted, clean/equilibrium xenon, ppm	715/493	4.3	719/452	881/611	820/590
Kinetic characteristics, range over life					
Moderator temperature coefficient, Δρ/F	See Table 4.3-4	4.3	See Table 4.3-4	-0.3×10^{-4} to -2.5×10^{-4}	-0.4×10^{-4} to -2.1×10^{-4}
Moderator pressure Coefficient, Δρ/psi	$+0.6 \times 10^{-6}$	4.3	$+0.7 \times 10^{-6}$	$+0.06 \times 10^{-6}$ to $+2.6 \times 10^{-6}$	$+0.49 \times 10^{-6}$ to $+2.55 \times 10^{-6}$
Moderator void coefficient, Δρ/% Void	-0.22×10^{-3}	4.3	-0.36×10^{-3}	-0.03×10^{-3} to -1.22×10^{-3}	-0.26×10^{-3} to -1.35×10^{-3}
Doppler coefficient, Δρ/F	See Figure 4.3-34	4.3	1.18 x 10 to 1.28×10^{-5}	-1.18×10^{-5} to -1.78×10^{-5}	-1.45×10^{-5} to -1.07×10^{-5}

TABLE 1.3-1 (Cont'd)

Item	St. Lucie Unit 2 (Cycle 1)	Reference Section	San Onofre Units 2 and 3	ANO-2	St. Lucie Unit 1 (Cycle 1)
Principal Design Parameters of the Reactor Coolant System					
Operating pressure, psig	2,235	5.1	2,235	2,235	2,235
Operating Reactor inlet temperature, F	550	5.1	553	553.5	539.7
Operating Reactor outlet temperature, F	604	5.1	611.2	612.5	595.7
Number of loops	2	5.1	2	2	2
Design pressure, psig	2,485	5.1	2,485	2,485	2,485
Design Temperature, F	650	5.1	650	650	650
Hydrostatic test pressure (cold), psig	3,110	5.1	3,110	3,110	3,110
Principal Design Parameters of the Reactor Vessel					
Material	See Table 5.2-3	5.2	See Table 5.2-2	SA-533, Grade B, Class I, low alloy steel, internally clad with Type 304 austenitic SS	SA-533,Grade B, Class 1, low alloy steel internally clad with Type 304 austenitic SS
Design pressure, psig	2,485	5.3	2,485	2,485	2,485
Design temperature, F	650	5.3	650	650	650
Operating pressure, psig	2,235	5.3	2,235	2,235	2,235
Inside diameter of shell, in.	172	5.3	172	157	172
Outside diameter across nozzles, in.	253	5.3	253	238	253
Overall height of vessel and enclosure head, ft-in. to top of CEDM nozzle	41-10-3/8	5.3	43-6-1/2	43-4-1/6	41-11-3/4
Minimum clad thickness, in.	1/8	5.3	1/8	1/8	5/16
Principal Design Parameters of the Steam Generators					
Number of Units	2	5.4	2	2	2

TABLE 1.3-1 (Cont'd)

Item	St. Lucie Unit 2 (Cycle 1)	Reference Section	San Onofre Units 2 and 3	ANO-2	St. Lucie Unit 1 (Cycle 1)
Principal Design Parameters of the Steam Generators (Cont'd)					
Type	Vertical U-tube with integral moisture separator	5.4	Vertical U-tube with integral moisture separator	Vertical U-tube with integral moisture separator	Vertical U-tube with integral moisture separator
Tube material	NiCrFe alloy	5.2	NiCrFe alloy	NiCrFe alloy	NiCrFe alloy
Shell material	SA-533 GR A&B, Class 1 and SA 516, Gr. 70	5.2	SA-533 Gr. B Class 1 and SA-516, Gr.70	SA-533 Gr. B Class 1 and SA-516, Gr. 70	SA-533 Gr. B Class 1 and SA-516, Gr. 70
Tube side design Pressure, psig	2,485	5.4	2,485	2,485	2,485
Tube side design temperature, F	650	5.4	650	650	650
Tube side design flow, lb/hr	61×10^6	5.4	74×10^6	60.2×10^6	61×10^6
Shell side design pressure, psia	1,000	5.4	1,100	1,100	1,000
Shell side design temperature, F	550	5.4	560	560	550
Operating pressure, tube side, nominal, psig	2,235	5.4	2,235	2,235	2,235
Operating Pressure, shell side, maximum, psig	885		985	985	885
Maximum moisture at outlet at full load, %	0.2	5.4	0.2	0.2	0.2
Hydrostatic test pressure, tube side (cold) psia	3,110		3,110	3,110	3,110
Steam Pressure at full power, psia	815	5.4	900	900	815
Steam temperature, at full power, F	520.3	5.4	532	531.95	520.3
Principal Design Parameters of the Reactor Coolant Pumps					
Number of units	4	5.4	4	4	4
Type	Vertical, single stage centrifugal with botton suction and horizontal discharge		Vertical, single stage radial flow with bottom suction and horizontal discharge	Vertical, single stage centrifugal with bottom suction and horizontal discharge	Vertical, single stage centrifugal with bottom suction and horizontal discharge
Design pressure, psig	2,485	5.4	2,485	2,485	2,485

TABLE 1.3-1 (Cont'd)

Item	St. Lucie Unit 2 (Cycle 1)	Reference Section	San Onofre Units 2 and 3	ANO-2	St. Lucie Unit 1 (Cycle 1)
Principal Design Parameters of the Reactor Coolant Pumps (Cont'd)					
Design temperature, F	650	5.4	650	650	650
Operating pressure, nominal psig	2,235	5.4	2,235	2,235	2,235
Suction temperature, F	550	5.4	553	553.5	540
Design capacity, gal/min	81,200	5.4	99,000	80,000	80,000
Design head, ft	310	5.4	310	275	250
Hydrostatic test pressure (cold), psig	3,110		3,110	3,110	3,110
Motor type	AC induction, single speed		AC induction, single speed	AC induction, single speed	AC induction, single speed
Motor rating, hp	6,500		9,700	6,500	6,500
Principal Design Parameters of the Reactor Coolant Piping					
Material	See Table 5.2-3		SA-516, Gr 70 with nominal 7/32 SS clad	SA-516, Gr 70 with nominal 3/16 SS clad	SA-516, Gr 70 with nominal 7/32 SS clad
Hot leg ID, in.	42	5.4	42	42	42
Cold leg ID, in.	30	5.4	30	30	30
Between pump and steam generator ID, in.	30	5.4	30	30	30
Engineered Safety Features					
High pressure safety injection pumps	2	6.3	3	3	3
Low pressure safety injection pumps	2	6.3	2	2	2
Safety injection tanks, number	4	6.3	4	4	4
Containment spray pumps	2	6.2	2	2	2
Containment fan coolers units	4	6.2	4	4	4
Air flow capacity, each at emergency conditions, ft^3 /min	39,600	6.2	31,000	50,000	55,800

TABLE 1.3-1 (Cont'd)

Item	St. Lucie Unit 2 (Cycle 1)	Reference Section	San Onofre Units 2 and 3	ANO-2	St. Lucie Unit 1 (Cycle 1)
Engineered Safety Features (Cont'd)					
Emergency power Diesel-generator unit	2	8.3	4 (for two units)	2	2
Containment System Parameters					
Type					
	Steel containment vessel with cylindrical shell, hemispherical dome and ellipsoidal bottom - ASME Code, Section III, Class MC, surrounded by reinforced concrete Shield Building.	3.8.2	Steel-lined prestressed post tensioned concrete cylinder, curve dome roof.	Steel-lined prestressed post tensioned concrete cylinder, curved dome roof.	Steel containment vessel with cylindrical shell, hemispherical dome and ellipsoidal bottom - ASME Code, Section III, Class B, surrounded by reinforced concrete Shield Building.
Inside Diameter, ft.	140	3.8	150	116	140
Height, ft.	232	3.8	172	207	232
Free volume, ft^3	2,500,000	6.2	2,335,000	1,780,000	2,500,000
Reference accident Pressure, psig	44	3.8	60	54	44
Steel Thickness, in.					
Vertical Wall	1.92	3.8	Not Applicable	Not Applicable	1.91
Hemispherical Head	0.96		Not Applicable	Not Applicable	0.95
Knuckles	2.125		Not Applicable	Not Applicable	225
Concrete Thickness, ft.					
Vertical Wall	Not Applicable	3.8	4 1/3	3 3/4	Not Applicable
Dome	Not Applicable		3 3/4	3 1/4	Not Applicable
Design Parameters - Shield Building		3.8	Not Applicable	Not Applicable	
Inside Diameter, ft.	148				148
Height, ft. (top of foundation to top of dome)	230.5				230.5
Concrete Thickness, ft.					
Vertical Wall	3				3
Dome	2.5				2.5

TABLE 1.3-1 (Cont'd)

Item	St. Lucie Unit 2 (Cycle 1)	Reference Section	San Onofre Units 2 and 3	ANO-2	St. Lucie Unit 1 (Cycle 1)
Containment Leak Prevention and Mitigation Systems	Leak-tight penetration, Automatic isolation where required.	6.2	Leak-tight penetration, and continuous steel liner. Automatic isolation where required.	Leak-tight penetration, and continuous steel liner. Automatic isolation where required.	Leak-tight penetration, Automatic isolation where required.
Gaseous Effluent Purge	Discharge through vent.	6.2	Discharge through vent.	Discharge through vent.	Discharge through vent.
RADIOACTIVE WASTE MANAGEMENT SYSTEM					
Liquid Waste Processing Systems					
Reactor Coolant Waste Holdup Tank (EMS)		11.2			
Number	4		1/2	4	4
Capacity (Gal.),each	40,000		6,000/25,000	51,270	40,000
Concentrators					
Number	1		1 (For 2 units)	1	1
Capacity (gpm)	20		50	20	2
Gaseous Waste Processing Systems					
Waste Gas Decay Tank		11.3			
Number	3		6 (For 2 units)	3	3
Capacity (ft^3), each	138		500	300	144
Pressure (psig)	190		150	380	190
Hold-up time (days)	25		30	30	30
ELECTRIC SYSTEMS					
Number of Offsite Circuits	3	8.1	8	3	3
Number of Incoming Lines to Startup Transformers	2	8.2	2	2	2
Number of Startup Transformers	2	8.2	4	1+1(shared)	2
Number of Main Unit Transformers (Three Phase)	2	8.2	1	3 (single phase)	2
Number of 4.16 KV Engineered Safety Features System Buses	3	8.3	3	2	3

TABLE 1.3-1 (Cont'd)

Item	St. Lucie Unit 2 (Cycle 1)	Reference Section	San Onofre Units 2 and 3	ANO-2	St. Lucie Unit 1 (Cycle 1)
ELECTRIC SYSTEMS (Cont'd)					
Number of 480V Engineered Safety Features System Buses	3	8.3	3	2	3
Number of 120V Safety Related Vital Buses	4	8.3	4	4	4
Number of Standby Diesel Generators	2	8.3	2	2	2
Diesel Generator Rating (KW)	3685	8.3	4700	2850	3500
INSTRUMENTATION SYSTEMS*					
Reactor Protective System	7.2	7.2	7.2	7.2	7.2
Reactor and Reactor Coolant System	7.7.1.1 7.6.1	7.7.1.1 7.1.1.2	7.7.1.1 7.7.1.2	7.7.1.1 7.7.1.2	7.7.1.1 7.7.1.2
Steam and Feedwater Control System	7.7.1.1	7.7.1.3	7.7.1.3	7.7.1.3	7.7.1.3
Nuclear Instrumentation	7.2.1.1 7.7.1.1	7.2.1.1	7.2.1.1	7.2.1.1	7.2.1.1
Non-Nuclear Process Instrumentation	7.7.1.1 7.5.1	7.5.1.5	7.5.1.5	7.5.1.5	7.5.1.5
CEA Position Instrumentation	7.7.1.1	7.5.1.3	7.5.1.3	7.5.1.3	7.5.1.3

*This section is not suited for tabular description. SAR section numbers have been included for the location of the detailed description of each system.

1.4 IDENTIFICATION OF AGENTS AND CONTRACTORS

Information contained herein was valid at the time the operating license for St. Lucie Unit 2 was issued, and is being retained in the updated FSAR for document completeness and historical record. No present or future update of this section is required.

The Florida Power & Light Company is the applicant for the operating license for St Lucie Unit 2. Florida Power & Light Company is responsible for the design and engineering review, construction and operation of the plant.

Florida Power & Light Company has engaged Combustion Engineering, Inc. (CE) to design, manufacture and provide the Nuclear Steam Supply System and nuclear fuel for the first core and the first three core reload batches. The Nuclear Steam Supply System includes the Reactor Coolant System, reactor auxiliary system components, nuclear and certain process instrumentation, and the reactor control and protective system. In addition, CE will furnish technical assistance for erection, initial fuel loading, testing and initial startup of the Nuclear Steam Supply System.

Ebasco Services Inc. has been engaged by the Applicant for engineering and procurement services for this project and as such has performed engineering and design work for the balance - of -plant equipment, systems and structures not included under the CE scope of supply. Ebasco has also provided supervision and craft labor for performance of construction as directed or required by Florida Power & Light Company.

These and other engineering firms with approved Quality Assurance Programs may perform backfit, retrofit, maintenance and construction activities during plant operation under the auspices of Florida Power & Light Company.

1.5 REQUIREMENTS FOR FURTHER TECHNICAL INFORMATION

Material contained herein were valid at the time the operating license for St Lucie 2 was issued, and are being retained in the Updated FSAR for document completeness and historical record. No present or future update of this section is required.

This section provides a description of safety related technical information relevant to this application. Combustion Engineering, Inc., (CE), is conducting research and development programs relating to the requirements of this section.

The St Lucie Unit 2 reactor incorporates a 16 x 16 fuel assembly design with five guide tubes. This design provides an increase in conservatism for loss -of -coolant accident (LOCA) considerations with a minimum change from previous CE fuel designs. Previous designs have undergone extensive testing, and operating experience is now being acquired.

The three test programs described in Subsections 1.5.1, 1.5.2, and 1.5.3 are considered necessary to confirm the adequacy of the 16 x 16 fuel assembly design.

References 1 to 6 present descriptions of development programs aimed at verifying the Nuclear Steam Supply System (NSSS) design and the anticipated performance characteristics, and at confirming the design margins. Other programs that apply to this plant are identified in Subsections 1.5.4 through 1.5.8.

1.5.1 FRETTING AND VIBRATIONS TESTS OF FUEL ASSEMBLIES

Extensive autoclave vibration and dynamic flow tests have been performed to characterize fuel rod and spacer grid fretting corrosion in CE fuel assemblies.

Tests have been completed using a full sized 16 x 16 fuel assembly. This assembly is similar to the 16 x 16 five guide tube design used on the St Lucie Unit 2 reactor. This assembly was subjected to flow testing under conditions of temperature, water chemistry, pressure, and flow velocities in excess of normal reactor conditions. Further information is provided in Subsections 4.2.3.1.1 and 4.2.3.1.2. EC296160

1.5.2 DEPARTURE FROM NUCLEATE BOILING (DNB) TESTING

Extensive heat transfer testing has been completed with electrically heated rod bundles representative of the CE 16 x 16 and 14 x 14 fuel assemblies. The program for each assembly geometry included tests to determine the effects on DNB of the control element assembly (CEA) guide tube, bundle heated length, and grid spacing, and lateral and axial power distributions. Each test yielded DNB data over a wide range of conditions of interest for pressurized water reactor (PWR) design. Those data were used with the TORC subchannel analysis code to develop and to verify the CE -1 DNB correlation for predicting DNB in fuel assemblies with standard spacer grids. The CE -1 correlation, which is discussed in more detail in Subsection 4.4.4.1, is used in computing margin to DNB for St. Lucie Unit 2.

For HTP™ fuel, DNBR analyses are performed using the XCOBRA-IIIC code (Reference 49) and the HTP CHF correlation (Reference 53 in Section 4.4). Details of the correlation development and testing campaigns are provided in Section 4.4 Reference 53. The Biasi correlation is used for Post-SCRAM MSLB analyses.

1.5.3 FUEL ASSEMBLY STRUCTURAL TESTS

The fuel assembly structural testing program was designed to verify the structural adequacy of the fuel assembly design under normal handling, normal operation, seismic excitation, and LOCA loadings. The test program provides the structural characteristics employed in the fuel assembly structural analyses.

A series of tests were conducted on a 14 x 14 fuel assembly to determine the combined axial and lateral load deflection characteristics of the fuel assembly. Axial compression tests and axial drop tests were performed. Measurements were made of axial loads, axial deflections, lateral deflections of all spacer grids, and strains in the guide tubes and fuel rods.

A series of structural tests on the 16 x 16 fuel assembly design was also conducted. The fuel assembly was subjected to both static and dynamic tests so as to determine basic structural characteristics. In addition, several 16 x 16 spacer grids were subjected to impact tests to determine dynamic load deflection characteristics and damage limits. These tests are also discussed in Subsection 4.2.3.1.3.

1.5.4 FUEL ASSEMBLY FLOW MIXING TESTS

The objective of the fuel assembly flow mixing program was to obtain information on the magnitude of coolant mixing in CE fuel assemblies. Several series of tests have been completed, and the data from these tests provide a sound basis for the treatment of coolant mixing in design thermal margin calculations.

The first series of single phase flow mixing tests was run in 1966 with a prototype CE PWR fuel assembly. The average level of coolant mixing was determined using dye injection and sampling equipment.

A second series of single phase mixing tests was conducted in 1968 with a model representing a portion of a 14 x 14 CEA type fuel assembly. Those tests, which also used dye injection and sampling techniques, are described in Reference 1.

More recently, tests were conducted in which coolant temperatures were measured in the subchannels of electrically heated rod bundles representative of the 14 x 14 or 16 x 16 fuel assemblies with standard spacer grids.

As discussed in Subsection 4.4.4.1, those data provide confirmation that the results from the previous dye sampling experiments are applicable for the fuel assembly design used in St Lucie Unit 2.

1.5.5 REACTOR FLOW MODEL TESTING AND EVALUATION

The objective of the reactor flow model test programs is to obtain information on:

a. Flow and pressure distributions in various regions of the reactor

b. Pressure loss coefficients

c. Hydraulic loads on certain vessel internal components

This information is used for establishing or verifying design hydraulic parameters.

Flow model testing, which began in 1966, was designed to obtain those reactor hydraulic design data not amenable to direct calculation. Scale model testing possesses the advantages, relative to actual reactor tests, of:

a. Providing the information early in the design stage

b. Being more suitable for extensive instrumentation

c. Being flexible so that proposed design modifications can be investigated

The reactor flow models used by CE are generally 1/5 true scale models. In the first four CE flow model programs, a closed core design was used. The closed core simulates the reactor fuel assemblies with individual closed wall tubes containing orifices to provide the correct axial hydraulic resistance.

Further discussion of the CE flow model test programs is provided in Subsection 4.4.4.2.1.

1.5.6 FUEL ASSEMBLY FLOW TESTS

The objectives of the fuel assembly flow test program included assessment of the effect of postulated flow maldistributions on thermal behavior and margin.

The program originated in 1967 with fuel assembly flow distribution testing. Both flow visualization and flow pattern measurements were generated on an overscale model of the lower portion of an early CE design fuel assembly.

A second test series was conducted for the CEA type fuel assembly. The second test series was designed to:

a. Determine the effect of flow obstructions on flow distribution within the fuel assembly

b. Determine the magnitude of the effect of the disturbed flow patterns on the thermal margin within a CEA type fuel assembly

The information from these tests, described further in Reference 1, has established the effect of flow obstructions within the fuel assembly. Additional information on the effects of postulated fuel coolant channel flow blockages is presented in Subsection 4.2.3.2.14.

1.5.7 CONTROL ELEMENT DRIVE MECHANISM (CEDM) TESTS

Performance testing of the magnetic jack CEDM is described in Subsection 3.9.4.4 and in Reference 1. The program has confirmed the operability of the drive assembly in normal and misaligned conditions as well as the load carrying capability and life characteristics. EC296160

1.5.8 DNB IMPROVEMENT

The DNB improvement program was initiated by CE in order to obtain empirical information on the departure from nucleate boiling (DNB) phenomenon and on other thermal and hydraulic characteristics of CE fuel assemblies. Testing has been performed with electrically heated rod bundles that correspond dimensionally to fuel rod configurations under in -reactor temperature pressure and flow conditions to obtain data on DNB, pressure drop, and coolant channel exit temperatures. These data were employed to verify that the CE thermal hydraulic design methods conservatively predict DNB.

The DNB improvement program is described in References 1, 2, 3, and 4. It is a continuing program providing improvements in the accuracy of CE thermal and hydraulic computer programs for predicting local coolant conditions and pressure drops and confirming the applicability of currently used DNB correlations to the CE fuel design. Additional information on the program and results applicable to St Lucie Unit 2 are presented in Subsection 4.4.4.1.

SECTION 1.5: REFERENCES

1. "Safety Related Research and Development for Combustion Engineering Pressurized Water Reactors, Program Summaries," CENPD -87 (Proprietary), January 1973, and CENPD -87, Rev 01, (Non -Proprietary), March 1973.

2. "Safety Related Research and Development for Combustion Engineering Pressurized Water Reactors, Program Summaries," CENPD -143 (Proprietary) and CENPD -143, Rev 01 (Non -Proprietary), May 1974.

3. "Safety Related Research and Development for Combustion Engineering Pressurized Water Reactors, 1974 Program Summaries," CENPD -184 -P (Proprietary) and CENPD -184 (Non -Proprietary), May 1975.

4. "Safety Related Research and Development for Combustion Engineering Pressurized Water Reactors, 1975 Program Summaries," CENPD -229 -P (Proprietary) and CENPD -229 (Non -Proprietary), June 1976.

5. "Safety Related Research and Development for Combustion Engineering Pressurized Water Reactors, 1976 Program Summaries," CENPD -258 (Non -Proprietary), October 1977.

6. "Safety Related Research and Development for Combustion Engineering Pressurized Water Reactors, 1977 -1978 Program Summaries," CENPD -262 (Non -Proprietary), December 1978.

1.6 MATERIAL INCORPORATED BY REFERENCE

Topical reports incorporated by reference were valid at the time of application to the NRC, and are being retained in the updated FSAR for document completeness and historical record. No present or future update of this section is required.

The following topical reports are incorporated by reference.

Report Number	Author and Title	Date to NRC	FSAR Section
CENPD-162 (with Suppl. 1)	Combustion Engineering, Inc. "CHF Correlation for C-E Fuel Assemblies with Standard Spacer Grids-Part I; Uniform Axial Power Distribution"	May 1975 (Approved Version, Sept.1976)	4.4, 15.0
CENPD-168 Rev. 1	Combustion Engineering, Inc. "Design Basis Pipe Breaks for the Combustion Engineering Two Loop Reactor Coolant System"	Oct. 1976 (Approved Version, Aug. 1977)	3.6
CENPD-178P and 178 Rev. 1	Combustion Engineering, Inc. "Structural Analysis of the 16 x 16 Fuel Assembly for Combined Seismic and Loss-of-Coolant-Accident Loadings"	August 1981	3.9, 4.2
CENPD-115 Suppl. 1	Combustion Engineering, Inc. "Comparison of Calvert Cliffs, Maine Yankee, and Fort Calhoun Design Parameters and Flow-Induced Structural Response"	April 1974	3.9
CENPD-182 Rev. 1	Combustion Engineering, Inc. "Seismic Qualification of C-E Instrumentation and Control Equipment"	June 1977	3.10, 7.2
CENPD-183	Combustion Engineering, Inc. "C-E Methods for Loss of Flow Analysis"	August 1975	15.3
CENPD-187 (with Suppl. 1)	Combustion Engineering, Inc. "Method of Analyzing Creep Collapse of Oval Cladding"	October 1975 and May 1975 (Approved Version, April 1976)	4.2
CENPD-26 (with Suppl. 1 through 3)	Combustion Engineering, Inc. "Description of Combustion Engineering Loss of Coolant Calculational Procedures"	August 1971	3.9

Report Number	Author and Title	Date to NRC	FSAR Section
CENPD-42	Combustion Engineering, Inc. "Dynamic Analysis of Reactor Vessel Internals Under Loss of Coolant Accident Conditions with Application to C-E 800 Mwe Class Reactors"	August 1972	3.9
CENPD-67 Rev. 1, Addenda 1 and 2	Combustion Engineering, Inc. "Iodine Decontamination Factors During PWR Steam Generation and Steam Venting"	September 1973 November 1974, August 1975	10.3
CENPD-98	Combustion Engineering, Inc. "Coast Code Description"	July 1973 (Approved Version, April 1974)	4.4, 15.0
CENPD-107 (with Suppl. 1 through 5)	Combustion Engineering, Inc. "CESEC"	August 1974, September 1974, Septem-1975, January 1976, June 1976	15.0
CENPD-105	Combustion Engineering, Inc. "Fast Neutron Attenuation by the ANISN-SHADRAC Analytical Method"	Nov. 1973	4.3
CENPD-132 (with Suppl. 1 and 2)	Combustion Engineering, Inc. "Calculative Methods for the C-E Large Break LOCA Evaluation Model"	September 1974, March 1975, August 1975.	6.2, 6.3, 15.6
CENPD-133 (with Suppl. 2)	Combustion Engineering, Inc. "CEFLASH-4A Fortran IV Digital Computer Program for Reactor Blowdown Analysis"	September 1974, March 1975	6.2, 6.3, 15.6
CENPD-134 (with Suppl. 1)	Combustion Engineering, Inc. "COMPERC-II A Program for Emergency Refill-Reflood of the Core"	September 1974, March 1975	6.2, 6.3, 15.6
CENPD-135 (with Suppl. 2, 4 and 5)	Combustion Engineering, Inc. "STRIKIN-II A Cylindrical Geometry Fuel Rod Heat Transfer Program"	September 1974, March 1975, September 1976, May 1977	4.2, 6.3, 15.6
CENPD-136	Combustion Engineering, Inc. "High Temperature Properties of Zircaloy and UO_2 for use in LOCA Evaluation Model"	August 1974	4.2, 6.3,15.6

Report Number	Author and Title	Date to NRC	FSAR Section
CENPD-137 (with Suppl. 1)	Combustion Engineering, Inc. "Calculative Methods for the C-E Small Break LOCA Evaluation Model"	September 1974	6.3, 15.6
CENPD-139 (with Suppl. 1)	Combustion Engineering, Inc. "C-E Fuel Evaluation Model"	September 1974 (Approved Version, April 1975)	4.1, 4.2, 4.3, 4.4, 6.3, 15.6
CENPD-145	Combustion Engineering, Inc. "A Method of Analyzing In-Core Detector Data in Power Reactors"	May 1975, February 1978	4.3
CENPD-148	Combustion Engineering, Inc. "Review of Reactor Shutdown System (PPS Design) for Common Mode Failure Susceptibility"	September 1974	4.6, 7.2
CENPD-153 with Amendments 1 through 3	Combustion Engineering, Inc. "Evaluation of Uncertainty in the Nuclear Form Factor Measured by Self Powered Fixed In-Core Detector Systems"	December 1974, August 1977, February 1978, April, 1979	4.3
CENPD-155	Combustion Engineering, Inc. "C-E Procedure for Design, Fabrication, Installation and Inspection of Surveillance Specimen Brackets Attached to Reactor Vessel Beltline Region"	October 1974 (Approved Version, August 1975)	5.3
CENPD-161 with Amendment 1.	Combustion Engineering, Inc. "TORC - A Computer Code for Determining the Thermal Margin of a Reactor Core"	June 1975, May 1976 (Approved Version, September 1978)	4.1, 4.2, 4.3, 4.4, 15.0
CENPD-190	Combustion Engineering, Inc. "C-E Method for Control Element Assembly Ejection Analysis"	January 1976 (Approved Version, August 1976)	15.4
CENPD-198 and Supplements 1 and 2	Combustion Engineering, Inc. "Zircaloy Growth-In-Reactor Dimensional Changes in Zircaloy-4 Fuel Assemblies"	December 1975 January 1978 November 1978	4.2
CENPD-206	Combustion Engineering, Inc. "Comparison of TORC Code Predictions with Experimental Data"	February 1977	4.4

Report Number	Author and Title	Date to NRC	FSAR Section
CENPD-207	Combustion Engineering, Inc. "Critical Heat Flux Correlation for C-E Fuel Assemblies with Standard Spacer Grids, Part 2, Non-Uniform Axial Power Distributions"	July 1976	4.4
CENPD-213 and Suppl.1	Combustion Engineering, Inc. "Application of FLECHT Reflood Heat Transfer Coefficients to Combustion Engineering 16 x 16 Fuel Bundles"	February 1976, March 1976	6.3, 15.6
CENPD-225	Combustion Engineering, Inc. "Fuel and Poison Rod Bowing”	October 1976	4,2, 4.4
CENPD-199	Combustion Engineering, Inc. "C-E Setpoint Methodology: Local Power Density and DNB LCSS and LCO Setpoint Methodology for Analog Protective System."	April 1976	4.3
CENPD-188	Combustion Engineering, Inc. "HERMITE, A Multi-Dimensional Space-Time Kinetics Code for PWR Transients"	April 1976 (Approved Version, September 1976	4.3
CENPD-254	Combustion Engineering, Inc. "Post-LOCA Long Term Cooling Evaluation Model"	August 1977	6.3
CENPD-252P-A	Combustion Engineering, Inc. "Method for Analysis of Blowdown Forces in a Reactor Vessel"	July 1979	3.9
CVI-TR-7301	CVI Design and Development of High Efficiency Charcoal Adsorbers and its Application in ESF Atmospheric Cleanup Systems	February 1975	6.5.1
AFF-TR-7101	American Air Filter "Design and Testing of Fan Cooler Filter Systems for Nuclear Applications"	November 1972	6.2.2
WCAP-7709-L	Westinghouse "Electric Hydrogen Recombiners for PWR Containments"	April 1972	6.2.5

Report Number	Author and Title	Date to NRC	FSAR Section
FPLTQAR 1-76A Revision 0 Revision 1 Revision 2	Florida Power & Light Co. Florida Power & Light Co. "Topical Quality Assurance Report"	January 1976 June 1976 September 1976 January 1977 (Approved by NRC September 1977)	17.2
ETR-1002 P	Ebasco Services, Inc. "Design Considerations for Protection from Effects of Pipe Rupture - Part I - Dynamic Analysis"	November 1975	3.6

1.7 DRAWINGS

Drawings contained herein were valid at the time the operating license for St Lucie 2 was issued, and are being retained in the Updated FSAR for document completeness and historical record. No present or future update of the section is required. Updated drawings are maintained at the St Lucie 2 site.

1.7.1 ELECTRICAL, INSTRUMENTATION, AND CONTROL DRAWINGS

Tables 1.7-1 and 1.7-2 are lists of electrical, instrumentation and control safety-related drawings prepared by the Architect/Engineer and NSSS supplier, respectively. There are no drawings considered proprietary.

1.7.2 PIPING AND INSTRUMENTATION DIAGRAMS

Tables 1.7-3 and 1.7-4 are lists of safety-related piping and instrumentation diagrams prepared by the Architect/Engineer and NSSS supplier, respectively,

TABLE 1.7-1

ARCHITECT/ENGINEER SUPPLIED
ELECTRICAL, INSTRUMENTATION AND CONTROL DRAWINGS
SAFETY RELATED

Drawing No.	Sheet No.	Revision No	Date	Prepared By	Title
B 271		0	3/3/78	E	ELECTRICAL GEN INSTALLATION NOTES
G 272		5	7/23/82	E	MAIN ONE LINE WIRING DIAGRAM
G 274		5	7/23/82	E	AUXILIARY ONE LINE WIRING DIAGRAM
B 325		7	11/14/80	E	BILL OF MATERIALS
B 328				E	CABLE & CONDUIT LIST
G 332		7	10/30/82	E	480V MISC 125V DC & VITAL AC ONE LINE WIRING DIAGRAM
G 332	2	0	10/30/82	E	480V MISC 125V DC & VITAL AC ONE LINE WIRING DIAGRAM SH 2
B 335		2	12/10/80	E	POWER DISTRIBUTION & MOTOR DATA SHEETS
B 337		2	1/4/79	E	ELECTRICAL PENETRATION SCHEDULE
G 340		10	1/8/83	E	TURBINE BUILDING GROUND FLOOR CONDUITS, TRAYS & GRDG SH2
C 348		11	1/8/80	E	MANHOLE & HANDHOLE DETAILS
G 352		4	8/4/82	E	ARRANGEMENT-SWITCHGEAR ROOM REACTOR AUX BLDG
G 354		4	1/18/83	E	CABLE TRAY ARRANGEMENT KEY PLAN
G 355		7	1/11/83	E	TURBINE AREA-UNDERGROUND CONDUIT& GROUNDING SH1
G 356		7	1/11/83	E	TURBINE AREA-UNDERGROUND CONDUIT & GROUNDING SH2
G 358		7	1/11/83	E	TURBINE AREA-UNDERGROUND CONDUIT & GROUNDING SH4

TABLE 1.7-1 (Cont'd)

Drawing No.	Sheet No.	No	Revision Date	Prepared By	Title
G 367		9	3/11/83	E	REACTOR CONTAINMENT BLDG-COND TRAYS & GRDG PLAN-EL 62'-0
G 364		8	4/12/83	E	REACTOR CONTAINMENT BLDG-COND & GRDG-PLAN-BELOW EL 18'-0
G 364	1	6	3/11/83	E	REACTOR CONTAINMENT BLDG CONDUIT LOCATION PLAN
G 365		9	11/22/82	E	REACTOR CONTAINMENT BLDG-COND TRAYS & GRDG PLAN EL-18'-0
G 366		8	11/22/82	E	REACTOR CONTAINMENT BLDG-COND TRAYS & GRDG PLAN EL-45'-0
G 368		8	10/29/82		REACTOR CONTAINMENT BLDG-COND SECTIONS & DETAILS SH-1
G 369	1	6	10/29/82	E	REACTOR CONTAINMENT BLDG-COND SECTIONS & DETAILS SH-2
G 369	2	5	10/6/82	E	REACTOR CONTAINMENT BLDG-COND SECTIONS & DETAILS SH-3
G 369	3	5	10/6/82	E	REACTOR CONTAINMENT BLDG-COND SECTIONS & DETAILS SH-4
G 372	1A	5	3/23/83	E	SUMMARY SHEET CABLE TRAY SUPPORT SH-1A
G 372	1B	1	8/11/78	E	SUMMARY SHEET CABLE TRAY SUPPORT SH-1B
G 372	2	4	11/18/82	E	REACTOR CONT BLDG EL 18.0 CABLE TRAY SUPPORT SH-2
G 372	3	3	8/4/82	E	REACTOR CONT BLDG EL 45.0 CABLE TRAY SUPPORT SH-3
G 372	4	5	8/31/81	E	RCB PEN AREA EL 23-0 CABLE TRAY SUPPORT SH-4
G 372	5	5	3/23/83	E	RCB PEN AREA EL 45-0 CABLE TRAY SUPPORT SH-5

TABLE 1.7-1 (Cont'd)

Drawing No.	Sheet No.	Revision No	Date	Prepared By	Title
G 372	6	6	11/18/82	E	REACTOR AUX BLDG PEN AREA CABLE TRAY SUPPORT SH-6
G 372	7	5	3/23/83	E	REACTOR AUX BLDG EL-05.0 CABLE TRAY SUPPORT SH-7
G 372	8	5	1/21/83	E	REACTOR AUX BLDG EL-05.0 CABLE TRAY SUPPORT SH-8
G 372	9	6	3/23/83	E	REACTOR AUX BLDG EL 19.5 CABLE TRAY SUPPORT SH-9
G 372	10	4	8/31/82	E	REACTOR AUX BLDG EL 19.5 CABLE TRAY SUPPORT SH-10
G 372	11	5	3/23/83	E	REACTOR AUX BLDG EL 43'-0 CABLE TRAY SUPPORT SH-11
G 372	12	4	11/18/82	E	REACTOR AUX BLDG EL 43'-0 CABLE TRAY SUPPORT SH-12
G 372	13	4	1/21/83	E	CABLE VAULT CABLE TRAY SUPPORT SH-13
G 372	14	4	8/31/82	E	REACTOR AUX BLDG EL 74.0 CABLE TRAY SUPPORT SH-14
G 372	15	2	8/31/82	E	CABLE VAULT-CABLE TRAY SUPPORT SH-15
G 372	16	2	8/31/82	E	PENETRATION AREA CABLE TRAY SUPPORT SH-16
G 374	1	6	11/18/82	E	REACTOR AUX BLDG PENETRATION AREA-COND-TRAYS & GRDG SH-1
G 374	3	3	7/28/82	E	REACTOR AUX BLDG PENETRATION AREA-SECTIONS & DETAILS
G 375	1	7	10/29/82	E	REACTOR CONT BLDG PEN AREA-CND, TRAYS & GRDG SH-1
G 375	3	5	1/21/83	E	REACTOR CONT BLDG PEN AREA-SECTIONS & DETAILS

TABLE 1.7-1 (Cont'd)

Drawing No.	Sheet No.	Revision No	Date	Prepared By	Title
G 375	4	3	8/31/82	E	REACTOR CONT BLDG PEN AREA-TRAYS - KEY PLAN
G 377	1	10	3/23/83	E	REACTOR AUXILIARY BUILDING UNDERGROUND COND GRDG SH-1
G 378	2	10	1/18/83	E	REACTOR AUXILIARY BUILDING UNDERGROUND COND & GRDG SH-2
G 380		8	1/18/83	E	OUTLYING AREA CONDUIT GROUNDING & LIGHTING
G 385		8	1/11/83	E	INTAKE STRUCTURE CONDUIT & LIGHTING
G 386		7	1/21/83	E	INTAKE STRUCTURE-LIGHTING SECTION & DETAILS
G 388		8	3/11/83	E	DIESEL GENERATOR BUILDING CONDUIT, GROUNDING & LIGHTING
G 390	1	9	3/23/83	E	REACTOR AUXILIARY BLDG EL-0.5 CONDUIT & TRAYS SH-1
G 391	2	9	1/11/83	E	REACTOR AUXILIARY BLDG EL-0.5 CONDUIT & TRAYS SH-2
G 392	1	5	1/28/83	E	REACTOR AUXILIARY BLDG EL 19'-6 CONDUIT TRAYS & GRDG SH-1
G 393	2	7	1/21/83	E	REACTOR AUXILIARY BLDG EL 19'-6 CONDUIT TRAYS & GRDG SH-2
G 394	1	6	1/15/83	E	REACTOR AUXILIARY BLDG EL 43'-0 & 62'-0 CND TRAYS & GRDG SH-1
G 394	3	6	1/21/83	E	REACTOR AUXILIARY BLDG EL 62'-0 CND & GRDG SH-3
G 395	2	6	1/21/83	E	REACTOR AUXILIARY BLDG EL 43'-0 & 62'-0 CND TRAYS & GRDG SH-2
G 396	1	7	12/15/82	E	REACTOR AUXILIARY BLDG EL 43'-0 SECTIONS & DETAILS SH-1

TABLE 1.7-1 (Cont'd)

Drawing No.	Sheet No.	Revision No	Date	Prepared By	Title
G 396	2	8	1/21/83	E	REACTOR AUXILIARY BLDG EL 43'-0 SECTIONS & DETAILS SH-2
G 396	3	5	11/15/82	E	REACTOR AUXILIARY BLDG SECTIONS & DETAILS SH-3
G 396	4	2	11/15/82	E	REACTOR AUXILIARY BLDG SECTIONS & DETAILS SH-4
G 396	5	2	11/15/82	E	REACTOR AUXILIARY BLDG SECTIONS & DETAILS SH-5
G 396	6	3	3/23/83	E	REACTOR AUXILIARY BLDG SECTIONS & DETAILS SH-6
G 396	7	2	11/15/82	E	REACTOR AUXILIARY BLDG SECTIONS & DETAILS SH-7
G 401	1	6	3/7/83	E	FUEL HANDLING BUILDING CONDUIT TRAYS & GROUNDING SH-1
G 401	2	6	3/7/83	E	FUEL HANDLING BUILDING CONDUIT TRAYS & GROUNDING SH-2
G 402		7	3/7/83	E	FUEL HANDLING BUILDING CONDUIT SECTIONS & DETAILS
B 404		0	5/30/78	E	BOX DETAILS
G 407		7	1/19/83	E	YARD DUCT RUNS & LIGHTING
G 407X		2	7/22/82	E	YARD DUCT RUNS & LIGHTING
G 408	1	6	2/7/83	E	YARD DUCT RUNS & LIGHTING SECTIONS & DETAILS SH-1
G 408 LS	2A	1	1/11/83	E	YARD DUCT RUNS & LIGHTING SECTIONS & DETAILS SH-2A
G 408	2B	3	1/11/83	E	STREAM TRESTLE AREA LTG & DETAILS

TABLE 1.7-1 (Cont'd)

Drawing No.	Sheet No.	Revision No	Date	Prepared By	Title
G 408	1B	2	7/22/82	E	COMPONENT COOLING LIGHTING
G 409		7	3/7/83	E	TRANSFORMER YARD CONDUIT GROUNDING & LIGHTING
G 409	2	4	1/11/83	E	XFRMR YD-PLAN XFRMR FIRE PROT & 5KV & 6.9KV NON-SEG PHASE BUS
G 409X		5	1/11/83	E	TRANSFORMER YARD CONDUIT GROUNDING & LIGHTING
G 409	2	4	1/11/83	E	XFMR YD-PLAN XFMR FIRE PROT & 5KV & 6-9KV NON-SEG PHASE BUS
G 410	1	5	3/7/83	E	CABLE VAULT TRAYS-PLAN & SECTIONS SH-1
G 410	2	4	11/18/82	E	RTG BOARDS-TRAY RISERS-PLAN
G 410	3	4	12/15/82	E	RTE BOARDS-TRAY RISERS-SECT
G 410	6	4	3/23/83	E	CABLE VAULT TRAYS - KEY PLAN
G 410	7	4	3/7/83	E	RAB EL 74.0 CONDUIT TRAYS & GRDG
G 410	8	3	1/18/83	E	RAB EL 62'-0 CONDUIT & GROUNDING
2998-G-386	2	2	11/19/82		INTAKE STRUCTURE LIGHTING SECTION & DETAILS
2998-G-415	1	4	5/6/83		RAB RADIATION MONITORING SYSTEM CONDUIT & EQUIPMENT
2998-G-415	2	4	5/6/83		RAB RADIATION MONITORING SYSTEM CONDUIT & EQUIPMENT
2998-G-415	3	4	5/6/83		RAB RADIATION MONITORING SYSTEM CONDUIT & EQUIPMENT
2998-G-415	4	4	5/6/83		RAB RADIATION MONITORING SYSTEM CONDUIT & EQUIPMENT
2998-G-415	5	4	4/18/83		RCB RADIATION MONITORING SYSTEM CONDUIT & EQUIPMENT
2998-G-415	6	4	4/18/83		RCB RADIATION MONITORING SYSTEM CONDUIT & EQUIPMENT
2998-G-415	7	4	2/22/83		RCB RADIATION MONITORING SYSTEM CONDUIT & EQUIPMENT

TABLE 1.7-1 (Cont'd)

Drawing No.	Sheet No.	Revision No	Date	Prepared By	Title
2998-G-415	8	3	11/19/82		RCB RADIATION MONITORING SYSTEM CONDUIT & EQUIPMENT
2998-G-420	1	2	1/11/83		HEAT TRACE SYSTEM CONDUIT, FLOOR PENETRATION & EQUIPMENT LOCATION
2998-G-420	2	2	1/11/83		HEAT TRACE SYSTEM CONDUIT & TRAY SECTIONS AND DETAILS
2998-G-420	3	2	12/22/82		HEAT TRACE SYSTEM CONDUIT & TRAY
2998-G-420	4	2	12/22/82		HEAT TRACE SYSTEM CONDUIT & TRAY SECTIONS AND DETAILS
2998-G-420	5	2	12/22/82		HEAT TRACE SYSTEM CONDUIT & TRAY
2998-G-420	6	3	12/22/82		HEAT TRACE SYSTEM THERMOCOUPLE & POWER JUNCTION BOXES
2998-G-420	7	3	1/18/83		HEAT TRACE SYSTEM THERMOCOUPLE & POWER JUNCTION BOXES
2998-G-420	8	2	1/18/83		HEAT TRACE SYSTEM THERMOCOUPLE & POWER JUNCTION BOXES
2998-G-420	9	3	1/18/83		HEAT TRACE SYSTEM THERMOCOUPLE & POWER JUNCTION BOXES
2998-G-420	10	3	1/18/83		HEAT TRACE SYSTEM THERMOCOUPLE & POWER JUNCTION BOXES
2998-G-420	11	3	1/18/83		HEAT TRACE SYSTEM THERMOCOUPLE & POWER JUNCTION BOXES
2998-G-420	12	3	5/23/83		HEAT TRACE SYSTEM THERMOCOUPLE & POWER JUNCTION BOXES
2998-G-420	13	3	5/23/83		HEAT TRACE SYSTEM THERMOCOUPLE & POWER JUNCTION BOXES
2998-G-420	14	1	7/29/82		HEAT TRACE SYSTEM THERMOCOUPLE & POWER JUNCTION BOXES
2998-G-420	15	1	7/29/82		HEAT TRACE SYSTEM THERMOCOUPLE & POWER JUNCTION BOXES
2998-G-420	17	0	1/24/83		HEAT TRACE SYSTEM THERMOCOUPLE & POWER JUNCTION BOXES

TABLE 1.7-1 (Cont'd)

Drawing No.	Sheet No.	Revision No	Date	Prepared By	Title
G 224	1	R8	8/24/82	E	TURBINE BUILDING INSTRUMENT ARR SH-1
G 226	1	R7	12/3/82	E	REACTOR BUILDING INSTRUMENT ARR SH-1
G 226	2	R8	3/7/83	E	REACTOR BUILDING INSTRUMENT ARR SH-2
G 226	3	R7	2/1/83	E	REACTOR BUILDING INSTRUMENT ARR SH-3
G 226	4	R7	2/1/83	E	REACTOR BUILDING INSTRUMENT ARR SH-4
G 226	5	R7	2/1/83	E	REACTOR BUILDING INSTRUMENT ARR SH-5
G 226	6	R7	10/24/82	E	REACTOR BUILDING INSTRUMENT ARR SH-6
G 226	7	R6	10/24/82	E	REACTOR BUILDING INSTRUMENT ARR SH-7
G 227	1	R8	3/7/83	E	REACTOR AUXILIARY BUILDING INSTRUMENT ARR SH-1
G 227	2	R8	3/7/83	E	REACTOR AUXILIARY BUILDING INSTRUMENT ARR SH-2
G 227	3	R4	6/21/81	E	REACTOR AUXILIARY BUILDING INSTRUMENT ARR SH-3
G 227	4	R9	12/3/82	E	REACTOR AUXILIARY BUILDING INSTRUMENT ARR SH-4
G 227	5	R6	3/7/83	E	REACTOR AUXILIARY BUILDING INSTRUMENT ARR SH-5
G 227	6	R7	12/3/83	E	REACTOR AUXILIARY BUILDING INSTRUMENT ARR SH-6
G 227	6	R6	1/22/82	E	REACTOR AUXILIARY BUILDING INSTRUMENT ARR SH-7
G 227	8	R6	3/7/83	E	REACTOR AUXILIARY BUILDING INSTRUMENT ARR SH-8

TABLE 1.7-1 (Cont'd)

Drawing No.	Sheet No.	Revision No	Date	Prepared By	Title
G 228	1	R6	3/7/83	E	FUEL HANDLING BUILDING INSTRUMENT ARR
G 229	1	R6	1/13/82	E	MISCELLANEOUS INSTRUMENT ARR
B 231	604 SHTS		VARIOUS	E	INSTRUMENT INSTALLATION DETAILS
G 232	4	R4	3/7/83	E	REACTOR AUX BLDG ANALYZER & SAMPLING LINES ARR
	5	R3	3/7/83	E	REACTOR AUX ALDG ANALYZER & SAMPLING LINES ARR
	7	R3	2/1/83	E	REACTOR BLDG ANALYZER & SAMPLING LINES ARR
	8	R2	10/22/81	E	REACTOR BLDG ANALYZER & SAMPLING LINES ARR
	9	R2	10/22/81	E	REACTOR BLDG ANALYZER & SAMPLING LINES ARR
G 233	1	R3	12/3/82	E	REACTOR AUXILIARY BUILDING LABORATORY GAS SYSTEM LAYOUT
G 278		R1	11/21/79	E	CONTROL & BLOCK DIAGRAM CONTAINMENT SPRAY & RECIRCULATION SYSTEM
B 326				E	SCHEMATIC DIAGRAMS
	103S	R2	5/26/83	E	OIL LIFT PUMPS FOR REACTOR COOLANT PUMP P-2A1 (2B1, 2A2, 2B2-TYP.)
	139S	R2	4/11/83	E	PRESSURIZER LEVEL CH-L-1110
	159	R2	5/23/83	E	VALVES V-2505, V2510, V2511 & V-2524
	163S	R1	5/26/83	E	VALVES FCV-2210X, FCV-2210Y & V-2512
	174S	R3	4/21/83	E	BORIC ACID MAKE-UP PUMP 2A
	175S	R3	5/26/83	E	BORIC ACID MAKE-UP PUMP 2B
	177S	R3	4/11/83	E	CHARGING PUMP 2A

TABLE 1.7-l (Cont'd)

Drawing No.	Sheet No.	Revision No	Date	Prepared By	Title
B 326					SCHEMATIC DIAGRAMS (Cont'd)
	178S	R3	5/26/83	E	CHARGING PUMP 2B
	179S	R3	5/26/83	E	CHARGING PUMP 2C
	187S	R2	4/21/83	E	CHARGING PUMPS SEAL LUBRI-CATION SYSTEM VALVES V-2627, V-2628, V-2629
	201S	R2	4/21/83	E	COMPONENT COOLING WATER PUMP 2A
	203S	R2	5/26/83	E	COMPONENT COOLING WATER SUCTION HDR VALVE MV-14-3 (MV-14-1, 14-2 & 14-4-TYP.)
	205S	R2	4/21/83	E	COMPONENT COOLING WATER PUMP 2B
	209S	R2	4/21/83	E	COMPONENT COOLING WATER PUMP 2C
	237S	R2	4/21/83	E	HP SAFETY INJECTION PUMP 2A
	238S	R2	4/21/83	E	HP SAFETY INJECTION PUMP 2B
	249S	R2	5/23/83	E	SHUTDOWN COOLING ISOLATION VALVE V-3480 (V-3481, V-3651, V-3652-TYP.)
	251S	R2	5/23/83	E	LP SAFETY INJECTION PUMP 2A
	252S	R2	5/23/83	E	LP SAFETY INJECTION PUMP 2B
	257S	R2	5/23/83	E	LP SAFETY INJECTION FLOW CONT VALVES (HCV-3615, 3626, 3637, 3625, 3616, 3617, 3635, 3636, 3637, 3645, 3646, 3647-TYP.)
	269S	R2	5/23/83	E	SAFETY INJECTION TANK 2A1 ISOL VALVE V-3624 (3614, 3634, 3644-TYP.)
	285S	R2	5/26/83	E	CONTAINMENT FAN COOLER 2-HVS-1A (-1B,-1C,-1D-TYP.)
	287S	R2	5/26/83	E	CONTAINMENT SPRAY PUMP 2A

TABLE 1.7-1 (Cont'd)

Drawing No.	Sheet No.	Revision No	Revision Date	Prepared By	Title
B 326					SCHEMATIC DIAGRAMS (Cont'd)
	289S	R2	5/26/83	E	CONTAINMENT SPRAY VALVES FCV-07-1A & FCV-07-1B
	290S	R2	5/26/83	E	CONTAINMENT SPRAY PUMP 2B
	297S	R1	5/28/83	E	REFUELING WATER TANK VALVE MV-07-1A (07-1B-TYP.)
	299S	R2	5/26/83	E	REACTOR SUMP VALVE MV-07-2A (07-2B-TYP)
	311S	R2	5/26/83	E	MAIN STEAM ISOLATION BYPASS VALVE MV-08-1A (08-113-TYP)
	312S	R3	5/26/83	E	MAIN STEAM ISOL VALVE HCV-08-1A OPENING, CLOSING & SOL TEST
	315S	R3	5/26/83	E	MAIN STEAM ISOL VALVE HCV-08-1B OPENING, CLOSING & SOL TEST
	411S	R2	5/26/83	E	REACTOR TRIP BKR. TCB-1
	482S	R1	6/24/83	E	REACTOR CONTAINMENT & SHLD BLDG DIFF PRESS
	490S	R2	6/24/83	E	CONTROL ROOM EMERG FILTRATION FAN 2HVE-13A (13B-TYP)
	492S	R2(0)	7/22/83	E	CONTROL ROOM AIR COND UNIT 2-HVA/ACC-3A(-3B,-3C TYP) SH 1
	493S	R2(0)	7/22/83	E	CONTROL ROOM AIR COND UNIT 2-HVA/ACC-3A(-3B,-3C TYP) SH 2
	503S	R2	6/24/83	E	REACTOR AUX BLDG EMERG EXHAUST FAN 2HVE-9A (9B-TYP)
	505S	R2	6/24/83	E	REACTOR AUX BLDG SUPPLY FAN 2HVS-4A (4B-TYP)
	507S	R1	6/24/83	E	CEDM COOLING FAN 2HVE-21A

TABLE 1.7-1 (Cont'd)

Drawing No.	Sheet No.	Revision No	Revision Date	Prepared By	Title
B 326					SCHEMATIC DIAGRAMS (Con't')
	508S	R1	6/23/83	E	CEDM COOLING FAN 2HVE-21F
	509S	R2	6/24/83	E	REACTOR CONTAINMENT PURGE EXHAUST FAN 2HVE-8A
	510S	R2	6/24/83	E	REACTOR CONTAINMENT PURGE EXHAUST FAN 2HVE-8B
	511S	R2	6/24/83	E	REACTOR CONTAINMENT PURGE ISOLATION VALVES - SH. 1
	512S	R2	6/24/83	E	REACTOR CONTAINMENT PURGE ISOLATION VALVES - SH. 2
	513S	R2	6/24/83	E	SHIELD BLDG VENT EXHAUST FAN 2HVE-6A
	516S	R2	5/24/83	E	SHIELD BLDG VENT EXHAUST FAN 2HVE-6B
	629S	R2	7/18/83	E	AUX FEEDWATER PUMP 2A
	630S	R2	7/18/83	E	AUX FEEDWATER PUMP 2B
	631S	R2	7/18/83	E	AUX FEEDWATER PUMP 2C TURBINE AND STM VLV MV-08-3
	711	R2(0)	7/22/83	E	EMERG TURBINE TRIP & TURBINE ALARMS
	832S	R2	7/18/83	E	INTAKE COOLING WATER PUMP 2A
	833S	R2	7/18/83	E	INTAKE COOLING WATER PUMP 2B
	834S	R2	7/18/83	E	INTAKE COOLING WATER PUMP 2C
	835S	R2	7/18/83	E	INTAKE COOLING WATER NON-EMERG HDR A ISOL VALVE MV-21-3 (MV-21-2-TYP)
	934S	R2	6/24/83	E	4160V SWGR 2A2 FDR TO BUS 2A3 (2B2 FDR TO BUS 2B3-TYP)

TABLE 1.7-1 (Cont'd)

Drawing No.	Sheet No.	Revision No	Prepared Date	By	Title
B 326					SCHEMATIC DIAGRAMS (Cont'd)
	936S	R2	6/24/83	E	4160V SWGR 2A3 INCOMING EEDER FROM BUS 2A2 (2B3 FDR ROM BUS 2B2-TYP)
	938S	R2	6/24/83	E	4160V SWGR 2A3 FDR TO BUS 2AB 2B3 FDR TO BUS 2AB-TYP)
	940S	R2	6/24/83	E	4160V SWGR 2AB INCOMING FEEDER FROM BUS 2A3 (2AB FDR FROM BUS 2B 3-TYP)
	949S	R2	6/24/83	E	4160V SWGR 2A3 LOAD SHEDDING RELAYS
	950S	R2	6/24/83	E	4160V SWGR 2B3 LOAD SHEDDING RELAYS
	951S	R1	6/24/83	E	4160V SWGR 2AB LOAD SHEDDING RELAYS
	953S	R2	6/24/83	E	DIESEL GENERATOR 2A BREAKER
	956S	R2	6/24/83	E	DIESEL GENERATOR 2A LOCKOUT RELAY
	957S	R2	6/24/83	E	DIESEL GENERATOR 2A START CKTS SH 1
	959S	R2	6/24/83	E	DIESEL GENERATOR 2A START SOLENOIDS
	963S	R2	6/24/83	E	DIESEL GENERATOR 2B BREAKER
	966S	R2	6/24/83	E	DIESEL GENERATOR 2B LOCKOUT RELAY
	967S	R2	6/24/83	E	DIESEL GENERATOR 2B START CKTS SH 1
	969S	R2	6/24/83	E	DIESEL GENERATOR 2B START SOLENOIDS
	1000S	R2	6/24/83	E	125V DC BUS TRANSFER CONTROL
	1170S	R2	6/24/83	E	CONTROL ROOM NORTH OUTSIDE AIR INSUL VA FCV-25-M

TABLE 1.7-1 (Cont'd)

Drawing No.	Sheet No.	Revision No	Prepared Date	By	Title
B 326					SCHEMATIC DIAGRAMS (Cont'd)
	1176S	R2	6/24/83	E	SHIELD BLDG VENT COOL AIR VA FCV-25-11 (FCV-25-12 TYP)
	1501S	R2(0)	7/22/83	E	SHUTDOWN COOLING ISOL, HEAT EXCH, WARM-UP & CONTROL VALVES (V-3545, 3664, 3665, 3456, 3457, 3517, 3658, 3536, 3539; HCV-3657, 2512, 3306, 3301)
	1601S	R2	6/24/83	E	DIESEL GENERATOR 2A START CKTS SH 2
	1602S	R2	6/24/83	E	DIESEL GENERATOR 2A START CKTS SH 3
	1603S	R2	6/24/83	E	DIESEL GENERATOR 2A START CKTS SH 4
	1604S	R2	6/24/83	E	DIESEL GENERATOR 2A START CKTS SH 5
	1605S	R2	6/24/83	E	DIESEL GENERATOR 2A START CKTS SH 6
	1611S	R2	6/24/83	E	DIESEL GENERATOR 2B START CKTS SH 2
	1612S	R2	6/24/83	E	DIESEL GENERATOR 2B START CKTS SH 3
	1613S	R2	6/24/83	E	DIESEL GENERATOR 2B START CKTS SH 4
	1614S	R2	6/24/83	E	DIESEL GENERATOR 2B START CKTS SH 5
	1615S	R2	6/24/83	E	DIESEL GENERATOR 2B START CKTS SH 6

TABLE 1.7-1 (Cont'd)

Drawing No.	Sheet No.	Revision No	Revision Date	Prepared By	Title
	1	R14	4/1/83		INDEX
	2	R14	4/1/83		INDEX
	3	R13	4/1/83		INDEX
	4	R14	4/1/83		INDEX
	5	R14	4/1/83		INDEX
	6	R14	4/1/83		INDEX
	7	R14	4/1/83		INDEX
	8	R14	4/1/83		INDEX
	8A	R14	4/1/83		INDEX
	8B	R14	4/1/83		INDEX
	8B-1	R4	4/1/83		INDEX
	8B-2	R3	4/1/83		INDEX
B-327					CONTROL WIRING DIAGRAM
					NUCLEAR INSTRUMENTATION
	8DS	R4	9/9/82	E	ANNUNCIATOR REFLASH MODULES SH 1
	8ES	R4	9/9/82	E	ANNUNCIATOR REFLASH MODULES SH 2
	8FS	R3	9/9/82	E	ANNUNCIATOR REFLASH MODULES SH 3
	8HS	R3	9/9/82	E	ANNUNCIATOR REFLASH MODULES SH 5
	8IS	R3	9/9/82	E	ANNUNCIATOR REFLASH MODULES SH 6
	5OS	R5	9/9/82	E	NUCLEAR INSTR SYS WIDE RANGE LOG CH-001A, 001B
	51S	R2	9/9/82	E	NUCLEAR INSTR SYS WIDE RANGE LOG CH-001C, 001D
	54S	R4	11/5/82	E	NUCLEAR INSTR. SYS. PWR RANGE SAF CH-003A/004A, 003B/004B, 003C/004C
	55S	R5	10/21/82	E	NUCLEAR INSTR. SYS.PWR RANGE SAF CH-003D/004D
	56S	R2	7/2/82	E	NUCLEAR INSTR. SYS.FLUX INDICATORS
	60S	R4	9/23/82	E	OUT-OF-CORE NEUTRON DETECTORS NO. 1, 2, 5 & 9

TABLE 1.7-1 (Cont'd)

Drawing No.	Sheet No.	Revision No	Date	Prepared By	Title
B-327					CONTROL WIRING DIAGRAM (Cont'd)
	61S	R4	9/23/82	E	OUT-OF-CORE NEUTRON DETECTORS NO. 4, 6, 10 & 11
	62S	R4	9/23/82	E	OUT-OF- CORE NEUTRON DETECTORS NO. 3 & 7
	63S	R4	9/23/82	E	OUT-OF-CORE NEUTRON DETECTORS NO. 8 & 12
	90S	R9	10/6/82	E	PRESSURIZER LEVEL CHANNEL L-1110 SH. 3
	91S	R1	5/26/83	E	MEASUREMENT CHANNELS P-1105 & P-1106
	101S	R13	10/20/82	E	REACTOR COOLANT PUMP 2A1
	103S	R11	5/26/83	E	OIL LIFT PUMPS FOR REACTOR COOLANT PUMP 2A1
	105S	R12	11/10/82	E	REACTOR COOLANT PUMP 2B1
	107S	R11	1/24/83	E	OIL LIFT PUMPS FOR REACTOR COOLANT PUMP 2B1
	109S	R10	10/20/82	E	REACTOR COOLANT PUMP 2A2
	111S	R11	1/24/83	E	OIL LIFT PUMPS FOR REACTOR COOLANT PUMP 2A2
	113S	R10	10/20/82	E	REACTOR COOLANT PUMP 2B2
	115S	R11	5/26/83	E	OIL LIFT PUMPS FOR REACTOR COOLANT PUMP 2B2
	118S	R6	2/26/82	E	PRESSURIZER RELIEF ISOLATION VALVE V-1477
	120S	R6	2/26/82	E	PRESSURIZER RELIEF ISOLATION VALVE V-1476
	136S	R9	1/8/83	E	REACTOR COOLANT LOOP TEMP CHT-1111Y, T-1111X & T-1115
	137S	R10	1/8/83	E	REACTOR COOLANT LOOP TEMP CHT-1121Y, T-1121X & T-1125

TABLE 1.7-1 (Cont'd)

Drawing No.	Sheet No.	Revision No	Date	Prepared By	Title
B-327					CONTROL WIRING DIAGRAM (Cont'd)
	139S	R9	1/31/83	E	PRESSURIZER LEVEL CH L-1110 SH2
	140S	R12	5/26/83	E	MEASUREMENT CHANNELS L-1103, L-1116 & P-1103
	141S	R9	4/14/83	E	REACTOR HEAD SEAL P-1118 & QUENCH TANK P-1116 - PRESS
					CHEMICAL & VOLUME SYSTEM
	146S	R4	12/20/82	E	CHEM & VOL CONTROL SYSTEM-BORIC ACID HEAT TRACE TRANSF 2A
	147S	R4	12/20/82	E	CHEM & VOL CONTROL SYSTEM-BORIC ACID HEAT TRACE TRANSF 2B
	150S	R9	1/8/83	E	MEASUREMENT CHANNELS F-2212, P-2212, P-2215, T-2229 & T-2221
	154S	R6	8/14/82	E	MEASUREMENT CHANNELS T-2225, P-2225, L-2227 & L-2226
	157S	R7	11/17/81	E	LETDOWN STOP VA V-2515 AND LET-DOWN CONTAINMENT ISOL VA V-2516
	159S	R5	8/17/82	E	VALVES V-2505, V-2510, V-2511& V-2524
	161S	R8	12/11/82	E	VOLUME CONTROL TANK DISCHARGE VALVE V-2501
	163S	R6	8/17/82	E	VALVES FCV-2210X, FCV-2210Y & V-2512
	165S	R9	12/15/82	E	BORIC ACID GRAVITY FEED VALVE V-2508
	166S	R8	9/18/82	E	BORIC ACID GRAVITY FEED VALVE V-2509
	167S	R8	9/17/82	E	MAKE-UP BYPASS TO CHARGING PUMPS VALVE V-2514
	174S	R9	2/28/83	E	BORIC ACID MAKE-UP PUMP 2A
	175S	R9	2/28/83	E	BORIC ACID MARE-UP PUMP 2B

TABLE 1.7-1 (Cont'd)

Drawing No.	Sheet No.	Revision No	Date	Prepared By	Title
B-327					CONTROL WIRING DIAGRAM (Cont'd)
	176S	R7	8/26/82	E	CHARGING LINES 2B1 & 2A2 VA'S I-SE-02-01 & I-SE-02-02 & RECIRC DRAIN TK VA V-3661
	177S	R10	12/6/82	E	CHARGING PUMP 2A
	178S	R10	12/6/82	E	CHARGING PUMP 2B
	179S	R10	12/6/82	E	CHARGING PUMP 2C
	180S	R6	9/9/82	E	FUEL POOL PUMP 2A
					FUEL POOL SYSTEM
	181S	R8	4/20/82	E	FUEL POOL PUMP 2B
	182S	R8	2/10/83	E	FUEL POOL PURIFICATION PUMP
					COMPONENT COOLING WATER SYSTEM
	187S	R5	8/25/82	E	CHARGING PUMP SEAL LUBE SYS VALVES V-2627, V-2628 & V-2629
	188S	R4	7/2/82	E	CHEMICAL & VOLUME CONTROL SYSTEM ANN REFLASH CIRCUITS
	189S	R5	1/10/83	E	AUX SPRAY VALVES 1-SE-02-03, 1-SE-02-04
	190S	R6	12/11/82	E	BORON LOAD CONTROL VALVE V-2525
	192S	R9	3/24/83	E	MAKE-UP SYSTEM CH F-2210
	194S	R7	8/4/82	E	LETDOWN CONTROL & CHARGING LINE ISOL VALVES V-2522 & V-2523
	196S	R5	9/29/82	E	CHARGING PUMP 2A BYPASS VALVE V-2555
	197S	R6	9/29/82	E	CHARGING PUMP 2B BYPASS VALVE V-2554
	198S	R6	1/12/83	E	CHARGING PUMP 2C BYPASS VALVE V-2553
	201S	R7	8/25/82	E	COMPONENT COOLING WATER PUMP 2A

TABLE 1.7-1 (Cont'd)

Drawing No.	Sheet No.	Revision No	Date	Prepared By	Title
B-327					CONTROL WIRING DIAGRAM (Cont'd)
	202S	R7	3/3/82	E	NORMAL SUPPLY HDR & NORMAL RE-TURN HDR ISOL VALVES
	203S	R6	9/18/82	E	COMPONENT COOLING WATER SUCTION HDR A VALVE MV-14-3
	204S	R7	9/18/82	E	COMPONENT COOLING WATER DISCH HDR A VALVE MV-14-1
	205S	R7	11/10/82	E	COMPONENT COOLING WATER PUMP 2B
	206S	R6	12/6/82	E	CCW FROM RCP'S
	207S	R6	9/18/82	E	COMPONENT COOLING WATER SUCTION HDR B VALVE MV-14-4
	208S	R7	8/27/82	E	COMPONENT COOLING WATER DISCH HDR B VALVE MV-14-2
	209S	R7	8/21/82	E	COMPONENT COOLING WATER PUMP 2C
	211S	R10	2/28/83	E	COMPONENT COOL WTR SHUTDN HT EXCH & SURGE TANK FILL VALVES
	212S	R5	6/3/82	E	CCW TO & FROM REACTOR COOL PUMPS HCV-14-1, 2 & HCV-14-6, 7
	217S	R7	8/21/82	E	COMP. COOL. WTR A FLOW & PRESSURE
	218S	R8	9/9/82	E	COMP. COOL. WTR B FLOW & PRESSURE
	220S	R7	9/18/82	E	COMP. COOL. WTR TO CONT COOL UNIT 2A VALVE MV-14-9
	221S	R5	9/18/82	E	COMP. COOL. WTR FROM CONT. COOL. UNIT 2A VALVE MV-14-10
	222S	R5	9/18/82	E	COMP. COOL. WTR TO CONT. COOL. UNIT 2B VALVE MV-14-11
	223S	R7	9/18/82	E	COMP. COOL. WTR FROM CONT. COOL. UNIT 2B VALVE MV-14-12
	224S	R5	9/18/82	E	COMP. COOL. WTR TO CONT. COOL. UNIT 2C VALVE MV-14-13

TABLE 1.7-1 (Cont'd)

Drawing No.	Sheet No.	Revision No	Date	Prepared By	Title
B-327					CONTROL WIRING DIAGRAM (Cont'd)
	225S	R5	9/18/82	E	COMP. COOL. WTR FROM CONT. COOL. UNIT 2C VALVE MV-14-14
	226S	R5	9/18/82	E	COMP. COOL. WTR TO COOL UNIT 2D VALVE MV-14-15
	227S	R6	9/27/82	E	COMP. COOL. WTR FROM CONT. COOL. UNIT 2D VALVE MV-14-16
	228S	R6	2/28/83	E	COMP. COOL. HDR B TO FUEL POOL HT EXCH VALVE MV-14-17
	229S	R6	12/11/82	E	COMP. COOL. HDR A TO FUEL POOL HT EXCH VALVE MV-14-18
	230S	R4	9/18/82	E	COMP. COOL. HDR B FROM FUEL POOL HT EXCH VALVE MV-14-19
	231S	R5	2/12/83	E	COMP. COOL. HDR A FROM FUEL POOL HT EXCH VALVE MV-14-20
					SAFETY INJECTION
	233S	R9	9/18/82	E	HP SAFETY INJECTION TO HOT LEG 2A VALVE V-3540
	234S	R8	9/18/82	E	HP SAFETY INJECTION TO HOT LEG 2A VALVE V-3550
	235S	R8	9/18/82	E	HP SAFETY INJECTION TO HOT LEG 2B VALVE V-3523
	236S	R8	9/18/82	E	HP SAFETY INJECTION TO HOT LEG 2B VALVE V-3551
	237S	R5	11/17/81	E	HP SAFETY INJECTION PUMP 2A
	238S	R6	9/3/82	E	HP SAFETY INJECTION PUMP 2B
	239S	R2	10/7/80	E	4160V SWGR 2AB SPARE
	242S	R8	8/17/82	E	SI TANK FILL & DRAIN VALVES I-SE-03-1A, I-SE-03-1B, I-SE-03-1C, I-SE-03-1D
	244S	R9	12/17/82	E	MINIMUM FLOW ISOLATION VALVE V-3659

TABLE 1.7-1 (Cont'd)

Drawing No.	Sheet No.	Revision No	Date	Prepared By	Title
B-327					CONTROL WIRING DIAGRAM (Cont'd)
	245S	R8	12/17/82	E	MINIMUM FLOW ISOLATION VALVE V-3660
	246S	R7	3/25/83	E	SAFETY INJECTION CHANNEL A-TRIP & BLOCK
	247S	R0	1/29/82	E	SAFETY INJECTION TANK VENT VALVES
	248S	R7	3/25/83	E	SAFETY INJECTION CHANNEL B-TRIP & BLOCK
	249S	R8	1/12/83	E	SHUTDOWN COOLING ISOLATION VALVE V-3480
	250S	R9	1/12/83	E	SHUTDOWN COOLING ISOLATION VALVE V-3481
	251S	R4	6/30/82	E	LP SAFETY INJECTION PUMP 2A
	252S	R4	8/17/82	E	LP SAFETY INJECTION PUMP 2B
	253S	R9	1/12/83	E	SHUTDOWN COOLING ISOLATION VALVE V-3651
	254S	R9	1/12/83	E	SHUTDOWN COOLING ISOLATION VALVE V-3652
	255S	R4	8/17/82	E	ISOL VALVES V-3614, V-3624, V-3634 & V-3644 POSITION INDICATORS
	256S	R3	8/17/82	E	N_2 TO SI TANK VALVES V-3612, V-3622, V-3632 & V-3642
	257S	R9	12/15/82	E	LOW PRESS SAFETY INJECT FLOW CONT VALVE HCV-3615
	258S	R7	9/18/82	E	HIGH PRESS SAFETY INJECT FLOW CONT VALVE HCV-3626
	259S	R8	1/14/83	E	AUX HIGH PRESS FLOW CONT VALVE HCV-3627
	260S	R10	1/14/83	E	LOW PRESS SAFETY INJECT FLOW CONT VALVE HCV-3625

TABLE 1.7-1 (Cont'd)

Drawing No.	Sheet No.	Revision No	Date	Prepared By	Title
B-327					CONTROL WIRING DIAGRAM (Cont'd)
	261S	R8	1/14/83	E	HIGH PRESS SAFETY INJECT FLOW CONT VALVE HCV-3616
	262S	R7	9/18/82	E	AUX HIGH PRESS FLOW CONT VALVE HCV-3617
	263S	R9	9/18/82	E	LOW PRESS SAFETY INJECT FLOW CONT VALVE HCV-3635
	264S	R7	9/18/82	E	HIGH PRESS SAFETY INJECT FLOW CONT VALVE HCV-3636
	265S	R9	1/14/83	E	AUX HIGH PRESS FLOW CONT VALVE HCV-3637
	266S	R8	9/18/82	E	LOW PRESS SAFETY INJECT FLOW CONT VALVE HCV-3645
	267S	R8	10/14/82	E	HIGH PRESS SAFETY INJECT FLOW CONT VALVE HCV-3646
	268S	R8	10/14/82	E	AUX HIGH PRESS FLOW CONT VALVE HCV-3647
	269S	R6	8/5/82	E	SAFETY INJECT TANK 2A1 ISOL VALVE V-3624
	270S	R6	8/5/82	E	SAFETY INJECT TANK 2A2 ISOL VALVE V-3614
	271S	R6	8/5/82	E	SAFETY INJECT TANK 2B1 ISOL VALVE V-3634
	272S	R7	2/28/83	E	SAFETY INJECT TANK 2B2 ISOL VALVE V-3644
	273S	R10	7/20/82	E	MEASUREMENT CHANNELS F-3305, P-3307, P-3308, P-3309, P-3303X & P-3303Y
	275S	R0	1/29/82	E	SI TANK VENT VALVES V-3736, V-3734, V-3738, V-3740
	277S	R7	10/14/82	E	HPSI PUMP DISCHARGE VALVE V-3654
	279S	R7	10/14/82	E	HPSI PUMP DISCHARGE VALVE V-3656

TABLE 1.7-1 (Cont'd)

Drawing No.	Sheet No.	Revision No	Date	Prepared By	Title
B-327					CONTROL WIRING DIAGRAM (Cont'd)
	280S	R6	8/4/82	E	SI TANK 2A2 INSTR & CHECK VA LEAKAGE DRAIN TO RWT HCV-3618
	281S	R8	12/15/82	E	SI TANK 2A1 INSTR & CHECK VA LEAKAGE DRAIN TO RWT HCV-3628
	282S	R6	8/4/82	E	SI TANK 2B1 INSTR & CHECK VA LEAKAGE DRAIN TO RWT HCV-3638
	283S	R6	8/4/82	E	SI TANK 2B2 INSTR & CHECK VA LEAKAGE DRAIN TO RWT HCV-3648
	284S	R7	8/24/82	E	HIGH PRESSURE SAFETY INJECTION FLOW & PRESSURE MONITORS
					CONTAINMENT COOLING
	285S	R5	2/18/83	E	CONTAINMENT FAN COOLER 2-HVS-1A
	286S	R5	2/18/83	E	CONTAINMENT FAN COOLER 2-HVS-1B
	287S	R5	6/30/82	E	CONTAINMENT SPRAY PUMP 2A
	288S	R6	5/2/83	E	IODINE REMOVAL SYSTEM INSTRU-MENTATION
	289S	R7	8/11/82	E	CONTAINMENT SPRAY VALVES FCV-07-1A & FCV-07-1B
	290S	R5	8/21/82	E	CONTAINMENT SPRAY PUMP 2B
	291S	R5	3/11/83	E	HYDRAZINE SYSTEM PUMP 2A
	292S	R5	3/11/83	E	HYDRAZINE SYSTEM PUMP 2B
	293S	R9	7/30/82	E	CONT PRESS, SPRAY HDR A PRESS & FLOW & REFUEL WTR TANK LEVEL
	294S	R9	7/30/82	E	CONT PRESS, SPRAY HDR B PRESS & FLOW & REFUEL WTR TANK LEVEL
	295S	R7	7/30/82	E	CONT PRESSURE & REFUELING WATER TANK LEVEL - 1

TABLE 1.7-1 (Cont'd)

Drawing No.	Sheet No.	Revision No	Revision Date	Prepared By	Title
B-327					CONTROL WIRING DIAGRAM (Cont'd)
	296S	R10	4/20/83	E	CONT PRESSURE, TEMP & REFUELING WATER TANK LEVEL
	297S	R8	12/11/82	E	REFUEL WATER TANK VALVE MV-07-1A
	298S	R8	12/11/82	E	REFUEL WATER TANK VALVE MV-07-1B
	299S	R6	12/11/82	E	REACTOR SUMP VALVE MV-07-2A
	300S	R6	12/11/82	E	REACTOR SUMP VALVE MV-07-2B
	302S	R5	12/6/82	E	CONTAINMENT SPRAY & RECIRC ACTUATION CH'S A MAN RESET
	303S	R7	12/6/82	E	CONTAINMENT SPRAY & RECIRC ACTUATION CH'S B MAN RESET
	304S	R6	2/18/83	E	CONTAINMENT FAN COOLER 2-HVS-1C
	305S	R6	2/18/83	E	CONTAINMENT FAN COOLER 2-HVS-1D
	307S	R2	8/26/82	E	MCC 2A9 FDR BKR (2-HVS-1A)
	308S	R2	3/30/82	E	MCC 2A9 FDR BKR (2-HVS-1B)
	309S	R2	6/3/82	E	MCC 2B9 FDR 8KR (2-HVS-1C)
	310S	R2	4/16/82	E	MCC 2B9 FDR BKR (2-HVS-1D)
					CONTAINMENT ISOLATION
	311S	R7	12/11/82	E	MAIN STEAM ISOLATION BYPASS VALVE MV-08-1A
	312S	R10	8/20/82	E	MAIN STEAM ISOL VALVE HCV-08-1A OPENING, CLOSING & SOL TEST
	313S	R7	3/25/83	E	MAIN STEAM ISOL VALVE HCV-08-1A STROKE TEST & SOLENOID TEST

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TABLE 1.7-1 (Cont'd)

Drawing No.	Sheet No.	Revision No	Revision Date	Prepared By	Title
B-327					CONTROL WIRING DIAGRAM (Cont'd)
	314S	R7	12/11/82	E	MAIN STEAM ISOLATION BYPASS VALVE MV-08-1B

TABLE 1.7-1 (Cont'd)

Drawing No.	Sheet No.	Revision No	Date	Prepared By	Title
B-327					CONTROL WIRING DIAGRAM (Cont'd) CONTAINMENT ISOLATION (Cont'd)
	315S	R8	6/3/82	E	MAIN STEAM ISOL VALVE HCV-08-1B OPENING, CLOSING & SOL TEST
	316S	R7	3/25/83	E	MAIN STEAM ISOL VALVE HCV-08-1B STROKE TEST & SOLENOID TEST
	317S	R4	10/21/82	E	INSTRUMENT AIR ISOLATION VALVE HCV-18-1
	319S	R8	5/11/83	E	STEAM GEN BLOWDOWN ISOL VALVES FCV-23, 3, 4, 5 & 6
	320S	R6	1/10/83	E	CONTAINMENT SAMPLE ISOLATION VALVES
	321	R1	3/12/82	E	CONTAINMENT ISOLATION VALVE I-SE-07-5A, -5C, -5E
	322	R1	3/12/82	E	CONTAINMENT ISOLATION VALVE I-SE-07-5B, -5D, -5F
	323	R1	3/12/82	E	CONTAINMENT PRESSURE CHANNELS P-07-4A1 & P-07-4B1
	324	R1	3/26/82	E	CONTAINMENT WATER LEVEL L-07-13A, -13B, -14A
	330S	R6	12/6/82	E	CONTAINMENT ISOL CH A-MAN RESET & MAIN STM ISOL VA BLOCK A
	331S	R7	12/6/82	E	CONTAINMENT ISOL CH B-MAN RESET & MAIN STM ISOL VA BLOCK B
	333	R3	2/10/82	E	CONT RADIATION MONITORS DETECTOR NO. RD-26-3 & RD-26-4
	334	R3	2/10/82	E	CONT RADIATION MONITORS DETECTOR NO. RD-26-5 & RD-26-6

TABLE 1.7-1 (Cont'd)

Drawing No.	Sheet No.	Revision No	Revision Date	Prepared By	Title
B-327					CONTROL WIRING DIAGRAM (Cont'd)
					SPENT FUEL POOL
	335S	R5	11/10/82	E	AREA RADIATION MONITOR DETECTOR NO.-RD-26-7
	336S	R5	11/24/82	E	AREA RADIATION MONITOR DETECTOR NO.-RD-26-8
	337	R2	2/26/82	E	AREA RADIATION MONITOR DETECTOR NO. RD-26-9
	338	R2	2/26/82	E	AREA RADIATION MONITOR DETECTOR NO. RD-26-10
	339	R2	2/26/82	E	AREA RADIATION MONITOR DETECTOR NO. RD-26-11
	340	R2	2/26/82	E	AREA RADIATION MONITOR DETECTOR NO. RD-26-12
					REACTOR PROTECTIVE SYSTEM
	369S	R5	12/6/82	E	STEAM GENERATORS 2A/2B PRESSURE & LEVEL
	370S	R5	7/30/82	E	PRESSURIZER PRESSURE & LEVEL
	371S	R3	4/16/82	E	STEAM GENERATORS 2A & 2B LEVEL
	372S	R6	12/6/82	E	PRESSURIZER PRESSURE P-1102A MEASUREMENT LOOP
	373S	R6	12/6/82	E	PRESSURIZER PRESSURE P-1102B MEASUREMENT LOOP
	374S	R7	12/6/82	E	PRESSURIZER PRESSURE P-1102C MEASUREMENT LOOP
	375S	R6	12/6/82	E	PRESSURIZER PRESSURE P-1102D MEASUREMENT LOOP

TABLE 1.7-1 (Cont'd)

Drawing No.	Sheet No.	Revision No	Date	Prepared By	Title
B-327					CONTROL WIRING DIAGRAM (Cont'd)
	376S	R7	7/30/82	E	STEAM GENERATOR 2A LEVEL
	377S	R7	7/30/82	E	STEAM GENERATOR 2B LEVEL
	378S	R6	7/30/82	E	STEAM GENERATOR 2A PRESSURE
	379S	R7	11/10/82	E	STEAM GENERATOR 2B PRESSURE
	381S	R6	10/21/82	E	REACTOR COOLANT TEMP CH T-1112A, T-1122A
	382S	R6	10/21/82	E	REACTOR COOLANT TEMP CH T-1112B, T-1122B
	383S	R6	10/21/82	E	REACTOR COOLANT TEMP CH T-1112C, T-1122C
	384S	R6	10/21/82	E	REACTOR COOLANT TEMP CH T-1112D, T-1122D
	385S	R7	10/21/82	E	REACTOR COOLANT DELTA FLOW CH P-1101A
	386S	R7	10/21/82	E	REACTOR COOLANT DELTA FLOW CH P-1101B
	387S	R7	10/21/82	E	REACTOR COOLANT DELTA FLOW CH P-1101C
	388S	R7	10/21/82	E	REACTOR COOLANT DELTA FLOW CH P-1101D
	392S	R3	8/7/82	E	RTGB-204 120V AC & 125V DC DISTRIBUTION
	393S	R3	8/31/82	E	RTGB-203 28V DC DISTRIBUTION
	395S	R4	9/9/82	E	RTGB-203 120V AC DISTRIBUTION SH 2
	396S	R6(0)	7/25/83	E	RTGB-203 125V DC DISTRIBUTION
					REACTOR REGULATING SYSTEM
	411S	R7	9/28/82	E	REACTOR TRIP BKP TCR-1

TABLE 1.7-1 (Cont'd)

Drawing No.	Sheet No.	Revision No	Date	Prepared By	Title
B-327					CONTROL WIRING DIAGRAM (Cont'd)
	412S	R6	9/28/82	E	REACTOR TRIP BKR TCB-5
	413S	R7	9/28/82	E	REACTOR TRIP BKR TCB-2
	414S	R6	9/27/82	E	REACTOR TRIP BKR TCB-6
	415S	R8	9/27/82	E	REACTOR TRIP BKR TCB-3
	416S	R6	9/28/82	E	REACTOR TRIP BKR TCB-7
	417S	R7	9/27/82	E	REACTOR TRIP BKR TCB-4
	418S	R6	9/28/82	E	REACTOR TRIP BKR TCB-8
	419S	R8	11/10/82	E	REACTOR TRIP BKR TCB-9
	424S	R6	9/27/82	E	REACTOR TRIP SWGR & CEDMC'S 120V AC & 125V DC DISTR
					AREA & PROCESS RADIATION MONITORING
	332	R2	2/10/82	E	POST-ACCIDENT MONITORS DETECTOR NOS. RD-26-38, RD-26-39
	438S	R2	1/29/82	E	CONTAINMENT SAMPLING VALVES SH 1
	439S	R2	1/29/82	E	CONTAINMENT SAMPLING VALVES SH 2
	440S	R2	1/29/82	E	CONTAINMENT SAMPLING VALVES SH 3
	441S	R2	1/29/82	E	CONTAINMENT SAMPLING VALVES SH 4
	442S	R1	2/10/82	E	PROCESS RADIATION MONITOR
	443S	R6	2/28/83	E	CONTAINMENT HIGH RANGE RAD MONITORS
	444S	R6	4/13/83	E	COMPONENT COOLING WATER RADIATION MONITORING
	445S	R2	11/24/82	E	PLANT VENT STACK & FUEL HANDLING BLDG VENT STACK RAD MONITORING
	446S	R5	2/12/83	E	ECCS EFFLUENT GAS & PLANT VENT GAS WIDE RAD MONITORING

TABLE 1.7-1 (Cont'd)

Drawing No.	Sheet No.	Revision No	Date	Prepared By	Title
B-327					CONTROL WIRING DIAGRAM (Cont'd)
	447S	R0	11/13/81	E	CCS EFFLUNET GAS (VENT A) WIDE RANGE RAD MONITORS SH 2
	448S	R1	2/10/82	E	ECCS EFFLUENT GAS (VENT B) WIDE RANGE RAD MONITORS SH 1
	449S	R0	11/13/81	E	ECCS EFFLUENT GAS (VENT B) WIDE RANGE RAD MONITORS SH 2
	452S	R1	2/10/82	E	CONTROL ROOM OA1 (NORTH) RADIATION MONITORS
	453S	R1	2/10/82	E	CONTROL ROOM OA1 (SOUTH) RADIATION MONITORS
	455S	R0	12/31/81	E	FUEL POOL RAD MONITORING 2-OUT-OF-3 LOGIC SH 1
	456S	R0	12/31/81	E	FUEL POOL RAD MONITORING 2-OUT-OF-3 LOGIC SH 2
	457S	R4	2/26/82	E	CONTAINMENT RADIATION
	461S	R5	12/6/82	E	STEAM GEN. BLOWDOWN SAMPLE ISOL. VALVES & SNUBBER OIL RESERVOIR LEVEL
					HEATING & VENTILATING
	462S	R1	10/26/78	E	AUX BLDG & ECCS SYSTEM DAMPERS SH 1 OF 6
	463S	R2	9/10/82	E	AUX BLDG & ECCS SYSTEM DAMPERS SH 2 OF 6
	464S	R2	9/10/82	E	AUX BLDG & ECCS SYSTEM DAMPERS SH 3 OF 6
	465S	R7	3/29/83	E	AUX BLDG & ECCS SYSTEM DAMPERS SH 4 OF 6
	466S	R4	3/30/82	E	AUX BLDG & ECCS SYSTEM DAMPERS SH 5 OF 6
	467S	R4	3/30/82	E	AUX BLDG & ECCS SYSTEM DAMPERS SH 6 OF 6
	468S	R7	2/22/83	E	ELEC EQUIPMENT ROOM FANS

TABLE 1.7-1 (Cont'd)

Drawing No.	Sheet No.	Revision No	Date	Prepared By	Title
B-327					CONTROL WIRING DIAGRAM (Cont'd)
	476S	R13	3/1/83	E	ELEC EQUP. RM SUPPLY FAN 2HVS-5A
	477S	R11	2/22/83	E	ELEC EQUIP. RM SUPPLY FAN 2HVS -5B
	478S	R4	3/20/81	E	TEMP RECORDER TR-25-2A MISC T/C'S
	479S	R5	12/15/82	E	TEMP RECORDER TR-25-2B MISC T/C'S
	481S	R4	12/15/82	E	AIRBORNE RADIOACTIVITY & ECCS VENT SYSTEM
	482S	R9	3/11/83	E	REACTOR CONTAINMENT & SHLD BLDG DIFF. PRESS
	483S	R7	8/26/82	E	TEMP RECORDERS TR-25-1A MISC T/C'S
	487S	R6	12/20/82	E	CONTAINMENT TO ANNULUS & ECCS ROOM DIFF PRESS
	490S	R12	4/7/83	E	CONTROL ROOM EMERG. FILTRATION FAB 2HVE-13A
	491S	R12	4/7/83	E	CONTROL ROOMEMERG. FILTRATION FAB 2HVE-13B
	492S	R11	1/24/83	E	CONTROL ROOM AIR COND. UNIT 2-HVA/ACC-3A
	494S	R11	1/24/83	E	CONTROL ROOM AIR COND UNIT 2-HVA/ACC-3B
	496S	R9	1/24/83	E	CONTROL ROOM AIR COND UNIT 2-HVA/ACC-3C
	499S	R4	2/18/83	E	CONTROL ROOM FILTER & FAN INLET DAMPERS
	500S	R8	1/14/83	E	CONTROL ROOM O.A.I. RADIATION DETECTORS
	503S	R7	3/1/83	E	REACTOR AUX BLDG EMER EXH FAN 2HVE-9A

TABLE 1.7-1 (Cont'd)

Drawing No.	Sheet No.	Revision No	Revision Date	Prepared By	Title
B-327					CONTROL WIRING DIAGRAM (Cont'd)
	504S	R7	12/20/82	E	REACTOR AUX BLDG EMERG EXH FAN 2HVE-9B
	505S	R7	12/20/82	E	REACTOR AUX BLDG SUPPLY FAN 2HVS-4A
	506S	R7	12/20/82	E	REACTOR AUX BLDG SUPPLY FAN 2HVS-4B
	507S	R8	11/10/82	E	CEDM COOLING FAN 2HVE-21A
	508S	R9	12/22/82	E	CEDM COOLING FAN 2HVE-21B
	509S	R8	2/10/83	E	REACTOR CONTAINMENT PURGE EXHAUST FAN 2HVE-8A
	510S	R7	2/18/83	E	REACTOR CONTAINMENT PURGE EXHAUST FAN 2HVE-8B
	511S	R7	7/16/82	E	REACTOR CONTAINMENT PURGE ISOLATION VALVES SH 1
	512S	R7	6/17/82	E	REACTOR CONTAINMENT PURGE ISOLATION VALVES SH 2
	513S	R7	11/10/82	E	SHIELD BLDG VENT EXH FAN 2HVE-6A
	516S	R7	11/10/82	E	SHIELD BLDG VENT EXH FAN 2HVE-6B
	517S	R4	9/17/82	E	FUEL POOL DIFF PRESS & HSCP ROOM FANS
	518S	R2	3/20/81	E	DIESEL GEN 2A BLDG FAN 2-RV-5
	519S	R2	3/20/81	E	DIESEL GEN 2B BLDG FAN 2-RV-6
	522S	R6	3/1/83	E	REACTOR CAVITY COOLING SYSTEM 2HVS-2A
	523S	R7	11/10/82	E	REACTOR CAVITY COOLING SYSTEM 2HVS-2B
	524S	R5	3/1/83	E	REACTOR SUPPORT COOLING SYSTEM 2HVE-3A

TABLE 1.7-1 (Cont'd)

Drawing No.	Sheet No.	Revision No	Date	Prepared By	Title
B-327					CONTROL WIRING DIAGRAM (Cont'd)
	525S	R4	5/11/82	E	REACTOR SUPPORT COOLING SYSTEM 2HVE-3B
	529S	R6	1/24/83	E	CONTAINMENT VACUUM RELIEF VALVES FCV-25-7 & FCV-25-8
					WASTE MANAGEMENT & SAMPLING
	532S	R8	4/21/83	E	SAFEGUARDS ROOM "A" SUMP PUMPS
	533S	R9	4/21/83	E	SAFEGUARDS ROOM "B" SUMP PUMPS
	536S	R1	4/28/80	E	DRAIN VALVES TO REACTOR AUXILIARY BUILDING SUMPS - SH 1
	542S	R4	9/7/82	E	REACTOR DRAIN PUMP 2A
	543S	R5	9/10/82	E	REACTOR DRAIN PUMP 2B
	563S	R4	9/7/82	E	RDT VENT STOP & CONT ISOL VALVES V-6300, V-6341, & V-6342
	564S	R6	12/6/82	E	WASTE GAS CONT ISOL & STOP VALVES V-6718, V-6750, & V-6565
	566S	R5	9/7/82	E	N2 HDR CONT ISOL & DISCH STOP VALVES V-6741 & V-6728
	576S	R7	4/1/83	E	REACTOR SUMP ISOL VALVES LCV-07-11A & LCV-07-11B AND REACTOR CAVITY LEAK DETECTORS
	578S	R5	7/20/82	E	PRIMARY COOLANT SAMPLES VALVES V-5200 & V 5203
	579S	R3	4/3/81	E	PRESSURIZER SURGE SAMPLE VALVES V-5201 & V-5204
	580S	R4	8/21/82	E	PRESSURIZER STEAM SAMPLE VALVES V-5202 & V-5205
	586S	R1	4/28/80	E	DRAIN VALVES TO REACTOR TO AUX BLDG SUMPS - SH 2

TABLE 1.7-1 (Cont'd)

Drawing No.	Sheet No.	Revision No	Revision Date	Prepared By	Title
B-327					CONTROL WIRING DIAGRAM (Cont'd)
					FEEDWATER
	601S	R4	7/2/82	E	AUX FW HDR'S A&B FLOW & PRESSURE
	602S	R7	2/28/83	E	AUX FW HDR C FLOW & PRESSURE & FWP 2A & 2B FLOW
	603S	R9	6/24/83	E	STM GEN 2A & 2B ATM STM DUMP FWP DISCH HDR PRESS SH 1
	608S	R8(0)	7/25/83	E	AUX FWP 2A DISCHARGE TO ST. GEN 2A MV-09-9
	609S	R9(0)	7/25/83	E	AUX FWP 2B DISCHARGE TO ST. GEN 2B MV-09-10
	610S	R9	12/11/82	E	AUX FWP 2A DISCHARGE TO ST. GEN 2B MV-09-13
	611S	R10	1/14/83	E	AUX FWP 2B DISCHARGE TO ST. GEN 2A MV-09-14
	612S	R8(0)	7/25/83	E	AUX FWP 2C DISCHARGE TO ST. GEN 2A MV-09-11
	613S	R7(0)	7/25/83	E	AUX FWP 2C DISCHARGE TO ST. GEN 2A MV-09-12
	629S	R8	3/1/83	E	AUX FEEDWATER PUMP 2A
	630S	R8	5/26/83	E	AUX FEEDWATER PUMP 2B
	631S	R7	5/26/83	E	AUX FEEDWATER PUMP 2C - TURBINE
	632S	R8	5/26/83	E	AUX FEEDWATER PUMP 2C - STEAM VALVE MV-08-3

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TABLE 1.7-1 (Cont'd)

Drawing No.	Sheet No.	Revision No	Date	Prepared By	Title
B-327					CONTROL WIRING DIAGRAM (Cont'd)
	638S	R6	3/30/82	E	SG 2A/2B TO AFWP 2C WARM-UP VALVES I-SE-08-1, 2
	639S	R4	8/24/82	E	RTGB-202 45VDC DISTRIBUTION
	643S	R5	8/5/82	E	RTGB-202 120VAC DISTRIBUTION SH.1
	645S	R5	8/26/82	E	RTGB-205 125VDC&120VAC DISTR.
	646S	R3	8/21/82	E	RTGB-206 125V DC DISTRIBUTION
	647S	R6	9/17/82	E	RTGB-206 120V AC DISTRIBUTION SH.1
	648S	R2	3/6/81	E	RTGB-206 120V AC DISTRIBUTION SH.2
	649S	R4	8/5/82	E	HOT SHUTDOWN CONTROL PANEL 120VAC DISTRIBUTION
	652S	R8	5/26/83	E	SG 2A TO AFWP 2C TURBINE MV-08-13
	653S	R7	1/17/83	E	SG 2B TO AFWP 2C TURBINE MV-08-12
	654S	R6	6/24/83	E	STM GEN 2A&2B ATM STM DUMP FWP DISCH HDR PRESS SH.2
	655S	R4	2/14/83	E	MAIN FEEDWATER ISOLATION VALVE HCV-09-1A
	656S	R4	2/14/83	E	MAIN FEEDWATER ISOLATION VALVE HCV-09-113
	657S	R5	4/1/83	E	RTGB 205, 125VDC&120VAC DISTRIBUTION SH.2
	658S	R3	9/2/82	E	RTGB 205, 120VAC DISTRIBUTION
	664S	R4	8/11/82	E	RTGB 206, 45VDC DISTRIBUTION
	671S	R5	5/11/83	E	MAIN FEEDWATER ISOLATION VALVE HCV-09-2A
	672S	R6	5/11/83	E	MAIN FEEDWATER ISOLATION VALVE HCV-09-2B

TABLE 1.7-1 (Cont'd)

Drawing No.	Sheet No.	Revision No	Revision Date	Prepared By	Title
B-327					CONTROL WIRING DIAGRAM (Cont'd) MAIN STEAM
	695S	R7	8/15/82	E	AUX FWP 2C & TURB INLET PRESS & STM GEN FLOW PRESSURE
					TURBINE
	709S	R3	1/18/82	E	TURBINE TRIP STEAM GEN. HIGH-HIGH LEVEL
	710S	R11	1/7/83	E	TURBINE AUTO-STOP-TRIP & TURBINE ALARMS
	743S	R8	9/7/82	E	CONDENSATE TRANSFER PUMP
	744S	R7	2/12/83	E	AUXILIARY STEAM
					SJAE STM&FEED PUMP SUCT HDR PRESS COND STM TK & HOTWELL LEVEL
					TURBINE INSTRUMENTATION
	800S	R5	6/1/81	E	RTGB-201, 125VDC&120VAC DISTRIBUTION
					TURBINE COOLING
	831S	R6	9/3/82	E	INTAKE COOL WTR DISCH HDR PRESS PUMP 2A & PUMP 2B
	832S	R7	7/18/83	E	INTAKE COOLING WATER PUMP 2A
	833S	R7	7/18/83	E	INTAKE COOLING WATER PUMP 2B
	834S	R6	7/18/83	E	INTAKE COOLING WATER PUMP 2C
	835S	R5	10/21/82	E	INTAKE COOL WTR NON EMER HDR A ISOL VALVE MV-21-3
	836S	R5	8/19/82	E	INTAKE COOL WTR NON EMER HDR B ISOL VALVE MV-21-2
	839S	R5	4/28/82	E	LUBE WATER SUPPLY STRAINERS

TABLE 1.7-1 (Cont'd)

Drawing No.	Sheet No.	Revision No	Revision Date	Prepared By	Title
B-327					CONTROL WIRING DIAGRAM (Cont'd) PRIMARY WATER
	849S	R4	12/17/82	E	PRIMARY WATER ISOLATION VALVE HCV-15-1
					STATION AUXILIARY POWER
	924S	R2	1/18/82	E	4160V SWGR 2A3 DIFF. RELAY
	925S	R2	10/6/82	E	4160V SWGR 2B3 DIFF. RELAY
	926S	R3	6/9/81	E	4160V SWGR 2AB DIFF. RELAY
	931S	R4	6/9/81	E	4160V SWGR 2A3 AC-DC DISTR & HEATERS
	932S	R4	6/9/81	E	4160 SWGR 2B3 AC-DC DISTR & HEATERS
	933S	R4	6/9/81	E	4160V SWGR 2AB AC-DC DISTR & HEATERS
	934S	R6	1/26/83	E	4160V SWGR 2A2 FDR TO BUS 2A3
	935S	R6	1/26/83	E	4160V SWGR 2B2 FDR TO BUS 2B3
	936S	R7	3/2/83	E	4160V SWGR 2A3 INCOMING FDR. FROM BUS 2A2
	937S	R6	3/2/83	E	4160V SWGR 2B3 INCOMING FEEDER FROM BUS 2B2
	938S	R4	1/26/83	E	4160V SWGR 2A3 FDR TO BUS 2AB
	939S	R4	1/26/83	E	4160V SWGR 2B3 FDR TO BUS 2AB
	940S	R5	1/26/83	E	4160V SWGR 2AB INCOMING FDR FROM BUS 2A3
	941S	R6	4/13/83	E	4160V SWGR 2AB INCOMING FDR FROM BUS 2B3
	942S	R2	9/7/82	E	4160V SWGR 2AB RELAYING & METERING
	943S	R8	6/24/83	E	PRESS HTR TRANSF 2A3 4160V FDR BKR

TABLE 1.7-1 (Cont'd)

Drawing No.	Sheet No.	Revision No	Revision Date	Prepared By	Title
B-327					CONTROL WIRING DIAGRAM (Cont'd)
	944S	R8	6/24/83	E	PRESS HTR TRANSF 2B3 4160V
	946S	R5	4/13/83	E	480V STA SERV TRANSF 2A2 4160V FDR BKR
	948S	R4	9/1/82	E	480V STA SERV TRANSF 2B2 4160V FDR BKR
	949S	R8	12/16/82	E	4160V SWGR 2A3 LOAD SHEDDING RELAYS
	950S	R5	12/16/82	E	4160V SWGR 2B3 LOAD SHEDDING RELAYS
	951S	R4	1/11/82	E	4160V SWGR 2AB LOAD SHEDDING RELAYS
					EMERGENCY DIESEL GENERATOR
	953S	R4	12/16/82	E	DIESEL GENERATOR 2A BREAKER
	954S	R7	12/16/82	E	DIESEL GENERATOR 2A RELAYING & METERING
	955S	R6	12/16/82	E	DIESEL GENERATOR 2A INSTR. & DIFF RELAYING
	956S	R5	6/24/83	E	DIESEL GENERATOR 2A LOCKOUT RELAY
	957S	R8	3/2/83	E	DIESEL GENERATOR 2A START CKT'S - SH.1
	958S	R5	3/2/83	E	DIESEL GENERATOR 2A REMOTE CONTROL
	959S	R1	1/28/80	E	DIESEL GENERATOR 2A START SOLENOIDS
	960S	R1	9/7/82	E	DIESEL GENERATOR 2A ANNUNCIA-TOR SH.1
	961S	R2	12/10/82	E	DIESEL GENERATOR 2A ANNUNCIA-TOR SH.2
	962S	R2	9/16/81	E	D-G 2A ENG CYL'S TEMP & EXH DIFF TEMP MONITORING

TABLE 1.7-1 (Cont'd)

Drawing No.	Sheet No.	Revision No	Revision Date	Prepared By	Title
B-327					CONTROL WIRING DIAGRAM (Cont'd)
	963S	R5	12/16/82	E	DIESEL GENERATOR 2B BREAKER
	964S	R8	12/16/82	E	DIESEL GENERATOR 2B RELAYING & METERING
	965S	R6	12/16/82	E	DIESEL GENERATOR 2B INSTR & DIFF RELAYING
	966S	R6	6/24/83	E	DIESEL GENERATOR 2B LOCKOUT RELAYS
	967S	R8	3/2/83	E	DIESEL GENERATOR 2B START CKT'S - SH.1
	968S	R5	3/2/83	E	DIESEL GENERATOR 2B REMOTE CONTROL
	969S	R2	1/8/82	E	DIESEL GENERATOR 2B START SOLENOIDS
	970S	R2	9/7/82	E	DIESEL GENERATOR 2B ANNUNCIA-TORS-SH.1
	971S	R2	12/10/82	E	DIESEL GENERATOR 2B ANNUNCIA-TORS-SH.2
	972S	R3	9/16/81	E	D-G 2B ENG CYL'S TEMP & EXH DIFF TEMP MONITORING
	974S	R2	4/16/82	E	D-G 2A ENG CYC'S TEMP EXH DIFF TEMP MONITORING
					480V AUXILIARY POWER
	977S	R3	9/7/82	E	480V SWGR 2A2 FDR
	978S	R4	8/3/82	E	480V SWGR 2A2 - 2AB TIE
	979S	R3	8/3/82	E	480V SWGR 2AB - 2A2 TIE
	980S	R2	1/24/83	E	480V SWGR 2B2 FDR
	981S	R3	8/3/82	E	480V SWGR 2B2-2AB TIE
	982S	R4	8/3/82	E	480V SWGR 2AB-2B2 TIE
	983S	R2	4/16/82	E	480V SWGR SPARE COMPARTMENTS
	984S	R3	3/30/82	E	480V SWGR 2A2 FDR TO FUEL HANDLING MCC 2A8

TABLE 1.7-1 (Cont'd)

Drawing No.	Sheet No.	Revision No	Revision Date	Prepared By	Title
B-327					CONTROL WIRING DIAGRAM (Cont'd)
	985S	R3	4/16/82	E	480V SWGR 2B2 FDR TO FUEL HANDLING MCC 2B8
	990S	R12	2/1/83	E	480V SWGR 2A2 MET REL & HTR'S
	991S	R4	8/25/82	E	480V SWGR 2A2 MCC FEEDERS
	992S	R12	2/1/83	E	480V SWGR 2B2 MET REL & HTR'S
	993S	R5	8/25/82	E	480V SWGR 2B2 MCC FEEDERS
	994S	R6	3/11/83	E	480V SWGR 2AV MEG. REL & HTR'S
	995S	R2	2/20/81	E	480V SWGR 2AB FEEDER TO REACTOR AREA MCC 2AB
	996S	R6	3/26/82	E	EMERGENCY DIESEL GEN NO. 2A LOADING LIGHTS
	997S	R8	3/25/83	E	EMERGENCY DIESEL GEN. NO. 2B LOADING LIGHTS
	998S	R6	9/27/82	E	EMERGENCY DIESEL GEN'S NO. 2A & NO. 2B LOADING LIGHTS
					MISCELLANEOUS ELECTRICAL
	999S	R7	1/10/83	E	BATTERY 2C & BATTERY CHARGER 2C
	1000S	R2	10/17/80	E	125 VDC BUS TRANSFER CONTROL
	1001S	R9	4/18/83	E	BATTERY 2A BATTERY CHARGER 2A
	1002S	R8	4/14/83	E	BATTERY 2B BATTERY CHARGER 2B
	1003S	R6	12/20/82	E	BATTERY CHARGER 2AB
	1004S	R4	4/20/83	E	ISOL CAB'S 125V DC POWER SUPPLY
	1005S	R4	9/1/82	E	MOTOR SPACE HEATER FEEDERS
	1006S	R3	3/1/83	E	MOTOR SPACE HEATER FEEDERS
	1007S	R6	4/28/82	E	MISC ANNUNCIATIONS
	1008S	R7	3/11/83	E	VITAL AC BUS POWER SUPPLY (SUPS)

TABLE 1.7-1 (Cont'd)

Drawing No.	Sheet No.	Revision No	Revision Date	Prepared By	Title
B-327					CONTROL WIRING DIAGRAM (Cont'd)
	1009S	R11	5/26/83	E	INSTRUMENT BUSES & INVERTERS 2MA & 2 MC
	1010S	R11	5/26/83	E	INSTRUMENT BUSES & INVERTERS 2MB & 2MD
	1024S	R1	11/10/82	E	MCC 2A1,2B1,2A3,2B3,2A9,2B9, SP 2HTRS
	1026S	R(0)	7/25/83	E	MCC 2A5,2B5,2AB,2A7,2B7 SP HTRS
	1027S	R2	11/10/82	E	MCC 2A6,2B6,2AB,2B SP HTRS
					EMERGENCY DIESEL GENERATOR
	1117S	R2	10/6/82	E	DIESEL GENERATOR 2A ANN CKT'S - SH 1
	1118S	R3	11/10/82	E	DIESEL GENERATOR 2A ANN CKT'S - SH 2
	1119S	R7	6/24/83	E	DIESEL GENERATOR 2A ANN CKT'S - SH 3
	1120S	R3	8/26/82	E	DIESEL GENERATOR 2A LUBE OIL CIRC. PUMP 2A1
	1121S	R3	8/26/82	E	DIESEL GENERATOR 2A LUBE OIL CIRC. PUMP 2A2
	1126S	R6	4/5/83	E	DIESEL GEN FUEL OIL TRANSFER PUMP 2A
	1127S	R2	1/8/82	E	DIESEL GENERATOR 2B ANN CKT'S SH 1
	1128S	R4	11/10/82	E	DIESEL GENERATOR 2B ANN CKT'S SH 2
	1129S	R6	6/24/83	E	DIESEL GENERATOR 2B ANN CKT'S SH 3
	1130S	R4	8/26/82	E	DIESEL GENERATOR 2B LUBE OIL CIRC. PUMP 2B1
	1131S	R4	8/26/82	E	DIESEL GENERATOR 2B LUBE OIL CIRC. PUMP 2B2

TABLE 1.7-1 (Cont'd)

Drawing No.	Sheet No.	Revision No	Revision Date	Prepared By	Title
B-327					CONTROL WIRING DIAGRAM (Cont'd)
					MISCELLANEOUS HVAC
	1136S	R5	4/5/83	E	DIESEL GEN FUEL OIL TRANSFER PUMP 2B
	1137S	R3	4/7/82	E	TEMPERATURE RECORDER TR-25-1B MISC THERMOCOUPLES
	1138S	R5	12/15/82	E	HYDRAMOTOR ACTUATORS FOR FANS 2HVE-9A & 2HVE-9B
	1139S	R3	1/17/83	E	HYDRAMOTOR ACTUATORS FOR FANS 2HVE-13A & 2HVE-13B
	1140S	R4	11/11/82	E	SHIELD BLDG VENT SYS D-23 DAMPER CONTROL
	1141S	R4	11/11/82	E	SHIELD BLDG VENT SYS D-24 DAMPER CONTROL
	1142S	R6	4/21/83	E	PLANT AUXILIARIES CONTROL BOARD, ANNUNCIATOR-LA
	1143S	R6	4/21/83	E	PLANT AUXILIARIES CONTROL BOARD ANNUNCIATOR-LB
	1149S	R4	3/6/82	E	PLANT AUX. CONTROL BOARD ANN. LA, LB INTER. WIRING
	1150S	R7	3/11/83	E	SHIELD BLDG VENT SYSTEM ELECTRIC HEATING COILS 2-HVE-6A1, 6A2
	1152S	R7	3/11/83	E	SHIELD BLDG VENT SYSTEM ELECTRIC HEATING COILS 2-HVE-6B1, 6B2
	1154S	R5	12/11/82	E	FUEL HANDLING BLDG EMERG. VENT VALVE FCV-25-30
	1155S	R4	12/11/82	E	FUEL HANDLING BLDG EMERG. VENT VALVE FCV-25-31
	1156S	R4	12/11/82	E	SHIELD BLDG VENT SYSTEM ISOL VALVE FCV-25-32
	1157S	R4	12/11/82	E	SHIELD BLDG VENT SYSTEM ISOL VALVE FCV-25-33

TABLE 1.7-1 (Cont'd)

Drawing No.	Sheet No.	Revision No	Revision Date	Prepared By	Title
B-327					CONTROL WIRING DIAGRAM (Cont'd)
	1158S	R6	4/5/80	E	CONT. CONTAIN./H2 PU DISCH. TO SHIELD BLDG. VENT SYS. FCV-25-29
	1159S	R5	10/21/82	E	CONT. CONTAIN./H2 PU DISCH. TO SHIELD BLDG. VENT SYS. FCV-25-34
	1160S	R6	12/17/82	E	CONT. CONTAIN./H2 PURGE ISOL VALVE FCV-25-20
	1161S	R5	12/17/82	E	CONT. CONTAIN./H2 PURGE ISOL. VALVE FCV-25-21
	1162S	R5	12/20/82	E	INTAKE STRUCTURE EXHAUST FAN 2HVE-41A
	1163S	R5	12/20/82	E	INTAKE STRUCTURE EXHAUST FAN 2 HVE-41B
	1164S	R4	7/20/82	E	CONT. CONTAIN./H2 PURGE ISOL. VALVE FCV-25-26
	1165S	R4	12/6/82	E	SHIELD BLDG. HEPA FILTERS & CHARCOAL ADSORBER DIFF PRESS
	1166S	R6	1/25/83	E	CONTROL ROOM DAMPERS D-39 & D-40 & DIFF PRESSURES
	1167S	R3	4/3/81	E	CONTROL ROOM HEPA FILTER DIFF PRESSURES
	1168S	R3	7/1/82	E	FUEL HDLG BLDG HEATING & VENT RM FAN 2HVE-17
	1169S	R9	12/20/82	E	ROOF VENTILATORS 2RV3 & 2RV4
	1170S	R6	10/14/82	E	CONTROL ROOM NORTH OAI ISOL VA FCV-25-14
	1171S	R5	10/14/82	E	CONTROL ROOM SOUTH OAI ISOL VA FCV-25-15
	1172S	R7	10/14/82	E	CONTROL ROOM NORTH OAI ISOL VA FCV-25-16
	1173S	R5	9/27/82	E	CONTROL ROOM SOUTH OAI ISOL VA FCV-25-17

TABLE 1.7-1 (Cont'd)

Drawing No.	Sheet No.	Revision No	Revision Date	Prepared By	Title
B-327					CONTROL WIRING DIAGRAM (Cont'd)
	1174S	R3	1/29/82	E	TOILET EXH FAN ISOL VA FCV-25-18
	1175S	R4	12/20/82	E	TOILET EXH FAN ISOL VA FCV-25-19
	1176S	R6	9/27/82	E	SHIELD BLDG VENT COOL AIR VALVE FCV-25-11
	1177S	R4	9/27/82	E	SHEILD BLDG VENT COOL AIR VALVE FCV-25-12
	1178S	R6	9/27/82	E	SHIELD BLDG VENT SYSTEM TIE VALVE FCV-25-13
	1182S	R3	4/6/82	E	FUEL HANDLING BLDG DAMPERS RAD SIGNAL A
	1183S	R3	4/6/82	E	FUEL HANDLING BLDG DAMPERS RAD SIGNAL B
	1189S	R1	4/6/82		
	1190S	R2	2/20/81	E	KITCHEN EXHAUST FAN ISOL VALVE FCV-25-24
	1191S	R2	2/20/81	E	KITCHEN EXHAUST FAN ISOL VALVE FCV-25-25
	1192S	R0	1/29/82		
	1196S	R4	2/28/83	E	CONTAINMENT ATMOSPHERE HYDROGEN ANALYZER SH.1
	1197S	R4	7/26/82	E	CONTAINMENT ATMOSPHERE HYDROGEN ANALYZER-2
	1204S	R2	7/26/82	E	CONTAINMENT ATMOSPHERE HYDROGEN ANALYZER-3
	1205S	R4	2/28/83	E	CONTAINMENT ATMOSPHERE HYDROGEN ANALYZER-4
	1217S	R3	7/27/81	E	ANNUNCIATOR REFLASH
	1219S	R4	12/20/82	E	BATTERY ROOM 2A ROOF VENTILA-TOR-2RV-1
	1220S	R4	12/20/82	E	BATTERY ROOM 2B ROOF VENTILA-TOR-2RV-2

TABLE 1.7-1 (Cont'd)

Drawing No.	Sheet No.	Revision No	Revision Date	Prepared By	Title
B-327					CONTROL WIRING DIAGRAM (Cont'd)
	1238S	R2	3/20/81	E	HVCB-45V DC DISTRIBUTION
	1239S	R3	4/6/82	E	HVCB-125V DC & 120V AC DISTRIBUTION SH 1
	1240S	R3	10/7/80	E	HVCB-120V AC DISTRIBUTION SH 2
	1253S	R3	9/2/82	E	MOTOR OPER VALVE SPACE HEATERS FEEDERS
	1254S	R3	10/6/82	E	MOTOR OPER VALVE SPACE HEATERS FEEDERS
	1255S	R6	7/26/82	E	MOTOR OPER VALVE SPACE HEATERS FEEDERS
	1256S	R4	7/26/82	E	MOTOR OPER VALVE SPACE HEATERS FEEDERS
	1257S	R4	7/26/82	E	MOTOR OPER VALVE SPACE HEATERS FEEDERS
	1260S	R3	12/1/81	E	MOTOR OPER VALVE SPACE HEATERS FEEDERS
	1276S	R1	3/28/80	E	480V SWGR SPACE COMPARTMENT
	1278S	R2 (0)	7/26/83	E	480V SWGR 2A2, 2B2 INTER-CONNECTIONS BETWEEN CUBICLES
	1279S	R4	8/25/81	E	480V SWGR 2AB INTERCONNEC-TIONS BETWEEN WIRING BOXES
					SAFETY INJECTION & SHUTDOWN COOLING
	1501S	R10	11/11/82	E	SHUTDOWN COOLING ISOL. VALVE V-3545
	1502S	R7	1/12/83	E	SHUTDOWN COOLING ISOL. VALVE V-3664
	1503S	R7	9/27/82	E	SHUTDOWN COOLING ISOL. VALVE V-3665
	1504S	R6	9/27/82	E	SHUTDOWN CLG FROM.HEAT EXCH 2A VALVE V-3456

TABLE 1.7-1 (Cont'd)

Drawing No.	Sheet No.	Revision No	Revision Date	Prepared By	Title
B-327					CONTROL WIRING DIAGRAM (Cont'd)
	1505S	R7	2/14/83	E	SHUTDOWN CLG FROM HEAT EXCH. - 2B VALVE V-3457
	1506S	R4	9/27/82	E	SHUTDOWN CLG HEAT EXCH. - 2A INLET VALVE V-3517
	1507S	R4	9/27/82	E	SHUTDOWN CLG HEAT EXCH.- 2B INLET VALVE V-3658
	1508S	R2	8/05/82	E	RECORDER DISTRIBUTION MODULE INTERCONNECTION SH 1
	1510S	R6	9/29/82	E	SHUTDN. CLG.LINE 2A WARM-UP VALVE V-3536
	1511S	R9	9/29/82	E	SHUTDN. CLG. LINE 2B WARM-UP VALVE V-3539
	1512S	R7	7/18/83	E	HP INJECTION TO HOT LOOP 2A FLOW & PRESS MONITORS
	1513S	R5	8/05/82	E	HP INJECTION TO HOT LOOP 2B FLOW & PRESS MONITORS
	1514S	R6	9/29/82	E	SHUTDN. COOLING CONTROL VALVE 2A HCV-3657
	1515S	R7	9/29/82	E	SHUTDN. COOLING CONTROL VALVE 2B HCV-3512
	1516S	R8(0)	7/26/82	E	SHUTDOWN COOLING & BYPASS VALVE FCV-3306
	1517S	R8(0)	7/26/82	E	SHUTDOWN COOLING & BYPASS VALVE FCV-3301
	1518S	R4	4/13/83	E	RECORDER DISTRIBUTION MODULE INTERCONNECTION-SH 2
	1519S	R7	8/21/82	E	HOT LEG HPSI LINE CHECK VLV LEAK'G DRAIN VA'S V-3571, V-3572, I-SE-03-2A, I-SE-03-2B
	1520S	R6	7/18/83	E	MINIMUM FLOW ISOLATION VALVES V-3495 & V-3496

TABLE 1.7-1 (Cont'd)

Drawing No.	Sheet No.	Revision No	Revision Date	Prepared By	Title
B-327					CONTROL WIRING DIAGRAM (Cont'd)
	1525S	R3	1/13/82	E	MEASUREMENT CHANNELS T-3351 X/Y, 3352 X/Y, 3303 W/X/Y/Z
	1526S	R6	3/10/82	E	MEASUREMENT CHANNELS P3301, P3302, P-3304, P-3307
	1527S	R4	10/14/82	E	SI TANKS 2A1, 2A2, 2B1, 2B2 SAMPLE VA'S I-SE-05-1A, 1B, 1C & 1D
	1528S	R11	1/13/83	E	SI TANKS SAMPLE FCV-03-1E MEASUREMENT CH'S F-3301, F-3306
	1529S	R2	2/26/82	E	CONTAINMENT SPRAY ISOLATION VALVE MV-07-161
	1530S	R2	2/26/82	E	CONTAINMENT SPRAY ISOLATION VALVE MV-07-164
	1531S	R2	2/26/82	E	LPSI PUMP 2A SUCTION VALVE V-3432
	1532S	R2	2/26/82	E	LPSI PUMP 2B SUCTION VALVE V-3444
					ANNUNCIATORS
	1551S	R5	10/14/82	E	ISOL CAB/ALC-1 INTERCONN DIAGRAM RTGB-201 ANN B SH 1
	1552S	R6	8/27/82	E	ISOL CAB/ALC-1 INTERCONN DISGRAM RTGB-201 ANN B SH 2
	1553S	R6	11/5/82	E	ISOL CAB/ALC-1 INTERCONN DIAGRAM RTGB-201 ANN A SH 1
	1554S	R6	11/5/82	E	ISOL CAB/ALC-1 ,2 INTERCONN DIAGRAM RTGB-201, 204 ANN A SH 2, L
	1555S	R6	11/5/82	E	ISOL CAB/ALC-2 INTERCONN DIAGRAM RTGB-202 ANN G
	1556S	R6(0)	7/26/82	E	ISOL CAB/ALC-2 INTERCONN DIAGRAM RTGB-202 ANN E

TABLE 1.7-1 (Cont'd)

Drawing No.	Sheet No.	Revision No	Revision Date	Prepared By	Title
B-327					CONTROL WIRING DIAGRAM (Cont'd)
	1557S	R5	9/10/82	E	ISOL CAB/ALC-3 INTERCONN DIAGRAM RTGB-205 ANN M
	1558S	R7	9/2/82	E	ISOL CAB/ALC-3 INTERCONN DIAGRAM RTGB-205 ANN N
	1559S	R7	5/23/83	E	ISOL CAB/ALC-3 INTERCONN DIAGRAM RTGB-206 ANN S SH 1
	1560S	R4	6/9/81	E	ISOL CAB/ALC-3 INTERCONN DIAGRAM RTGB-206 ANN S SH2
	1561S	R5	6/9/81	E	ISOL CAB/ALC-2, 3 INTERCONN DIAGRAM RTGB-205, 206, ANN B, SH 3
	1563S	R3	3/6/81	E	ISOL CAB/ALC-3 INTERCONN DIAGRAM RTGB-206 ANN R SH 2
	1564S	R4	10/21/82	E	ISOL CAB/ALC-3 INTERCONN DIAGRAM RTGB-206 ANN R SH 3
	1565S	R5	10/14/82	E	ISOL CAB/ALC-3 INTERCONN DIAGRAM RTGB-206 ANN Q SH 1
	1566S	R4	8/5/82	E	ISOL CAB/ALC-3 INTERCONN DIAGRAM RTGB-206 ANN Q SH 2
	1567S	R6	2/23/83	E	ISOL CAB/ALC-2, 3 INTERCONN DIAGRAM RTGB-204, 206, ANN K, Q, SH 3
	1568S	R7	12/15/82	E	ISOL CAB/ALC-3 INTERCONN DIAGRAM RTGB-206 ANN P SH 1
	1569S	R4	10/11/82	E	ISOL CAB/ALC-3 INTERCONN DIAGRAM RTGB-206 ANN P SH 2
	1570S	R4	9/9/82	E	ISOL CAB/ALC-3 INTERCONN DIAGRAM RTGB-206 ANN P SH 3
	1571S	R5	3/6/81	E	ISOL CAB/ALC-1 INTERCONN DIAGRAM HVCB ANN "T"
	1572S	R4	3/6/81	E	ISOL CAB/ALC-1 INTERCONN DIAGRAM HVCB ANN "U"

TABLE 1.7-1 (Cont'd)

Drawing No.	Sheet No.	Revision No	Revision Date	Prepared By	Title
B-327					CONTROL WIRING DIAGRAM (Cont'd)
	1573S	R6	12/20/82	E	ISOL CAB/ALC-1 INTERCONN DIAGRAM HVCB ANN "V" SH 1
	1574S	R5	1/24/83	E	ISOL CAB/ALC-1 INTERCONN DIAGRAM HVCB ANN "V" SH 2
	1575S	R4	3/6/81	E	ISOL CAB/ALC-1 INTERCONN DIAGRAM HVCB ANN "W"
	1576S	R7	2/22/83	E	ISOL CAB/ALC-1 INTERCONN DIAGRAM HVCB ANN "X"
	1577S	R5	8/30/82	E	ISOL CAB/SEQ-OF EVENTS CAB INTERCONN DIAGRAM SH 1
	1578S	R4	6/14/82	E	ISOL CAB/SEQ OF EVENTS CAB INTERCONN DIAGRAM SH 2
	1580S	R1	7/16/82	E	ESC/ISOL CAB/ALC-3 INTER-WIRING
	1583S	R5	4/6/82	E	BYPASS INDICATION SYSTEM A SH.2
	1584S	R5	4/6/82	E	BYPASS INDICATION SYSTEM A SH.3
	1587S	R4	5/22/81	E	BYPASS INDICATION SYSTEM B SH.2
	1588S	R4	4/6/82	E	BYPASS INDICATION SYSTEM B SH.3
					EMERGENCY DIESEL GENERATORS
	1601S	R6	6/24/83	E	DIESEL GEN. 2A START CKT'S SH. 2
	1602S	R5	6/24/83	E	DIESEL GEN. 2A START CKT'S SH. 3
	1603S	R1	1/28/80	E	DIESEL GEN. 2A START CKT'S SH. 4
	1604S	R2	10/11/82	E	DIESEL GEN. 2A START CKT'S SH. 5

TABLE 1.7-1 (Cont'd)

Drawing No.	Sheet No.	Revision No	Revision Date	Prepared By	Title
B-327					CONTROL WIRING DIAGRAM (Cont'd)
	1605S	R2	9/17/82	E	DIESEL GEN. 2A START CRT'S SH. 6
	1606S	R5	10/21/82	E	DIESEL GEN 2A GROUNDING & METERING
	1607S	R4	1/11/82	E	DIESEL GEN 2A IMMERSION HEATERS
	1608S	R6	3/11/83	E	DIESEL GEN 2A VOLTAGE
	1609S	R0	7/31/81	E	REGULATOR
	1611S	R6	6/24/83	E	DIESEL GEN. 2B START CSTS SH. 2
	1612S	R5	6/24/83	E	DIESEL GEN. 2B START CKTS SH. 3
	1613S	R1	1/28/80	E	DIESEL GEN 2B START CKTS SH. 4
	1614S	R2	1/13/82	E	DIESEL GEN 2B START CKTS SH. 5
	1615S	R3	9/17/82	E	DIESEL GEN 2B START CKTS SH. 6
	1616S	R6	10/21/82	E	DIESEL GEN 2B GROUNDING & METERING
	1617S	R4	1/13/82	E	DIESEL GEN 2B IMMERSION HEATERS
	1618S	R7	3/11/83	E	DIESEL GEN 2B VOLTAGE
	1619S	R0	7/31/81	E	REGULATOR
	1621S	R3	1/17/83	E	ATMOS STM DUMP ISOL VA MV-08-15
	1622S	R4	1/17/83	E	ATMOS STM DUMP ISOL VA MV-08-14

TABLE 1.7-1 (Cont'd)

Drawing No.	Sheet No.	Revision No.	Date	Prepared By	Title
B-327					CONTROL WIRING DIAGRAM (Cont'd)
	1623S	R3	1/17/83	E	ATMOS STM DUMP ISOL VA MV-08-17
	1624S	R3	1/17/83	E	ATMOS STM DUMP ISOL VA MV-08-18
	1625S	R3	1/10/83	E	STM GEN 2A ATMOS STM DUMP VA MV-08-19A
	1626S	R3	1/10/83	E	STM GEN 2A ATMOS STM DUMP VA MV-08-18A
	1627S	R4	1/10/83	E	STM GEN 2A ATMOS STM DUMP VA MV-08-19B
	1628S	R3	1/10/83	E	STM GEN 2A ATMOS STM DUMP VA MV-08-18B
	1629	R2	1/10/83	E	RELIEF VALVE V-1474
	1630	R2	1/10/83	E	RELIEF VALVE V-1475
	1631	R5	7/18/83	E	AFWP-2A DISCH TO SG-2A I-SE-09-2
	1632	R6	7/18/83	E	AFWP-2B DISCH TO SG-2B I-SE-09-3
	1633	R6	7/18/83	E	AFWP-2C DISCH TO SG-2A I-SE-09-4
	1634	R5	7/18/83	E	AUX FW PUMP 2C DISCH TO STEAM GEN 2B I-SE-09-5
	1635	R4	2/12/83	E	FEEDWATER, HEADER PRESS 9B-9C-9D-10A-10B-10C-10D
	1636	R3	12/20/82	E	STEAM GEN 2A & 2B LEVEL/ PRESSURE
	1637	R2	7/18/83	E	REMOTE MANUAL INITIATE AFAS-1, AFAS-2
	1638	R3	2/18/83	E	AFAS ANNUNCIATORS SH 1
	1639	R3	7/18/83	E	AFAS ANNUNCIATORS SH 2

TABLE 1.7-1 (Cont'd)

Drawing No.	Sheet No.	Revision No.	Date	Prepared By	Title
B-327					CONTROL WIRING DIAGRAM (Cont'd)
	1641S	R4	11/21/83	E	RADIATION MONITORING 120V AC DISTRIBUTION
	1642	R5	1/10/83	E	120V AC DISTRIBUTION SH 3
	1643	R9	3/11/83	E	120V AC DISTRIBUTION SH 4
	1648	R1	2/26/82	E	LOOP NO. 2 SH 1
	1649	R1	2/26/82	E	LOOP NO. 2 SH 2
	1650	R1	1/29/82	E	LOOP NO. 2 SH 3
	1653	R2	11/24/82	E	LOOP NO. 3 SH 1
	1654	R2	11/15/83	E	LOOP NO. 3 SH 2
	1655	R2	7/18/83	E	LOOP NO. 3 SH 3
	1656	R2	11/24/82	E	LOOP NO. 3 SH 4
	1657	R2	11/24/82	E	LOOP NO. 3 SH 5
	1658	R1	2/26/82	E	LOOP NO. 4 SH 1
	1659	R1	2/26/82	E	LOOP NO. 4 SH 2
	1668	R2	3/25/83	E	RAD MONITORING LOOP 3 SH 6
	1691S	R3	2/14/83	E	REACTOR COOLANT VENT SYSTEM-1
	1692	R5	12/14/82	E	REACTOR COOLANT VENT SYSTEM-2
	1694	R2	12/14/82	E	PLANT AUX CONTROL BOARD-2 120V AC & 125V DC DISTRIBUTION
	1695	R1	3/12/82	E	PLANT AUX CONTROL BOARD 45V DC DISTRIBUTION
	1701S	R3	2/1/83	E	480V SWGR 2A-5 MOT REL & HTRS
	1702S	R1	8/14/82	E	480V SWGR 2A-5 FEEDER
	1703S	R1	3/26/82	E	REACTOR AREA MCC-2A6

TABLE 1.7-1 (Cont'd)

Drawing No.	Sheet No.	Revision No.	Date	Prepared By	Title
B-327					CONTROL WIRING DIAGRAM (Cont'd)
	1711	R4	2/1/83	E	480V SWG 2B5 METERING RELAYS & HEATERS
	1712	R2	8/15/82	E	480V SWGR 2B5 FDR
	1713	R1	4/8/82	E	480V SWGR 2B5 FEEDWATER TO REACTOR AREA MCC-2B6
	1751	R4	8/17/83	E	QSPDS INTERCONNECTION
	1755	R3	1/24/83	E	REACTOR COOLANT TEMP. SAS-QSPDS INPUTS SH 1
	1756	R4	3/11/83	E	REACTOR COOLANT TEMP. SAS-QSPDS INPUTS SH 2
	1757	R1	8/13/82	E	PRESSURIZED PRESSURE ICC-INPUTS
	1810	R2	8/17/83	E	PRESSURIZER PRESSURE ICC-INPUTS
	1829	R4	5/12/83	E	PASS VALVES SH 4
	1831	R3	4/12/83	E	PSB-1 UNDERVOLTAGE 4160V BUS 2A3
	1833	R3	4/12/83	E	PSB-1 UNDERVOLTAGE 480V BUS 2A2/2A5
	1834	R3	4/12/83	E	PSB-1 UNDERVOLTAGE PROTECTION 480V BUS 2B2/2B5
	1836	R3	4/11/83	E	PSB-1 UNDERVOLTAGE PROTECTION BUS 2A3/2A2/2A5 RELAY
	1837	R3	4/11/83	E	PSB-1 UNDERVOLTAGE PROTECTION BUS 2B3/2B2/2B3 RELAY
	1851	R2	11/30/82	E	INCORE MONITOR DETECTORS L18, L20, R16, R18
	1852	R2	11/30/82	E	INCORE MONITOR DETECTORS C18, E13, E16, E18
	1853	R2	11/30/82	E	INCORE MONITOR DETECTORS C6, C13, E2, G4

TABLE 1.7-1 (Cont'd)

Drawing No.	Sheet No.	Revision No.	Date	Prepared By	Title
B-327					CONTROL WIRING DIAGRAM (Cont'd)
	1854	R2	11/30/82	E	INCORE MONITOR DETECTORS G6, L4, L6, R2
	1855	R2	11/30/82	E	INCORE MONITOR DETECTORS R9, R15, T4, T6
	1856	R2	11/30/82	E	INCORE MONITOR DETECTORS T9, W4, W9, W13
	1857	R2	11/30/82	E	INCORE MONITOR DETECTORS R20, T20, Y8, Y14
	1858	R2	11/30/82	E	INCORE MONITOR DETECTORS A8, C4, E4, G2
	1859	R2	11/30/82	E	INCORE MONITOR DETECTORS A14, C9, E6, E9
	1860	R2	11/30/82	E	INCORE MONITOR DETECTORS C16, E20, G9, G13
	1861	R2	11/30/82	E	INCORE MONITOR DETECTORS G16, G18, L13, L16
	1862	R2	11/30/82	E	INCORE MONITOR DETECTORS G20, T18, W16, W18
	1863	R2	11/30/82	E	INCORE MONITOR DETECTORS L9, T13, T16, W6
	1864	R2	11/30/82	E	INCORE MONITOR DETECTORS L2, R4, R6, T2
	1865	R3	3/11/83	E	HEATER JUNCTION THERMOCOUPLES 1A, 2A, 3A, 4A
	1866	R3	3/11/83	E	HEATER JUNCTION THERMOCOUPLES 5A, 6A, 7A, 8A
	1867	R3	3/11/83	E	HEATER JUNCTION THERMOCOUPLES 1B, 2B, 3B, 4B
	1868	R3	3/11/83	E	HEATER JUNCTION THERMOCOUPLES 5B, 6B, 7B, 8B

TABLE 1.7-1 (Cont'd)

Drawing No.	Sheet No.	Revision No.	Date	Prepared By	Title
B-327					CONTROL WIRING DIAGRAM (Cont'd)
	-1692S	R5	2/14/83	E	REACTOR COOLANT VENT SYSTEM-2
G-878		R7	5/19/83	E	HVAC - CONTROL DIAGRAMS SH. 1
G-879	2	R7	5/19/83	E	HVAC - CONTROL DIAGRAMS SH. 2
G-879	3	R9	5/19/83	E	HVAC - CONTROL DIAGRAMS SH. 3

TABLE 1.7-2

NSSS SUPPLIED ELECTRICAL INSTRUMENTATION AND CONTROL DRAWINGS
SAFETY RELATED

Drawing No.	Sheet No.	Revision No.	Date	Prepared By	Title
B-13172-412-330	1-9	R2	7/26/77	CE	ELEMENTARY W/D REAC TRIP CKT BKR
E-13172-413-130		R2	7/20/77	CE	REACTOR TRIP SWITCHGEAR ARRANGEMENT
E-13172-411-022		R3	10/28/82	CE	NUCLEAR INSTRUMENTATION AND RPS CABINET ASSY
E-13172411012	1	R6	10/27/82	CE	RPS TERMINAL BLOCK WIRING DIAGRAM
E-13172411012	2	R6	10/28/82	CE	RPS TERMINAL BLOCK WIRING DIAGRAM SH. 2
E-13172411012	3	R6	10/28/82	CE	RPS TERMINAL BLOCK WIRING DIAGRAM SH. 3
E-13172411012	4	R6	10/28/82	CE	RPS TERMINAL BLOCK WIRING DIAGRAM SH. 4
805B1		R0	8/18/75	CE	RTSG HEATER ELEMENTARY
805B13		R2	12/3/75	CE	GE AK-2-25 CIRCUIT BREAKER ELEMENTARY & CONN DIAG
805B12		R1	3/24/76	CE	RTSG DC ELEMENTARY TCB-9
8055-B10		R2	3/24/76	CE	RTSG DC ELEMENTARY TCB-7
805B11		R2	3/24/76	CE	RTSG DC ELEMENTARY TCB-8
805B8		R2	3/24/76	CE	RTSG DC ELEMENTARY TCB-5
805B9		R2	3/24/76	CE	RTSG DC ELEMENTARY TCB-6
805B6		R2	3/24/76	CE	RTSG DC ELEMENTARY TCB-3
805B7		R2	3/24/76	CE	RTSG DC ELEMENTARY TCB-4
805B5		R2	3/24/76	CE	RTSG DC ELEMENTARYTCB-2
805B4		R2	3/24/76	CE	RTSG DC ELEMENTARY TCB-1
805B3		R0	8/19/75	CE	RTSG CURRENT MONITOR ELEMENTARY

TABLE 1.7-2 (Cont'd)

Drawing No.	Sheet No.	Revision No.	Date	Prepared By	Title
805B2		R1	9/26/75	CE	RTSG CURRENT MONITOR ELEMENTARY
805E6		R1	3/24/76	CE	RTSG WIRING DIAGRAM SECTION 04
805E7		R1	3/24/76	CE	RTSG WIRING DIAGRAM SECTION 05
805E5		R1	3/24/76	CE	RTSG WIRING DIAGRAM SECTION 03
805E4		R1	3/24/76	CE	RTSG WIRING DIAGRAM SECTION 02
805E3		R1	3/23/76	CE	RTSG WIRING DIAGRAM SECTION 01
805E2		R1	10/1/75	CE	RTSG ARRANGEMENT & DETAILS
805E1		R1	10/1/75	CE	RTSG ARRANGEMENT & DETAILS
E-13172-411-071		R3	7/26/82	CE	CORE PROTECT CALCULATOR NO. 1 SCHEMATIC
E-13172-411-072	1	R3	1/28/83	CE	CORE PROTECT CALCULATOR No. 2 SCHEMATIC
E-13172-411-086	1	R2	1/28/83	CE	RPS ISOLATION LOGIC & WIRING DIAGRAM
E-13172-411-013	4	R2	7/28/82	CE	RPS SCHEMATIC SH4 of 4
E-13172-411-018		R2	7/28/82	CE	TRIP INHIBIT MODULE WIRING DIAGRAM
E-13172-411-400		R1	3/14/79	CE	RPS CALIB.& INDIC. PNL. SCHEMATIC
E-13172-411-040		R3	1/28/83	CE	REACTOR TRIP BYPASS SCHE-MATIC
E-13172-411-325		R1	3/14/79	CE	RPS MISC. SCHEMATICS
E-13172-411-013	1	R2	7/28/82	CE	RPS MISC. SCHEMATICS SH1 OF 4

TABLE 1.7-2 (Cont'd)

Drawing No.	Sheet No.	Revision No.	Revision Date	Prepared By	Title
E-13172-411-013	3	R2	7/28/82	CE	RPS MISC. SCHEMATICS SH3 OF 4
E-13172-411-013	2	R2	7/28/82	CE	RPS MISC. SCHEMATICS SH2 OF 4
E-13172-411-401		R1	3/14/79	CE	RPS CALIB & IND. PNL ASSEMBLY
E-13172-411-011		R2	7/28/82	CE	RPS BIN ASSEMBLIES WIRING DIAG.
E-13172-411-324		R1	3/14/79	CE	AUX. LOGIC WIRING DIAGRAM
E-13172-411-072	2	R2	1/28/83	CE	CORE PROT. CALC NO.2 SCHEMATIC-SH2 OF 2
E-13172-411-024		R2	10/28/82	CE	RPS BIN ASSEMBLY
E-13172-411-015		R0	12/23/77	CE	TRAC 1 WIRING & ASSEMBLY DIAGRAM
E-13172-411-029		R2	1/28/83	CE	TRIP TEST CABLE PNL ASSEMB.
D-13172-411-366		R2	3/14/79	CE	TRIP UNIT BIN ASSEMB. PERSPECT.
E-13172-411-085		R2	1/28/82	CE	POWER RATIO SIGNAL CALC. SCHEM.
E-13172-411-310		R2	3/14/79	CE	AUX. LOGIC SCHEMATIC
E-13172-411-025		R2	10/28/82	CE	TRIP INHIBIT MODULE ASSY
E-13172-411-034	1	R3	1/28/83	CE	RPS TRIP STATUS PNL. SCHEM & W/D
E-13172-411-302		R4	7/28/82	CE	TRIP UNIT INTERCONN. MODULE W/D
E-13172-411-021		R2	7/28/82	CE	NUC. INST. RPS CAB. ASSY. FRNT PNL LAYOUT
E-13172-411-039		R2	1/28/83	CE	SCHEM. INPUT SIG. CONN.TO TRIP UNITS
E-13172-411-350		R1	3/14/79	CE	AUX. LOGIC ASSEMBLY

TABLE 1.7-2 (Cont'd)

Drawing No.	Sheet No.	Revision No.	Date	Prepared By	Title
E-13172-411-003		R2	7/28/82	CE	RPS FUNCTIONAL DIAGRAM
E-13172-411-033		R2	1/28/83	CE	RPS TRIP STATUS PNL. ASSY
E-13172-411-043		R1	12/23/77	CE	LOW FLOW PROT. SYS. FUNCT. DIAG.
E-13172-411-376		R2	3/14/79	CE	TRIP UNIT INTERCONN. MODULE ASSY
D-13172-411-091		R1	3/14/79	CE	BISTABLE TRIP UNIT MODULE ASSEMBLY
D-13172-411-035		R2	1/28/83	CE	RPS/NI INTERFACE
D-13172-411-092		R1	3/14/79	CE	AUXILIARY TRIP UNIT MODULE ASSEMBLY
E-13172-411-034	2	R2	1/28/83	CE	RPS TRIP STATUS PANEL SCHEM & WIRING DIAGRAM
E-13172-411-086	2	R2	1/28/83	CE	RPS ISOLATION LOGIC & WIRING DIAGRAM AW 20
D-13172-413-412		R5	11/29/82	CE	STEAM GENERATOR-B PROTEC-TIVE CHANNEL BLOCK DIAGRAM
D-13172-413-411		R4	1/29/82	CE	STEAM GENERATOR-A PROTEC-TIVE CHANNEL BLOCK DIAGRAM
D-13172-416-214		R5	3/18/83	CE	INTERC/D CHGNG PMP DISCH HDR PRES CHAN P2212
D-13172-416-121	3	R5	5/02/83	CE	INTERCONN DIAG PRESS LEVEL CHANNEL L-1110 SH3
D-13172-416-121	2	R7	5/02/83	CE	INTERCONN DIAG PRESS LEVEL CHANNEL L-1110 SH2
D-13172-416-121	1	R6	4/06/83	CE	INTERCONN DIAG PRESS LEVEL CHANNEL L-1110 SH1
D-13172-416-112		R4	3/18/83	CE	INTERC/D-PRESSURIZER PRES-SURE CHANNEL P1102
D-13172-416-131		R5	4/06/83	CE	I/D-REAC COOL DELTA PRES FLOW CHANS P1101A-D

TABLE 1.7-2 (Cont'd)

Drawing No.	Sheet No.	Revision No.	Date	Prepared By	Title
D-13172-416-217		R2	11/21/80	CE	I/D-CHARGING PUMP SUCT PRESS CHANNEL P-2224
D-13172-416-311	1	R2	4/06/83	CE	ID-HPSI,.LPSI HEADER PRESS CHANNELS P-3308,9
D-13172-416-311	2	R3	4/06/83	CE	ID-HPSI,.LPSI HDR PRESS CHANNELS P3304-7
D-13172-416-103	1	R3	6/25/82	CE	ID-RCS LOOP TEMP CHANNELS T1111 & 1121 SH1OF2
D-13172-416-103	2	R4	4/06/83	CE	ID-RCS LOOP TEMP CHANNELS T1115 & 1125 SH2OF2
D-13172-416-104		R3	4/06/83	CE	INTERCONN DIAG-RCS LOOP TEMP CHS T1112&1122
D-13172-416-113	1	R5	4/06/83	CE	ID-PRESSURIZER PRESS LO RNGE CH P1103,5 SH1/2 877
D-13172-416-113	2	R5	4/06/83	CE	ID-PRESSURIZER PRESS LO RNGE CH P1103,5 SH2/2
D-13172-416-115		R1	2/26/80	CE	INTERCONN DIAG-RCP SEAL PRE SS CHANNELS
D-13172-416-132		R1	2/26/80	CE	ID-RCP CONT BLEEDOFF FLO CHS F1150,60,70,80
D-13172-416-401		R4	4/06/83	CE	ID-STEAM GENERATOR STEAM PRESS CHANNEL P8013
D-13172-416-402		R4	4/06/83	CE	INTERCONN DIAG-STEAM GENERATOR LEVEL CH L9013 877
D-13172-416-651		R3	11/24/81	CE	I/D WMS MISC LOCAL & REMOTE ALARMS 8770-1938
D-13172-416-470	1	R3	1/29/82	CE	I/D-CONT BRD MTD NUCLEAR INST SH1 OF 4
D-13172-416-470	4	R2	7/12/81	CE	I/D CONT BRD MTD NUCLEAR INST SH 4 OF 4
D-13172-416-331		R2	4/06/83	CE	ID HI & LO PRES SI FLOW CHANNELS F3311,F3312

TABLE 1.7-2 (Cont'd)

Drawing No.	Sheet No.	Revision No.	Date	Prepared By	Title
D-13172-416-470	2	R4	9/27/82	CE	ID CONTR BOARD MNTD NUC INSTR SH2 8770-1511-13
D-13172-416-470	3	R2	1/19/81	CE	INTERCONN DIAG CONTROL BOARD MNTD INSTR SH 3
D-13172-416-105		R4	10/12/82	CE	INTERCONN DIAG RCS TEMP CHANNEL T-1102
D-13172-416-222		R2	7/10/81	CE	ID-BA TANKS 2A & 2B LEVEL CHS L-2206,7,8,9

TABLE 1.7-3

ARCHITECT/ENGINEER SUPPLIED
FLOW DIAGRAMS, PIPING AND INSTRUMENTATION DIAGRAMS
SAFETY RELATED

Drawing No.	Sheet No.	Revision No.	Date	Prepared By	Title
G 079	1	R11	6/15/83	E	FLOW DIAG - MAIN, EXTRACTION AUXILIARY STEAM & AIR EVACUATION SYSTEMS
G 079	2	R10	4/12/83	E	FLOW DIAG - MAIN, EXTRACTION AUXILIARY STEAM & AIR EVACUATION SYSTEMS
G 080	1	R9	6/15/83	E	FLOW DIAG - FDWTR & CONDENSATE SYSTEMS
G 080	2	R9	6/15/83	E	FLOW DIAG - FDWTR & CONDENSATE SYSTEMS
G 081	1	R9	4/12/83	E	FLOW DIAG - HEATER DRAIN & VENT SYSTEMS
G 081	2	R9	4/12/83	E	FLOW DIAG - HEATER DRAIN & VENT SYSTEMS
G 082		R11	6/15/83	E	FLOW DIAG - CRLG & INTAKE COOLING WATER SYSTEMS
G 083		R11	12/27/82	E	FLOW DIAG - COMPONENT COOLANT SYSTEM
G 084		R11	6/15/83	E	FLOW DIAG - FIREWATER DOMESTIC AND MAKEUP SYSTEMS
G 085	1	R11	6/15/83	E	FLOW DIAGRAM - SERVICE INSTRUMENT AIR SYSTEM
G 085	2	R10	6/15/83	E	FLOW DIAGRAM - INSTRUMENT INSTRUMENT AIR SYSTEM
G 086		R11	6/15/83	E	FLOW DIAG - MISCELLANEOUS SYSTEMS SH-1
G 087		R11	6/15/83	E	FLOW DIAG - MISCELLANEOUS SYSTEMS SH-2
G 088		R11	6/15/83	E	FLOW DIAG - CONTAINMENT SPRAY & REFUELING WATER SYSTEMS

TABLE 1.7-3 (Cont'd)

Drawing No.	Sheet No.	Revision No.	Date	Prepared By	Title
G 089		R9	10/4/82	E	FLOW DIAG - TURBINE COOLING WATER SYSTEM
G 090		R7	10/11/82	E	REACTOR COOLANT - PRESSURE BOUNDARY DIAGRAM
G 091		R9	6/15/83	E	FLOW DIAG MISC SYSTEMS
G 092		R7	6/15/83	E	FLOW DIAG MISC SAMPLING SYSTEMS
G 862		R6	11/19/82	E	HVAC - AIR FLOW DIAGRAM
G 863		R5	5/19/83	E	HVAC - REFRIGERANT PIPING

Table 1.7-4

NSSS SUPPLIED FLOW DIAGRAMS, PIPING AND INSTRUMENTATION DIAGRAMS
SAFETY RELATED

Drawing No.	Revision No.	Date	Prepared By	Title
E-13172-310-100	13	7/1/82	CE	PIPING & INSTRUMENTATION DIAGRAM SYMBOLS
E-13172-310-110	17	5/17/83	CE	REACTOR COOLANT SYSTEM P&I DIAGRAM
E-13172-310-111	17	7/7/83	CE	REACTOR COOLANT PUMP P&I DIAGRAM
E-13172-310-120	17	3/16/83	CE	CHEMICAL & VOLUME CONTROL SYSTEM P&I DIAG
E-13172-310-121	18	5/17/83	CE	CHEMICAL & VOLUME CONTROL SYSTEM P&I DIAG
E-13172-310-130	19	5/17/83	CE	SAFETY INJECTION SYS P&I DIAGRAM
E-13172-310-131	17	5/17/83	CE	SAFETY INJECTION SYS P&I DIAGRAM
E-13172-310-140	19	7/7/83	CE	FUEL POOL SYS P&I DIAGRAM
E-13172-310-150	18	7/7/83	CE	SAMPLING SYSTEM P&I DIAGRAM
E-13172-310-160	19	5/17/83	CE	WASTE MANAGEMENT SYS P&I DIAGRAM
E-13172-310-161	19	7/7/83	CE	WASTE MANAGEMENT SYS P&I DIAGRAM
E-13172-310-162	19	7/7/83	CE	WASTE MANAGEMENT SYS P&I DIAGRAM
E-13172-310-163	20	7/7/83	CE	WASTE MANAGEMENT SYS P&I DIAGRAM (SHEET 4)
E-13172-310-164	16	5/17/83	CE	WASTE MANAGEMENT SYS P&I DIAGRAM
E-13172-310-105	17	7/7/83	CE	AUXILIARY PUMPS P&I DIAGRAM
E-13172-310-165	13	3/18/83	CE	BORIC ACID CONCENTRATOR 2A P&ID

TABLE 1.7-4 (Cont'd)

Drawing No.	Revision No.	Revision Date	Prepared By	Title
E-13172-310-166	13	3/16/83	CE	BORIC ACID CONCENTRATOR B P&ID
E-13172-310-167	13	3/16/83	CE	RADIOACTIVE WASTE CON-CENTRATOR P&ID
E-13172-310-168	11	11/22/82	CE	WASTE MANAGEMENT SYS P&I DIAGRAM
E-13172-310-109	13	7/7/83	CE	REACTOR COOLANT SYS P&I DIAGRAM
E-13172-310-122	13	7/7/83	CE	CHEMICAL & VOLUME CONTROL SYS P&I DIAGRAM
E-13172-310-107	06	3/16/83	CE	REACTOR COOLANT SYSTEM P&I DIAGRAM
E-13172-310-108	04	5/17/83	CE	REACTOR COOLANT SYSTEM P&I DIAGRAM
E-13172-310-153	09	7/7/83	CE	SAMPLING SYSTEM P&I DIAGRAM
E-13172-310-132	07	5/17/83	CE	SAFETY INJECTION SYSTEM P&I DIAGRAM
E-13172-310-152	08	7/7/83	CE	SAMPLING SYSTEM P&I DIAGRAM
E-13172-310-169	08	5/18/83	CE	WASTE MANAGEMENT SYSTEM P&I DIAGRAM
E-13172-310-171	04	9/23/82	CE	WASTE MANAGEMENT SYSTEM P&I DIAGRAM
E-13172-310-101	03	9/23/82	CE	STEAM GENERATOR SUPPORT SNUBBER PIPING SYSTEM VALVE IDENTIFICATION
E-13172-310-145	03	11/23/82	CE	REFUELING EQUIPMENT VALVE IDENTIFICATION
E-13172-310-115	06	7/7/83	CE	R.C. PUMP SEAL INJECTION ADDITION P & I DIAGRAM

1.8 NRC REGULATORY GUIDES

Information contained herein were valid at the time the Construction Permit for St. Lucie 2 was issued, and are being retained in the Updated FSAR for document completeness and historical record. No present or future update of this section is required.

Subject to the implementation dates therein, Regulatory Guides issued on or before May 2, 1977 (Construction Permit date for St. Lucie Unit 2) are considered to contain the recommendations that are applicable to the design of this plant. Table 1.8-1 is a listing of all such Regulatory Guides, with corresponding dates and revision numbers. Cross-references are provided in Table 1.8-1 for those regulatory guide subjects discussed in particular subsections.

In specific instances, later revisions to Regulatory Guides listed in Table 1.8-1 are addressed where following such guidance is deemed proper.

Other NRC staff requirements are discussed in Section 1.9.

TABLE 1.8-1

APPLICABLE NRC REGULATORY GUIDES

Number	Title	Date	Revision	Discussion in Subsection(s)	Remarks
1.1	Net Positive Suction Head for Emergency Core Cooling and Containment Heat Removal System Pumps	11/70	0	6.2.2.3.1 6.3.4.1.1	
1.2	Thermal Shock to Reactor Pressure Vessels	11/70	0	5.3.1	
1.3	Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Boiling Water Reactors.	6/74	2	Not Applicable	
1.4	Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Pressurized Water Reactors	6/74	2	2.3.4	
1.5	Assumptions Used for Evaluating the Potential Radiological Consequences of a Steam Line Break Accident for Boiling Water Reactors	3/71	0	Not Applicable	
1.6	Independence Between Redundant Standby (Onsite) Power Sources and Between Their Distribution Systems	3/71	0	8.3.1.2	
1.7	Control of Combustible Gas Concentrations in Containment Following a Loss of Coolant Accident	11/78	2	6.2.5.3.2	
1.8	Personnel Selection and Training	5/77	1-R	12.5.1/12.5.3 13.1.3, 17.2	
1.9	Selection of Diesel Generator Set Capacity for Standby Power Supplies	3/71	0	8.3.1.2	
1.10	Mechanical (Cadweld) Splices in Reinforcing Bars of Category I Concrete Structures	1/73	1	3.8.3.2	
1.11	Instrument Lines Penetrating Primary Reactor Containment	3/71	0	7.1.2.2 6.2.4	
1.12	Instrumentation for Earthquakes	4/74	1	3.7.4	
1.13	Spent Fuel Storage Facility Design Basis	12/75	1	9.1.1.3/9.1.2.3 9.1.3.3	
1.14	Reactor Coolant Pump Flywheel Integrity	10/71	0	5.4.1.1	Regulatory Position C.4.6 of Revision 1 (8/75) is applicable to in-service inspections con ducted on all plants after January 1, 1976.

TABLE 1.8-1 (Cont'd)

Number	Title	Date	Revision	Discussion in Subsection(s)	Remarks
1.15	Testing of Reinforcing Bars for Category I Concrete Structures	12/72	1	3.8.3.2/3.8.3.6	
1.16	Reporting of Operating Information-Appendix A Technical Specifications	8/75	4	12.5.3	
1.17	Protection of Nuclear Plants Against Industrial Sabotage	6/73	1	13.6	A proprietary St Lucie Plant Security Plan is submitted under separate cover.
1.18	Structural Acceptance Test for Concrete Primary Reactor Containments	12/72	1	Not Applicable	
1.19	Nondestructive Examination of Primary Containment Liner Welds	8/72	1	Not Applicable	
1.20	Comprehensive Vibration Assessment Program for Reactor Internals During Preoperational and Initial Startup Testing	5/76	2	3.9.2.4	
1.21	Measuring, Evaluating, and Reporting Radioactivity in Solid Wastes and Release of Radioactivity in Liquid and Gaseous Effluents from Light Water-Cooled Nuclear Power Plants	6/74	1	11.5.1.2 12.3.4	
1.22	Periodic Testing of Protection System Actuation Functions	2/72	0	7.2.1.1.9/7.5.2.9 7.3.1.1.1/7.6.2 7.4.2.2	
1.23	Onsite Meteorological Programs	2/72	0	2.3.3	
1.24	Assumptions Used for Evaluating the Potential Radiological Consequences of a Pressurized Water Reactor Gas Storage Tank Failure	3/72	0	15.7.1	
1.25	Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facility for Boiling and Pressurized Water Reactors	3/72	0	15.7.3	
1.26	Quality Group Classifications and Standards for Water-Steam- and Radio-Waste-Containing Components of Nuclear Power Plants	2/76	3	3.2.2	
1.27	Ultimate Heat Sink for Nuclear Power Plants	1/76	2	9.2.5	

TABLE 1.8-1 (Cont'd)

Number	Title	Date	Revision	Discussion in Subsection(s)	Remarks
1.28	Quality Assurance Program Requirements (Design and Construction)	6/72	0	Not Applicable	This regulatory guide is applicable during the design and construction phases of nuclear power plants and as such is discussed in PSARs, not FSARs.
1.29	Seismic Design Classification	9/78	3	3.2.1	
1.30	Quality Assurance Requirements for the Installation, Inspection, and Testing of Instrumentation and Electric Equipment	-	-	17.2	The revision and date of this document endorsed for St. Lucie Unit 2 is governed by the latest revision of the FP&L Topical Quality Assurance Report as referenced in Section 17.2 of the FSAR.
1.31	Control of Ferrite Content in Stainless Steel Weld Metal	5/77	2 & 3	5.2.3.4.2 6.1.1.1 10.3.6.2	Subsections 6.1.1.1 and 10.3.6.2 address Revision 1 (6/73) of this regulatory guide also.
1.32	Criteria for Safety-Related Electric Power Systems for Nuclear Power Plants	8/72	0	8.3.1.2	
1.33	Quality Assurance Program Requirements (Operations)	-	-	13.5.1, 17.2	The revision and date of this document endorsed for St. Lucie Unit 2 is governed by the latest revision of the FP&L Topical Quality Assurance Report as referenced in Section 17.2 of the FSAR.
1.34	Control of Electroslag Weld Properties	12/72	0	5.2.3.3.2 5.2.3.4	
1.35	Inservice Inspection of Ungrouted Tendons in Prestressed Concrete Containment Structures	1/76	2	Not Applicable	
1.36	Nonmetallic Thermal Insulation for Austenitic Stainless Steel	2/73	0	5.2.3.2 6.1.1.1	
1.37	Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water-Cooled Nuclear Power Plants	-	-	6.1.1.1, 17.2	The revision and date of this document endorsed for St. Lucie Unit 2 is governed by the latest revision of the FP&L Topical Quality Assurance Report as referenced in Section 17.2 of the FSAR.
1.38	Quality Assurance Requirements for Packaging, Shipping, Receiving, Storage, and Handling of Items for Water-Cooled Nuclear Power Plants	-	-	17.2	The revision and date of this document endorsed for St. Lucie Unit 2 is governed by the latest revision of the FP&L Topical Quality Assurance Report as referenced in Section 17.2 of the FSAR.

TABLE 1.8-1 (Cont'd)

Number	Title	Date	Revision	Discussion in Subsection(s)	Remarks
1.39	Housekeeping Requirements for Water-Cooled Nuclear Power Plants	-	-	17.2	The revision and date of this document endorsed for St. Lucie Unit 2 is governed by the latest revision of the FP&L Topical Quality Assurance Report as referenced in Section 17.2 of the FSAR.
1.40	Qualification Tests of Continuous-Duty Motors Installed Inside the Containment of Water-Cooled Nuclear Power Plants	3/73	0	3.11	
1.41	Preoperational Testing of Redundant Onsite Electric Power Systems to Verify Proper Load Group Assignments	3/73	0	8.3.1.2	
1.42	Withdrawn 3/76				
1.43	Control Stainless Steel Weld Cladding of Low-Alloy Steel Components	5/73	0	5.2.3.3.2	
1.44	Control of the Use of Sensitized Stainless Steel	5/73	0	5.2.3.4.1 6.1.1.1 10.3.6.2	
1.45	Reactor Coolant Pressure Boundary Leakage Detection Systems	5/73	0	5.2.5	
1.46	Protection Against Pipe Whip Inside Containment	5/73	0	3.6.1.1/3.6.2.3.2 3.6.2.1.1	
1.47	Bypassed and Inoperable Status Indication for Nuclear Power Plant Safety Systems	5/73	0	7.5.2.7	
1.48	Design Limits and Loading Combinations for Seismic Category I Fluid System Components	5/73	0	3.9.1.4 3.9.3.1.1	
1.49	Power Levels of Nuclear Power Plants	12/73	1	6.2, 15.0	The guidance provided in this regulatory guide is utilized in accident analyses performed.
1.50	Control of Preheat Temperature for Welding of Low-Alloy Steel	5/73	0	5.2.3.3.2 10.3.6.2	
1.51	Withdrawn	7/75			
1.52	Design, Testing, and Maintenance Criteria for Post-accident Engineered-Safety-Feature Atmosphere Cleanup System Air Filtration and Absorption Units of Light-Water-Cooled Nuclear Power Plants	3/78	2	6.5.1	

TABLE 1.8-1 (Cont'd)

Number	Title	Date	Revision	Discussion in Subsection(s)	Remarks
1.53	Application of the Single-Failure Criterion to Nuclear Power Plant Protection Systems	6/73	0	7.1.2.2 7.2.1.2	
1.54	Quality Assurance Requirements for Protective Coatings Applied to Water-Cooled Nuclear Power Plants	6/73	0	6.1.2	
1.55	Concrete Placement in Category I Structures	6/73	0	3.8.3.2/3.8.3.6	
1.56	Maintenance of Water Purity in Boiling Water Reactors	6/73	0	Not Applicable	
1.57	Design Limits and Loading Combinations for Metal Primary Reactor Containment System Components	6/73	0	3.8.2.3	
1.58	Qualification of Nuclear Power Plant Inspection,	-	-	13.1, 17.2	The revision and date of this document endorsed for St. Lucie Unit 2 is governed by the latest revision of the FP&L Topical Quality Assurance Report as referenced in Section 17.2 of the FSAR.
1.59	Design Basis Flood for Nuclear Power Plants	4/76	1	2.4.2.2/2.4.3 3.4.1	
1.60	Design Response Spectra for Seismic Design of Nuclear Power Plants	12/73	1	3.7.1.1	
1.61	Damping Values for Seismic Design of Nuclear Power Plants	10/73	0	3.7.1.3	
1.62	Manual Initiation of Protective Actions	10/73	0	7.1.2.2, 8.3.1.2	
1.63	Electric Penetration Assemblies in Containment Structures for Light-Water-Cooled Nuclear Power Plants	10/73	0	8.3.1.2	
1.64	Quality Assurance Requirements for the Design of Nuclear Power Plants	-	-	17.2	The revision and date of this document endorsed for St. Lucie Unit 2 is governed by the latest revision of the FP&L Topical Quality Assurance Report as referenced in Section 17.2 of the FSAR.
1.65	Materials and Inspection for Reactor Vessel Closure Studs	10/73	0	5.3.1.7	
1.66	Withdrawn 10/77				
1.67	Installation of Overpressure Protective Devices	10/73	0	3.9.3.3	

TABLE 1.8-1 (Cont'd)

Number	Title	Date	Revision	Discussion in Subsection(s)	Remarks
1.68	Initial Test Programs for Water-Cooled Nuclear Power Plants	1/77	1		Section 14.0 addressed Revision 0.
1.68.1	Preoperational and Initial Startup Testing of Feed-water and Condensate Systems for Boiling Water Reactor Power Plants	1/77	1	Not Applicable	
1.68.2	Initial Startup Test Program to Demonstrate Remote Shutdown Capability for Water-Cooled Nuclear Power Plants	7/78	1		Initially discussed in Section 14.0
1.69	Concrete Radiation Shields for Nuclear Power Plants	12/73	0	12.3.2.4	
1.70	Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants-LWR Edition	9/75	2		Revision 3 (11/78) of this regulatory guide was used insofar as to the extent practicable in developing the St Lucie Unit 2 Final Safety Analysis Report.
1.71	Welder Qualification for Areas of Limited Accessibility	12/73	0	5.2.3.3.2 10.3.6.2	
1.72	Spray Pond Piping Made From Fiberglass-Reinforced Thermosetting Resin	12/73	0	Not Applicable	
1.73	Qualification Tests of Electric Valve Operators Installed Inside the Containment of Nuclear Power Plants	1/74	0	3.11	
1.74	Quality Assurance Terms and Definitions	-	-	17.2	The revision and date of this document endorsed for St. Lucie Unit 2 is governed by the latest revision of the FP&L Topical Quality Assurance Report as referenced in Section 17.2 of the FSAR.
1.75	Physical Independence of Electric Systems	1/75	1	7.1.2.2 8.3.1.2	
1.76	Design Basis Tornado for Nuclear Power Plants	4/74	0		The Design Basis Tornado for St Lucie Unit 2 is discussed in Subsection 3.3.2.
1.77	Assumptions Used for Evaluating a Control Rod Ejection Accident for Pressurized Water Reactors	5/74	0	15.4.3	
1.78	Assumptions for Evaluating the Habitability of a Nuclear Power Plant Control Room During a Postulated Hazardous Chemical Release	6/74	0	2.2.3.2	EC284401
1.79	Preoperational Testing of Emergency Core Cooling Systems for Pressurized Water Reactors	9/75	1	6.3.4.1.1	
1.80	Preoperational Testing of Instrument Air Systems	6/74	0	14.2	

TABLE 1.8-1 (Cont'd)

Number	Title	Date	Revision	Discussion in Subsection(s)	Remarks
1.81	Shared Emergency and Shutdown Electric Systems for Multi-Unit Nuclear Power Plants	1/75	1	8.3.1.2	St Lucie Units 1 and 2 have separate and independent onsite emergency and shutdown electric systems.
1.82	Sumps for Emergency Core Cooling and Containment Spray Systems	6/74	0	6.2.2.2.3	
1.83	Inservice Inspection of Pressurized Water Reactor Steam Generator Tubes	7/75	1	5.4.2.2	RG 1.83 withdrawn November 2009. (Ref: NRC document NRC-2009-0488: ML13066A546) EC283094
1.84	Design and Fabrication Code Case Acceptability - ASME Section III Division 1	3/77	9	3.9.3.1.1	
1.85	Materials Code Case Acceptability - ASME Section III Division 1	3/77	9	-	Materials acceptability is discussed in various subsections of the FSAR which deal with this topic for various structures, systems, and components.
1.86	Termination of Operating Licenses for Nuclear Reactors	6/74	0	Not Applicable	The regulatory guide is applicable when a licensee decides to terminate the nuclear reactor operating license.
1.87	Guidance for Construction of Class 1 Components in Elevated-Temperature Reactors (Supplement to ASME Section III Code Classes 1592, 1593, 1594, 1595, and 1596)	6/74	0	Not Applicable	
1.88	Collection, Storage, and Maintenance of Nuclear Power Plant Quality Assurance Records	-	-	17.2	The revision and date of this document endorsed for St. Lucie Unit 2 is governed by the latest revision of the Quality Assurance FP&L Topical Report as referenced in Section 17.2 of the FSAR.
1.89	Qualification of Class IE Equipment for Nuclear Power Plants	11/74	0	8.3.1.2 3.11	
1.90	In-service Inspection of Prestressed Concrete Containment Structures With Grouted Tendons	11/74	0	Not Applicable	
1.92	Combining Modal Responses and Spatial Components in Seismic Response Analysis	12/74	0	3.7.2.6/3.7.2.7 3.7.3.6/3.7.3.7	
1.93	Availability of Electric Power Sources	12/74	0	8.3.1.2	The applicable recommendations of Regulatory Guide 1.93 are also a part of the Technical Specifications.

TABLE 1.8-1 (Cont'd)

Number	Title	Date	Revision	Discussion in Subsection(s)	Remarks
1.94	Quality Assurance Requirements for Installation, Inspection, and Testing of Structural Concrete Structural Steel, Soils and Foundations During the Construction Phase of Nuclear Power Plants	4/76	1	Not Applicable	The revision and date of this document endorsed for St. Lucie Unit 2 is governed by the latest revision of the FP&L Topical Quality Assurance Report as referenced in Section 17.2 of the FSAR.
1.95	Protection of Nuclear Power Plant Control Room Operators Against an Accidental Chlorine Release	2/75	0	2.2.3.2	EC284401
1.96	Design of Main Steam Isolation Valve Leakage Control Systems for Boiling Water Reactor Nuclear Power Plants	6/76	1	Not Applicable	
1.98	Assumptions Used for Evaluating the Potential Radiological Consequences of a Radioactive Offgas System Failure in a Boiling Water Reactor	3/76	0	Not Applicable	
1.99	Effects of Residual Elements on Predicted Radiation Damage to Reactor Vessel Materials	7/75	0	5.3.1.6.7	
1.101	Emergency Planning for Nuclear Power Plants	3/77	1	13.3	A St. Lucie Plant Emergency Plan is submitted under separate cover.
1.102	Flood Protection for Nuclear Power Plants	9/76	1	3.4.1	
1.104	Withdrawn	8/79			
1.107	Qualifications for Cement Grouting for Prestressing Tendons in Containment Structures	2/77	1	Not Applicable	
1.108	Periodic Testing of Diesel Generator Units used as Onsite Electric Power Systems at Nuclear Power Plants	8/77	1	8.3.1.2	This regulatory guide was issued after the CP issuance date of May 2, 1977 on St. Lucie Unit 2.
1.109	Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I	3/76	0	11.2.3/11.3.3	
1.111	Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine Releases from Light-Water-Cooled Reactors	3/76	0	2.3.5 11.2.3/11.3.3	
1.112	Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Light-Water-Cooled Power Reactors	5/77	O-R	11.2.3/11.3.3	
1.113	Estimating Aquatic Dispersion of Effluents from Accidental and Routine Reactor Releases for the Purpose of Implementing Appendix I	4/77	1	11.2.3	

TABLE 1.8-1 (Cont'd)

Number	Title	Date	Revision	Discussion in Subsection(s)	Remarks
1.114	Guidance on Being Operator at the Controls of a Nuclear Power Plant	11/76	1		Operator training is discussed in Chapter 13.
1.116	Quality Assurance Requirements for Installation, Inspection, and Testing of Mechanical Equipment and Systems	-	-	17.2	The revision and date of this document endorsed for St Lucie Unit 2 is governed by the latest revision of the FP&L Topical Quality Assurance Report as referenced in Section 17.2 of the FSAR.
1.119	Withdrawn 6/77				
1.121	Bases for Plugging Degraded PWR Steam Generator Tubes	8/76	0		Steam generator tube corrosion allowance is addressed in Subsection 5.4.4.2.
1.123	Quality Assurance Requirements for Control of Procurement of Items and Services for Nuclear Power Plants	-	-	17.2	The revision and date of this document endorsed for St. Lucie Unit 2 is governed by the latest revision of the FP&L Topical Quality Assurance Report as referenced in Section 17.2 of the FSAR.
1.126	An Acceptable Model and Related Statistical Methods for the Analysis of Fuel Densification	3/77	0		The subject of fuel density is discussed in Subsection 4.2.1.2.4.3.
1.127	Inspection of Water-Control Structures Associated with Nuclear Power Plants	4/77	0	Not Applicable	There are no water-control structures specifically built for use in conjunction with this plant and whose failure could have radiological consequences adversely affecting the public health and safety.
1.129	Maintenance, Testing, and Replacement of Large Lead Storage Batteries for Nuclear Power Plants	4/77	0		The implementation section for this regulatory guide states that this regulatory guide is used in the evaluation of CP applicants docketed after December 1, 1977.
1.144	Auditing of Quality Assurance Programs for Nuclear Power Plants	-	-	17.2	The revision and date of this document endorsed for St. Lucie Unit 2 is governed by the latest revision of the FP&L Topical Quality Assurance Report as referenced in Section 17.2 of the FSAR.
1.146	Qualification of Quality Assurance Program Audit Personnel for Nuclear Power Plants	-	-	17.2	The revision and date of this document endorsed for St. Lucie Unit 2 is governed by the latest revision of the FP&L Topical Quality Assurance Report as referenced in Section 17.2 of the FSAR.
1.183	Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors	7/00	0		Provides guidance on the performance of AST dose analyses for design basis accidents.

1.9 OTHER CONCERNS AND COMMITMENTS

1.9.1 TMI ACTION PLAN

Appendix 1.9A depicts those TMI Action Plan[1] requirements as described in NUREG-0737[2] for St. Lucie Unit 2. UFSAR Subsections discussing TMI "Lessons Learned" are delineated in Appendix 1.9A.

1.9.2 UNDERGROUND CABLE REVIEW

Kerite insulated power and control cables have been reviewed and approved by the NRC for underground wet/dry environmental qualification.[3]

1.9.3 REPLACEMENT STEAM GENERATORS

As a result of tube degradation, Florida Power & Light Company replaced the original steam generators (OSGs) with two replacement steam generators (RSGs) manufactured by AREVA. Specific UFSAR text pertinent to the installation and operation of the RSGs was updated as necessary.

SECTION 1.9: REFERENCES

1. NUREG - 0660, May 1980 "NRC Action Plan Developed as a Result of the TMI-2 Accident.”

2. NUREG - 0737, Letter dated October 31, 1980, D G Eisenhut (NRC) to all Licensees of Operating Plants and Applicants for Operating Licenses and Holders of Construction Permits, Subject "Post-TMI Requirements."

3. Letter dated January 31, 1978, K Kniel (NRC) to R E Uhrig (FP&L), "Use of Kerite Insulated Cable."

APPENDIX 1.9A

1.9A TMI RELATED REQUIREMENTS

The following item numbers correspond to those listed in NUREG-0737 "Clarification of TMI Action Plan Requirements" (October, 1980)[1] . NRC staff documented reviews and approval of these TMI related requirements are given by references to the Safety Evaluation Report[2] and its supplements.[3-6]

I.A.1.1 SHIFT TECHNICAL ADVISOR

Florida Power & Light Co (FP&L) programs in response to this requirement have been developed for St. Lucie Unit 1 (Docket No. 50-335) and are also applicable to St. Lucie Unit 2.

I.A.1.2 SHIFT SUPERVISOR ADMINISTRATIVE DUTIES

FP&L programs in response to this requirement have been developed for St. Lucie Unit 1 (Docket No. 50-335) and are also applicable to St. Lucie Unit 2.

I.A.1.3 SHIFT MANNING

Procedures reflecting the requirements of NUREG-0737 and Generic Letter 82-16 in limiting overtime, hours of work and minimum shift complement have been generated for St. Lucie Unit 1 and also apply to St. Lucie Unit 2.

I.A.2.1 IMMEDIATE UPGRADING OF OPERATOR AND SENIOR OPERATOR TRAINING AND QUALIFICATIONS

Unit Staff qualifications are delineated in the plant Technical Specifications Section 6.3.

I.A.2.3 ADMINISTRATION OF TRAINING PROGRAMS

Training is covered by Section 6.4 of the plant Technical Specifications.

I.A.3.1 REVISE SCOPE AND CRITERIA FOR LICENSING EXAM

FP&L initial and requalification training program revisions to address the increased scope of the license exams have been developed for St. Lucie 1 (Docket No. 50-335) and are also applicable to St. Lucie Unit 2.

I.B.1.2 EVALUATION OF ORGANIZATION AND MANAGEMENT

The FP&L organization is provided in the FPL Quality Assurance Topical Report discussed in Section 17.2. The principal function of the Independent Safety Engineering Group as indicated by NUREG-0737 is assessment of operating experience. This function is the responsibility of the Engineering Manager and the Performance Improvement Manager.

I.C.1 SHORT TERM ACCIDENT ANALYSIS AND PROCEDURE REVISION

The Combustion Engineering (CE) Owners' Group revised analysis and guidelines contained in CEN-152[7] were reviewed. Meetings were held with representatives of the CE Owners' Group in

Bethesda, Maryland, on June 23, 24, and 29, 1982 to discuss NRC's preliminary comments on the analysis and guidelines. At a follow-up meeting in Bethesda on August 20, 1982, a revised CEN-152 was submitted which addressed a majority of the NRC staff concerns discussed at the June meetings. This revised document is now under review. Until the revised analysis and guidelines are approved, CEN-117 and CEN-128 are being used as interim technical bases for the St. Lucie Plant Unit No. 2 emergency operating procedures.

Based on their review of selected emergency operating procedures and their observation of these procedures being exercised on a simulator and in a control room walk-through, as described in Item I.C.8, NRC has concluded that the interim guidelines have been adequately incorporated into the procedures. Further revision to the procedures is expected to be necessary when the revised analysis and guidelines are approved. This satisfies the requirements of Item I.C.1, as per SER Supplement 4.[5]

I.C.2 SHIFT RELIEF AND TURNOVER PROCEDURES

The FP&L program in response to this requirement has been developed for St. Lucie 1 (Docket No. 50-335) and also is applicable to St. Lucie Unit 2.

I.C.3 SHIFT SUPERVISOR RESPONSIBILITIES

The FP&L program in response to this requirement has been developed for St. Lucie 1 (Docket No. 50-335) and also is applicable to St. Lucie Unit 2.

I.C.4 CONTROL ROOM ACCESS

The FP&L program in response to this requirement has been developed for St. Lucie 1 (Docket No. 50-335) and also is applicable to St. Lucie Unit 2. Access limitations are also addressed in the site Security Plan.

I.C.5 PROCEDURES FOR FEEDBACK OF OPERATING EXPERIENCE TO PLANT STAFF

Procedures have been generated to reflect the requirements of NUREG-0737. Administrative controls are addressed in the FPL Quality Assurance Topical Report discussed in Section 17.2.

I.C.6 VERIFY CORRECT PERFORMANCE OF OPERATING ACTIVITIES

Performance and procedures currently in effect at St. Lucie Unit 1 reflect the requirements of NUREG-0737. This requirement is also met at St. Lucie Unit 2. Reviews and audits are covered in the FPL Quality Assurance Topical Report discussed in Section 17.2.

I.C.7 NSSS VENDOR REVIEW OF PROCEDURES

The NRC reviewed selected emergency operating procedures as described in SER Supplement 2 and concluded that the NSSS vendor's comments have been acceptably incorporated into the selected emergency operating procedures.

I.C.8 PILOT MONITORING OF SELECTED EMERGENCY PROCEDURES FOR NTOL APPLICANTS

Any deficiencies identified by an NRC audit were corrected.

I.C.9 LONG TERM PROGRAM PLAN FOR UPGRADING PROCEDURES

Generic Letter 82-33[8] requests that each licensee and applicant develop and submit to the NRC by April 15, 1983 its own plant-specific schedule for completion of the upgrading and implementation of Emergency Operating Procedures (EOPs). FP&L has upgraded and implemented the EOPs.

I.D.1 CONTROL ROOM DESIGN

Generic Letter 82-33 requests that each licensee and applicant develop and submit to the NRC by April 15, 1983 its own plant-specific schedule for submittal of the Control Room Design Review Program Plan and of the Summary Report. FPL has submitted the Summary Report of the Detailed Control Room Design Review (DCRDR), dated October 1983. The history and methodology of the DCRDR is presented in UFSAR Section 7.7.3.

I.D.2 PLANT SAFETY PARAMETER DISPLAY SYSTEM

The Safety Assessment System (SAS)/Emergency Response Data Acquisition And Display System (ERDADS) (refer to Appendix 7.5A) provides the Safety Parameter Display System (SPDS) and all other data required in the control room. The ERDADS system also provides data to Technical Support Center (TSC), Emergency Offsite Facility (EOF) and the Nuclear Data Link (NDL) through the PI server.

Generic Letter 82-33 requests that each licensee and applicant develop and submit to the NRC by April 15, 1983 its own plant-specific schedule for completion of the SPDS and submittal of the SAR and SPDS Implementation Plan. By letter L-83-238 dated April 15, 1983, FP&L indicated the following:

a. The SPDS is operable and the operators were trained by the end of the first refueling outage.

b. The SAR and SPDS Implementation Plan have been submitted.[15]

I.G.1 TRAINING DURING LOW - POWER TESTING

This training is in accordance with Robert L. Tedesco, Assistant Director for Licensing to Dr. Robert E. Uhrig letter dated June 12, 1981. (Subject, TMI-2 Action Plan Item I.G.1). Since testing was accomplished at a comparable prototype plant, SONGS-2, only the training required by this letter need be accomplished.

II.B.1 REACTOR COOLANT SYSTEM VENTS

A description of the Reactor Coolant System Vents is provided in Subsection 9.3.7.

II.B.2 PLANT SHIELDING

A design review was conducted to evaluate the radiological environment of the plant following an accident in which significant core damage has occurred. The evaluation provides for access to vital areas and equipment needed for post-accident operations. A detailed description and results of this design review is provided in Appendix 12.3A.

For environmental qualification of safety-related equipment for post-accident conditions refer to Section 3.11.

II.B.3 POST-ACCIDENT SAMPLING

A description of the Post-Accident Sampling System is provided in Subsection 9.3.6.

II.B.4 TRAINING FOR MITIGATING CORE DAMAGE

Training criteria are discussed in Section 6.4 of the plant Technical Specifications.

II.D.1 RELIEF AND SAFETY VALVE TEST REQUIREMENTS

The design and testing of these valves are summarized in Table 5.4-9 and Subsection 5.4.13.

FP&L's letter of March 22, 1983 from Mr. R Uhrig, FP&L to Mr. D Eisenhut, NRC, references two Combustion Engineering Topical Reports [9, 10] as documentation as to how the EPRI/NSAC test results are applicable to the St. Lucie 2 relief and safety valves.

The staff finds that the general approach in the reports of using the EPRI test results to demonstrate plant specific operability of the relief and safety valves is acceptable (see SER Supplement 3).

NUREG-0737 required utilities to evaluate the functional performance capabilities of PWR safety, PORV, and block valves and to verify the piping systems for normal, transient, and accident conditions. Reference 18 documents the NRC review and acceptance of performance capabilities of pressurizer safety valves, PORVs, and block valves. Qualification of the plant specific piping systems by performing the appropriate analyses is still under evaluation.

II.D.3 RELIEF AND SAFETY VALVE POSITION INDICATION

Acoustic flow monitors are used for the indication of pressurizer safety relief and power operated relief valve position. Design information is presented in Subsection 7.6.3.10.

II.E.1.1 AUXILIARY FEEDWATER SYSTEM RELIABILITY EVALUATION

a. A standard deterministic type of safety review has been performed using as principal guidance the acceptance criteria specified in Standard Review Plan 10.4.9 "Auxiliary Feedwater System" (R1) and Branch Technical Position ASB 10-1, "Design Guidelines for Auxiliary Feedwater System Pump Drive and Power Supply Diversity for PWR Plants" (R0). The results of this review are provided in Appendix 10.4.9A.

b. The guidelines of Enclosure 2 of NRC letter to pending OL applicants dated March 10, 1980[11] has been addressed to describe the design basis accident and transients and the corresponding acceptance criteria for Auxiliary Feedwater System in Appendix 10.4.9A.

c. Event tree and fault tree logic techniques have been conducted as part of a reliability analysis to determine dominant failure modes and assess Auxiliary Feedwater System reliability levels. The results of this reliability evaluation are provided in Appendix 10.4.9B.

II.E.1.2 AUXILIARY FEEDWATER INITIATION AND INDICATION

Safety Grade Auxiliary Feedwater Flow indication and automatic initiation is implemented for St. Lucie Unit 2 and is described in Subsections 10.4.9, 7.3.1.1.8, and 7.5.

II.E.3.1 EMERGENCY POWER SUPPLY FOR PRESSURIZER HEATERS

St. Lucie Unit 2 employs a Combustion Engineering (CE) pressurized water nuclear steam supply system. An analysis performed by CE for St. Lucie Unit 2 has determined that 150 kilowatts of pressurizer heater capacity is needed to maintain hot standby conditions when offsite power is lost. CE recommends this minimum pressurizer heater capacity be available within two hours following loss of offsite power.

The St. Lucie Unit 2 design provides two heater banks each rated 200 kilowatts which are connected to separate 400-volt emergency power trains. The emergency power trains are energized from separate and independent diesel generators upon loss of offsite power. Each of the two heater banks has access to only one Class 1E diesel generator and their controls are likewise supplied from separate safety-grade power supplies. The pressurizer heaters are automatically shed from the Class 1E power system upon the occurrence of a Safety Injection Actuation Signal (SIAS). Procedures for manually loading the pressurizer heaters onto the emergency power sources following an SIAS are available to the operator, and identify under what conditions selected loads can be shed from the emergency bus to prevent overloading of the diesel generators when the pressurizer heaters are connected. The connection of the pressurizer heater elements and controls to the Class 1E buses is through safety-grade circuit breakers.

Based on NRC review, the staff concludes that the power supplies for pressurizer heaters are capable of being powered from both offsite and onsite emergency power systems. This is consistent with the staff positions and clarifications and is acceptable, as per the Safety Evaluation Report.

II.E.4.1 DEDICATED HYDROGEN PENETRATIONS

This requirement is not applicable to St. Lucie Unit 2. EC295353

II.E.4.2 CONTAINMENT ISOLATION DEPENDABILITY

The following items address corresponding NRC positions contained in NUREG-0737:

a. As discussed in Subsection 7.3.1.1 the containment isolation actuation signal (CIAS) is initiated upon high containment pressure, high containment radiation or on SIAS actuation. Therefore, the CIAS complies with the recommendation in Standard Review Plan 6.2.4 "Containment Isolation System" (R1) with respect to diversity in the parameters sensed for initiation of containment isolation.

b. Using the definition in Appendix A to the Branch Technical Position APCSB 3-1 (11/24/75) (attached to Standard Review Plan 3.6.1), essential system and components are defined as those systems and components required to shutdown the reactor and mitigate the consequences of an accident. Table 6.2-52 identifies the essential penetrations as ESF penetrations. As indicated in Subsection 6.2.4, containment penetrations associated with nonessential systems are either administratively locked closed or automatically isolated upon a CIAS. Penetrations for systems like post-accident monitoring instrumentation and RCS sampling however are provided with manual override of the CIAS to enable the operator to open the containment isolation valves and activate the systems as necessary.

c. The St. Lucie Unit 2 containment isolation system complies with General Design Criteria (GDC) 55, 56, and 57. A CIAS is used to isolate nonessential systems. GDC 57 permits the use of one containment isolation valve located outside containment which is capable of automatic or remote manual operation and does not require closure on a CIAS. The penetrations that fall into this category are main steam and feedwater which are automatically isolated upon receipt of a MSIS. However, with the diversity of high containment pressure or low steam generator pressure, an MSIS is generated and isolates the main steam isolation valves and Main Feedwater isolation valves. The component cooling water lines to and from the reactor coolant pump fall under the requirements of GDC 56. An SIAS isolates these penetrations and is initiated by diverse parameters: low pressurizer pressure or high containment pressure.

d. The present design of control systems for automatic containment isolation valves is such that resetting the isolation signal does not result in the automatic reopening of containment isolation valves. Certain valves (e.g., post-accident sampling, instrument air) which are required to open during an accident are provided with the capability of manually overriding the automatic isolation signal. Reopening of these containment isolation valves requires deliberate operator action, and is accomplished only on a valve-by-valve basis. The containment isolation design does not utilize “ganged” control switches for containment isolation valves.

e. A review of the operating history of containment pressure for St. Lucie Unit 1 was performed. (St. Lucie Units 1 & 2 have similar containment volumes and thermal power ratings). Pressure increases of up to two psi can be expected to occur from time to time during plant operation. The instrument loop error, including setpoint variances, effects of line voltage fluctuations, temperature effects and instrument drift is incorporated in the plant Technical Specification setpoint values.

f. The containment purge valves comply with the operability criteria provided in Branch Technical Position CSB 6-4 (R1) and are maintained and surveyed pursuant to the plant Technical Specifications.

   The 48 inch purge valves are verified to be closed at least every 31 days.

g. The continuous containment purge valves close on a CIAS which, as stated in Item 1, is initiated upon a high radiation or high pressure inside containment.

II.F.1 ADDITIONAL ACCIDENT MONITORING INSTRUMENTATION

Generic Letter No. 82-33 requests that each licensee and applicant develop and submit to the NRC by April 15, 1983 its own plant specific schedule for submittal of the Regulatory Guide (RG) 1.97 Evaluation Report describing how RG 1.97 has been met. FP&L submitted this material in Letter L-83-573.

For a discussion of the FPL compliance with RG 1.97 see Section 7.5.2.9

II.F.2 INSTRUMENTS FOR CORE COOLING

Description of Instruments for Core Cooling is provided in Subsections 3.9.5.1.5 and 7.5.4.

II.G.1 EMERGENCY POWER FOR PRESSURIZER EQUIPMENT

The description of the operation of the PORV and PORV block valves is found in Subsection 5.2.6.

The PORVs are powered from safety-related 125V dc buses 2A and 2B and are available continuously. The PORV block valves are powered from safety-related 480V ac motor control centers which are powered through the onsite distribution system. Upon loss of offsite power, the diesel generator is started and powers the onsite system (refer to Section 8.3). Therefore, the PORV block valves receive reliable power in the event they are required to operate during a loss of offsite power. The design is acceptable to NRC as per the SER.

II.K.1 IE BULLETINS ON MEASURES TO MITIGATE SMALL-BREAK LOCAS AND LOSS OF FEEDWATER ACCIDENTS

As per the requirements of NUREG-0737, only two concerns under this item are applicable to St. Lucie Unit 2. These concerns are addressed below.

II.K.1.5 REVIEW ESF VALVES

All safety-related valve positions, positioning requirements, and positive controls were reviewed, and documented in Table 1.9A-1, to assure that valves remain positioned (open or closed) in a manner to ensure the proper operation of engineered safety features.

The provision of complete display of instrumentation is an integral part of the design of systems required for safe shutdown and accident mitigation. A major component of the display information provided in the control room is position indication for valves and HVAC dampers. Table 1.9A-1 lists all active valves and dampers that may be required to operate to achieve safe shutdown or mitigate the consequences of an accident. For most valves and dampers position indicating lights are provided on control panels in the control room. For all other valves and dampers whose failure might have adverse consequences, sufficient information is available for position determination in the control room (refer to Table 1.9A-1).

The related procedures for maintenance, testing, plant and system startup and supervisory periodic surveillance require that these valves are returned to their correct positions following necessary manipulation and are maintained in their proper position during all operational modes. These procedures have been developed in response to this NUREG-0737 requirement for St. Lucie Unit 1 (Docket No. 50-335) and also are applicable to St. Lucie Unit 2.

II.K.1.10 OPERABILITY STATUS

FP&L programs in response to this requirement have been developed for St. Lucie Unit 1 (Docket No. 50-335) and also are applicable to St. Lucie Unit 2. As indicated in NUREG-0660 (not clarified by NUREG-0737) for units applying for operating licenses, this item is addressed in Items I.D.2 and I.C.6 above.

II.K.2.13 THERMAL MECHANICAL REPORT-EFFECT OF HIGH PRESSURE

FP&L is participating in CE Owners Group generic efforts to evaluate the effect of high pressure safety injection on reactor vessel integrity in response to Item II.K.2.13 of NUREG-0737 (see Subsection 5.3.3.8). FP&L concurs with the CE Owners Group evaluation as reported in CEN 189 and CEN 189 Appendix F, December 1981. Staff review of this item is covered in their Unresolved Safety Issues program, issue A-49, “Pressurized Thermal Shock." See SER Supplement 2.

II.K.2.17 POTENTIAL FOR VOIDING IN THE REACTOR COOLANT SYSTEM DURING TRANSIENTS

II.K.2.17.1 DESCRIPTION

In the event a void formation is identified in the Reactor Coolant System the operators are trained to implement a procedure to mitigate voiding. The NSSS vendor has completed an extensive analysis of voiding in the Reactor Coolant System. The results show that rapid refill and drain of the reactor vessel head does not cause stress levels in excess of those occurring during a normal cooldown at 100°F/hour. The results of this analysis for St. Lucie Unit 1, which is applicable to St. Lucie Unit 2, are provided in Appendix 5.2C.

Reactor Coolant System Cooldown rate is addressed in Amendment 4, Subsection 5.4.7.5. FP&L is also participating in the CE Owners Group effort to address item II.K.2.17; FP&L concurs with the evaluation as reported in CEN-199[12].

II.K.2.19 SEQUENTIAL AUXILIARY FEEDWATER FLOW ANALYSIS

As indicated by the NRC (letter from R A Clark, Chief Operating Reactor Branch 3, Division of Licensing to R E Uhrig, Vice President, Florida Power & Light Co., dated July 2, 1981), this item is not applicable to CE supplied steam generators which utilize inverted U tubes.

II.K 3.1 INSTALLATION AND TESTING OF AUTOMATIC PORV ISOLATION SYSTEM

II.K.3.2 REPORT ON OVERALL SAFETY EFFECT OF PORV ISOLATION SYSTEM

FP&L has participated in CE Owners Group activities conducted since the Three Mile Island accident to address various aspects of PORV design and operation. These activities have included review of operating experience with PORVs on CE reactors, development of input to the EPRI program for testing these valves, review of requirements for emergency power to the PORVs and the associated block valves, development of a recommendation for PORV position indication, review and updating of emergency procedure guidelines to assure PORV operation is adequately addressed, and development of associated operator training materials. The requirements of Action Plan Item II.K.3.2 have also been addressed as a CE Owners Group activity in CEN-145(13).

It has been concluded based on the CE Owners Group activities that the addition of an automatic PORV isolation system on St. Lucie Unit 2 to further decrease the probability of a small-break loss-of-coolant accident caused by a stuck-open PORV is not necessary. This conclusion is based on the following considerations. First, the design of the PORV actuation logic is such that the valves are only actuated coincident with the high pressurizer pressure trip of the reactor. The PORV cases are not used prior to the Reactor Protection System actuation in an attempt to avoid the reactor trip. Thus, challenges to the PORVs are reduced because the margin between the normal operating pressure and the high pressure reactor trip is maximized. The success of this design approach is evident based on the operating experience compiled to date which has only 19 challenges to the PORVs in 29 reactor-years of operation on CE plants (data from a recent survey of the CE Owners Group). It should be noted that 11 of these 19 challenges were caused by a turbine runback feature which has been removed. The PORVs successfully reclosed in each case where they were challenged.

The second consideration for not needing an automatic PORV isolation system is that various actions have been taken which significantly improve the reliability of the PORVs and associated block valves. The elimination of the turbine runback feature mentioned previously, and the provision of a direct reliable means for indicating PORV position to the operator reduce the recurrence frequency of a small break LOCA due to PORV failure by an estimated factor of 15. Improved operator training programs, improved emergency procedures, and the provision of emergency power to the PORVs and block valves reduce the small break LOCA recurrence frequency further although the exact magnitude has not been quantified.

The final consideration for not needing an automatic PORV isolation system is that the recurrence frequency of a small break LOCA due to PORV failure has been substantially reduced by the actions mentioned previously to an estimated value which falls well within the uncertainty band of the recurrence frequencies for a LOCA due to a small pipe rupture estimated in WASH-1400.

Thus, the recurrence frequency is now at an acceptably low value. The incorporation of an automatic PORV isolation system would further increase PORV system reliability. However, this action is not considered to be necessary since the recurrence frequency of PORV system failures without this feature is small.

II.K.3.3 REPORTING SAFETY VALVE AND PORV FAILURE AND CHALLENGES

FP&L assures that any failure of a PORV or safety valve to close is reported to the NRC promptly. All challenges to the PORVs or safety valves are documented in a Special Report.

II.K.3.5 AUTOMATIC TRIP OF REACTOR COOLANT PUMPS DURING A LOCA

FP&L is a member of the CE Owners Group. The CE Owners Group has selected an operational strategy which will close out TMI Action Plan Item II.K.3.5, "Automatic Trip of Reactor Coolant Pumps." Following a current review of several possible strategies, the strategy chosen is to trip two pumps initially followed by the trip of the remaining two pumps at the same time a LOCA has been diagnosed. The "trip two, leave two" strategy has been discussed in the past as a preferred approach. Based on the currently available information, it remains the preference of CE, the CE Owners Group, and FP&L. A program is being developed whose goal will be to provide information which both meets the NRC guidelines stated in the reference letter and provides the operational requirements for participating utilities to use in developing emergency operating procedures and conducting training. The expectation is that the selected operational strategy for the RCPs will make use of manual operator actions. The operational strategy currently in use on St. Lucie Unit No. 2 is to trip all RCPs during the initial phase of a depressurization transient followed by pump restart when it is confirmed that the event is not a LOCA. This strategy will remain in effect until replaced by the new approach which will be implemented with supported by appropriate documentation and operator training.

The NRC, in Reference 17, has concluded that the CE Owner's Group methodology significantly improves reactor safety. The adoption and implementation of this methodology resolves TMI Action Plan Item II.K.3.5 satisfactorily.

II.K.3.17 REPORT ON OUTAGES OF EMERGENCY CORE-COOLING SYSTEMS LICENSEE REPORT AND PROPOSED TECHNICAL SPECIFICATION CHANGES

Reports on ECCS outages will follow the guidelines of 10 CFR 50.73 for the development and content of License Event Reports which will document any significant problems with the ECCS equipment. Other ECCS equipment failures are reported via the Institute of Nuclear Power Operations (INPO) Equipment Performance and Information Exchange System (EPIX), formally known as Nuclear Plant Reliability Data System (NPRDS). These two methods provide an on-line reporting system which satisfies the requirements of NUREG-0737, Item II.K.3.17.

These methods were accepted by the NRC in Reference 16.

II.K.3.25 EFFECT OF LOSS OF AC POWER ON PUMP SEALS

FP&L has conducted a test of RCP seals under simulated loss of ac power conditions of full temperature and pressure. After approximately 50 hours at coolant conditions of 550°F and 2250 psig, the RCP seal cartridge still performed satisfactorily with the pump idle. Some seal damage was observed during the post-test inspection; however, the maximum seal leakage during the test was only 16 gph (Reference: FP&L letter L-81-107, March 10, 1981).

PCM 98021 replaced the RCP SU mechanical seals with N-9000 seals. An aged N-9000 seal has been rigorously tested by Flowserve (OEM) in a test fixture to simulate the conditions imposed by a station blackout for an eight (8) hour period. During this test downward shaft movements and pressure changes were imposed.

II.K.3.30 REVISED SMALL BREAK LOCA METHODS TO SHOW COMPLIANCE WITH 10 CFR PART 50, APPENDIX K

NRC Generic Letter 83-10b[(14)] documents NRC evaluation of the analyses of LOFT Test L3-6 performed by the CE Owners Group and concludes that the evaluations acceptably predict the test results, and finds the currently approved CE evaluation model for small LOCAs in continued conformance with 10 CFR 50 Appendix K for the case of limited RCP operation after reactor trip, and for the range of licensed CE reactor designs.

II.K.3.31 PLANT SPECIFIC CALCULATIONS TO SHOW COMPLIANCE WITH 10 CFR PART 50.46

See Item II.K.3.30 of NUREG-0737.

III.A.1.1 UPGRADE EMERGENCY PREPAREDNESS

The St. Lucie Plant Emergency Plan discussed in Section 13.3 incorporates the requirements of this task.

III A.1.2 UPGRADE EMERGENCY SUPPORT FACILITIES

FP&L programs in response to this requirement have been or are being developed for St. Lucie Unit 1 (Docket No. 50-335) and also are applicable to St. Lucie Unit 2.

Generic Letter No. 82-33 requests that each licensee and applicant develop and submit to the NRC by April 15, 1983 its own plant-specific schedule for completion of the Emergency Response Facilities (ERFs). By letter L-83-238 dated April 15, 1983, FP&L indicated the ERFs schedule is as follows:

a. Technical Support Center (TSC)

   The TSC is operational.

b. Operational Support Center (OSC)

   The OSC is operational.

c. Emergency Operation Facility (EOF)

   The EOF is operational.

III.D.1.1 INTEGRITY OF SYSTEMS OUTSIDE CONTAINMENT LIKELY TO CONTAIN RADIOACTIVE MATERIAL

In the unlikely event of an accident, the Containment Isolation Actuation Signal (CIAS) isolates all non-essential systems, thereby eliminating all large radioactive leakage paths from containment. The only means of leakage into the Reactor Auxiliary Building is through ESF system components (i.e., pump seals, valve leakage, etc.) and post-accident monitoring sample lines. Liquid leakages collected in the ECCS room sumps are normally routed to the equipment drain tank in the Waste Management System (WMS). The normal operational mode of the

ECCS room sump pumps has not been modified. On high sump water level, the pumps discharge to the equipment drain tank. To prevent radioactive contaminants from entering the WMS, the ESF Leakage Collection and Return System (see Subsection 9.3.5) provides operators with a method to direct ESF leakage to the containment. This system eliminates highly radioactive liquid from entering normally "Low activity" waste hold-up tanks. Likewise, all sources of high activity sample gas (e.g., hydrogen sampling) are re-routed to the containment, thus eliminating contamination of the Waste Gas System. The above described design precludes the use of Liquid and Gaseous Waste Management Systems during an unlikely event of an accident.

The following systems contain high activity fluid during a postulated accident:

a. Shutdown Cooling System

b. High Pressure Safety Injection (Recirculation Phase)

c. Containment Spray (Recirculation Phase)

d. Sampling System.

Periodic integrated leak testing, at intervals not to exceed each refueling cycle, is established for these systems. A program is established to evaluate results and initiate leakage reduction measures as appropriate.

III.D.3.3 IN PLANT RADIATION MONITORING

FP&L programs in response to this requirement have been developed for St. Lucie Unit 1 (Docket No. 50-335) and are applicable to St. Lucie Unit 2. The Health Physics procedures address detailed radioiodine assessment. These are generally described in Subsection 12.5.3. Training is an integral part of the non-licensed training program is covered in the plant Technical Specifications.

III.D.3.4 CONTROL ROOM HABITABILITY

Potential hazards in the vicinity of the site have been identified and evaluated to confirm that operators in the control room are adequately protected (refer to Section 2.2). In addition, radioactive releases have been analyzed for their effects on control room operators (refer to Section 6.4). Liquid source terms from within the Reactor Auxiliary Building, although not factored into the dose rate to the operators presented in Section 6.4, would have insignificant impact in terms of doses because the control room itself is located on top of the Reactor Auxiliary Building and is well separated from liquid source terms.

REFERENCES: APPENDIX 1.9A

1. U.S. Nuclear Regulatory Commission, "Clarification of TMI Action Plan Requirements," USNRC Report NUREG-0737, October, 1980.

2. NUREG-0843, Safety Evaluation Report related to the operation of St. Lucie Plant, Unit No. 2, Docket No. 50-389; October 1981.

3. NUREG-0843, Supplement No. 1; December 1981.

4. NUREG-0843, Supplement No. 2; September 1982.

5. NUREG-0843, Supplement No. 3; April 1983.

6. NUREG-0843, Supplement No. 4; June 1983.

7. CEN-152, "Combustion Engineering Emergency Procedure Guidelines," dated November 22, 1982.

8. NRC Generic Letter 82-33, Supplement 1 to NUREG-0737 - Requirements for Emergency Response Capability, dated December 17, 1982.

9. CEN-227, "Summary Report on the Operability of Pressurizer Safety Valves in CE Designed Plants," December 1982.

10. CEN-213, "Summary Report on the Operability of Powered Operated Relief Valves," July 1982.

11. Letter from D. F. Ross Jr., NRC to All Pending Operating License Applicants of Nuclear Steam Supply Systems Designed by Westinghouse and Combustion Engineering, Subject: Actions Required from Operating License Applicants of Nuclear Steam Supply Systems Designed by Westinghouse and Combustion Engineering Resulting from the NRC Bulletins and Orders Task Force Review Regarding the Three Mile Island Unit 2 Accident, dated March 10, 1980.

12. CEN-199, "Effects of Vessel Head Voiding During Transients and Accidents in CE-NSSS's," March 1982.

13. CEN-145, "PORV Failure Reduction Methods-Final Report," December 1980.

14. NRC Generic Letter No. 83-10b, Resolution of TMI Action Items II.K.3.5, "Automatic Trip of Reactor Coolant Pumps," dated February 8, 1983.

15. FPL Letter L-84-49 dated March 1, 1984 from Mr. J. W. Williams, Jr. to Mr. D.G. Eisenhut, "SPDS Implementation Plan and Parameter Selection Report."

16. Letter, from E. G. Tourigny (NRC) to W. F. Conway (FPL), "Emergency Core Cooling Systems (ECCS) Outages, 5-Year Report - St Lucie Plant Unit No. 2," dated May 11, 1988.

17. Letter, from J. A. Norris/G.E. Edison (NRC) to W.F. Conway (FPL), "Closing of Multiplant Action G-01-Reactor Coolant Pump Trip (NUREG-0737 Item II.K.3.5)," dated March 15, 1989

18 Letter, from J. A. Norris (NRC) to C.O. Woody (FPL), "NUREG-0737 Item II.D.1 Performance Testing of Relief and Safety Valves," dated May 11,1989.

19. NRC letter, J. A. Norris to J. H. Goldberg, "Instrumentation To Follow The Course Of An Accident (Regulatory Guide 1.97)", dated April 1, 1992

20. FPL Letter to NRC, L-92-194, "St. Lucie Unit 1 and 2 Docket No. 50-335 and 50-389 Regulatory Guide 1.97", dated July 14, 1992

21. FPL Letter L-92-28 from D. A. Sager to NRC, dated February 10, 1992

22. NRC letter, J. A. Norris to J. H. Goldberg, "ST. Lucie Units 1 And 2 - Proposed Modifications Related To Regulatory Guide 1.97 (TAC Nos. 64333 And 64334)", dated November 12, 1991

TABLE 1.9A-1

SAFETY RELATED VALVE POSITION AND POSITION INDICATION

SYSTEM	VALVE	FUNCTION	TYPE	OPERATOR	ACTUATION SIGNAL	NORMAL VALVE POSITION	ACCIDENT VALVE[a] POSITION	FAILURE MODE	METHOD OF[b] POSITION INDICATION
Reactor Coolant	V1474	LTOP	Ang. Glb.	Sol.	---	Closed	Closed	Closed	1
	V1475	LTOP	Ang. Glb.	Sol.	---	Closed	Closed	Closed	1
	V1476	LTOP Isol.	Gate	Motor	---	Open	Closed	As Is	1
	V1477	LTOP Isol.	Gate	Motor	---	Open	Closed	As Is	1
	V1460	RV Head Vent	Glb.	Solenoid	---	Closed	Closed	Closed	1
	V1461	RV Head Vent	Glb.	Solenoid	---	Closed	Closed	Closed	1
	V1462	RV Head Vent	Glb.	Solenoid	---	Closed	Closed	Closed	1
	V1463	RV Head Vent	Glb.	Solenoid	---	Closed	Closed	Closed	1
	V1464	RV Head Vent	Glb.	Solenoid	---	Closed	Closed	Closed	1
	V1465	RV Head Vent	Glb.	Solenoid	---	Closed	Closed	Closed	1
	V1466	RV Head Vent	Glb.	Solenoid	---	Closed	Closed	Closed	1
Chemical and Volume Control	V2522	Cont. Isol.	Glb.	Pneu.	CIAS	Open	Closed	Closed	1
	V2508	BAMT. Isol.	Gate	Motor	SIAS	Closed	Open	As Is	1
	V2509	BAMT. Isol.	Gate	Motor	SIAS	Closed	Open	As Is	1
	V2514	BAMP. Disch.	Gate	Motor	SIAS	Closed	Open	As Is	1
	V2525	PMW Supply	Gate	Motor	SIAS	Closed	Closed	As Is	1
	V2504	RWT Supply	Gate	Motor	---	Closed	Open	As Is	1
	V2515	Cont. Isol.	Glb.	Pneu.	SIAS	Open	Closed	Closed	1
	V2516	Cont. Isol.	Glb.	Pneu.	SIAS/CIAS	Open	Closed	Closed	1
	SE-02-3	Aux. Spray	Glb.	Sol.	---	Locked Closed	Open	Closed	1
	SE-02-4	Aux. spray	Glb.	Sol.	---	Locked Closed	Open	Closed	1
	SE-02-1	Charging	Glb.	Sol.	---	Open	Open	Open	1
	SE-02-2	Charging	Glb.	Sol.	---	Open	Open	Open	1
	V2553	Charg. Bypass	Glb.	Motor	---	Open ©	Closed	As Is	2
	V2554	Charg. Bypass	Glb.	Motor	---	Open ©	Closed	As Is	2
	V2555	Charg. Bypass	Glb.	Motor	---	Open ©	Closed	As Is	2
	V2523	Charg. Isol.	Glb.	Pneu.	---	Locked Open	Open	Open	1
	FCV-2210Y	BAMT Supply	Glb.	Pneu.	SIAS	Closed	Closed	Closed	1
	V2524	Cont.Isol.	Glb.	Pneu.	CIAS	Open	Closed	Closed	1
	V2505	Cont.Isol.	Glb.	Pneu.	CIAS	Open	Closed	Closed	1
	V2501	VCT Isol.	Gate	Motor	SIAS	Open	Closed	As Is	1
	V2650	BAMT Recirc	Glb.	Pneu.	SIAS	Open	Closed	Closed	1
	V2651	BAMT Recirc	Glb.	Pneu.	SIAS	Open	Closed	Closed	1
Safety Injection	FCV-3301	SDC	BFY	Motor	---	Locked Open	Open	As Is	1
	FCV-3306	SDC	BFY	Motor	---	Locked Open	Open	As Is	1
	HCV-3512	SDC	BFY	Motor	---	Locked Closed	Open	As Is	1
	HCV-3657	SDC	BFY	Motor	---	Locked Closed	Open	As Is	1
	V3456	SDC Isol.	Gate	Motor	---	Locked Closed	Open	As Is	1
	V3457	SDC Isol.	Gate	Motor	---	Locked Closed	Open	As Is	1
	V3517	SDC Isol.	Gate	Motor	---	Locked Closed	Open	As Is	1
	V3658	SDC Isol.	Gate	Motor	---	Locked Closed	Open	As Is	1
	V3540	Hot Leg Inj.	Glb.	Motor	---	Locked Closed	Open	As Is	1
	V3550	Hot Leg Inj.	Glb.	Motor	---	Locked Closed	Open	As Is	1

TABLE 1.9A -1(Cont'd)

SYSTEM	VALVE	FUNCTION	TYPE	OPERATOR	ACTUATION SIGNAL	NORMAL VALVE POSITION	ACCIDENT VALVE[a] POSITION	FAILURE MODE	METHOD OF[b] POSITION INDICATION
Safety Injection(Cont'd)	V3523	Hot Leg Inj.	Glb.	Motor	---	Locked Closed	Open	As Is	1
	V3551	Hot Leg Inj.	Glb.	Motor	---	Locked Closed	Open	As Is	1
	V3656	HPSI Isol.	Gate	Motor	---	Locked Open	Open	As Is	1
	V3654	HPSI Isol.	Gate	Motor	---	Locked Open	Open	As Is	1
	SE-03-2A	Cont. Isol.	Glb.	Sol.	SIAS/CIAS	Closed	Closed	Closed	1
	SE-03-2B	Cont. Isol.	Glb.	Sol.	SIAS/CIAS	Closed	Closed	Closed	1
	V3659	Recirc.	Gate	Motor	RAS	Locked Open	Closed	As Is	1
	V3660	Recirc.	Gate	Motor	RAS	Locked Open	Closed	As Is	1
	V3495	Recirc.	Glb.	Sol.	RAS	Locked Open	Closed	Closed	1
	V3611	SIT Drain	Glb.	Pneu.	SIAS	Closed	Closed	Closed	1
	V3621	SIT Drain	Glb.	Pneu.	SIAS	Closed	Closed	Closed	1
	V3631	SIT Drain	Glb.	Pneu.	SIAS	Closed	Closed	Closed	1
	V3641	SIT Drain	Glb.	Pneu.	SIAS	Closed	Closed	Closed	1
	V3496	Recirc.	Glb.	Sol.	RAS	Locked Open	Closed	Closed	1
	HCV-3615	Inj.	Glb.	Motor	SIAS	Closed	Open	As Is	1
	HCV-3625	Inj.	Glb.	Motor	SIAS	Closed	Open	As Is	1
	HCV-3635	Inj.	Glb.	Motor	SIAS	Closed	Open	As Is	1
	HCV-3645	Inj.	Glb.	Motor	SIAS	Closed	Open	As Is	1
	HCV-3616	Inj.	Glb.	Motor	SIAS	Closed	Open	As Is	1
	HCV-3626	Inj.	Glb.	Motor	SIAS	Closed	Open	As Is	1
	HCV-3636	Inj.	Glb.	Motor	SIAS	Closed	Open	As Is	1
	HCV-3646	Inj.	Glb.	Motor	SIAS	Closed	Open	As Is	1
	HCV-3617	Inj.	Glb.	Motor	SIAS	Closed	Open	As Is	1
	HCV-3627	Inj.	Glb.	Motor	SIAS	Closed	Open	As Is	1
	HCV-3637	Inj.	Glb.	Motor	SIAS	Closed	Open	As Is	1
	HCV-3647	Inj.	Glb.	Motor	SIAS	Closed	Open	As Is	1
	V3480	SDC Isol.	Gate	Motor	---	Locked Closed	Open	As Is	1
	V3481	SDC Isol.	Gate	Motor	---	Locked Closed	Open	As Is	1
	V3651	SDC Isol.	Gate	Motor	---	Locked Closed	Open	As Is	1
	V3652	SDC Isol.	Gate	Motor	---	Locked Closed	Open	As Is	1
	V3545	SDC X-Tie	Gate	Motor	---	Locked Open	Open	As Is	1,8
	V3664	SDC Isol.	Gate	Motor	---	Locked Closed	Open	As Is	1
	V3665	SDC Isol.	Gate	Motor	---	Locked Closed	Open	As Is	1
	V3536	SDC Warmup	Glb.	Motor	---	Locked Closed	Open	As Is	1
	V3539	SDC Warmup	Glb.	Motor	---	Locked Closed	Open	As Is	1
	V3614	SIT Isol.	Gate	Motor	SIAS	Locked Open	Open	As Is	1,8
	V3624	SIT Isol.	Gate	Motor	SIAS	Locked Open	Open	As Is	1,8
	V3634	SIT Isol.	Gate	Motor	SIAS	Locked Open	Open	As Is	1,8
	V3644	SIT Isol.	Gate	Motor	SIAS	Locked Open	Open	As Is	1,8
	SE-03-1A	SIT Drain	Glb.	Sol.	SIAS	Closed	Closed	Closed	1
	SE-03-1B	SIT Drain	Glb.	Sol.	SIAS	Closed	Closed	Closed	1
	SE-03-1C	SIT Drain	Glb.	Sol.	SIAS	Closed	Closed	Closed	1
	SE-03-1D	SIT Drain	Glb.	Sol.	SIAS	Closed	Closed	Closed	1
	HCV-3618	CV Leakage	Glb.	Pneu.	SIAS	Closed	Closed	Closed	1
	HCV-3628	CV Leakage	Glb.	Pneu.	SIAS	Closed	Closed	Closed	1

TABLE 1.9A -1(Cont'd)

SYSTEM	VALVE	FUNCTION	TYPE	OPERATOR	ACTUATION SIGNAL	NORMAL VALVE POSITION	ACCIDENT VALVE[(a)] POSITION	FAILURE MODE	METHOD OF[(b)] POSITION INDICATION
Safety Injection(Cont'd)	HCV-3638	CV Leakage	Glb.	Pneu.	SIAS	Closed	Closed	Closed	1
	HCV-3648	CV Leakage	Glb.	Pneu.	SIAS	Closed	Closed	Closed	1
	V3571	Inj. Relief	Glb.	Pneu.	SIAS	Closed	Closed	Closed	1
	V3572	Inj. Relief	Glb.	Pneu.	SIAS	Closed	Closed	Closed	1
	V3444	RWT Isol	Gate	Motor	---	Locked Open	Closed	As Is	1
	V3432	RWT Isol	Gate	Motor	---	Locked Open	Closed	As Is	1
Sampling	SE-05-1A	Cont. Isol.	Glb.	Sol.	CIAS	Closed	Closed	Closed	1
	SE-05-1B	Cont. Isol.	Glb.	Sol.	CIAS	Closed	Closed	Closed	1
	SE-05-1C	Cont. Isol.	Glb.	Sol.	CIAS	Closed	Closed	Closed	1
	SE-05-1D	Cont. Isol.	Glb.	Sol.	CIAS	Closed	Closed	Closed	1
	SE-05-1E	Cont. Isol.	Glb.	Sol.	CIAS	Closed	Closed	Closed	1
	V5200	Cont. Isol.	Glb.	Sol.	CIAS	Closed	Closed	Closed	1
	V5201	Cont. Isol.	Glb.	Sol.	CIAS	Closed	Closed	Closed	1
	V5202	Cont. Isol.	Glb.	Sol.	CIAS	Closed	Closed	Closed	1
	V5203	Cont. Isol.	Glb.	Pneu.	CIAS	Closed	Closed	Closed	1
	V5204	Cont. Isol.	Glb.	Pneu.	CIAS	Closed	Closed	Closed	1
	V5205	Cont. Isol.	Glb.	Pneu.	CIAS	Closed	Closed	Closed	1
SIT Vent Valves	V3733	SIT Vent to Atm.	Glb.	Sol.	---	Closed	Closed	Closed	1
	V3734	SIT Vent to Atm.	Glb.	Sol	---	Closed	Closed	Closed	1
	V3735	SIT Vent to Atm.	Glb.	Sol.	---	Closed	Closed	Closed	1
	V3736	SIT Vent to Atm.	Glb.	Sol.	---	Closed	Closed	Closed	1
	V3737	SIT Vent to Atm.	Glb.	Sol.	---	Closed	Closed	Closed	1
	V3738	SIT Vent to Atm.	Glb.	Sol.	---	Closed	Closed	Closed	1
	V3739	SIT Vent to Atm.	Glb.	Sol.	---	Closed	Closed	Closed	1
	V3740	SIT Vent to Atm.	Glb.	Sol.	---	Closed	Closed	Closed	1
Waste Management	V6341	Cont. Isol.	Diaph.	Pneu.	CIAS	Open	Closed	Closed	1
	V6342	Cont. Isol.	Diaph.	Pneu.	CIAS	Open	Closed	Closed	1
	V6718	Cont. Isol.	Diaph.	Pneu.	CIAS	Open	Closed	Closed	1
	V6750	Cont. Isol.	Diaph.	Pneu.	CIAS	Open	Closed	Closed	1
	V6741	Cont. Isol.	Glb.	Pneu.	CIAS	Open	Closed	Closed	1
Main Steam	HCV-08-1A	Cont. Isol	Glb.	Pneu.	MSIS	Open	Closed	As Is	1
	HCV-08-1B	Cont. Isol	Glb.	Pneu.	MSIS	Open	Closed	As Is	1
	MV-08-1A	Warmup	Glb.	Motor	MSIS	Closed	Closed	As Is	1
	MV-08-1B	Warmup	Glb.	Motor	MSIS	Closed	Closed	As Is	1
	MV-08-18A	ADV	Glb.	Motor	---	Closed	Open	As Is	1
	MV-08-18B	ADV	Glb.	Motor	---	Closed	Open	As Is	1
	MV-08-19A	ADV	Glb.	Motor	---	Closed	Open	As Is	1
	MV-08-19B	ADV	Glb.	Motor	---	Closed	Open	As Is	1
	MV-08-12	Aux. Stm	Gate	Motor	AFAS	Closed	Open	As Is	1
	MV-08-13	Aux. Stm	Gate	Motor	AFAS	Closed	Open	As Is	1

TABLE 1.9A - 1(Cont'd)

SYSTEM	VALVE	FUNCTION	TYPE	OPERATOR	ACTUATION SIGNAL	NORMAL VALVE POSITION	ACCIDENT VALVE[a] POSITION	FAILURE MODE	METHOD OF[b] POSITION INDICATION
Main Steam (Cont'd)	MV-08-3	Aux.Stm	Glb.	Motor	---	Locked Open	Open	As Is	1
	MV-08-14	ADV Isol.	Gate	Motor	---	Open	Open	As Is	1
	MV-08-15	ADV Isol.	Gate	Motor	---	Open	Open	As Is	1
	MV-08-16	ADV Isol	Gate	Motor	---	Open	Open	As Is	1
	MV-08-17	ADV Isol.	Gate	Motor	---	Open	Open	As Is	1
Main Feed Water	HCV-09-1A	Cont. Isol.	Gate	Hyd.	MSIS/AFAS	Open	Closed*	As Is	1
	HCV-09-1B	Cont. Isol.	Gate	Hyd.	MSIS/AFAS	Open	Closed*	As Is	1
	HCV-09-2A	Cont. Isol.	Gate	Hyd.	MSIS/AFAS	Open	Closed*	As Is	1
	HCV-09-2B	Cont. Isol.	Gate	Hyd.	MSIS/AFAS	Open	Closed*	As Is	1
	MV-09-9	Aux. Feed	Glb.	Motor	AFAS	Closed	Open/Closed	As Is	1
	MV-09-10	Aux. Feed	Glb.	Motor	AFAS	Closed	Open/Closed	As Is	1
	MV-09-11	Aux. Feed	Glb.	Motor	AFAS	Closed	Open/Closed	As Is	1
	MV-09-12	Aux. Feed	Glb.	Motor	AFAS	Closed	Open/Closed	As Is	1
	MV-09-13	Aux. Feed	Gate	Motor	---	Closed	Open	As Is	1
	MV-09-14	Aux. Feed	Gate	Motor	---	Closed	Open	As Is	1
	SE-09-2	Aux. Feed Isol.	Glb.	Sol.	AFAS	Closed	Open/Closed	Closed	1
	SE-09-3	Aux. Feed Isol.	Glb.	Sol.	AFAS	Closed	Open/Closed	Closed	1
	SE-09-4	Aux. Feed Isol.	Glb.	Sol.	AFAS	Closed	Open/Closed	Closed	1
	SE-09-5	Aux. Feed Isol.	Glb.	Sol.	AFAS	Closed	Open/Closed	Closed	1
Intake Cooling Water	MV-21-2	Sys. Isol.	BFY	Motor	SIAS	Open	Closed	As Is	1
	MV-21-3	Sys. Isol.	BFY	Motor	SIAS	Open	Closed	As Is	1
Component Cooling Water	HCV-14-8A	Sys. Isol.	BFY	Pneu.	SIAS	Open	Closed	Closed	1
	HCV-14-8B	Sys. Isol.	BFY	Pneu.	SIAS	Open	Closed	Closed	1
	MV-14-17	FP. Isol.	BFY	Motor	SIAS	Open	Closed	As Is	1
	MV-14-18	FP. Isol.	BFY	Motor	SIAS	Closed	Closed	As Is	1
	MV-14-19	FP. Isol.	BFY	Motor	---	Open	Closed	As Is	1
	MV-14-20	FP. Isol.	BFY	Motor	---	Closed	Closed	As Is	1
	MV-14-9	Fan Isol.	BFY	Motor	---	Open	Open	As Is	1
	MV-14-10	Fan Isol.	BFY	Motor	---	Open	Open	As Is	1
	MV-14-11	Fan Isol.	BFY	Motor	---	Open	Open	As Is	1
	MV-14-12	Fan Isol.	BFY	Motor	---	Open	Open	As Is	1
	MV-14-13	Fan Isol.	BFY	Motor	---	Open	Open	As Is	1
	MV-14-14	Fan Isol.	BFY	Motor	---	Open	Open	As Is	1
	MV-14-15	Fan Isol.	BFY	Motor	---	Open	Open	As Is	1
	MV-14-16	Fan Isol.	BFY	Motor	---	Open	Open	As Is	1

*The AFAS maybe overridden and the valve re-opened by the control room operator only during 2-EOP-06, Total Loss of Feedwater.

TABLE 1.9A - 1 (Cont'd)

SYSTEM	VALVE	FUNCTION	TYPE	OPERATOR	ACTUATION SIGNAL	NORMAL VALVE POSITION	ACCIDENT VALVE[a] POSITION	FAILURE MODE	METHOD OF[b] POSITION INDICATION
Component Cooling Water (Cont'd)	HCV-14-1	RCP Isol.	BFY	Pneu.	SIAS	Open	Closed	Closed	1
	HCV-14-2	RCP Isol.	BFY	Pneu.	SIAS	Open	Closed	Closed	1
	HCV-14-6	RCP Isol.	BFY	Pneu.	SIAS	Open	Closed	Closed	1
	HCV-14-7	RCP Isol.	BFY	Pneu.	SIAS	Open	Closed	Closed	1
	HCV-14-9	Sys. Isol.	BFY	Pneu.	SIAS	Open	Closed	Closed	1
	HCV-14-10	Sys. Isol.	BFY	Pneu.	SIAS	Open	Closed	Closed	1
	HCV-14-3A	SDC HX	BFY	Pneu.	SIAS	Closed	Open	Open	1
	HCV-14-3B	SDC HX	BFY	Pneu.	SIAS	Closed	Open	Open	1
	MV-14-1	CCW Pump Isol.	BFY	Motor	---	Open (1)	Open (1)	As Is	1
	MV-14-2	CCW Pump Isol.	BFY	Motor	---	Closed (2)	Closed (2)	As Is	1
	MV-14-3	CCW Pump Isol.	BFY	Motor	---	Open (1)	Open (1)	As Is	1
	MV-14-4	CCW Pump Isol.	BFY	Motor	---	Closed (2)	Closed (2)	As Is	1
Primary Water	HCV-15-1	Cont. Isol.	Glb.	Pneu.	CIAS	Closed	Closed	Closed	1
Instr. Air	HCV-18-1	Cont. Isol.	Glb.	Pneu.	CIAS	Open	Closed	Closed	1
Station Air	HCV-18-2	Cont. Isol.	Glb.	Pneu.	CIAS	Closed	Closed	Closed	1
Steam Generator Blowdown	FCV-23-3	Cont. Isol.	Gate	Pneu.	CIAS	Open	Closed	Closed	1
	FCV-23-5	Cont. Isol.	Gate	Pneu.	CIAS	Open	Closed	Closed	1
	FCV-23-7	Cont. Isol.	Glb.	Pneu.	CIAS	Open	Closed	Closed	1
	FCV-23-9	Cont. Isol.	Glb.	Pneu.	CIAS	Open	Closed	Closed	1
Diesel Oil	SE-59-1A1	Oil Supply	Glb.	Sol.	---	Closed	Open	Closed	3
	SE-59-1A2	Oil Supply	Glb.	Sol.	---	Closed	Open	Closed	3
	SE-59-1B1	Oil Supply	Glb.	Sol.	---	Closed	Open	Closed	3
	SE-59-1B2	Oil Supply	Glb.	Sol.	---	Closed	Open	Closed	3
HVAC	FCV-25-1	Cont. Isol.	BFY	Pneu.	CIAS	Closed	Closed	Closed	1
	FCV-25-2	Cont. Isol.	BFY	Pneu.	CIAS	Closed	Closed	Closed	1
	FCV-25-3	Cont. Isol.	BFY	Pneu.	CIAS	Closed	Closed	Closed	1
	FCV-25-4	Cont. Isol.	BFY	Pneu.	CIAS	Closed	Closed	Closed	1
	FCV-25-5	Cont. Isol.	BFY	Pneu.	CIAS	Closed	Closed	Closed	1
	FCV-25-6	Cont. Isol.	BFY	Pneu.	CIAS	Closed	Closed	Closed	1
	FCV-25-20	Cont. Isol.	BFY	Pneu.	CIAS	Open	Closed	Closed	1
	FCV-25-21	Cont. Isol.	BFY	Pneu.	CIAS	Open	Closed	Closed	1
	FCV-25-26	Cont. Isol.	BFY	Pneu.	CIAS	Open	Closed	Closed	1
	FCV-25-36	Cont. Isol.	BFY	Pneu.	CIAS	Open	Closed	Closed	1
	FCV-25-7	Vac. Relief	BFY	Pneu.	Cont. Press	Closed	Open	Closed	1
	FCV-25-8	Vac. Relief	BFY	Pneu.	Cont. Press	Closed	Open	Closed	1
	FCV-25-29	SBVS Isol.	BFY	Motor	---	Locked Closed	Closed	As Is	1
	FCV-25-30	Cont. Isol.	BFY	Motor	CIAS	Open	Closed	As Is	1
	FCV-25-31	Cont. Isol.	BFY	Motor	CIAS	Open	Closed	As Is	1
	FCV-25-32	Cont. Isol.	BFY	Motor	CIAS	Closed	Open	As Is	1
	FCV-25-33	Cont. Isol.	BFY	Motor	CIAS	Closed	Open	As Is	1
	FCV-25-34	SBVS Isol.	BFY	Motor	---	Locked Closed	Closed	As Is	1

TABLE 1.9A - 1(Cont'd)

SYSTEM	VALVE	FUNCTION	TYPE	OPERATOR	ACTUATION SIGNAL	NORMAL VALVE POSITION	ACCIDENT VALVE[(a)] POSITION	FAILURE MODE	METHOD OF[(b)] POSITION INDICATION
HVAC (Cont'd)	FCV-25-14	CRECS Isol.	BFY	Motor	CIAS	Open	Closed	As Is	1
	FCV-25-15	CRECS Isol.	BFY	Motor	CIAS	Open	Closed	As Is	1
	FCV-25-16	CRECS Isol.	BFY	Motor	CIAS	Open	Closed	As Is	1
	FCV-25-17	CRECS Isol.	BFY	Motor	CIAS	Open	Closed	As Is	1
	FCV-25-18	CRECS Isol.	BFY	Motor	CIAS	Open	Closed	As Is	1
	FCV-25-19	CRECS Isol.	BFY	Motor	CIAS	Open	Closed	As Is	1
	FCV-25-24	CRECS Isol.	BFY	Motor	CIAS	Open	Closed	As Is	1
	FCV-25-25	CRECS Isol.	BFY	Motor	CIAS	Open	Closed	As Is	1
	FCV-25-11	SBVS Isol.	BFY	Motor	Diff. Pres.	Closed	Open	As Is	1
	FCV-25-12	SBVS Isol.	BFY	Motor	Diff. Pres.	Closed	Open	As Is	1
Containment Spray	MV-07-1A	RWT Isol.	BFY	Motor	RAS	Open	Closed	As Is	1
	MV-07-1B	RWT Isol.	BFY	Motor	RAS	Open	Closed	As Is	1
	MV-07-2A	Sump Isol.	BFY	Motor	RAS	Closed	Open	As Is	1
	MV-07-2B	Sump Isol.	BFY	Motor	RAS	Closed	Open	As Is	1
	FCV-07-1A	Cont. Isol.	BFY	Pneu.	CSAS	Closed	Open	Open	1
	FCV-07-1B	Cont. Isol.	BFY	Pneu.	CSAS	Closed	Open	Open	1
	LCV-07-11A	Cont. Isol.	Glb.	Pneu.	CIAS/SIAS	Closed	Closed	Closed	1
	LCV-07-11B	Cont. Isol.	Glb.	Pneu.	CIAS/SIAS	Closed	Closed	Closed	1
	MV-07-3	Cont. Spray Isol.	Gate	Motor	---	Open	Open	As Is	1
	MV-07-4	Cont. Spray Isol.	Gate	Motor	---	Open	Open	As Is	1
	SE-07-5A thru 5D	Cont. Pressure	Globe	Sol.	---	Open	Open	Open	1
	SE-07-5E, 5F	Cont. Pressure	Globe	Sol.	---	Open	Open	Open	1
Containment Air Monitoring	FCV-26-1	Cont. Isol.	Glb.	Pneu.	CIAS	Open	Closed	Closed	1
	FCV-26-2	Cont. Isol.	Glb.	Pneu.	CIAS	Open	Closed	Closed	1
	FCV-26-3	Cont. Isol.	Glb.	Pneu.	CIAS	Open	Closed	Closed	1
	FCV-26-4	Cont. Isol.	Glb.	Pneu.	CIAS	Open	Closed	Closed	1
	FCV-26-5	Cont. Isol.	Glb.	Pneu.	CIAS	Open	Closed	Closed	1
	FCV-26-6	Cont. Isol.	Glb.	Pneu.	CIAS	Open	Closed	Closed	1
Hydrogen Sampling	FSE-27-8	Cont. Isol.	Glb.	Sol.	---	Closed	Open	Closed	1
	FSE-27-9	Cont. Isol.	Glb.	Sol.	---	Closed	Open	Closed	1
	FSE-27-10	Cont. Isol.	Glb.	Sol.	---	Closed	Open	Closed	1
	FSE-27-11	Cont. Isol.	Glb.	Sol.	---	Closed	Open	Closed	1
	FSE-27-12	Cont. Isol.	Glb.	Sol.	---	Closed	Open	Closed	1
	FSE-27-13	Cont. Isol.	Glb.	Sol.	---	Closed	Open	Closed	1
	FSE-27-14	Cont. Isol.	Glb.	Sol.	---	Closed	Open	Closed	1
	FSE-27-15	Cont. Isol.	Glb.	Sol.	---	Closed	Open	Closed	1
	FSE-27-16	Cont. Isol.	Glb.	Sol.	---	Closed	Open	Closed	1
	FSE-27-17	Cont. Isol.	Glb.	Sol.	---	Closed	Open	Closed	1
	FSE-27-18	Cont. Isol.	Glb.	Sol.	---	Closed	Open	Closed	1

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TABLE 1.9A - 1(Cont'd)

SYSTEM	VALVE	FUNCTION	TYPE	OPERATOR	ACTUATION SIGNAL	NORMAL VALVE POSITION	ACCIDENT VALVE[(a)] POSITION	FAILURE MODE	METHOD OF[(b)] POSITION INDICATION
HVAC	D-17A	Cont. Room	N/A	Motor	CIAS [(d)]	Closed	Open	Open	1
	D-17B	Cont. Room	N/A	Motor	CIAS [(d)]	Closed	Open	Open	1
	D-18	Cont. Room	N/A	Motor	CIAS [(d)]	Closed	Open	Open	1
	D-19	Cont. Room	N/A	Motor	CIAS [(d)]	Closed	Open	Open	
	D-29	FHB Isol.	N/A	Motor	High Rad	Open	Closed	Closed	1
	D-30	FHB Isol.	N/A	Motor	High Rad	Open	Closed	Closed	1
	D-31	FHB Isol.	N/A	Motor	High Rad	Open	Closed	Closed	1
	D-32	FHB Isol.	N/A	Motor	High Rad	Open	Closed	Closed	1
	D-33	FHB Isol.	N/A	Motor	High Rad	Open	Closed	Closed	1
	D-34	FHB Isol.	N/A	Motor	High Rad	Open	Closed	Closed	1
	D-35	FHB Isol.	N/A	Motor	High Rad	Open	Closed	Closed	1
	D-36	FHB Isol.	N/A	Motor	High Rad	Open	Closed	Closed	1
	D-23	SBVS Cont.	N/A	Motor	Diff. Pres.	Open	Open	Open	4
	D-24	SBVS Cont.	N/A	Motor	Diff. Pres.	Open	Open	Open	4
HVAC	D-1	RAB Isol.	N/A	Motor	SIAS	Open	Open	Open	5
	D-2	RAB Isol.	N/A	Motor	SIAS	Open	Open	Open	5
	D-3	RAB Isol.	N/A	Motor	SIAS	Open	Open	Open	5
	D-4	RAB Isol.	N/A	Motor	SIAS	Open	Open	Open	5
	D-9A	RAB Isol.	N/A	Motor	SIAS	Open	Closed	Closed	1
	D-9B	RAB Isol.	N/A	Motor	SIAS	Open	Closed	Closed	1
	D-12A	RAB Isol.	N/A	Motor	SIAS	Open	Closed	Closed	1
	D-12B	RAB Isol.	N/A	Motor	SIAS	Open	Closed	Closed	1
	D-7A	RAB Isol.	N/A	Motor	SIAS	Open	Closed	Closed	6
	D-7B	RAB Isol.	N/A	Motor	SIAS	Open	Closed	Closed	6
	D-8A	RAB Isol.	N/A	Motor	SIAS	Open	Closed	Closed	6
	D-8B	RAB Isol.	N/A	Motor	SIAS	Open	Closed	Closed	6
	D-5A	RAB Isol.	N/A	Motor	SIAS	Open	Closed	Closed	1
	D-5B	RAB Isol.	N/A	Motor	SIAS	Open	Closed	Closed	1
	D-6A	RAB Isol.	N/A	Motor	SIAS	Open	Closed	Closed	1
	D-6B	RAB Isol.	N/A	Motor	SIAS	Open	Closed	Closed	1
	D-13	RAB Isol.	N/A	Motor	SIAS [(d)]	Open	Open	Open	1
	D-14	RAB Isol.	N/A	Motor	SIAS [(d)]	Open	Open	Open	1
	D-15	RAB Isol.	N/A	Motor	SIAS [(d)]	Open	Open	Open	1
	D-16	RAB Isol.	N/A	Motor	SIAS [(d)]	Open	Open	Open	1
	L-7A	RAB Isol.	N/A	Motor	SIAS [(d)]	Open	Open	Open	7
	L-7B	RAB Isol.	N/A	Motor	SIAS [(d)]	Open	Open	Open	7

TABLE 1.9A-1 (Cont'd)

Notes:

a) <u>Accident Valve Position</u>

The designation "open" or "closed" indicates the position as a result of an ESFAS signal or a position that may be manually selected as part of a post accident procedure.

1) These valves will be closed if the “C” CCW pump is supplying the “B” CCW header.
2) These valves will be open if the “C” CCW pump is supplying the “B” CCW header.

b) <u>Method of Position Indication in the Control Room</u>

1) Position Indicating Lights.
2) Failure of valve to close would result in low flow indication by flow transmitter FIA-22l2.
3) Failure of valve to open would result in a low-low alarm for Diesel Generators Day tank.
4) Failure of damper to open would result in low flow indication by flow transmitter FIS-25-20A1 or 20B1 for D23 and D24 respectively.
5) Failure of damper to open would result in high differential pressure indication by pressure transmitter PDIS-25-16A or 16B.
6) Each damper is backed up by redundant counterpart. Failure of one damper to close would result in no adverse consequence.
7) Failure of damper to open would result in low flow indication by flow transmitter FIS-25-21A1 or 21B1 for 2L-7A and 7B respectively.
8) Analog Position indicator; Indicator power separate from control power.

c) Valve position is dependent on charging pump running status. See section 9.3.4.2.2g for details.

d) Damper is actuated to its accident position by the start signal of its associated fan.

e) Normal Valve Position

1) These valves will be closed if the “C” CCW pump is supplying the “B” CCW header.
2) These valves will be open if the “C” CCW pump is supplying the “B” CCW header.