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* The listed drawings are included as “General References” only; i.e., refer to the drawings to obtain additional detail or to obtain background information. These drawings are not part of the UFSAR. They are controlled by the Controlled Documents Program.

CHAPTER 1.0 - INTRODUCTION AND GENERAL DESCRIPTION OF PLANT

1.1 INTRODUCTION

This Updated Final Safety Analysis Report (UFSAR) is submitted for the nuclear power station designated as the LaSalle County Station (LSCS) Unit 1, in accordance with the requirements of 10 CFR 50 Section 50.71(e) as published in the Federal Register on May 9, 1980.

Written as if LaSalle is a single unit plant, but applying to both units unless expressly written for Unit 1 or Unit 2, the original LSCS Final Safety Analysis Report (FSAR) was submitted in April 1976. The FSAR was written in accordance with Regulatory Guide 1.70, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants," Revision 2, September 1975. The last amendment to the original FSAR prior to the 10CFR50.71(e) update for Unit 1 is number 64. The original FSAR up through and including amendment 64 will be referred to herein as the "FSAR." The responses to the NRC questions comprise three volumes of the FSAR. The information in the response(s) was current at the time the LaSalle Operating License was granted, and has not since been revised. However, the responses have been reviewed and applicable updated information has been incorporated into the UFSAR text as required. Therefore, the three FSAR volumes of responses to NRC questions are now “historical information” pursuant to the guidance provided in NEI 98-03, Revision 1.

This UFSAR is the updated version of the FSAR and follows the same format as the FSAR (with allowed content criteria as specified in NEI 98-03 Revision 1, and as endorsed by Regulatory Guide 1.181 09/99). The UFSAR contains a description of LSCS Unit 1 which is up-to-date as of not more than 6 months prior to the latest revision date. The latest UFSAR revision date is specified in the document control section at the beginning of Volume 1. The UFSAR is revised in accordance with 10CFR50.71(e).

The Nuclear Regulatory Commission approved the transfer of the facility licenses from Commonwealth Edison (ComEd) Company to Exelon Generation Company, LLC (EGC) on August 3, 2000 (Reference 1). References in the UFSAR to ComEd, CECo, and Commonwealth Edison have been retained, as appropriate, instead of being changed to EGC to properly preserve the historical content.

The LSCS Preliminary Safety Analysis Report (PSAR) was submitted on November 3, 1970 (Docket Nos. 50-373 and 50-374). The station was constructed under construction permits CPPR-99 and CPPR-100 which were issued on September 10, 1973. Unit 1 was authorized to commence power operation under license No. NPF-11 which was granted on April 17, 1982. Unit 2 was authorized to commence power operation under license No. NPF-18.

This power generating station is located in the agricultural area of Brookfield Township, LaSalle County, Illinois. It is approximately 55 direct-line miles southwest of Chicago and 20 miles west of Dresden Nuclear Power Station. The plant is on flat terrain about 220 feet above the Illinois River channel which traverses north central Illinois some 3-1/2 miles to the north of the site.

The station utilizes two single-cycle forced-circulation boiling water reactors, rated at 3546 MWt and designed for 3559 MWt. The gross electric output of each unit is approximately 1207 MWe; the net output of each unit is approximately 1178 MWe from each General Electric (GE) turbine-generator. The NSSS supplier was GE (Nuclear Energy Division). The plant, except for the NSSS, was designed by Sargent & Lundy (S&L) Engineers.

The containment design employs the BWR Mark II concept of over-under pressure suppression with multiple downcomers connecting the reactor drywell to the water-filled pressure suppression chamber. The primary containment is a steel-lined, post-tensioned, concrete enclosure, housing the reactor and the suppression pool. This primary containment is entirely enclosed in the reinforced concrete reactor building which is the secondary containment structure.

The power generation complex includes several contiguous buildings, two reactor buildings, an auxiliary building (housing the control room), the turbine building, diesel-generator buildings, the radwaste building, the service building, and the off-gas building. Other buildings such as the gatehouse, warehouses, etc., are also located in the general plant area. A lake screen house on the intake flume is located about 800 feet east of the main building complex. A small river screen house, located on the Illinois River, provides makeup water to the cooling lake for the LaSalle County Station.

Condenser cooling for the station is provided from a perched cooling lake of 2058 acres. The ultimate heat sink for emergency core cooling is a submerged pond and intake flume that underlies the cooling lake and the natural grade of the site.

The station utilizes a single vent stack for elevated release of all gaseous waste. Liquid radwaste is stored for decay or concentrated to solid waste for controlled disposal at regulated storage sites. The shielding design and plant layout incorporate 16 years of reactor operating experience at CECo to restrict radiological exposures to as low as reasonably achievable levels. Estimated radiological doses for normal operations and classical postulated accidents are all fractional parts of the federal radiological guidelines for siting and operation of nuclear power plants.

1.1.1 References

1. Letter from D. M. Skay (NRR) to O. D. Kingsley (ComEd), dated August 3, 2000, and the associated NRC safety evaluation report.

2. Letter from M.D. Jesse (Exelon) to U. S. NRC, dated May 13, 2010, "Additional Information Supporting Request for License Amendment Regarding Measurement Uncertainty Recapture Power Uprate".

1.2 GENERAL PLANT DESCRIPTION

For the purposes of this UFSAR, the LaSalle County Station is described in terms of its safety functions via safety criteria, and in terms of its power generation functions via nonsafety or power generation criteria. These general criteria define the design approach for safety and for power generation objectives of the nuclear power plant. Although the distinctions between safety design criteria and power generation design criteria are not always clear-cut, this arbitrary division of criteria facilitates the safety analyses while also enabling a portrayal of the plant equipment in sufficient detail to assist in the understanding of its functional purpose. As a secondary categorization technique in this report, functionally related equipment is further grouped into "systems" which are discussed primarily for their importance to safety and secondly as they relate to power generation objectives.

The summary overview in this chapter provides brief descriptions of the site, its environs, and the arrangement of the station building complex. This is followed by the specific safety features of the NSSS, the power conversion system, the electrical and instrumentation equipment, and the radioactive waste and auxiliary support systems. A glossary of terms is provided in Section 1.2.4.

1.2.1 Principal Design Criteria

LaSalle County Station was designed, fabricated, erected, and is operated in such a manner that the release of radioactivity to the environment does not exceed the limits and guideline values of applicable government regulations pertaining to the release of radioactive materials for normal operations and abnormal transients and accidents.

The station is designed in conformance with applicable government regulations, ASME Codes, IEEE Codes, and other appropriate standards as noted herein. Compliance with NRC Regulatory Guides is discussed specifically in Appendix B.

The electrical design for essential safety equipment is of such redundancy and independence that no single failure of active or passive components can prevent the required safety actions.

The mechanical design for the equipment which makes up the primary pressure boundary conforms with Sections II, III, VIII, IX, and XI of the ASME Boiler and Pressure Vessel Code. Mechanical separation criteria were also incorporated in the plant design.

The classification of structures, components, and systems is discussed in Section 3.2. Specific conformance to the 56 general design criteria of 10 CFR 50,

Appendix A, is discussed in Section 3.1. Other criteria are discussed in Chapter 3.0 and throughout this UFSAR.

Single failures are considered for applicable safety situations.

The plant is designed to produce steam for direct use in a turbine-generator unit that feeds CECo's electrical network.

1.2.1.1 Safety Design Criteria

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

b. The reactor is designed so that there is no tendency for divergent oscillation of any operating characteristic, considering the interaction of the reactor with other appropriate station systems.

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

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

e. Sufficient indications are provided to allow determination that the reactor is operating within the envelope of conditions considered in this safety analysis.

f. Those portions of the nuclear system that form part of the reactor coolant pressure boundary are designed to retain integrity as a radioactive material barrier following abnormal operational transients and credible accidents.

g. A primary containment is provided that completely encloses the reactor system. The containment employs the pressure-suppression concept.

h. It is possible to test primary containment integrity and leaktightness at periodic intervals.

i. A secondary containment completely encloses the primary containment. This secondary containment includes the capability to control the release of radioactive materials from the primary containment.

j. Provisions are made to remove long-term energy from the primary containment as necessary to maintain the integrity of the containment following accidents which release energy into the containment.

k. Piping that penetrates the primary containment and could serve as a path for the uncontrolled release of any radioactive leakage to the environs is automatically isolated whenever such an uncontrolled release of radioactive material is threatened. Such isolation is effected in time to limit radiological effects to significantly less than the prescribed radiation limits.

l. The primary and secondary containments, in conjunction with other engineered safety features, limit the radiological effects of accidents resulting from the release of radioactive material within these containment volumes to less than prescribed radiation limits.

m. The control room is shielded against radiation so that continued occupancy is possible under accident conditions.

n. In the event that the control room becomes uninhabitable, it is possible to bring the reactor from power range operation to cold shutdown conditions by a remote shutdown system located outside the control room.

o. A backup reactor shutdown system, independent of normal reactivity control provisions, has the capability to shut down the reactor from any normal operating condition and subsequently to maintain the shutdown condition.

p. Interlocks or other automatic equipment are provided as backup to procedural controls to avoid conditions requiring needless functioning of nuclear safety systems or engineered safety features.

q. Faulted equipment is detected and isolated from the electrical systems with a minimum of disturbance via activation of protective relaying in the event of equipment failure.

r. The Class 1E power systems are designed as triple-bus systems, with any two buses being adequate to safely shut down the unit.

s. Standby electrical power sources are provided to allow prompt reactor shutdown and removal of decay heat under circumstances where normal auxiliary power is not available.

t. Where positive, precise action is immediately required in response to abnormal operational transients and accidents, such action is automatic and requires no decision or manipulation of controls by station operations personnel.

u. Voltage relays are used on the emergency equipment buses to isolate these buses from the normal electrical system in the event of loss of offsite power and concurrently to initiate starting of the standby emergency power system generators.

v. Standby electrical power sources have sufficient capacity to power all nuclear safety systems and engineered safety features requiring electrical power.

w. The design of nuclear safety systems and engineered safety features includes design allowances for unusual natural phenomena such as earthquakes, floods, and storms on the site.

x. Nuclear safety systems and engineered safety features act to ensure that no violation of the reactor coolant pressure boundary results from internal pressures caused by abnormal operational transients or accidents.

y. Provisions are made for control of active components of nuclear safety systems and engineered safety features from the control room during normal operations. During a remote shutdown condition, this control is purposely removed from the control room.

z. Engineered safety features are designed to permit demonstration of their performance.

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

bb. Emergency core cooling systems are provided to limit fuel cladding temperatures to less than the fragmentation temperature in the event of a loss-of-coolant accident.

cc. The emergency core cooling systems (ECCS) provide for continuity of core cooling over the complete range of postulated break sizes in the reactor coolant pressure boundary.

dd. Operation of the ECCS is initiated automatically when required, regardless of the availability of offsite power supplies and the normal generating system of the station.

ee. Auxiliary systems, such as emergency cooling water, required heating and ventilating, communications, and lighting, are designed to function during normal and accident conditions.

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

gg. Gaseous, liquid, and solid waste disposal facilities are designed so that the discharge and offsite shipment of radioactive effluents can be made in accordance with applicable regulations.

hh. The radwaste systems are designed to minimize the release of radioactive materials from the station to the environs. Such releases as may be necessary during normal operations are limited to values that meet the requirements of 10 CFR 20 and 10 CFR 50.

ii. The design of the systems provides means by which station operations personnel can be informed whenever specified limits on the release of radioactive material may be approached.

jj. The control room is shielded against radiation so that occupancy is possible under accident conditions and so that radiation doses are less than those set by 10 CFR 50.67.

kk. Fuel handling and storage facilities are designed to prevent inadvertent criticality of new and spent fuel and to maintain shielding and cooling of spent fuel.

1.2.1.2 Power-Generation Criteria

a. Reactor power level is manually controllable.

b. Control of the reactor is possible from a single location.

c. Reactor controls, including alarms, are arranged to allow rapid operator assessment of reactor conditions and the location of reactor system malfunctions.

d. Control equipment is provided to allow the reactor to respond automatically to both minor and major load changes including abnormal operational transients.

e. Reactor controls, including alarms, are arranged to allow rapid operator assessment of reactor conditions and to locate reactor system malfunctions.

f. Backup heat-removal systems are provided to remove decay heat generated in the core under circumstances wherein the normal operational heat removal systems become inoperative. The capacity of such systems is adequate to prevent fuel cladding damage.

g. A means is provided by which station operators can be informed when limits on the release of radioactivity are approached.

h. The power conversion system is designed to ensure that any fission products or radioactivity associated with the steam and condensate during normal operation are contained safely inside the system or are released under controlled conditions in accordance with appropriate regulations and waste disposal procedures.

i. Sufficient normal and standby auxiliary sources of electrical power are provided to attain prompt shutdown and continued maintenance of the station in a safe condition under all credible circumstances.

j. Control of the nuclear system and the power-conversion equipment is possible from a central location.

k. Control equipment is provided to control the reactor pressure throughout its operating range.

l. Control equipment in the feedwater system maintains the water level in the reactor vessel at the optimum level required by steam separators.

m. Metering for essential generators, transformers, and circuits is monitored in the control room.

n. Components of the power-conversion systems are designed to produce electrical power from the steam coming from the reactor, condense the steam into water, and return the water to the reactor as heated feedwater with a major portion of its gases and particulate impurities removed.

o. Gaseous, liquid, and solid radioactive waste disposal systems are designed so that in-plant processing, discharge of effluents, and offsite shipments are in accordance with all applicable federal regulations.

p. Auxiliary systems that are not required to effect safe shutdown of the reactor or maintain it in a safe condition are designed so that a failure of these systems does not prevent the essential auxiliary systems from performing their design functions.

q. Radiation shielding is designed and access control provisions are made to minimize radiation levels and provide the means to control radiation doses within the limits of published regulations.

r. Radiation shielding is provided and access control patterns are established to allow a properly trained operating staff to control radiation doses within the limits of applicable regulations in any mode of normal station operations.

1.2.2 Station Description of Plant Features Important to Safety

This section provides an overview of those plant features of the LaSalle County Station that are important to safety considerations. The following subsections describe:

a. site characteristics - acreage, location, environs, meteorology, hydrology, seismology, and site dependent design bases;

b. general arrangement of structures and equipment.

Descriptive symbols appearing in P&ID's referenced by the UFSAR, and in UFSAR Figures which are based upon P&ID's, are defined on Drawing M-54 and M-5054.

The two side-by-side power generating units are essentially independent, although certain components are shared, such as the common control room, common radwaste facility, the station vent stack, etc.

1.2.2.1 Site Characteristics

1.2.2.1.1 Site Location and Size

LSCS is located on an irregular pentagonally shaped site (see Drawing No. M-1). Approximately 3060 acres lie within the site boundaries, with 2058 acres being used for a cooling lake. A pipeline corridor, consisting of 815 acres, extends north from the site to the Illinois River, which is approximately 5.0 miles north of the reactor (see Drawing No. M-2).

1.2.2.1.2 Description of Site Environs

Human population near the site is sparse. Isolated farm homes and small groupings of houses typify the inhabited areas. The site environs are further described in Subsection 2.1.3 and Section 2.2.

1.2.2.1.3 Meteorology

The site is subject to typical continental meteorology characterized by high variability and a wide range of temperature extremes. The average annual precipitation at Ottawa based upon 89 years of record is 34 inches. This includes an annual average of 27 inches of snow. Thunderstorms occur on an average of 49 days per year.

The prevailing winds of this area are primarily south by southwest at an average of 10 mph. The probability of tornado occurrence at the site is 0.0016 for any given year, which converts to a recurrence interval of 625 years.

Dispersion of normal releases from the elevated station vent stack is further discussed in Section 2.3.

1.2.2.1.4 Hydrology

The LSCS site is located in the Illinois River basin. The Illinois River is a perennial stream with a drainage area of approximately 7640 mi^2 surrounding the plant site. The normal pool elevation of the Marseilles pool is 483.25 feet MSL (USGS datum 1912 adjustment, which is 0.462 foot lower than USGS datum 1929 adjustment).

The plant grade is 710 feet MSL (1929 datum). Therefore, the station site may be described as "floodproof" or "dry" with regard to floods in the Illinois River. Flood effects on the river screen house are discussed in Section 2.4.

1.2.2.1.5 Geology and Seismology

The site is located in the Central Lowland Physiographic Province and in one of the most stable tectonic areas of the North American Craton. The regional structure consists of a system of sedimentary basins, arches, and domes of Paleozoic age.

The depth to Precambrian rock is approximately 4200 feet at the site. Cambrian and Ordovician sandstones and dolomites form most of the sedimentary column overlying the Precambrian basement.
Pennsylvanian cyclothems, mainly shale, form a cap about 120 feet in thickness over the Ordovician strata in the site area.

Approximately 170 feet of predominantly Wisconsinan glacial drift overlies the bedrock surface in the site area. The nearest major fault zone is the Sandwich Fault Zone, which is located approximately 26 miles northeast of this site and is noncapable. There are no geologic features at or near the site which would preclude its use for the construction and operation of the nuclear power station.

For seismic design of Seismic Category I structures, the maximum horizontal acceleration caused by the safe shutdown earthquake (SSE) is 20% of gravity at the free field foundation level. The operating-basis earthquake (OBE) is a horizontal acceleration of 10% of gravity at the foundation level. For additional information concerning geology and seismology consult Section 2.5.

1.2.2.1.6 Design Bases Dependent on Site Environs

An elevated, 370-foot, station vent stack common to both units is provided for the continuous release of all gaseous effluents. In addition, a recombiner and charcoal bed adsorber system are employed to limit gaseous effluent releases from normal operations. This subject is further discussed in Subsection 11.3.2.

1.2.2.1.6.1 Liquid Waste Effluents

Liquid waste releases are controlled to ensure that concentrations at the point of discharge do not exceed 10 CFR 20 limits. This subject is further discussed in Section 11.2.

1.2.2.1.6.2 Wind Loading and Seismic Design

The structures and components whose failure might conceivably contribute to an uncontrolled release of fission products are designed to resist tornado loads possessing a maximum wind velocity of 360 mph and an internal differential pressure of 3 psi in 3 seconds. This subject is further discussed in Section 3.3.

1.2.2.1.6.3 Flooding

The plant design accounts for safety static water head pressures on plant structures. Consult Sections 2.4 and 3.4 and Subsection 1.2.2.1.4 for additional information.

1.2.2.2 General Arrangement of Structures and Equipment

Station equipment is housed in the following principle structures:

a. reactor building - the nuclear steam supply system, the drywell, suppression pool, primary containment, new and spent fuel pools, refueling equipment, and emergency core cooling equipment;

b. auxiliary building - the control room, the HVAC equipment, the station vent stack, and much of the station electrical switchgear;

c. turbine building - the power conversion equipment and feedwater cleanup equipment;

d. off-gas filter building - off-gas filters and associated equipment;

e. diesel-generator buildings - the standby diesel generators, diesel oil storage tanks, CSCS cooling water pumps and strainers, and associated controls and instrumentation;

f. service building - the machine shop, offices, warehouses, and training rooms;

g. lake screen house - the service and circulating water pumps with their accompanying equipment and instrumentation;

h. river screen house - the lake makeup equipment and control instrumentation;

i. solid radwaste building - all solid radwaste disposal equipment;

j. switchyard;

k. security gatehouse; and

l. interim radwaste storage facility

The arrangement of these structures on the station site is shown in Drawing No. M-3. The arrangement of the equipment inside the main buildings is shown in Drawing Nos. M-4 through M-22.

1.2.2.3 Nuclear System

The nuclear system includes a direct-cycle, forced-circulation, General Electric boiling water reactor that produces steam for direct use in the steam turbine. A heat balance showing the major parameters of the nuclear system for the rated power conditions is shown in Figure 1.2-1. This system is discussed in Chapter 4.0.

1.2.2.3.1 Reactor Core and Control Rods

Fuel for the reactor core consists of slightly enriched uranium dioxide pellets sealed in Zircaloy tubes. These fuel rods are assembled into individual fuel assemblies. Gross control of the core is achieved by movable, bottom-entry control rods which are positioned by individual control rod drives.

When a scram is signaled by the reactor protection system, the high-pressure water stored in an accumulator in the hydraulic control unit forces its control rod into the core. This system is discussed in Subsection 4.2.3.

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

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

Each fuel assembly has several fuel rods with a burnable poison, gadolina (Gd_2O_3) mixed in solid solution with UO_2.

The initial core and reload fuel were provided by General Electric (GE). Beginning in 1999, reload fuel was provided by Siemens Power Corporation, Nuclear Division (SPC). While the SPC fuel differs slightly from the GE fuel, the basic design requirements and description remain the same. Where design features and analytical methods differ substantially between the two fuel vendors, the UFSAR test has been revised to describe or reference either the appropriate method, or both methods.

1.2.2.3.2 Reactor Vessel and Internals

The reactor vessel contains: the core and supporting structures; the steam separators and dryers; the jet pumps; the control rod guide tubes; the distribution lines for the feedwater, core sprays, and liquid level control; the incore instrumentation, and other components. The main connections to the vessel include: steamlines, coolant recirculation lines, feedwater lines, control rod drive and incore nuclear instrument housings, high- and low- pressure core spray lines, low-pressure core injection lines, standby liquid control line, jet pump pressure-sensing lines, water level instrumentation, and control rod drive system return lines.

The SS-clad low alloy steel reactor vessel is designed and fabricated in accordance with applicable codes for a pressure of 1250 psig. The nominal operating pressure in the steam space above the separators is 1020 psia.

The reactor core is cooled by demineralized water that enters the lower portion of the core and boils as it flows upward around the fuel rods. The steam leaving the core is dried by steam separators and dryers located in the upper portion of the reactor vessel. The steam is then directed to the turbine through the main steamlines. Each steamline is provided with two isolation valves in series, one on each side of the primary containment barrier. This system is described further in Subsection 5.2.2.

1.2.2.3.3 Reactor Recirculation System

The reactor recirculation system pumps reactor coolant through the core. This is accomplished by two recirculation loops external to the reactor vessel but inside the primary containment. Each external loop contains one high capacity, motor-driven recirculation pump, a flow control valve, and two motor-operated gate valves for suction shutoff and discharge shutoff purposes. Each pump suction line contains a flow measuring system. The variable-position flow control valve in the main recirculation pipe allows control of reactor power level through the effects of coolant flow rate on moderator void content. The pumps can be operated at either high speed or low speed. Low speed operation of the pumps provides capability for reduced recirculation flow during startup, shutdown, or other times of reduced power operation.

Jet pumps provide a continuous internal circulation path for the major portion of the core coolant flow. The jet pumps are located in the annular region between the core shroud and the vessel's inner wall; thus any recirculation line break would still allow core flooding to approximately two-thirds of the core height--the level of the inlet of the jet pumps.

A detailed, comprehensive description of the reactor recirculation system is provided in Appendix G of the UFSAR.

1.2.2.3.4 Residual Heat Removal System

The residual heat removal (RHR) system is a set of pumps, heat exchangers, and piping that fulfills the cooling functions under various configurations and conditions as follows:

a. Shutdown cooling and reactor vessel head spray - to remove residual heat (decay heat and sensible heat) from the nuclear boiler system after a normal shutdown and cooldown.

b. * Steam condensing mode deleted per AIR 373-160-92-00108. (E01-2-9500158)

c. Low-pressure coolant injection mode - This capability is discussed in Subsection 1.2.2.5.3.

d. The primary containment cooling mode limits temperature, hence the pressure, by water spray action inside the primary containment when activated during an isolation event.

e. The suppression pool cooling mode limits the water temperature of the suppression pool following a design-basis LOCA or following testing of the safety/relief valves and the RCIC system which discharge to the suppression pool.

This system is discussed further in Subsection 5.4.7.

1.2.2.3.5 Primary Reactor Water Cleanup System

The reactor water cleanup system recirculates a portion of reactor coolant through a filter-demineralizer to remove particulate and dissolved impurities from the reactor system under controlled conditions (see Subsection 5.4.8).

1.2.2.3.6 Reactor Protection System

The reactor protection system (RPS) is an electric logic network which initiates a rapid, automatic shutdown of the reactor. It acts in time to prevent fuel clad damage and any nuclear system process barrier damage associated with abnormal operational transients. The reactor protection system overrides all operator actions and process controls. It uses a logic of one-out-of-two taken twice for protective actions. The design is based on a fail-safe philosophy that allows appropriate protective action even when a single failure occurs. Some of the neutron monitors function uniquely as part of this nuclear safety system. The high neutron flux signals are used for this scram protection.

The source range monitors (SRM's) and the intermediate range monitors (IRM's) provide flux level indications during reactor startup and low-power operation.

This system is further discussed in Section 7.2.

1.2.2.3.7 Main Steamline Flow Restrictors

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

This system is further discussed in Subsection 5.4.4.

1.2.2.3.8 Refueling Interlocks

A system of interlocks that restricts movement of refueling equipment and control rods when the reactor is in the refueling and startup modes is provided to prevent inadvertent criticality during refueling operations. The interlocks back up procedural controls that have the same objective. The interlocks affect the refueling platform, refueling platform hoists, fuel grapple, and control rods. This system is discussed in Section 7.7.

1.2.2.3.9 Nuclear System Pressure Relief System

A pressure relief system consisting of safety/relief valves mounted on the main steamlines prevents excessive nuclear boiler pressure following either abnormal operational transients or accidents. This system is discussed in Subsection 5.2.2.

1.2.2.3.10 Reactor Core Isolation Cooling System

Although not a safety system, the reactor core isolation cooling (RCIC) system provides makeup water to the reactor vessel when the vessel is isolated. It uses a steam-driven turbine-pump unit and operates automatically to maintain adequate water level in the reactor vessel. The RCIC pump takes water from the condensate storage tank or directly from the suppression pool or from the suppression pool via the RHR heat exchangers, depending on reactor conditions, and discharges it through the head spray nozzle of the reactor vessel to maintain reactor water level. This system is also discussed in Subsection 5.4.6.

1.2.2.4 Containment

The containment is a set of leaktight barriers which prohibit the release of fission products to the environs. Although these barriers include the fuel cladding and the reactor pressure vessel, the word "containment" connotes the structures in which the reactor pressure vessel and the nuclear process equipment operate. The primary containment, utilizing the pressure suppression concept, and the secondary containment, including the reactor buildings with their atmospheric ventilation systems and the standby gas treatment system (SGTS), have the capability of minimizing rapid pressure transients. The containment provides an isolation function for the lines penetrating the primary containment. Ventilation dampers on secondary containment are provided to inhibit leakage. Both primary and secondary containments are designed to Class I seismic standards.

1.2.2.4.1 Primary Containment

The primary containment is designed to limit the release of radioactivity to the environs subsequent to the postulated loss-of-coolant accident. The vapor suppression concept for the reduction of internal pressure is utilized in the LSCS design. The drywell is constructed above the wetwell in a single concrete vessel shaped like the frustrum of a cone on top of a right circular cyclinder. Unique features of the LSCS primary containment are as follows:

a. The drywell is lined with carbon steel.

b. The wetwell is lined in its entirety with stainless steel; this includes the central pedestal, the supporting columns, and the ceiling.

c. There are no projections, equipment, or galleries inside the wetwell; the wetwell is "structurally clean" internally.

d. Four vacuum breaker lines externally connect the wetwell and the drywell to provide internal pressure relief from the initial air pressurization of the wetwell. These vacuum breakers are serviced from within the secondary containment.

e. During normal operations, the suppression pool water volume is level-controlled; however, during servicing when the reactor head is removed, the suppression pool water is used to fill the reactor cavity and pool. (Suppression pool water is demineralized, filtered, and pumped for this dual usage.)

The drywell and wetwell are separated by a reinforced concrete floor which is penetrated by 98 stainless steel downcomers. The primary containment is a posttensioned concrete and steel structure which houses the reactor vessel, the reactor coolant recirculation loops, and other principal connections of the reactor fluid loops making up the primary pressure boundary.

Cooling systems are provided to remove heat from the reactor core, the drywell, and the water in the suppression chamber, and thus provide continuous cooling of the primary containment under postulated accident conditions. Isolation valves are used to ensure that radioactive materials which otherwise might be released from the reactor during the course of an accident are contained within the primary containment.

This subject is addressed in Section 6.2 and Reference 1.

1.2.2.4.2 Secondary Containment

The reactor building completely surrounds the primary containment and functions as a secondary containment when the primary containment is closed and in service. The reactor building also houses refueling and reactor servicing equipment, new and spent fuel storage facilities, and other reactor safety and auxiliary systems.

The design of the reactor building includes provisions for seismic load resistance and low infiltration and exfiltration rates. The building consists of poured-in-place, reinforced concrete exterior walls up to the refueling floor. Above this level, the building structure is steel frame with insulated metal siding with sealed joints. Access to the secondary containment is through interlocked double doors.

This subject is addressed in Section 6.2.

1.2.2.4.3 Standby Gas Treatment System

The standby gas treatment system (SGTS) consists of two identical filter trains and interconnecting piping and ductwork. The individual trains are available to each reactor building (M-89).

Either train by itself is capable of exchanging both reactor building volumes once in a 24-hour period.

The system maintains a slightly negative internal building pressure and processes all gaseous effluent prior to its discharge via the station vent stack.

All SGTS equipment is powered from the essential buses and is started either automatically or manually from the control room. This system is further discussed in Subsection 6.5.1.

1.2.2.4.4 Containment and Reactor Vessel Isolation Control System

The primary containment and reactor vessel isolation control system automatically initiates closure of isolation valves to close off all potential leakage paths for radioactive material to the environs. This action is taken upon indication of a potential breach in the nuclear system process barrier. A containment and isolation status panel is provided in the control room to display the status and operations of the isolation control system. This system is further discussed in Subsection 6.2.4.

Although all pipelines that both penetrate the containment and offer a potential release path for radioactive material are provided with redundant isolation capabilities, the main steamlines, because of their large size and large mass flow rates, are given special isolation consideration. Automatic isolation valves are provided in each main steamline. Each is powered by both air pressure and spring force.

1.2.2.4.5 Main Steamline Isolation Valve Leakage Control System (U2 deleted, U1 abandoned-in-place)

The main steamline isolation valve leakage control system (MSIV-LCS) provided originally has been deleted. The valve leakages are processed through the main steam lines, main steamline drains, and the main condenser. The system is discussed in Section 6.8.

1.2.2.4.6 Reactor Building Isolation Dampers

The reactor building heating, ventilation, and air-conditioning system supply and discharge ducts are each supplied with two isolation dampers in series. These dampers are designed to maintain secondary containment isolation and are automatically closed whenever the standby gas treatment system is initiated. These isolation dampers may also be manually closed from the local control panel (see Drawing Nos. M-1455 and M-1456). This system is further discussed in Subsections 6.2.4 and 7.3.7.

1.2.2.4.7 Containment Vent and Purge System

Although not a safety system, a separate dual-train containment vent and purge system is connected, via isolation valving, in parallel with the SGTS filter trains. This vent and purge equipment includes charcoal and HEPA filters with exhaust fans and ducting to the station vent stack. This equipment is to be used to clean up the primary and secondary containment atmospheres when low-level airborne contamination exists, thereby attaining as low as reasonably achievable worker exposures. The SGTS is therefore reserved for the accident case and need not be operated for routine atmospheric cleanup.

1.2.2.5 Emergency Core Cooling Systems

Four emergency core cooling systems are provided to maintain fuel cladding below fragmentation temperature in the event of a breach in the reactor coolant pressure boundary that results in a loss of reactor coolant. The systems are:

a. high-pressure core spray (HPCS) system;

b. automatic depressurization system (ADS);

c. low-pressure core spray (LPCS) system; and

d. low-pressure coolant injection (LPCI), an operating mode of the residual heat removal system.

These systems are further discussed in Section 6.3

1.2.2.5.1 High-Pressure Core Spray System

The HPCS system provides and maintains an adequate coolant inventory inside the reactor vessel to maintain fuel cladding temperatures below fragmentation temperature in the event of breaks in the reactor coolant pressure boundary. The system is initiated by either high pressure in the drywell or low water level in the vessel. It operates independently of all other systems over the entire range of pressure differences from greater-than-normal operating pressure to zero. The HPCS system pump motor is powered by a diesel generator if auxiliary power is not available, and the system may also be used as a backup for the RCIC system. This system is further discussed in Subsection 6.3.2.

1.2.2.5.2 Automatic Depressurization System

The automatic depressurization system rapidly reduces reactor vessel pressure in a LOCA situation in which the HPCS system fails to maintain the reactor vessel water level. The depressurization provided by the system enables the low-pressure emergency core cooling systems to deliver cooling water to the reactor vessel. The ADS will not be activated unless either the LPCS or LPCI pumps are operating. This is to ensure that adequate coolant will be available to maintain reactor water level after the depressurization. This system is further discussed in Subsection 6.3.2.

1.2.2.5.3 Low-Pressure Core Spray System

The LPCS system consists of one independent pump and the valves and piping to deliver cooling water to a spray sparger over the core. The system is actuated by conditions indicating that a breach exists in the reactor coolant pressure boundary, but water is delivered to the core only after reactor vessel pressure is reduced. This system provides the capability to cool the fuel by spraying water into the fuel channels. In conjunction with the HPCS, ADS, AND LPCI mode of RHR, the LPCS can maintain the fuel cladding below final acceptance criteria limits for the entire spectrum of breaks. This system is further discussed in Subsection 6.3.2.

1.2.2.5.4 Low-Pressure Coolant Injection

Low-pressure coolant injection is an operating mode of the residual heat removal (RHR) system, but it is discussed here because the LPCI mode acts as an engineered safety feature in conjunction with the other emergency core cooling systems. LPCI uses the pump loops of the RHR system to inject cooling water directly into the pressure vessel. LPCI is actuated by conditions indicating a breach in the reactor coolant pressure boundary, but water is delivered to the core only after reactor vessel pressure is reduced. LPCI operation provides the capability of core reflooding, following a loss-of-coolant accident, in time to maintain the fuel cladding below final acceptance criteria limits. This system is further discussed in Subsection 6.3.2.

1.2.2.6 Auxiliary Systems

Certain supportive equipment have functions which relate indirectly to the safety performance of those systems previously described in Subsections 1.2.2.1 through 1.2.2.5. Some supportive equipment regulates the internal environments in which the engineered safety systems normally operate, hence they contribute to the assurance of a "ready status" for these ESF systems. In case of accident, they provide standby power, added heat sink capacity for thermal control, and an added assurance of reactor shutdown capability. For completeness, this auxiliary equipment is briefly noted here because it indirectly supports safety objectives.

1.2.2.6.1 Reactor Building Closed Cooling Water System

The reactor building closed cooling water system consists of five pumps, five heat exchangers, and control and instrumentation to provide adequate cooling for the reactor auxiliary systems. Spare equipment is provided to ensure adequate cooling capacity during normal conditions. This system is further discussed in Subsection 9.2.3.

1.2.2.6.2 CSCS Equipment Cooling

The CSCS equipment cooling water system supplies cooling water to the RHR heat exchangers, diesel-generator coolers, core standby cooling system (CSCS) area coolers, and the LPCS and RHR pumps. Each unit's CSCS consists of three separate electrical and physical divisions, one of which is shared between units.

Each division is provided with separate pumps and draws cooling water from the CSCS cooling pond through separate intake pipes. Cooling water is returned to the station from the CSCS cooling pond through three discharge pipes corresponding to the three divisions of each unit.

1.2.2.6.3 Shielding Building

The Mark II containment concept does not require a shielding building.

1.2.2.6.4 Reactor Building Ventilation Radiation Monitoring System

The reactor building ventilation radiation monitoring system consists of radiation detectors which monitor the activity level of the normal exhaust from the reactor building en route to the station vent stack. Upon detection of high radiation due to an accidental release, the reactor building is automatically isolated and the standby gas treatment system is started. For small, nonaccident releases of radioactivity, the drywell purge unit is utilized to exhaust to the station vent stack. This monitoring system is further discussed in Subsection 7.6.1.

1.2.2.6.5 Main Steamline Radiation Monitoring System

The main steamline radiation monitoring system consists of four gamma radiation monitors located external to the main steamlines just outside the primary containment. The monitors are designed to detect a gross release of fission products from the fuel.

Upon detection of high radiation, the signals generated by these monitors are used to provide an alarm in the control room. This system is further discussed in Subsection 7.6.1.

1.2.2.6.6 Nuclear Leak-Detection System

The nuclear leak-detection system consists of temperature, pressure, flow, and radioactivity sensors and associated instrumentation with alarms used to detect and annunciate leakage from the following system:

a. nuclear boiler system,

b. reactor water cleanup (RWCU) system,

c. residual heat removal (RHR) system,

d. reactor core isolation cooling (RCIC) system,

e. fuel pool cooling system,

f. feedwater system

g. fuel pool cooling system, and

h. instrument lines associated with the above systems.

Small leaks are detected by temperature and pressure changes, fill-up rates of drain sumps, and fission-product concentration inside the primary containment. Large leaks are also detected by changes in reactor water level and changes in flow rates in process lines. This system is further discussed in Subsection 5.2.5.

1.2.2.6.7 Standby A-C Power Supply

Standby a-c power is supplied from five diesel generators. Two diesel-generators are provided for each Unit 1 and 2. The other diesel-generator is arranged to serve essential auxiliaries for either Unit 1 or Unit 2.

This subject is further discussed in Subsection 8.3.1.

1.2.2.6.8 D-C Power Supply

D-c power supplies consist of storage batteries of ample capacity for all essential emergency loads for a minimum period of 4 hours. D-c power supplies of three different voltage levels are provided for each of the two units.

Two independent 24-volt batteries are provided for each unit for neutron-monitoring instrumentation.

A three battery 125-volt system is provided for each unit for circuit breaker controls and other essential control systems. Each independent battery feeds its respective ESF division.

A separate 250-volt system is also provided for each unit for essential power required for valve operators and emergency pump motors.

These battery systems are described in Subsection 8.3.2.

1.2.2.6.9 Standby Liquid Control (SLC) System

Although not intended to provide prompt reactor shutdown, as the control rods do, the standby liquid control system provides a redundant, independent, and different way to bring the nuclear fission reaction to subcriticality and to maintain subcriticality as the reactor cools. The system makes possible an orderly and safe shutdown in the event that not enough control rods can be inserted into the reactor core to accomplish shutdown in the normal manner. The system is sized to counteract the positive reactivity effect from rated power to the cold shutdown condition. This system is discussed in Subsection 9.3.5.

1.2.2.6.10 Station Equipment and Safe Shutdown from Outside the Control Room

A separate remote shutdown control panel is provided in the auxiliary-electric equipment room, with sufficient indication to knowledgeably shut down the reactor from outside the control room. This item is discussed in Subsection 7.4.4.

1.2.2.6.11 Combustible Gas Control

The combustible gas control system consists of a hydrogen recombiner for each unit, with a crosstie between units for redundancy. In the event of a LOCA, the recombiner system can be actuated to prevent the hydrogen-oxygen level within the primary containment from reaching the flammability limit. The hydrogen recombining function of the hydrogen recombiners is abandoned in place. This system is further discussed in Subsection 6.2.5.

1.2.3 Station Description of Features Important to Power Generation

This section provides an overview of those plant features which are important to the power generation objective for LaSalle County Station. The power conversion equipment has the primary function of converting the internal steam energy to electricity. The power conversion system (PCS), with its associated process control and instrumentation, and the connected radwaste treatment systems are all important to power generation. Additionally, certain auxiliary systems which support the power equipment are briefly discussed in the following subsections. Examples of such auxiliaries include: new and wet spent fuel storage facilities, the fuel pool cleanup system, the reactor makeup water demineralizer, and the HVAC systems that condition the facilities in which these PCS functions take place.

1.2.3.1 Power Conversion System

The power conversion system actually includes five interrelated systems: the turbine-generator, the main steamlines with control valving, the main condenser, the circulating water system, and the condensate and feedwater system. A brief description follows for each constituent system.

1.2.3.1.1 Turbine-Generator

The turbine is an 1800-rpm, tandem-compound, six-flow, reheat unit with an electrohydraulic governor for normal operation. The turbine-generator is provided with an emergency trip system for turbine overspeed. The approximate rating of the turbine-generator is 1,206,500 kW at approximately 2 in Hg abs exhaust pressure.

The generator is a direct-driven, 3-phase, 60-Hz, 25,000-Volt, 1800-rpm, hydrogen inner-cooled, synchronous generator rated at 1,355,400 kVA at .904 power factor, 0.58 short circuit ratio at a maximum hydrogen pressure of 75 psig.

The turbine-generator is discussed in Section 10.2.

A turbine gland seal subsystem is provided to minimize air in-leakage or radioactive steam out-leakage. The subsystem consists of a steam evaporator, steam seal pressure regulator, steam seal header, two full-capacity gland seal steam condensers with the associated piping, valves, and instrumentation. This subsystem is further discussed in Subsection 10.4.3.

A steam bypass subsystem is provided which passes steam directly to the main condenser under the control of the pressure regulator. Steam is bypassed to the condenser whenever the reactor steaming rate exceeds the turbine-generator load (such as during generator synchronization or following a large electrical load rejection). The capacity of the turbine steam bypass subsystem is 23% of the reactor rated steam flow. This subsystem is further discussed in Subsection 10.4.4.

1.2.3.1.2 Main Steamlines

In the context of the power conversion sytem, the main steamlines consist of four 26-inch-diameter lines from the outermost main steamline isolation valves to the main turbine stop valves. The use of four main steamlines permits testing of the turbine stop valves and main steamline isolation valves during station operation without load reduction. The design pressure and temperature of the main steamlines from the outermost MSIV to the turbine valve is 1250 psig at 575°F. This component is further discussed in Subsection 5.4.9.

1.2.3.1.3 Main Condenser

The main condenser is a single-shell, single-pass, deaerating-type condenser with a divided water box. The condenser includes provisions for accepting up to 23% of the main steam flow at design conditions from the turbine bypass system and serves as a heat sink for several other flows, such as exhaust steam from the feed pump turbines, cascading heater drains, and feedwater heater shell operating vents. This item is discussed in Subsection 10.4.1.

A main condenser evacuation subsystem is provided to remove noncondensable gases from the condenser, including air inleakage and radiolytic dissociation products originating in the reactor, and to exhaust them to the gaseous radwaste system. The subsystem consists of two 100%-capacity, twin-element, two-stage, steam jet air ejector (SJAE) units complete with intercondensers for normal plant operation and a mechanical vacuum pump for use during startup and shutdown. This subsystem is discussed in Subsection 10.4.2.

1.2.3.1.4 Circulating Water System

The circulating water system provides the condenser with a continuous supply of cooling water. The circulating water system takes water from a man-made perched cooling lake. Makeup water to the lake is provided from the Illinois River.

1.2.3.1.5 Condensate and Feedwater System

The condensate and feedwater system delivers condensate from the condenser hotwell to the reactor pressure vessel. Condensate is pumped by four condensate pumps (one spare) through the intercondenser of the steam jet air ejector, the off-gas condenser, and the gland steam condenser. After leaving the gland steam condenser, the condensate is pumped through a full-flow condensate demineralizer system. The demineralizer effluent is then pumped by four condensate booster pumps (one spare) through the low-pressure heaters. The heaters are split into three one-third capacity parallel streams each stream consisting of five low pressure heaters in series. The last low-pressure heater discharges to the suction of the reactor feedwater pumps. The discharge from the two turbine-driven reactor feedwater pumps and/or the motor-driven feedwater pump passes through the sixth stage of feedwater heating and then to the reactor pressure vessel. Feedwater flow is controlled by varying the speed of the turbine-driven feedwater pump or the position of the regulating valves on the motor-driven reactor feedwater pump. This system is further discussed in Subsection 10.4.7.

The condensate demineralizer subsystem is discussed in Subsection 10.4.6.

1.2.3.2 Electrical Systems and Instrumentation Control

This subsection provides a general overview of the electrical subsystems and of instrumentation and control. All safety systems are supplied with redundant power supplies. This subject is further discussed in Chapters 7.0 and 8.0.

1.2.3.2.1 Electrical Power System

The plant consists of two main generator units designated as Unit 1 and Unit 2. Each main generator is directly connected to a main power transformer through an isolated phase electrical bus duct. The main power transformers transform the output of each generator from the generator voltage to a nominal 345-kV for transmission.

The output of each main power transformer is connected to a 345-kV switchyard consisting of circuit breakers, disconnect switches, buses, and associated equipment.

Overhead 345-kV transmission lines distribute power to various points on the transmission network. This system is further discussed in Chapter 8.0.

1.2.3.2.2 Electrical Power System Process Control and Instrumentation

Main generator electrical controls are located in the station control room. These include the main generator circuit breaker controls, the synchronizing equipment, the generator excitation and voltage control equipment, and the circuit breaker controls for all main supply circuits to the auxiliary power system.

High-speed protective relaying equipment is provided for the main generators, main and auxiliary transformers, main buses, transmission lines, and interconnecting cables and bus ducts so as to provide proper clearing of this equipment in the event of electrical faults. The protective relay system includes breaker failure protection and backup relaying to ensure proper clearing of electrical faults in the event of a failure of the primary protective relaying.

Instrumentation is provided in the main control room for the main generator equipment. This includes indicating instruments for voltage, current, Megawatt (MW), megavolt ampere reactive (MVAR), and frequency. Recording instruments are provided for generator-MW output. KWh meters are provided for main generator outputs and for auxiliary power system loads.

Instrumentation is also provided for monitoring the generator and transformer performance.

Control of transmission line circuit breakers is by remote action from the station control room.

Electrical instrumentation is discussed in Chapter 7.0.

1.2.3.2.3 Nuclear System Process Control and Instrumentation

1.2.3.2.3.1 Rod Control Management System

The rod control management system provides the means by which control rods are positioned from the control room to regulate reactor power. The system operates valves in each hydraulic control unit to change control rod position. Only one control rod can be manipulated at a time. The system includes these hydromechanical blocks that restrict control rod movement under certain conditions as a backup to procedural controls. This system is discussed in Subsection 7.7.2.

1.2.3.2.3.2 Deleted

1.2.3.2.3.3 Recirculation Flow Control System

The recirculation flow control system adjusts the variable-position flow control discharge valve. This changes the coolant flow rate through the core and thereby changes the core power level. The system automatically matches the reactor power output to the load demand. This system is discussed in Subsection 7.7.3.

1.2.3.2.3.4 Neutron Monitoring System

The neutron monitoring system is a system of incore neutron detectors and out-of-core electronic monitoring equipment. The system provides indication of neutron flux, which can be correlated to thermal power level for the entire range of flux conditions that can exist in the core. The local power range monitors (LPRM's) and average power range monitors (APRM's) allow assessment of local and overall flux conditions during power range operation. Automatic control rod blocks, based on input signals from the neutron monitoring system, prevent rod withdrawal beyond the point of limited local reactor power for the existing reactor coolant flow rate. The traversing incore probe (TIP) system provides a means to calibrate the individual LPRM sensors.

1.2.3.2.3.5 Reactor Vessel Instrumentation

In addition to instrumentation for the nuclear safety systems and engineered safety features, instrumentation is provided to monitor and transmit information that can be used to assess conditions existing inside the reactor vessel and the physical condition of the vessel itself. This instrumentation monitors reactor vessel pressure, water level, coolant temperature, reactor core differential pressure, coolant flow rates, and reactor vessel head inner seal ring leakage. This topic is further discussed in Subsection 7.7.1.

1.2.3.2.3.6 Process Computer System

An on-line process computer is provided for each unit to monitor and log process variables and to make certain analytical computations. This system is further discussed in Subsection 7.7.7.

1.2.3.3 Power Conversion Systems Process Control and Instrumentation

The power conversion systems are controlled by the equipment described in the following. Instrumentation is provided to sense a need for a controlling action.

1.2.3.3.1 Pressure Regulator and Turbine-Generator Control

The pressure regulator and turbine-generator instrumentation is classified as non-safety-related. It includes the remote turbine-generator controls, a redundant electrical supply, computer-operated automatic controls, and bypass valves and lines to relieve reactor vessel pressure.

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

The turbine-generator speed-load controls act to maintain constant turbine speed (generator frequency) and to respond to load changes by adjusting the reactor recirculation flow controller and the pressure regulator operating points.

The turbine-generator speed-load controls can initiate rapid closure of the turbine control valves (rapid opening of the turbine bypass valves) to prevent turbine overspeed upon loss of generator electric load. This is necessary to compensate for the delay of the nuclear boiler to respond to turbine-generator load fluctuations.

This item is discussed further in Subsections 7.7.5.

1.2.3.3.2 Feedwater System Control

A three-element controller is used to regulate the feedwater system so that proper water level is maintained in the reactor vessel. The controller uses main steam flow rate, feedwater flow rate, and reactor water level error signals. The feedwater control signal maintains a programmed level by varying the speed of the turbine-driven feedwater pumps and/or by varying the flow control valve position on the discharge of the constant speed motor-driven feedwater pump. Alternatively, operation in single element control is available.

1.2.3.4 Radioactive Waste Systems

The radioactive waste systems provide a means to monitor, remove, treat, and dispose of radioactive wastes in a manner consistent with the applicable sections of 10 CFR 20 and 10 CFR 50, Appendix I.

1.2.3.4.1 Gaseous Radwaste System

Each unit has a completely independent gaseous radwaste system discharging to the station vent stack. All radioactive gaseous effluents are controlled and released via the station vent stack.

The diffusion and dispersal characteristics of the station vent stack enable release without processing of low-level effluents.

Steam for the turbine gland sealing system is provided by an auxiliary steam seal evaporator. Because clean water is used as feedwater to the evaporator, the expected release of radionuclides from the gland seal condenser is expected to be minimal. The system is fully described in Subsection 10.4.3.

The main condenser is the largest volumetric source of gaseous radioactive effluent. Treatment of these gases includes high-temperature catalytic recombining, holdup for decay, high-efficiency particulate filtration, and charcoal adsorption. Effluent monitoring is provided to ensure that the released activity is well within federal limits.

This system is discussed in Section 11.3.

1.2.3.4.2 Liquid Radwaste System

This system collects, treats, stores, and disposes of or recycles all radioactive liquid wastes. Liquid wastes are accumulated in sumps and drain tanks at various locations throughout the plant and are then transferred to collection tanks in the radwaste facility for subsequent treatment, storage, and transport to the solid radwaste system for ultimate disposal. Wastes are processed on a batch basis, with each batch being processed by methods appropriate for the particulate type and quantity of isotopic materials present. Processed liquid wastes are routed to the cycled condensate system or by the waste discharge piping to the river. The liquid wastes in the discharge piping are sufficiently diluted with cooling lake water to achieve a concentration for discharge into the Illinois River well within state and federal concentration limits. A design dilution factor of approximately 670 prior to discharge to the river is typical.

Radwaste equipment is selected, arranged, and shielded to permit operation, inspection, and maintenance with minimum personnel exposure. Processing equipment is selected and designed to require a minimum of maintenance.

Protection against accidental discharge of liquid radioactive waste is provided by instrument redundancy, for detection and alarm of abnormal conditions, and by procedural controls.

This system is discussed in Section 11.2.

1.2.3.4.3 Solid Radwaste System

Solid radioactive wastes are collected, processed, and packaged for storage. These wastes are generally stored on the site until the isotopes with short half-lives have decayed. Outside storage on turning pads 4 & 5 is used for low-level waste casks. Ultimately, the waste is loaded and shipped to a burial site.

The solid radwaste system is designed to maintain radiation exposures to personnel "as low as reasonably achievable" during system operation and maintenance.

This system is discussed in Section 11.4.

1.2.3.5 Radiation Monitoring and Control

Radiation monitoring systems are provided to monitor and control radioactivity in process and effluent streams and to activate appropriate alarms and controls.

A process radiation monitoring system is provided for indication and recording radiation levels associated with plant process streams and effluent paths leading to the environment. All effluents from the plant which are potentially radioactive are monitored.

Process radiation monitoring is also discussed in Sections 9.3 and 11.5.

1.2.3.6 Miscellaneous Power Generation

1.2.3.6.1 New and Spent Fuel Storage

New and spent fuel storage racks are designed to prevent inadvertent criticality and load buckling. Sufficient coolant and shielding are maintained to prevent overheating and excessive personnel exposure, respectively. The design of the fuel pool provides for corrosion resistance, adherence to Seismic Category I requirements, and prevention of k_{eff} from reaching 0.95 under flooded conditions. The new fuel vault design prevents k_{eff} from reaching 0.90 under dry conditions, and 0.95 under flooded conditions. This subject is further discussed in Section 9.1.

Spent nuclear fuel may also be stored in Dry Casks as part of an Independent Spent Fuel Storage Installation (ISFSI) within the Protected Area (PA). This form of storage is discussed in Section 9.1.

1.2.3.6.2 Fuel Pool Cleanup System

The fuel pool cooling and cleanup subsystem provides the removal of decay heat from stored spent fuel and maintains specified water temperature, purity, clarity, and level. This prevents spent fuel overheat and the buildup of excessive radioactive materials in the cooling water, thereby minimizing possible exposures to plant personnel.

1.2.3.6.3 Service Water System

The normal service water system supplies cooling water for turbine-generator and miscellaneous HVAC loads, fuel pool cooling, and the heat exchangers in the turbine building and reactor building closed cooling water systems. Service water for traveling screen wash and fire protection is also provided by this system. Gland water to the circulating water pumps (Unit 1 only) is also provided by this system. This system is further discussed in Subsection 9.2.2.

1.2.3.6.4 Demineralized Water Makeup System

The demineralized water makeup system is abandoned-in-place and has been replaced with a vendor trailer. The demineralized water makeup system for LSCS, Units 1 and 2, provides demineralized water for plant usage. The system consists of a vendor trailer, which is capable of producing 72,000 gallons of demineralized water per day. A detailed discussion of the demineralized water makeup system is in Subsection 9.2.4.

1.2.3.6.5 Station HVAC

The ventilation for the radwaste building is provided by a once-through system which uses evaporative coolers. Evaporative coolers (abandoned-in-place) are shutdown and controlled administratively. Exhaust air is filtered through HEPA filters en route to the station vent stack. The station vent stack has a full-time stack monitoring system for radioactivity.

The HVAC systems are described in Section 9.4.

1.2.3.6.6 Heating, Ventilating, and Air-Conditioning (HVAC) Systems

Separate HVAC systems exist for the control room, the auxiliary electric equipment room and the rooms standby diesel generators. These HVAC systems, the CSCS equipment area coolers, and the switchgear heat-removal systems are designed to operate under all station conditions.

The CSCS equipment area cooling system consists of four water cooled air blowers for each primary containment that supplies cool air to respective CSCS pump cubicles.

All air distribution systems are designed so that airflow is directed from areas of lower contamination to areas of progressively higher potential contamination.

1.2.4 Glossary

1.2.4.1 Definitions

The following definitions apply to the terms used in the LaSalle County Station, Units 1 and 2, Updated Final Safety Analysis Report:

Accident -- A single event, not reasonably expected during the course of station operation, that has been hypothesized for analysis purposes or postulated from unlikely but possible situations, and that causes or threatens a rupture of a radioactive material barrier.

Active Component -- A safety related component characterized by an automatically initiated change of state or discernible mechanical action in response to an imposed demand.

Active Failure -- The failure of an active component to perform its function when called upon to do so by an initiating signal.

Administrative Controls -- The provisions relating to organization and management, personnel function procedures, recordkeeping, review and audit, and reporting necessary to ensure responsible operation of the facility.

Anticipated Operational Occurrences -- Those abnormal conditions of operation that are expected to occur one or more times during the life of the nuclear power unit, whose consequences do not affect safety.

Auxiliary Building -- A Seismic Category I building adjacent to the secondary containment (reactor building).

Availability -- The probability that a component will be operable when called upon to perform its specified function.

Available Reactor Power -- The steam power available for the turbine and other heat cycle equipment.

Boiling Length -- In a heated fuel bundle, the length that is producing net steam generation.

Channel -- An arrangement of one or more sensors and associated components used to evaluate station variables and produce discrete outputs used in logic. A channel terminates and loses its identity where individual channel outputs are combined in logic.

Cold Shutdown -- The condition of the reactor when the reactor is shut down; the reactor coolant is maintained at less than 212° F, and the reactor vessel is near atmospheric pressure.

Components -- Items from which a functional system is assembled.

Design Basis -- That information which identifies the specific functions to be performed by a structure, system, or component, and the specific values or ranges of values chosen for controlling parameters as reference bounds for design.

Design-Basis Accident -- A hypothesized accident, the characteristics and radiological consequences of which are utilized in the design of those systems and components pertinent to the preservation of radioactive material barriers.

Design Power -- Refers to the power level at which the reactor is producing 102% of reactor vessel rated steam flow.

Diesel-Generator Building -- That Seismic Category I building which houses the standby diesel generator systems.

Drywell --A pressure and radioactive material barrier, surrounding the reactor vessel and its recirculation loops, that conveys steam resulting from a postulated LOCA to the suppression pool for condensation.

Emergency Core Cooling Systems -- The systems which furnish cooling water to the core to compensate for a loss of normal cooling capability during the postulated loss-of-coolant accidents.

Engineered Safety Features -- Systems provided to mitigate the consequences of postulated accidents.

Excursion -- A sudden, very rapid rise in the reactor power level.

Functional Test -- The intentional operation or initiation of a system, subsystem, or component to verify that it operates within design tolerances.

Hot Shutdown -- The reactor condition when the mode switch is in the shutdown position and the reactor coolant temperature is greater than 212° F.

Hot Standby Mode -- The condition of the reactor when it is operating with the coolant temperature greater than 212° F, the system pressure less than 1060 psig, and the mode switch in the startup position.

Logic -- That array of components which combines individual bistable output signals to produce decision outputs.

Loss-of-Coolant Accidents -- Those postulated accidents that result from the loss of reactor coolant at a rate in excess of the capability of the reactor coolant makeup system, and from breaks in the reactor coolant pressure boundary, up to and including a break equivalent in size to the double-ended rupture of the largest pipe of the reactor coolant system.

Minimum Critical Power Ratio (MCPR) -- The lowest ratio of that power which results in onset of transition boiling to the actual bundle power at the same location.

Module -- Any assembly of interconnected components that constitutes an identifiable device, instrument, or piece of equipment.

Nuclear Power Unit -- A nuclear power reactor and associated equipment necessary for electric power generation, including those structures, systems, and components required to provide reasonable assurance that the facility can be operated without undue risk to the health and safety of the public.

Nuclear Steam Supply System (NSSS) -- A contractual term which designates those components of the nuclear power system and their related engineered safety features and instrumentation furnished by the nuclear steam supply system supplier (GE).

Operator Error -- An active deviation from written operating procedures or nuclear station standard operating practices.

Passive Component -- A safety related component characterized by no change of state nor mechanical motion.

Passive Failure -- Loss of function of a passive component.

Power Generation Design Basis -- The unique design requirements that establish the power generation objective.

Power Generation Evaluation -- A comparison to show how the system satisfies the power generation design bases.

Power Generation System -- Any system not essential to safety, but essential to power generation.

Power Operation -- A time reference which begins where "heatup" ends and includes continued operation of the station at power levels in excess of heatup power.

Primary Containment -- The drywell in which the reactor vessel is located, the pressure suppression chamber, and the process lines out to the second isolation valve.

Rated Reactor Power -- Refers to the power level at which the reactor is producing 100% steam flow.

Reactor Building -- The Seismic Category I structure comprising the secondary containment.

Reactor Isolated -- A condition wherein the reactor is isolated from the condenser.

Reactor Mode Switch Positions -- Four modes of reactor operation for which switch positions are available as follows:

a. Shutdown Mode -- Condition of the reactor when it is shut down, the reactor mode switch is in the shutdown mode position, and all operable control rods are fully inserted.

b. Startup Mode -- Condition of the reactor when the reactor mode switch is in the startup mode position.

c. Run Mode -- Condition of the reactor when the reactor mode switch is in the run mode position.

d. Refuel Mode -- Condition of the reactor when the reactor mode switch is in the refuel mode position.

Safety Design Basis -- The unique design requirements that establish the safety objective.

Safety Evaluation -- A comparison to show how the system satisfies the safety design basis.

Safety Related -- Those structures, and equipment necessary to maintain the integrity of the reactor coolant pressure boundary, to shut down the reactor and maintain it in a safe shutdown condition, and/or to prevent or mitigate the consequences of accidents.

Scram -- The simultaneous rapid insertion of all control rods into the core.

Secondary Containment or Reactor Building -- A Seismic Category I building that completely encloses the primary containment.

Sensor -- That part of a channel used to monitor a measurable power plant variable.

<u>Setpoint</u> -- That value of a monitored plant variable that results in a channel trip when the monitored variable reaches or exceeds this value.

<u>Shutdown</u> -- The reactor condition when the effective multiplication factor is sufficiently less than 1.0 such that the withdrawal of any one control rod could not produce criticality under the most restrictive potential conditions of temperature, pressure, burnup, and fission-product concentration.

<u>Single Failure</u> -- An occurrence that results in the loss of capability of a safety related component to perform its intended safety fuctions.

<u>Source Material</u> -- Uranium or thorium or any combination thereof, in any physical or chemical form; or ores which contain by weight one-twentieth of one percent (0.05%) or more of uranium, thorium, or any combination thereof. Source material does not include special nuclear material.

<u>Special Nuclear Material</u> -- Plutonium, uranium-233, uranium enriched in the isotope 235, and any other material that the NRC, pursuant to the provisions of Section 51 of the Atomic Energy Act of 1954, as amended, determines to be special nuclear material, or any material artificially enriched by any of the foregoing. Special nuclear material does not include source material.

<u>Standby Gas Treatment System (SGTS)</u> -- An engineered safety system in the reactor building that processes leakage from the primary containment and discharges it after treatment to the atmosphere.

<u>Station Vent Stack</u> -- The common exhaust providing elevated release (370 feet above grade) for all gaseous effluents and plant ventilation air.

<u>Suppression Pool</u> -- A pool of water, located in the suppression chamber under the drywell, which normally provides the water seal between the drywell and containment.

Technical Specifications -- A set of detailed technical requirements and limits which establish the operational envelope for the plant, based on safety considerations.

<u>Test Interval</u> -- The elapsed time between the initiation of sequential identical tests.

<u>Trip</u> -- The change of state of a bistable device from a normal condition.

<u>Turbine Cycle Rated Power</u> -- Rated power available for the turbine.

<u>Unit</u> -- A nuclear steam supply system, turbine-generator, and supporting facilities.

1.2.4.2 ACRONYMS USED IN LSCS-UFSAR

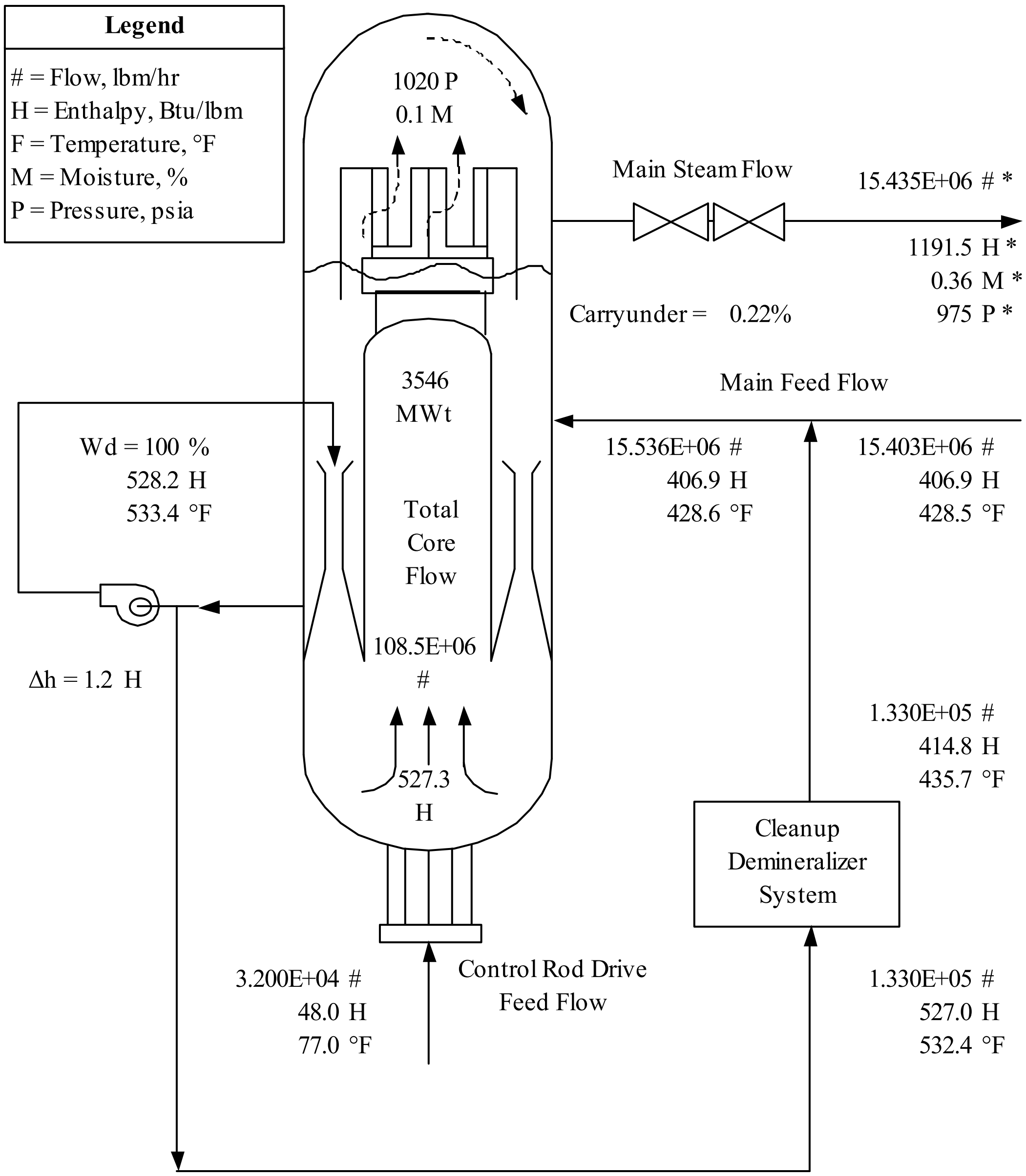
ADS	Automatic Depressurization System
AEER	Auxiliary Electric Equipment Room
APRM	Average Power Range Monitor
ARI	Alternate Rod Insertion
ARM	Area Radiation Monitor
ATWS	Anticipated Transients Without Scram
BWR	Boiling Water Reactor
CCW	Closed Cooling Water
ComEd	Commonwealth Edison Company
CECo	Commonwealth Edison Company
CHF	Critical Heat Flux
CRD	Control Rod Drive
CRPI	Control Rod Position Indication
CSCS-ECWS	Core Standby Cooling System - Equipment Cooling Water System
DBA	Design-Basis Accident
DG	Diesel Engine-Generator
DIB	Digital Isolation Block
ECCS	Emergency Core Cooling Systems
EFCV	Excess Flow Check Valve
EGC	Exelon Generation Company, LLC
EHC	Electrohydraulic Control
ESF	Engineered Safety Feature
FA	Full Arc (mode of TCV operation)
FLECHT	Full-Length Emergency Cooling Heat Transfer
FPCC	Fuel Pool Cooling and Cleanup
FSAR	Final Safety Analysis Report
GE	General Electric Company
HCU	Hydraulic Control Unit
HEPA	High-Efficiency Particulate Air/Absolute (referring to filters)
HPCS	High-Pressure Core Spray
HX	Heat Exchanger
H&V	Heating and Ventilating
HVAC	Heating, Ventilating, and Air-Conditioning
IGSCC	Intergranular Stress Corrosion Cracking
HWC	Hydrogen Water Chemistry
IAC	Interim Acceptance Criteria (NRC)
IRM	Intermediate Range Monitor
IRSF	Interim Radwaste Storage Facility
LCO	Limiting Condition of Operation
LDS	Leak-Detection System
LOCA	Loss-of-Coolant Accident
LPCS	Low-Pressure Core Spray
LPRM	Local Power Range Monitor
LRCP	Liquid Radwaste Control Panel
LSCS	LaSalle County Station
LSSS	Limiting Safety System Setting

LPZ	Low Population Zone
M/A	Manual/Auto
MCC	Motor Control Center
MCPR	Minimum Critical Power Ratio
MDRFP	Motor Driven Reactor Feed Pump
MG	Motor-Generator Set
MLD	Mean Low Water Datum
MSL	Mean Sea Level
MSIV	Main Steam Isolation Valve
MSIV-ICLTM	Main Steam Isolation Valve Isolated Condenser Leakage Treatment Method
MSIV-LCS	Main Steam Isolation Valve Leakage Control System
NB	Nuclear Boiler
NBR	Nuclear Boiler Rated (power)
NED	Nuclear Energy Division (GE)
NMS	Neutron-Monitoring System
NSSS	Nuclear Steam Supply System
NSSSS	Nuclear Steam Supply System Shutoff
NSOA	Nuclear Safety Operational Analysis
OBE	Operating Basis Earthquake
OPRM	Oscillation Power Range Monitor
PA	Public Address (System)
PMF	Probable Maximum Flood
PMP	Probable Maximum Precipitation
P&ID	Piping and Instrumentation Diagram
PRM	Power Range Monitor
PSAR	Preliminary Safety Analysis Report
PCS	Process Computer System
RBM	Rod Block Monitor
RCM	Rod Control Management
RCPB	Reactor Coolant Pressure Boundary
RCIC	Reactor Core Isolation Cooling
RHR	Residual Heat Removal
RMC	Reactor Manual Control
RPS	Reactor Protection System
RPV	Reactor Pressure Vessel
RWCU	Reactor Water Cleanup
RWM	Rod Worth Minimizer
SAR	Safety Analysis Report
SGTS	Standby Gas Treatment System
SJAE	Steam Jet Air Ejector
S&L	Sargent & Lundy
SLC	Standby Liquid Control
SPC	Siemens Power Corporation, Nuclear Division
SPF	Standard Project Flood
SPS	Standard Project Storm
SRM	Source Range Monitor
SRV	Safety/Relief Valve

SSE	Safe Shutdown Earthquake
SW	Service Water
TBCCW	Turbine Building Closed Cooling Water
TCV	Turbine Control Valve
TDRFP	Turbine Driven Reactor Feed Pump
TG	Turbine-Generator
TIP	Traversing Incore Probe
URC	Ultrasonic Resin Cleaner

1.2.5 References

1. LaSalle County Station, "Mark II - Design Assessment Report," Commonwealth Edison Company, February 1976.



*Conditions at upstream side of TSV

Core Thermal Power	3546.0	
Pump Heating	12.4	
Cleanup Losses	-4.4	
Other System Losses	-1.1	
Turbine Cycle Use	3552.9	MWt

LASALLE COUNTY STATION
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FIGURE 1.2-1

REACTOR SYSTEM - RATED POWER HEAT BALANCE

1.3 COMPARISON TABLES

1.3.1 Comparison with Similar Facility Designs

A comparison of the principal design features of the LaSalle County Station (LSCS) with those of other boiling water reactor facilities was included in Section 1.3 of the FSAR which compared LSCS with Zimmer 1, Washington Public Power Supply System (WPPSS) 2, and Hatch 1, listing the design characteristics of the following:

a. nuclear steam supply,

b. power conversion,

c. engineered safety features,

d. containment,

e. radioactive waste management,

f. structural,

g. instrumentation and electrical, and

h. standby gas treatment.

This information was current at the time the LSCS Unit 1 operating license (OL) was granted and has not since been revised.

1.3.2 Comparison of Final and Preliminary Information

Table 1.3-9 of the FSAR provided a list of significant differences between the final and preliminary designs of the LaSalle County Station. This information was current at the time the LSCS Unit 1 OL was granted and has not since been revised.

1.4 IDENTIFICATION OF AGENTS AND CONTRACTORS

The identification of the principal agents and contractors involved in the design and construction of the LaSalle County Station is included in Section 1.4 of the FSAR. This information was current at the time the LSCS-1 operating license was granted and has not since been revised.

1.5 REQUIREMENTS FOR OTHER TECHNICAL INFORMATION - CURRENT CONCERNS FROM LSCS ACRS LETTER

The concerns of the Advisory Committee on Reactor Safeguards pertaining to LSCS at the time the LSCS-1 OL was granted were addressed in Section 1.5 of the FSAR. Modifications made in response to those concerns have been identified and were documented in amendments to the FSAR.

1.6 MATERIAL INCORPORATED BY REFERENCE

Table 1.6-1 of the FSAR provided a list of all GE topical reports and any other report or document which was incorporated in whole or in part by reference in the FSAR and had been previously filed with the NRC. Topical reports and other documents incorporated by reference in the FSAR and in annual UFSAR revisions are included in the reference sections of the applicable chapters in the UFSAR.

Additional documents were incorporated into the UFSAR by reference in the appropriate sections when nuclear reload fuel fabricated by SPC was introduced into the reactor cores with technical support shared by SPC and Commonwealth Edison (ComEd). Technical Requirements Manual also contains pertinent SPC licensing topical reports.

Additional documents were incorporated into the UFSAR by reference in the appropriate sections when the UFSAR was revised due to NRC approval of the Power Uprate License Amendment.