Quad Cities Nuclear Power Station

Updated Final Safety Analysis Report

(UFSAR)

Rev 11 - October 2011

Quad Cities Nuclear Power Station, Unit 1 and 2
Renewed Facility Operating License Nos. DPR-29 (Unit 1) and DPR-30 (Unit 2)
NRC Docket Nos. STN 50-254 (Unit 1) and 50-265 (Unit 2)

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1.1 INTRODUCTION

1.1.1 Background

This Updated Final Safety Analysis Report (UFSAR) is submitted by the Commonwealth Edison Company (CECo) for Quad Cities Station in accordance with the requirement of 10 CFR 50.71(e).

The original Final Safety Analysis Report (FSAR) was submitted in support of the application, as amended, for facility licenses for Units 1 and 2 at Quad Cities Station, under Section 104(B) of the Atomic Energy Act of 1954, as amended, and provided the technical information required by 10 CFR 50.34 and 50.36 of the Regulations of the United States Atomic Energy Commission (AEC) governing the licensing of production and utilization facilities. [1.1.1]

This UFSAR is an updated version of the FSAR and follows a different format than the FSAR. The FSAR was written before the issuance of Regulatory Guide 1.70, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants." In an effort to provide consistency with the industry, the UFSAR was rebaselined and reformatted in 1992 using Regulatory Guide 1.70, Revision 3, November 1978 as guidance. This UFSAR contains a description of Quad Cities Unit 1 and Unit 2. The UFSAR will be revised periodically in accordance with 10 CFR 50.71(e). [1.1.2]

Construction of Units 1 and 2 was authorized by the AEC by issuance to CECo of provisional construction permits on February 15, 1967 for Unit 1, CPPR-23, in AEC Docket 50-254, and for Unit 2, CPPR-24, in AEC Docket 50-265. On July 18, 1968, such construction permits were amended by the AEC to authorize CECo to own an undivided 75% interest and MidAmerican Energy Company (MEC) to own an undivided 25% interest in Units 1 and 2. Units 1 and 2 are identical in virtually all respects, including design concepts and criteria, capacity, and components. The few differences which exist are discussed in the applicable sections of this report. Units 1 and 2 were completed and ready for power operation in 1972. The commercial service dates for Units 1 and 2 were February 18, 1973 and March 10, 1973 respectively. [1.1.3]

This UFSAR contains results from the analysis performed for each Unit; these analyses allow Unit 1 and Unit 2 to operate at a thermal power up to 2957 MWt. The Plant Design and Analysis Report (PDAR) previously filed (AEC Dockets 50-254 and 50-265) states that each of the Units 1 and 2 were designed to permit ultimate operation at power levels of about 2600 MWt. The initial licensed power level for Units 1 and 2 was 2511 MWt. Analyses have been performed as part of the extended power uprate to support operation of the Units at 2957 MWt.

1.1.2 Organization of the Report

The UFSAR is divided into 17 chapters. Each chapter is divided into numbered sections, e.g., the fourth section in Chapter 1 is numbered 1.4. Pages are numbered with two digits corresponding to the chapter and first-level section numbers followed by a hyphen and a sequential number within the section, e.g., the third page in Section 4.1 of Chapter 4 is numbered 4.1-3.

Tables and figures that are referenced in the text appear at the end of the section in which they are referenced; first tables, then figures. Tables and figures are numbered with the chapter and first-level section numbers followed by a hyphen and a sequential number within the section, e.g., the second table in Section 2.4 of Chapter 2 is numbered Table 2.4-2.

The appendices to the UFSAR describe and evaluate material which should be included in the UFSAR but does not fit into the format, content, or level of detail under Regulatory Guide 1.70.

Abbreviations contained in the UFSAR which reference the entity "Commonwealth Edison Company" include: CECO, CECo, and ComEd. The Nuclear Regulatory Commission approved the transfer of the facility licenses from Commonwealth Edison Company to Exelon Generation Company, LLC (Exelon Generation Company or EGC) on August 3, 2000 (Reference 1). References in the UFSAR to Commonwealth Edison, ComEd, and CECo have been retained, as appropriate, instead of being changed to Exelon Generation Company or EGC to properly preserve the historical context. [1.1.4]

1.1.3 References

1. NRC letter, “Braidwood, Byron, Dresden, LaSalle, Quad Cities, and Zion – Orders Approving Transfer to Licenses from Commonwealth Edison Company to Exelon Generation Company, LLC, and Approving Conforming Amendments, “dated August 3, 2000. The issuance of the conforming license amendments (197/193) to reflect the transfer of ownership was transmitted in a NRC letter dated January 12, 2001 (“Quad Cities Nuclear Power Station Units 1 and 2 – Issuance of Conforming Amendment RE: Transfer of Licenses to Exelon Generation Company, LLC”).

1.2 GENERAL PLANT DESCRIPTION

1.2.1 Principal Design Criteria

The principal criteria for design and construction of Units 1 and 2 are summarized below. Specific design criteria and design features are detailed in later sections. [1.2-1]

A. The units are designed, fabricated, erected, and operated to produce electrical power in a safe and reliable manner and, as a minimum, in accordance with applicable codes and regulations.

B. The design of those components which are important to the safety of the units and the station includes allowances for environmental phenomena at the site.

C. The design of those components which are important to the safety of the units and the station permits safe operation of the units and accommodates serious accidents.

1.2.1-1 Reactor Core

A. The reactor core is designed as part of a boiling water reactor (BWR) to produce steam for direct use in a turbine-generator. [1.2-2]

B. The reactor core, in conjunction with other equipment, is designed and operated to prevent the occurrence of uncontrolled power oscillations.

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

D. Power excursions which could result from any credible reactivity addition accident will not cause damage, either by motion or rupture, to the pressure vessel or impair operation of required safeguards.

E. The reactor core is designed so that control rod action, with the highest worth rod fully withdrawn and unavailable for use, shall be capable of bringing the core subcritical and maintaining it so at any point in the operating cycle.

F. A backup reactor shutdown system is provided independent of normal reactivity control provisions. This system has the capability, with adequate margin, to shut down the reactor from any operating condition.

G. The fuel rod cladding is designed to contain the fission gas released from the fuel material throughout the design life of the fuel rod.

H. Thermal characteristics of the reactor core preclude fuel clad surface heat flux or fuel material center temperatures which could cause sudden fuel cladding ruptures.

I. The reactor core and associated systems are designed to accommodate transients and maneuvers which might be expected without compromising safety and without fuel damage.

1.2.1.2 Reactor Core Cooling Systems

A. Heat removal systems are provided to remove heat generated in the reactor core to cover the full range of normal operational conditions from unit shutdown to maximum thermal output. The capacities of such systems are adequate to prevent fuel clad damage. [1.2-3]

B. Heat removal systems are provided to remove decay heat generated in the reactor core under circumstances wherein the normal operational heat removal systems become inoperative. The capabilities of such systems are adequate to prevent fuel clad damage.

C. Redundant heat removal systems are provided to preserve core heat transfer geometry following various postulated loss of coolant accidents.

D. Independent means are provided to prevent overpressure conditions which could jeopardize primary system and reactor core cooling system integrity.

1.2.1.3 Containment

A. The primary containment system is designed, fabricated, and constructed to accommodate, without failure, the pressures and temperatures resulting from or subsequent to the double-ended rupture or equivalent failure of any coolant pipe within the primary containment. [1.2-4]

B. Provisions are made both for the removal of heat from within the primary containment and/or for such other measures as may be necessary to maintain the integrity of the containment system as long as necessary following a loss of coolant accident.

C. The reactor building, encompassing the primary containment system, provides secondary containment when the primary containment is closed and in service, and provides primary containment when the primary containment system is open.

D. Provisions were made for preoperational pressure and leak rate testing of the primary containment system and for subsequent leak testing at periodic intervals after the unit commenced operation. Provision is also made for leak testing selected penetrations and for demonstrating the functional integrity of reactor building containment.

E. The integrity of the complete containment system and other associated engineered safeguards, as may be necessary, are designed and maintained so that offsite doses resulting from postulated accidents will be below the guides presented in 10 CFR 100 (or 10 CFR 50.67 as applicable).

1.2.1.4 Control and Instrumentation

A. The station is provided with a control room having adequate shielding and ventilation facilities to permit occupancy during any design basis accident situation. [1.2-5]

B. Interlocks or other protective devices are provided so that procedural controls are not the only means of preventing serious accidents.

C. A reliable reactor protection system (RPS), independent from the reactor process control system, is provided to automatically initiate a reactor scram whenever plant conditions approach pre-established limits. Periodic testing capability is provided. Sufficient redundancy is provided so that failure or removal from service of any one component or portion of the system will not preclude appropriate actuation of the reactor protection system when required.

1.2.1.5 Electrical Power

Sufficient normal and standby auxiliary sources of electrical power are provided to attain prompt shutdown and continued maintenance of the plant in a safe condition under all credible circumstances. The capacity of the power sources is adequate to accomplish all required safeguards functions under postulated accident conditions for which the plant is designed. [1.2-6]

1.2.1.6 Radioactive Waste Disposal

A. Gaseous, liquid and solid waste disposal facilities are designed so that discharge of effluents and offsite shipments will be in accordance with 10 CFR 20, and within the requirements of the Interstate Commerce Commission and other regulatory agencies having jurisdiction. [1.2-7]

B. Process and discharge streams are appropriately monitored and such features incorporated as may be necessary to maintain releases below the permissible limits of 10 CFR 20. Off-gas monitors are provided on the main condenser air ejector capable of automatically isolating the off-gas system from the stack when abnormally high radiation levels are sensed. This isolation is controlled by a time delay which can be manually overridden by a manual control switch in the control room permitting immediate closure of the holdup valve.

1.2.1.7 Shielding an Access Control

The radiation shielding and the station access control patterns are such that the personnel doses are less than the limits in 10 CFR 20. [1.2-8]

1.2.1.8 Fuel Handling and Storage

Appropriate fuel handling and storage facilities are provided to preclude accidental criticality and to provide cooling for spent fuel. [1.2-9]

1.2.2 Summary Design Description and Safety Analysis

1.2.2.1 Site Characteristics

Information relating to the Quad Cities site and environs is summarized in Chapter 2.0. This information as applicable was used in the design of Quad Cities Units 1 and 2. [1.2-10]

1.2.2.1.1 Design Bases Dependent Upon the Site and Environs Characteristics

1.2.2.1.1.1 Gaseous Waste Effluents

Units 1 and 2 off-gas systems are designed to use a 310-foot chimney for release of the treated radioactive gaseous effluents. The reactor building ventilation discharges through the reactor building ventilation exhaust stack. Release limits are in accordance with 10 CFR 20. [1.2-11]

1.2.2.1.1.2 Liquid Waste Effluents

Units 1 and 2 use diffuser pipes which discharge into the Mississippi River. The radioactive material concentrations at the outlets of the diffuser pipes are in compliance with 10 CFR 20. [1.2-12]

1.2.2.1.1.3 Wind Loading Design

The reactor building, turbine building, service building, radwaste building, crib house, 310-foot chimney, and the control room structures are designed to withstand the maximum potential loadings resulting from a wind velocity of 110 mph, in accordance with standard codes and normal engineering practice. [1.2-13]

Structures whose failure could affect the operation and functions of the primary containment and process systems are designed to assure that safe shutdown of the reactor can be achieved, considering the effects of possible damage when subjected to the forces of tornado loading. [1.2-14]

This subject is further discussed in Section 3.3.1.1.3.

1.2.2.1.1.4 Geology

The plant is supported on Niagaran dolomite. When core borings disclosed the presence of some cavities and crevices in the rock, a procedure of uncovering and filling the crevices with concrete, grouting to fill voids, and a special foundation design to span questionable support areas was undertaken. These construction techniques have resulted in a foundation which will adequately support all design loads without question of safety or undue settlement.

1.2.2.1.1.5 Seismic Design

The following design criteria apply only to Class I items. Class I items are defined as structures (building and equipment) whose failure could cause significant release of radioactivity or are vital to a safe plant shutdown. [1.2-15]

The seismic design for Class I structures and equipment for Quad Cities Station are based on dynamic analyses using acceleration or velocity response spectrum curves which are based on a ground motion of 0.12 g.

The natural periods of vibration are calculated for buildings and equipment which are vital to the safety of the plant. Damping factors are based upon the materials and methods of construction used.

Earthquake design is based on ordinary allowable stress as set forth in the applicable codes, but is more conservative because the usual one-third increase in allowable working stresses due to earthquake loadings is not used. As an additional requirement the design is such that a safe shutdown can be made during a ground motion of 0.24 g.

1.2.2.1.1.6 Conclusions Respecting Site and Environs

The Quad Cities site meets the reactor site criteria in 10 CFR 100 (or 10 CFR 50.67 as applicable) for the following reasons: [1.2-16]

A. The site, consisting of approximately 784 acres, provides the requisite exclusion area. [1.2-17]

B. There are no residences on the site or within a radius of 1/2 mile of the Units. [1.2-18]

C. Units 1 and 2 are independent of each other to the extent that an accident in one would not initiate an accident in the other. The simultaneous operation of the two units will not result in total radioactive effluent releases beyond allowable limits.

D. The calculated total radiation doses under postulated accident conditions to an individual at the boundary of the exclusion area or at the outer boundary of the low population zone are within the limits prescribed by 10 CFR 100 (or 10 CFR 50.67 as applicable).

E. Activities which are permitted on the site, but the unrelated to the operation of any unit, do not present hazards to the public.

F. There are numerous access roads, including State Highway 84, within the low population zone, permitting rapid evacuation.

G. The population density and use characteristics of the site environs in the low population zone are compatible with plant operation.

H. The geological, hydrological, meteorological, and seismological characteristics of the site and environs are suitable.

1.2.2.2 Station Arrangements

The turbine building is shared by Units 1 and 2. The Unit 1 turbine-generator, exciter, condenser, feedwater heaters, feedwater and condensate pumps, condensate polishing filter-demineralizer system, condenser circulating water system, and electrical switchgear occupy the south half of the turbine building. Duplicate equipment and systems for Unit 2 are located in the north half of the building. [1.2-19]

The reactor building is shared by Units 1 and 2 and abuts the east wall of the turbine building. The south half of the reactor building houses the Unit 1 reactor vessel, recirculation system, primary containment, reactor auxiliary systems, refueling equipment, and spent fuel storage, as well as the common new fuel storage vault used for both Units 1 and 2. Except as noted, duplicate equipment for Unit 2 occupies the north half of the reactor building. Below the main refueling floor a concrete wall separates the two units.

Units 1 and 2 are designed to use the same radioactive waste building, which is centrally located adjacent to the west side of the turbine building. This building is a concrete structure containing the control, processing, packaging, and storage facilities necessary to process the solid and liquid waste.

The turbine, reactor, and radwaste buildings are shown in plan views (P&IDs M-7, M-3, M-4, M-5, and M-6) and in elevation views on M-8, M-9 and M-10. Both plan and elevation views of the crib house are contained in M-11. The chimney is shown on M-6. Locations of the various major equipment can be fond on these P&IDs. [1.2-20]

The control room, which serves both units, is located adjacent to the south wall of the turbine building (P&ID M-4). The service building is also south of the turbine building and is coupled to the control room. The technical support center is a separate building located to the south of the south of the service building. The access control building is located a short distance to the east of the service building. [1.2-21]

1.2.2.3 Reactor System

The reactor is a single-cycle, forced-circulation boiling water reactor (BWR) producing steam for direct use in the steam turbine. The reactor core includes the gadolinium-bearing fuel assemblies and control rods. The mechanical, thermal-hydraulic, and nuclear design of this reactor is similar to that of other BWRs designed and built by the General Electric Company (GE). [1.2-22]

The core is assembled in modules, each consisting of four fuel assemblies set in the interstices of a cruciform control rod. This modular core form, common to all GE BWRs, permits substantial increase in thermal power over earlier designs with only a small increase in core diameter. At the same time, desired reactivity control characteristics of earlier designs were preserved.

The reactor pressure vessel contains the reactor core and structure, steam separators and dryers, jet pumps, control rod guide tubes, feedwater spargers, core spray spargers, and other components as shown in Figure 3.9-10. The inside diameter of the vessel is approximately 21 feet and the inside height between heads is approximately 68 1/2 feet. The main connections to the reactor vessel include the steam lines, jet pump motive flow recirculation lines, feedwater lines, and control rod drive (CRD) thimbles. Other connections are provided for the standby liquid control system (SBLC), Emergency Core Cooling System (ECCS), CRD hydraulic, and instrumentation systems. [1.2-23]

The fuel for the reactor core consists of uranium dioxide pellets contained in sealed Zircaloy-2 tubes. These fuel rods are assembled into individual fuel assemblies. Each fuel assembly is fitted with a Zircaloy-2 or Zircaloy-4 flow channel. Water serves as both the moderator and coolant for the core. [1.2-24]

The systems for reactivity control consists of moveable control rods, burnable absorber (gadolinium) in the fuel, and a variable recirculation flow rate. These systems accommodate fuel burnup, load changes and long-term reactivity changes. A standby liquid control system is provided as an independent reactor shutdown system. [1.2-25]

The original control rods consist of assemblies of 3/16-inch diameter, sealed, stainless steel tubes filled with compacted boron carbide (B4C) powder and held in a cruciform array by a stainless steel sheath of 1/16 inch wall thickness fitted with castings at each end. Newer control rod designs use B4C and/or hafnium as the control materials in the rods. For a description of control rod designs see Section 4.6.2.1. The control rods are of the bottom-entry type and are moved vertically within the core by individual, hydraulically-operated, locking piston-type control rod drives. [1.2-26]

The CRD hydraulic system is designed to allow control rod withdrawal or insertion at a limited rate, on rod at a time, for power level control and flux shaping during reactor operation. Stored energy available from gas-charged accumulators and from reactor pressure provides hydraulic power for rapid simultaneous insertion of all control rods for reactor shutdown. Each CRD has its own separate control and scram devices. [1.2-27]

Reactor coolant enters the bottom of the core and flows upward through the fuel assemblies where boiling produces steam. The steam-water mixture is separated by steam separators and dryers located within the reactor vessel. The steam passes through steam lines to the turbine. The separated water mixes with the incoming feedwater and is returned to the core inlet through jet pumps located within the reactor vessel. The motive force for the jet pumps is supplied by the water from the two reactor recirculation loops. Each loop has a variable-speed centrifugal pump with mechanical seals, motor-operated gate valves (for isolation of pumps for maintenance), and instrumentation for recirculation flow measurements.

1.2.2.4 Containment Systems

The primary containment consisting of a drywell, a pressure suppression chamber, and interconnecting vent pipes provides the first containment barrier surrounding the reactor pressure vessel and coolant recirculation system. Any leakage from the primary containment system is to the secondary containment system which consists of the reactor building, the standby gas treatment system (SBGTS), vent stack and chimney. The integrated containment systems and their associated engineered safeguards are designed so that offsite doses resulting from postulated accidents are well below the reference values stated in 10 CFR 100 (or 10 CFR 50.67 as applicable). [1.2-28]

1.2.2.4.1 Primary Containment System

The primary containment is designed to accommodate the pressures and temperatures which would result from, or occur subsequent to, a failure equivalent to a circumferential rupture of a major recirculation line within the primary containment resulting in the loss of reactor water at the maximum rate. The pressure suppression chamber is a steel, torus-shaped pressure vessel approximately half filled with water, and located below and encircling the drywell. The vent system from the drywell terminates below the water level of the pressure suppression chamber so that in the event of a pipe failure in the drywell, the released steam would pass directly to the water where it would be condensed. This transfer of energy to the water pool would reduce rapidly (within 30 seconds) the residual pressure in the drywell. [1.2-29]

Isolation valves are provided on piping penetrating the drywell and the suppression chamber to provide integrity of the containment when required. These primary containment system isolation valves are actuated automatically. These valves on the auxiliary systems are left open, or are closed depending upon the functional requirements of the system, without reducing the integrity of the primary containment system.

Two features are included in the primary containment design to aid in maintaining the integrity of the primary containment system indefinitely in the event of a loss-of-coolant accident (LOCA). The first feature is the containment cooling mode of the residual heat removal (RHR) system which provides redundant, cooling capability for the removal of heat within the drywell and the pressure suppression chamber. The second feature is the capability provided in the containment structure design to withstand the forces exerted in the event that it is necessary to flood the containment vessel to a level which would flood the reactor core.

A preoperational test was performed after complete installation of all penetrations in the drywell and suppression chamber. The containment vessels were pressurized to the calculated peak accident pressure and measurements taken to verify that the integrated leakage rate from the vessels did not exceed 0.5% per day of the combined volumes.

All containment closures which are fitted with resilient seals or gaskets are separately testable to verify leak tightness in accordance with Technical Specification Primary Containment Leakage Rate Testing Program. The covers on flanged closures, such as the equipment access hatch cover, the drywell head, access manholes, and personnel air lock doors, are provided with double seals and a test tap which will allow pressurizing the space between the seals without pressurizing the entire containment system. In addition, provisions are made so that the space between the air lock (interlock) doors can be pressurized to test for leak tightness.

Electrical penetrations through the drywell are also provided with double seals and are separately testable in accordance with Technical Specification Primary containment Leakage Rate Testing Program. The test taps and the seals are located so that tests of the electrical penetrations can be conducted without entering or pressurizing the drywell or suppression chamber. [1.2-30]

Those pipe penetrations which must accommodate thermal movement are provided with expansion bellows. The bellows expansion joints are designed for the containment system design pressure and can be checked for leak tightness when the containment system is pressurized. In addition, these joints are provided with a second seal and test tap so that the space between the seals can be pressurized to the calculated accident design pressure to permit testing the individual penetrations for leakage. [1.2-31]

1.2.2.4.2 Secondary Containment System

The primary safeguard functions of the secondary are to minimize ground level release of airborne radioactive materials, and to provide for a controlled, filtered, elevated release of the building atmosphere under accident conditions. The reactor building provides secondary containment when the primary containment is in service, and primary containment during periods when the primary containment is open. For these reasons, the reactor building is designed as a controlled-leakage structure. Units 1 and 2 are designed to use the same reactor building. The reactor building is constructed to provide a single operating floor without separation barriers above that level. Beneath the operating floor the reactor building is provided with a common wall separating Unit 2 operating and equipment areas from those of Unit 1. Door between the separate areas are provided for access control. [1.2-32]

A standby gas treatment system is provided to filter the reactor building ventilation exhaust and discharge it to the 310-foot cubicle chimney during containment isolation conditions.

1.2.2.5 Auxiliary and Standby Cooling Systems

In addition to the turbine-generator and main condenser system, multiple independent auxiliary systems are provided for the purpose of cooling the reactor and primary containment system under various normal and abnormal conditions. [1.2-33]

A. A reactor core isolation cooling (RCIC) system is provided for a continuous supply of makeup cooling water to the reactor core when the reactor is isolated. The system is sized to allow for plant shutdown under conditions of loss of the normal feedwater system by maintaining sufficient reactor water inventory until the reactor is depressurized to where the shutdown cooling system can be placed in operation.

B. Two core spray trains are provided. Each train is designed to pump water from the suppression chamber pool directly to the reactor core by a separate spray header or sparger mounted in the reactor vessel above the core rapidly enough and in a manner which will prevent fuel clad melting after depressurization following a postulated loss-of-coolant accident.

C. An RHR system is provided which serves the following functions:

1. To inject water into the reactor vessel after depressurization following a postulated LOCA in order to rapidly reflood the core and prevent fuel clad melting (low pressure coolant injection mode).

2. To remove heat from the water in the suppression chamber (containment cooling mode).

3. To spray water into the primary containment as an augmented means of removing energy from the drywell, as required, subsequent to a postulated LOCA (containment cooling mode).

4. To remove decay heat and sensible heat from the primary system so that the reactor can be shutdown for a normal refueling and servicing operation (shutdown cooling mode). [1.2-34]

Additional functions of the RHR system are described in Section 5.4.7.

D. A high pressure coolant injection (HPCI) system is provided for removal of decay heat and to provide coolant inventory control and heat dissipation from the core to the suppression chamber during a slow-depressurizing LOCA. [1.2-35]

E. An automatic pressure relief system is provided which, together, with core spray subsystem or the low pressure coolant injection (LPCI) mode of the RHR slow-depressurizing LOCA.

The core cooling provisions itemized above are designed to prevent fuel clad melting for the full range of primary system pipe size ruptures which may be postulated to occur without reliance upon offsite sources of power.

In addition, a standby coolant supply system is provided by a crosstie between the service water system and the condenser hotwell which makes available an inexhaustible supply of cooling water from the river to the reactor core and containment via the condensate and feedwater system independent of all other cooling water sources. [1.2-36]

1.2.2.6 Unit Control and Instrumentation

1.2.2.6.1 Unit Control

Reactor power is controlled by the movement of control rods and by regulation of the recirculation flow rate. Control rods are used to bring the reactor through the full range of power and to shape the core power distribution. Changing recirculation flow rate provides a second method for controlling reactor power. Load following adjustments in reactor power level are accomplished with recirculation flow control. [1.2-37]

Procedural controls backed up by protective devices are used so that thermal performance does not exceed established limits.

Reactor pressure is automatically controlled by the pressure regulator which adjusts steam flow to the turbine to control pressure in the reactor. As a result, the turbine-generator power output follows the reactor power output.

Reactor water level is controlled in either single element control (water level) or three element control (water level, steam flow and feed flow). When the unit is in three element control, a change in steam flow is immediately sensed and the feedwater control system adjusts the opening of the feedwater control valves to balance the feedwater flow with the steam flow and maintain level.

A bypass system having a capacity of approximately 33.3% of steam flow at rated load is supplied with the turbine to restrict overpressure transients resulting from sudden turbine control valve or stop valve closure. The bypass valves are operated on an overpressure signal from the initial pressure regulator. A rapid partial load rejection of 33.3% can be accommodated with the bypass system.

The reactor protection system overrides the previously described controls to shutdown the reactor and a redundant standby liquid control system is provided as an independent backup control mechanism to be used in the remote event that the control rod system becomes inoperative.

1.2.2.6.2 Reactor Protection System

A reactor protection system is provided which automatically shuts down the reactor whenever the plant parameters monitored by the system approach pre-established limits. [1.2-38]

The reactor protection system consists of two trip systems of relay contacts that are actuated by sensors from the parameters being monitored. The trip systems are energized during normal operation, and de-energization of both systems in the reactor scram circuit results in the opening of the scram valves in the control rod hydraulic system causing rapid insertion (scram) of the control rods. Each system has at least two independent tripping devices for each measured variable which initiates a scram but only one device must operate to trip the system in which it is connected. Both trip systems must be de-energized to produce a scram. The reactor protection system is designed to cause a scram on loss of power to the protection system. Components of the reactor protection system can be removed from service for testing and maintenance without interrupting plant operations and without negating the ability of the protection system to perform its protective functions upon receipt of appropriate signals.

An alternate means for reactor shutdown can be implemented by the alternate rod insertion system (ARI). This system can be used if an anticipated transient without scram (ATWS) occurs.

1.2.2.6.3 Radiation Monitoring Systems

Instrumentation is provided for continuous monitoring of the radioactivity of certain processes. Processes, significantly high in radioactivity, are monitored for variation from normal. Certain nonradioactive processes are monitored to provide alarm in the event of contamination. Process and effluent radiological monitoring and sampling is discussed in Section 11.5. [1.2-39]

1.2.2.6.4 Emergency Core Cooling System Initiation and Control

The ECCS initiation and control instrumentation is divided in two parts, the detection circuitry and the control instrumentation. The detection circuitry includes the instrument channels that detect a need for core cooling system operation and the corresponding trip systems that initiate the proper cooling system response. The control instrumentation includes the balance of ECCS instrumentation which is utilized in control and testing. [1.2-40]

1.2.2.6.5 Primary Containment and Reactor Vessel Isolation Control System

The primary containment and reactor vessel isolation control system automatically initiates closure of isolation valves to close off all potential leakage paths for radioactive material to the environs. This action is taken upon indication of a potential breach in the nuclear system process barrier. [1.2-41]

1.2.2.7 Fuel Handling and Storage

The refueling procedure is generally referred to as “wet” refueling since all irradiated fuel is always kept under water. The wet refueling procedure allows visual control of operations at all times. This feature is instrumental in producing a safe, efficient refueling sequence. [1.2-42]

Spent fuel discharged from the reactor is transferred under water through the spent fuel storage pool canal into racks provided in the storage pool. Spent fuel may also be stored in the Dry Cask Storage (DCS) system discussed in Section 9.1.2.4. The storage pool is designed to accommodate the channel stripping operation and the many other fuel maintenance operations that are required. The spent fuel racks are designed and arranged so that the risk of criticality is obviated under any condition. Storage space is also provided in the pool for irradiated fuel assembly channels and control rods, fuel shipping cask, blade guides, and small internal components of the reactor.

New fuel is brought through the equipment entrance of the reactor building and hoisted to the upper floor utilizing the reactor building crane. The new fuel for both Units 1 and 2 is stored in the new fuel vault located adjacent to the refueling pool area within the reactor building. Subsequently, new fuel is transferred to the spent fuel pool prior to loading into the reactor.

Alternatively, new fuel may be transferred directly from shipping containers to the fuel pool (via appropriate unpacking and inspection stands).

Refer to Section 9.1.2.4 for a description of spent fuel storage and handling using the DCS system and the Independent Spent Fuel Storage Installation (ISFSI).

1.2.2.8 Turbine System

The saturated steam leaving the rector vessel flows through the steam lines to the turbine located in the turbine building. After passing through the turbine, the low pressure exhaust steam is condensed, the noncondensible gases are removed, and the condensate is filtered and demineralized before being returned to the reactor through the feedwater heaters. [1.2-43]

1.2.2.9 Electrical System

The electrical output of the units is fed into a 345-kV switchyard, and from the yard to the CECo, MidAmerican Energy Company (MEC), and Alliant Energy network grid system via five 345-kV transmission lines. Auxiliary power is supplied from Units 1 and 2, and from the 345-kV switchyard. A diesel generator system provided standby auxiliary power. [1.2-44]

Batteries are used for all controls which are vital to unit and station safety and to power certain functions required for unit shutdown, such as closing of isolation valves, providing lighting, driving motors and opening valves. A separate battery supplies the neutron monitoring equipment to monitor the core during shutdown.

1.2.2.10 Shielding, Access Control, and Radiation Protection Procedures

Control of radiation exposure of plant personnel and people external to the plant is accomplished by a combination of radiation shielding, control of access into certain areas, and administrative procedures. The requirements of 10 CFR 20 are used as a basis for establishing the basic criteria and objective for normal operations. [1.2-45]

Shielding is used to reduce radiation dose rates in various parts of the plant to acceptable limits consistent with operational and maintenance requirements. Access control and administrative procedures are used to limit the integrated dose received by plant personnel to that set forth in 10 CFR 20. Access control and procedures are also used to limit the potential spread of contamination from various areas, particularly areas where maintenance occurs. Table 1.2-1 summarizes the design bases for shielding to assure that radiation levels in various areas of the plant are consistent with operational requirements. [1.2-46]

The design bases summarized in Table 1.2-1 are at the shield walls. Generally, areas away from a shield wall receive lesser dose rates and this plus occupancy factors reduce the integrated dose received. Personnel involved in all phases of operation and maintenance normally receive less than the permissible dose.

Operating, shutdown, and accident conditions are considered in establishing the shielding design.

Shielding is also used as necessary to protect equipment from radiation damage. Of principal concern are organic materials such as insulation, linings, and gaskets. The basic dose limit established for such components is generally 10^6 rads over the life of the equipment or parts thereof. The design levels are adjusted to accommodate the radiation damage resistance of specific materials.

1.2.2.11 Radioactive Waste Control

A gaseous radioactive waste control system is provided to filter, monitor, and record the process off-gases as appropriate before release through the 310-foot chimney during normal and abnormal plant operation. A system is also provided to monitor and record the activity of the reactor building air which is discharged during normal operation through the reactor building vent stack. [1.2-47]

A liquid radioactive waste control system is provided for collection, treatment, temporary storage, and discharge of liquid wastes from both Units 1 and 2. Wastes are collected in sumps and drain tanks and transferred to the radwaste facility further treatment, temporary storage, return to the system, or discharge.

In the radwaste, facility, liquid wastes to be discharged from the system are handled on a batch basis, with each batch being analyzed and disposed of as required. Treated liquid waste is either returned to the condensate system, stored awaiting disposition offsite, or released to the Mississippi River after dilution in the south diffuser pipes or discharge flume. [1.2-48]

Solid radioactive wastes are treated, stored, packaged, and shipped offsite for burial. [1.2-49]

1.2.2.12 Summary Evaluation of Safety

1.2.2.12.1 General

The general safeguards objectives of the design of these units are to protect the equipment and to limit radiation exposures to a small fraction of established limits, to any person on or off the station premises, during normal operation and under accident conditions. [1.2-50]

In order to meet these objectives, the design and operation include the following:

A. A means for positive control of plant process parameters important to safety;

B. Inherent safety features and automatic devices to prevent an operator error or equipment malfunction from causing an accident (tests are conducted periodically to assure proper functioning of such devices);

C. Multiple barriers to contain the radioactive materials (the core is conservatively designed to operate with thermal parameters significantly below those which could cause fuel damage); and

D. Operating personnel thoroughly knowledgeable in the operating characteristics of each unit, trained to follow written procedures to minimize the occurrence of operating errors.

1.2.2.12.2 Summary of Offsite Doses

The radioactive waste control systems for the combined normal operation of Units 1 and 2 are designated to limit the radiation exposure of the neighboring population to levels significantly below those doses set forth in 10 CFR 20. The expected maximum concentrations and discharge rates of radioactive waste effluents, and the calculated offsite doses under various abnormal operations (postulated accidents) are tabulated in Table 1.2-2. [1.2-51]

1.2.3 Summary of Plant Data

Design features and nominal data appropriate to achieve a reactor thermal output of 2957 MWt are summarized in Table 1.2-3. [1.2-52]

1.2.4 Interaction of Units 1 and 2

The criteria followed in designing Units 1 and 2 as a contiguous station is that each unit would operate independently of the other. A malfunction of equipment or operator error in either of the two units will not affect the continued operation of the other unit. A high degree of station reliability is accrued from the standpoint of continuity of power for the operation of standby equipment through the operation of a two-unit generating station. [1.2-53]

1.2.4.1 Gaseous Waste Effluents

A 310-foot concrete chimney discharges Units 1 and 2 off-gas, the turbine building ventilation air, and the effluent from the SBGTS. A reactor building ventilation stack located on top of the turbine building is provided for the discharge of the reactor building ventilation air. [1.2-54]

The chimney release control is accomplished by radiation monitors. These monitors provide:

A. Continuous monitoring and control of the air ejector off-gas radioactivity for each Unit by level indication, recording, high level annunciation, and a high level trip closing the off-gas system isolation valve; and

B. continuous monitoring of the Units 1 and 2 chimney gas radioactivity level by indication, recording, and high level annunciation in the control room.

The reactor building ventilation stack release control is accomplished by use of radiation monitors which continuously monitor radioactivity release rate. These monitors read out and are recorded in the control room. A high-level trip closes the outlet ventilation valves, trips the ventilation fans, and starts the SBGTS.

1.2.4.2 Liquid Waste Effluents

Units 1 and 2 share a common intake canal from the Mississippi River to supply water to their respective turbine condensers. However, Unit 1 and Unit 2 each has its own circulating water system. [1.2-55]

Units 1 and 2 use a single discharge bay which discharges into the Mississippi River just downstream of the station through two diffuser pipes which extend across the river. Liquid wastes from the radioactive waste facility are analyzed and discharged on a batch basis when necessary. The discharge limit meets the requirements of 10 CFR 20, State of Illinois regulations, and State of Iowa regulations. The activity level of the water discharge from the station is monitored and recorded. [1.2-56]

1.2.4.3 Unit Auxiliary Power Supplies

The Unit 1 auxiliary power supply is split between the unit auxiliary power transformer, which is connected to the generator leads, and the reserve auxiliary power transformer, which is connected to the 345-kV switchyard at Quad Cities; either transformer has sufficient capacity to carry the total auxiliary power requirements of Unit 1. The normal auxiliary power supply for Unit 2 is split between the unit auxiliary power transformer, which is connected to its generator leads, and the reserve auxiliary power transformer which is connected to the 345-kV switchyard. Either transformer has sufficient capacity to carry the total auxiliary power requirements of Unit 2. [1.2-57]

1.2.4.4 Common Auxiliary Systems

In those instances where a system serving one unit is interconnected with its counterpart in the other unit, the effect of the intertie upon the function of each system has been evaluated to assure that the criteria as stated in Section 1.2.4 have not been compromised. On some systems the effect of an intertie is beneficial to both units since it provides additional redundancy of equipment. [1.2-58]

1.2.4.4.1 Fire Protection Systems

The fire protection systems of Units 1 and 2 are interconnected. Therefore, through the use of crosstie valving and the additional pumping capacity, the protection afforded to each unit is increased. [1.2-59]

1.2.4.4.2 Service Air Systems

The service air systems of the two units are interconnected. This crosstie provides a backup air supply for either unit and also provides operating flexibility between the units with regard to maintenance of the service air compressors on either of the units. [1.2-60]

1.2.4.4.3 Service Water System

The service water systems of the two units are interconnected. This crosstie provides a backup supply for either unit and also provides operating flexibility between the units with regard to maintenance of the equipment. [1.2-61]

1.2.4.4.4 Reactor Building Closed Cooling Water System

There are interconnections between Units 1 and 2 reactor building closed cooling water systems. The operating flexibility of both cooling systems is enhanced by the use of interties. This crosstie is provided with locked closed valves. [1.2-62]

1.2.4.4.5 Turbine Building

The Units 1 and 2 turbines are housed in a single turbine building. The turbine building supply and exhaust ventilation systems are operated as a combined system. [1.2-63]

1.2.4.4.6 Reactor Secondary Containment

Units 1 and 2 have separate primary containments and pressure suppression systems. The secondary containment for each unit is constructed to serve its own unit. Units 1 and 2 share the same standby gas treatment, ventilation and heating systems, with each 2 share the same standby gas treatment, ventilation and heating systems, with each having capacity to accommodate the combined secondary containment volume. [1.2-64]

1.2.4.4.7 Demineralized Water Makeup System

A portable makeup demineralizer system serves both units. The makeup demineralizer obtains water from a 200,000-gallon well water tank. This demineralized water is discharged to the two 350,000-gallon contaminated condensate storage tanks or the 100,000-gallon clean condensate storage tank which serve both Units 1 and 2. The operation of the makeup demineralizer as a common system does not raise any safety or operational problems. [1.2-65]

1.2.4.4.8 Control Rooms

The control rooms for Unit 1 and 2 are adjacent and open to each other. The equipment and panels are arranged and spaced so that each control room occupies a definite and separate area. [1.2-66]

1.2.4.4.9 Radioactive Waste Systems

Units 1 and 2 share a common radioactive waste system which is designed to collect, process, control, and dispose of potentially radioactive waste in a safe manner without limiting unit or station operations or availability. [1.2-67]

1.2.4.4.10 Control Rod Drive System

There is an interconnection between the Unit 1 and Unit 2 control rod drive pumps. This provides an additional source of hydraulic water for insertion of control rods if both pumps on one unit should fail. [1.2-68]

1.2.4.4.11 Fuel Pools

The fuel pools of both units are connected by a canal. This allows fuel to be transferred between pools without the use of a shipping cask. The station is licensed to permit the storage of spent fuel assemblies from either Unit 1 or Unit 2 reactor in either Unit 1 or Unit 2 spent fuel pool. [1.2-69]

1.2.4.4.12 RHR Service Water Subsystem

There is an interconnection between the Unit 1 and Unit 2 RHR service water pumps. This provides an alternate source of cooling water to the RHR heat exchangers in the event both pumps in either loop on one unit should become inoperable. This interconnection is normally locked closed. [1.2-70]

1.2.4.4.13 Safe Shutdown Makeup Pump System

The safe shutdown makeup pump (SSMP) system consists of one pump which serves both units. The pump discharge header is divided into two segments which inject into the feedwater piping of Unit 1, and the HPCI piping of Unit 2. The SSMP system was installed as a backup to the RCIC system to meet 10 CFR 50, Appendix R, Section III.G requirements. [1.2-71]

1.2.4.4.14 Process Computer

Unit 1 and Unit 2 have separate process computers. This system is discussed in Section 7.5.2.1.

1.2.4.4.15 Miscellaneous Common Facilities

Several facilities common to Units 1 and 2, Which are necessary, but are not critical to the safe startup, operation and shutdown of the plant, are listed below: [1.2-72]

A. Service Building,

B. Instrument Maintenance Shop,

C. Machine Shop,

D. Storeroom,

E. Laundry,

F. Laboratory,

G. Security facilities,

H. New fuel storage,

I. Technical Support Center/Operations Support Center,

J. Emergency Operations Facility, and

K. Wastewater treatment facility, [1.2-73]

L. Warehouses,

M. Interim radwaste storage facility,

N. Industrial gas storage facilities,

O. Lift Station, and

P. Training building.

1.2.4.5 Inter-Plant Effects of Accident

An accident in either of the units, up t6o and including the maximum postulated accidents will not prohibit control room access or prevent safe operation or shutdown of the other unit. Provisions are made to assure that control room personnel shall not be subject to doses under postulated accident conditions during occupancy of, ingress to and egress from the control room, which would exceed limits specified in General Design Criterion 19, Standard Review Plan 6.4 and NUREG 0737. [1.2-74]

1.2.5 New Features

The design of the Quad Cities Units 1 and 2 is essentially the same as that of Dresden Units 2 and 3. New features which were included in the Dresden units were also included in the Quad Cities units. Additional features have been developed by GE for use in the Quad Cities units. The following paragraphs summarize these new features which were different between the Dresden units and Quad Cities units at the time of licensing. Additional differences that developed since the licensing date are not included. Further discussions of these features are presented in other sections of this report. [1.2-75]

1.2.5.1 Residual Heat Removal System

The RHR system can be subdivided into three modes which are functionally independent, but utilize the same equipment. The RHR system is used under both normal and emergency conditions. It operates in different modes as required. Following are descriptions of the functions of each subsystem: [1.2-76]

A. The LPCI mode of the RHR system restores and maintains the water level (after a LOCA) in the reactor vessel to at least two-thirds the core height by flooding. This mode of the RHR system performs the function which was previously achieved by the LPCI subsystem in Dresden Units 2 and 3.

B. The containment cooling mode of the RHR system is designed to limit the suppression pool water temperature so that it can still achieve containment capability in the event of a reactor vessel blowdown. Under post-accident conditions, water coolant may be diverted, if necessary, to spray-headers located in the drywell and suppression chamber for the purpose of condensing steam. This mode of the RHR system performs the function which was previously achieved by the containment cooling subsystem in Dresden Units 2 and 3.

C. The shutdown cooling mode of the RHR system functions to remove the residual heat (decay heat and sensible heat) from the reactor vessel so that the reactor can be refueled and serviced. This mode of the RHR system performs the function previously achieved by the reactor shutdown cooling system in Dresden Units 2 and 3.

At one time it was possible to divert part of the RHR flow to a spray nozzle in the reactor vessel head to condense steam generated from the hot walls of the reactor vessel while it was being flooded, thereby lowering pressure and temperature. This head spray mode of operation was rarely used and was not required for normal operation or to mitigate accidents. Due to intergranular stress corrosion cracking in the piping, the head spray lines were removed to eliminate possible pathways for leakage. [1.2-77]

1.2.5.2 The Reactor Core Isolation Cooling System

The RCIC System, with the aid of the automatic pressure relief subsystem, performs the same function as the isolation condenser system in Dresden Units 2 and 3. In the event that offsite power is lost during the time that the main steam lines are isolated, the automatic pressure relief subsystem would vent the reactor steam to the suppression pool. [1.2-78]

During this venting operation the RCIC system would maintain the reactor water level by providing makeup water from the contaminated condensate storage tank or the suppression pool.

1.2.5.3 Use of Gadolinia in Fuel Assemblies

Gadolinia is present in all fuel assemblies as a burnable absorber to supplement the control rod system. The burnable absorber I uniformly distributed in the UO2 fuel pellets in various concentrations. Fuel rods can have various gadolinia concentrations. These concentrations can also be varied axially within the rod. A full description of the burnable neutron absorber is given in Section 4.2 of the UFSAR. [1.2-79]

1.2.6 General Conclusions

Based on the favorable site characteristics, on the design of the Quad Cities Station Units 1 and 2 herein analyzed, on the criteria, principles, and design requirements of all major systems related to safety, on the calculated potential consequences of routine and accidental release of radioactive materials to the environs, on the scope of the testing programs which have been conducted, on the technical competence of the applicants and their contractors, there is reasonable assurance that the Quad Cities Station Units 1 and 2 can be operated at the present site without endangering the health and safety of the public. [1.2-80]

Table 1.2-1

DESIGN BASES FOR SHIELDING REQUIREMENTS

Degree of Access Required	Design Radiation Dose Rate (mrem/hr)
Continuous occupancy:	
Outside controlled access areas	0.5
Inside controlled access areas	1
Occupancy to 10 hr/week	6
Occupancy to 5 hr/week	12

Table 1.2-2

SUMMARY OF MAXIMUM OFFSITE DOSES FROM POSTULATED ACCIDENTS

(Original Analysis – Historical Information)

Accident		Maximum Total Offsite Exposure - Rads	
		Whole Body	Thyroid
Rod drop	6.2×10^{4} curies noble gases 1.8 curies halogens released to condenser	1.2×10^{-2}	1.2×10^{-3}
Fuel loading	5.7×10^{3} curies noble gases 3.5×10^{3} curies halogens released to reactor water	5.9×10^{-3}	4.1×10^{-3}
Steam line rupture	5.4 curies noble gases 116 curies (principally) halogens released to reactor water	4.1×10^{-3}	5.2×10^{-1}
Loss-of-coolant	5.2×10^{5} curies noble gases 2.7×10^{4} curies halogens airborne in primary containment at 30 minutes	5.3×10^{-4}	1.3×10^{-4}

Table 1.2-3
PRINCIPAL FEATURES

<u>Site</u>	
Location	Quad Cities Site, County of Rock Island, State of Illinois
Size of site	784 acres
Site and plant ownership	Exelon Generation Company and MidAmerican Energy Company
<u>Plant</u>	
MVA Rating	1068 MVA
Gross electrical output	1009 Mwe
Net heat rate	10, 962 Btu/kW-hr
Feedwater temperature	356°F
<u>Reactor</u>	
Thermal output	2957 MWt
Operating pressure (dome)	1020 psia
Total core flow rate	98 x 10^6 lb/hr
Steam flow rate	11.7 x 10^6 lb/hr
<u>Core Size</u>	
Circumscribed core diameter	189.7 in
Active length	145.28 in
<u>Fuel Assembly</u>	
Number of fuel assemblies	724
Cladding material	Zircaloy-2
Fuel material	UO_2 plus gadolinium adsorber
Fuel channel material	Zircaloy-2 SVEA-96 Optima2
<u>Fuel Rod Array</u>	<u>SVEA-96 Optima2</u>
Active fuel length	‡
Number of fuel rods	96
Number of water rods/water channels	‡‡
Cladding outside diameter	‡
Cladding thickness	‡
<u>Core Design Operating Conditions</u>	<u>SVEA-96 Optima2</u>
Core type	
Power density	‡
Heat transfer surface area/bundle	‡
Peak rod power	13.1 kw/ft

Table 1.2-3 (Continued)

PRINCIPAL FEATURES

Minimum critical power ratio safety limit at 2957 MWt	See the Safety Limit Minimum Critical Power Ratio Report for Optima2 reloads for this cycle specific value.
k_{eff} all control rods fully inserted, except one of maximum worth	0.99 (0.99 k_{eff} core design is based on projected conditions for the previous cycle)
<u>Control Systems</u>	
Number of movable control rods	177
Shape of movable control rods	Cruciform
Pitch of movable control rods	12.0 in
Control materials in movable control rods	B_4C or B_4C and Hafnium
Type of control drives	Bottom entry, hydraulic actuated
Control of reactor power output	Movement of control rods and variations of coolant flow rate
<u>Reactor Vessel</u>	
Inside diameter	20 ft. 11 in
Overall length inside	68 ft. 7 5/8 in
Design pressure	1250 psig
<u>Coolant Recirculation Loops</u>	
Location of recirculation loops	Primary containment
Number of recirculation loops	2
Pipe size	28 in
Pump capacity	45,000 gal/min each
Number of jet pumps	20
Location of jet pumps	Inside reactor vessel
<u>Primary Containment</u>	
Type	Pressure suppression
Design pressure	56 psig
Maximum allowable pressure	62 psig
Design leakage rate (before installation of penetrations)	0.5% free volume per day at calculated peak accident pressure
<u>Secondary Containment</u>	
Type	Reinforced concrete and steel superstructure with metal siding
Internal design pressure	0.25 psig
Inleakage rate	100% free volume per day at 0.25 in water negative pressure

Table 1.2-3 (Continued)

PRINCIPAL FEATURES

Structural Design	
Seismic resistance	0.12 g
Sustained wind loading	110 mph
Control room shielding	Dose not to exceed 500 mrem in 8 hours under design basis accident
Electrical Systems	
Number of incoming power sources	5 345 kV
Separate power sources provided	6 Auxiliary transformers 1 Unit transformer per unit 1 Reserve transformer per unit 2 Spray canal transformers 3 Standby diesel generators Unit 1 DG Unit 2 DG Unit 1/2 DG 10 Station batteries 1 250 V per unit 1 125 V per unit 1 125 V spare per unit 2 24/48 V per unit 7 Balance of plant batteries 1 Lift station 1 Computer UPS 2 Switchyard 2 NonEssential 250 V 1 Security 1 Security diesel generator
Reactor Instrumentation System	
Location of neutron monitor system	Incore
Ranges of nuclear instrumentation	
Startup range	Source to 0.001% rated power
Intermediate range	0.0001 - 40% rated power
Power range	1 - 125% rated power
Reactor Protection System	
Number of channels in reactor protection system	2
Number of channels required to scram or effect other protective functions	2
Number of sensors per monitored variable in each channel	2
Method to prevent unwarranted withdrawal of control rods	Automatic interlocks

Table 1.2-3 (Continued)

PRINCIPAL FEATURES

Radioactive Waste Control Systems Liquid, gaseous, and solid radioactive wastes disposed of in accordance with the requirements of 10 CFR 20.	
Summary of Other Safety Systems and Functions	
ECCS	These subsystems provide core cooling continuity over the entire range of postulated loss-of-coolant accidents to prevent cladding melt.
Control rod velocity limiter	To limit the free fall of a control rod from the core to less than 3.11 ft/s.
Control rod drive housing support	A grid work is provided to prevent control rod ejection should the control rod housing fail.
Main steam line flow restrictors	A simple venturi is welded into each main steam line for the purpose of limiting the steam discharge through a break in the steam line.
Isolation valves	To effect reactor containment automatically when required under accident conditions.
Core flooding capability	To provide a means, in conjunction with the jet pump configuration, to permit reflooding the core subsequent to a postulated loss-of-coolant accident.
Standby gas treatment system	To provide means for removal of particulates and halogens from Units 1 and 2 reactor building air under accident conditions prior to discharge of the filtered air through the 310-foot concrete chimney. Also, to provide a means for maintaining the reactor building at a negative pressure so that leakage is into the reactor building and thus prevents ground level release of building air under accident conditions.

Table 1.2-3 (Continued)

PRINCIPAL FEATURES

Standby liquid control system	To provide a redundant, independent backup reactivity control mechanism in the event that the control rod system becomes inoperable.

‡ This information is proprietary to Westinghouse Electric Company and is located in WCAP-15942-P-A (included in OPTIMA2-TR032QC-FUEL_ASSY).

‡‡ Additional water (moderation) for the interior of the bundle is provided by the watercross in the fuel channel.

1.3 COMPARISON TABLES

Certain original design features of Quad Cities Unit 1 and 2 are similar to those of other BWRs designed in the same time frame as Quad Cities, especially Dresden Units 2 and 3 and other GE BWR/3 type plants. These similarities in addition to subtle plant differences are documented in the original FSAR and Amendments. A discussion of features developed by GE for use in Quad Cities Station original design is provided in Section 1.2.5.

1.4 IDENTIFICATION OF AGENTS AND CONTRACTORS

Although the Quad Cities facility is jointly owned by Exelon Generation Company (EGC) and MidAmerican Energy Company (MEC), complete responsibility for the development, construction and operation of the station is vested in EGC. [1.4.1]

Commonwealth Edison Company engaged, or approved the engagement of, the contractors identified below in the construction of both Units. Commonwealth Edison Company was the sole applicant for the construction permit and operating license for both Units and was responsible for the design, construction, and operation of the station.

The Quad Cities station was designed and built by GE as prime contractor for CECo. General Electric Company engaged the architect-engineering services of Sargent and Lundy, Incorporated (S&L), Chicago, Illinois, to provide the design of the nonnuclear portions of the units and to prepare specifications for the purchase and construction thereof. Commonwealth Edison Company reviewed the designs and construction and purchase specifications prepared by S&L and GE to assure that the general plant arrangements, equipment and operating provisions were satisfactory. The units were constructed under the general direction of GE and through a construction management organization at the site, United Engineers and Constructors, Inc., utilizing appropriate construction, erection, and equipment subcontracts.

Preoperational testing of equipment and systems and initial operation were performed by CECo personnel under the technical direction of GE. Personnel provided by CECo for operation were drawn largely from experienced operating staffs of the Dresden station, trained and qualified in the startup and having several years of operational experience on boiling water reactors. The units were turned over to the owners, and responsibility for their subsequent operation was assumed by CECo after completion of a demonstration of unit operational capability at a specified output.

1.5 REQUIREMENTS FOR FURTHER TECHNICAL INFORMATION

For a licensed operating facility such as Quad Cities Station, requirements for further technical information are regularly promulgated by the NRC at both the plant-specific and generic levels. Responses to these requests are documented in docketed correspondence to the NRC. The results of any NRC-requested or EGC-initiated studies or analyses, to the extent they impact the plant design or safety analysis, are reflected in plant modifications, changes to procedures, and changes to the Technical Specifications, as appropriate. These results are documented in special or periodic submittals to the NRC and periodic updates of the UFSAR.

1.6 MATERIAL INCORPORATED BY REFERENCE

1.6.1 Special Topical Reports

Incorporated into the design of these units are features to improve both operational performance and overall safety which have been presented in special topical reports. These reports which have been provided to the AEC or NRC for review are listed below. [1.6.1]

1.	APED 5286	Design Basis for Critical heat Flux in Boiling Water Reactors (September 1966)
2.	APED 5446	Control Rod Velocity Limiter (March 1967)
3.	APED 5449	Control Rod Worth Minimizer (March 1967)
4.	APED 5450	Design Provisions for In-Service Inspection (April 1967)
5.	APED 5453	Vibration Analysis and Testing of Reactor Internals (April 1967)
6.	APED 5555	Impact Testing on Collet Assembly for Control Rod Drive Mechanism 7RDB144A (November 1967)
7.	Deleted	
8.	APED 5608	General Electric Company Analytical and Experimental Program for Resolution of ACRS Safety Concerns (April 1968)
9.	APED 5455	The Mechanical Effects of Reactivity Transients (January 1968)
10.	APED 5528	Nuclear Excursion Technology (August 1967)
11.	APED 5448	Analysis Methods of Hypothetical Super-Prompt Critical Reactivity Transients in Large Power Reactors (April 1968)
12.	APED 5458	Effectiveness of Core Standby Cooling Systems for General Electric Boiling Water Reactors (March 1968)
13.	APED 5640	Xenon Consideration in Design of Large Boiling Water Reactors (June 1968)
14.	APED 5454	Metal Water Reactions – Effects on Core Cooling and Containment (March 1968)
15.	APED 5460	Design and Performance of General Electric Boiling Water Reactor Jet Pumps (September 1968)
16.	APED 5654	Consideration Pertaining to Containment Inerting (August 1968)

17. APED 5696 Tornado Protection for the Spent Fuel Storage Pool (November 1968)

18. APED 5706 In-Core Neutron Monitoring System for General Electric Boiling Water Reactors, Rev. 1 (April 1969)

19. APED 5703 Design and Analysis of Control Rod Drive Reactor Vessel Penetrations (November 1968)

20. APED 5698 Summary of Results Obtained from a Typical Startup and Power Test Program for a General Electric Boiling Water Reactor (February 1969)

21. APED 5750 Design and Performance of General Electric Boiling Water Reactor Main Steam Line Isolation Valves (March 1969)

22. APED 5756 Analytical Methods for Evaluating the Radiological Aspects of the General Electric Boiling Water Reactor (March 1969)

23. APED 5652 Stability and Dynamic Performance of the General Electric Boiling Water Reactor (April 1969)

24. APED 5736 Guidelines for Determining Safe Test Intervals and Repair Times for Engineered Safeguards (April 1969)

25. APED 5447 Depressurization Performance of the General Electric Boiling Water Reactor High Pressure Coolant Injection System (June 1969)

26. NEDO 10017 Field Testing Requirements for Fuel, Curtains and Control Rods (June 1969)

27. NEDO 10029 An Analytical Study of Brittle Fracture of GE-BWR Vessel Subject to the Design Basis Accident (July 1969)

28. NEDO 10045 Consequences of a Steam Line Break for a General Electric Boiling Water Reactor (October 1969)

29. NEDE 22290-A Supplement 2, Safety Evaluation of the General Electric Advanced Longer Life Control Rod Assembly (January 1985)

30. NEDE 22290-A Supplement 3, Safety Evaluation of the General Electric Duralife 230 Control Rod Assembly (May 1988)

31. NEDE 30931-2-P Revision 2, General Electric BWR Control Rod Lifetime (May 1988)

32.	NEDC 20634	Reactor Protection System Common Mode Failure Analysis
33.	TR UR 85-225A	Topical Report ASEA-ATOM BWR Control Blades for US BWRs (October 1985)
34.	UR 87-102	Revision 1, ABB-ATOM Control Rods for BWR 2/3/4/5/6 Service Life Limits Recommendations (April 1987)
35.	NEDO 30130-A	Steady State Nuclear Methods (May 1985)
36.	NEDE 24011-P-A-10	General Electric Standard Application for Reactor Fuel (GESTAR II) (February 1991)
37.	NEDO 20964	Generation of Void and Doppler Reactivity Feedback for Application to BWR Design (December 1975)
38.	NO-AA-10	Exelon Nuclear Quality Assurance Topical Report (current revision)
39.	E-Plan	Exelon Nuclear Standardized Radiological Emergency Plan

1.6.2 Technical Requirements Manual

The Technical Requirements Manual (TRM) is incorporated by reference into the UFSAR as a result of the implementation of the Improved Technical Specifications (ITS). Changes to the TRM can be made pursuant to 10 CFR 50.59. [1.6.2]

1.7 DRAWINGS AND OTHER DETAILED INFORMATION

A list of drawings provided to the AEC as part of the license application was not included in the FSAR and therefore has not been developed for this updated report.

Applicable drawings, pictures, and sketches are included at the end of the sections in which they are referenced, or at the end of the related sections in the case of duplicate drawing references.

1.8 CONFORMANCE TO NRC REGULATORY GUIDES

Quad Cities Station was designed and partially constructed before the issuance of the first Regulatory Guides in 1970. During this time frame the AEC issued Safety Guides for utility guidance. Therefore, Quad Cities was not designed specifically to conform to Regulatory Guides. Conformance to the provisions of Regulatory Guides is generally6 indicated under three categories, full compliance or compliance with intent or objectives of the Regulatory Guide via an alternate approach, or partial compliance. Full compliance indicates that the provisions of the Regulatory Guides are met by direct conformance or by the assessed capability of the design. [1.8.1]

In certain cases, CECo has assessed the design against a particular Regulatory Guide or specifically committed to the NRC to conform in part or in whole to a particular Regulatory Guide. Where appropriate these Regulatory Guides are discussed in the applicable sections of the UFSAR. Table 1.8-1 provides a list of the Regulatory Guides and Safety Guides discussed and the sections in which they are discussed. This table is not a listing of Regulatory Guides that have been committed to by CECo.

Table 1.8-1

REGULATORY GUIDE REFERENCE SECTIONS

Commitment to or conformance with the identified Regulatory or
Safety Guides is to the extent identified in the referenced UFSAR sections.

Regulatory Guide	Title	UFSAR Section(s)
1.3	Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss-of-Coolant Accident for Boiling Water Reactors	7.5 15.6
1.5	Assumptions Used for Evaluating the Potential Radiological Consequences of a Steam Line Break Accident for Boiling Water Reactors	15.6.4
1.7	Control of Combustible Gas Concentrations in Containment Following a Loss of Coolant Accident	6.2
1.8 (Safety Guide 8, Mar. 1971)	Qualification and Training of Personnel for Nuclear Power Plants	T.S. 5.3 [Note 1]
1.21	Measuring, Evaluating, and Reporting Radioactivity in Solid Wastes, and Releases of Radioactive Materials in Liquid, and Gaseous Effluents from Light-Water-Cooled Nuclear Power Plants	11.2 T.S. 5.6.3
1.23	Onsite Meteorological Programs	2.3
1.26	Quality Group Classifications and Standards for Water-,Steam-, and Radioactive Waste Containing Components of Nuclear Power Plants (for Comment)	3.9, 5.2, 6.6
1.28 Rev. 3, Aug. 1985	Quality Assurance Program Requirements — Design and Construction	[Note 1]

Table 1.8-1 (Continued)

REGULATORY GUIDE REFERENCE SECTIONS

Regulatory Guide	Title	UFSAR Section(s)
1.30 (Safety Guide 30, Aug. 1972)	Quality Assurance Program Requirements for the Installation, Inspection, and Testing of Instrumentation and Electrical Equipment	[Note 1]
1.33 (Safety Guide 33, Nov. 1972)	Quality Assurance Program Requirements - Operation	13.5
1.34	Control of Electroslag Weld Properties	5.2, 5.3
1.36	Nonmetallic Thermal Insulation for Austenitic Stainless Steel	6.1
1.37 Mar. 1973	Quality Assurance Requirements for Cleaning Fluid Systems and Associated Components of Water-Cooled Nuclear Power Plants	[Note 1]
1.38 Mar. 1973	Quality Assurance Requirements for Packaging, Shipping, Receiving, Storage, and Handling of Items for Water-Cooled Nuclear Power Plants	[Note 1]
1.39 Mar. 1973	Housekeeping Requirements for Water-Cooled Nuclear Power Plants	[Note 1]
1.44	Control of the Use of Sensitized Stainless Steel	5.3
1.45	Reactor Coolant Pressure Boundary Leakage Detection Systems	5.2
1.49	Power Levels of Nuclear Power Plants	15.4.10, 15.6.5, 15.7.2, & T.S. 2.0
1.50	Control of Preheat Temperature for Welding of Low-Allow Steel	5.3

Table 1.8-1 (Continued)

REGULATORY GUIDE REFERENCE SECTIONS

Regulatory Guide	Title	UFSAR Section(s)
1.52	Design, Testing and Maintenance Criteria for Postaccident Engineered-Safety-Feature Atmosphere Cleanup System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants	6.5
1.54 June 1973	Quality Assurance Requirements for Protective Coatings Applied to Water-Cooled Nuclear Power Plants	[Note 1]
1.61	Damping Values for Seismic Design of Nuclear Power Plants	3.9
1.70 Rev 3, Nov. 1978	Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants LWR Edition	1.1, 5.3, 12.2
1.75	Physical Independence of Electric Systems	7.5
1.77 May, 1974	Assumptions Used for Evaluating a Control Rod Ejection Accident for Pressurized Water Reactors	3.2, 4.3, 15.8
1.78	Assumptions for Evaluating the Habitability of a Nuclear Power Plant Control Room During a Postulated Hazardous Chemical Release	6.4
1.97	Instrumentation for Light-Water Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident	7.1, 7.5, 9.1 3.11
1.99 Rev. 2	Effects of Residual Elements on Predicted Radiation Damage to Reactor Vessel Materials	5.2, 5.3

Table 1.8-1 (Continued)

REGULATORY GUIDE REFERENCE SECTIONS

Regulatory Guide	Title	UFSAR Section(s)
1.100	Seismic Qualification of Electric Equipment for Nuclear Power Plants	3.10
1.101 Rev. 2	Emergency Planning and Preparedness for Nuclear Power Reactors	13.3
1.109	Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluation Compliance with 10 CFR Part 50, Appendix I	11.3
1.111	Methods for Estimating Atmosphere Transport and Dispersion of Gaseous Effluents in Routine Releases from Light-Water-Cooled Reactors	11.3
1.113	Estimating Aquatic Dispersion of Effluent from Accidental and Routine Reactor Releases for the Purpose of Implementing, Appendix I	T.S. 5.5.4
1.120	Fire Protection Guidelines for Nuclear Power Plants	6.4
1.145	Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants	2.3.6, 15.4.10, 15.6.5, 15.7.2
1.155	Station Blackout	8.3.1.9
1.163	Performance-Based Containment Leak Test Program	6.2.6
1.183	Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors	15.4.10, 15.6.4, 15.6.5, 15.7.2
1.194	Atmospheric Relative Concentrations for Control Room Radiological Habitability Assessments at Nuclear Power Plants	15.4.10, 15.6.5, 15.7.2
4.8 Table 1, Dec. 1975	Environmental Technical Specifications for Nuclear Power Plants	T.S. 5.5.4
8.15	Acceptable Programs for Respiratory Protection	12.5

Table 1.8-1 (Continued)

REGULATORY GUIDE REFERENCE SECTIONS

Regulatory Guide	Title	UFSAR Section(s)
Safety Guide 3	Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Boiling Water Reactors	15.6
Safety Guide 7	Control of Combustible Gas Concentrations in Containment Following a Loss of Coolant Accident	6.2

Note 1 — These items are committed to in Topical Report NO-AA-10 for Quad Cities Station, but not specifically referenced in the text of the rebaselined UFSAR. Exceptions or alternatives identified in the UFSAR take precedence over commitments in the Topical Report.