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CHAPTER 1 - INTRODUCTION AND GENERAL DESCRIPTION OF PLANT

1.1 INTRODUCTION

This Updated Safety Analysis Report (USAR) for the Power Station (CPS), Unit 1, complies with the "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants, LWR Edition," issued by the Nuclear Regulatory Commission (NRC) as Regulatory Guide 1.70, Revision 3, November 1978.

A discussion of the format of the USAR is presented in Subsection 1.1.9.

1.1.1 Type of License Required

The FSAR was submitted to support the application for a Class 103 operating license to operate a one-unit nuclear power facility, designated as the Clinton Power Station, at a rated core thermal power level of 2894 MWt (100% steam flow). The FSAR was submitted under Section 103(b) of the Atomic Energy Act of 1954, as amended, and the regulations of the Nuclear Regulatory Commission set forth in Part 50 of Title 10 to the Code of Federal Regulations (10 CFR 50).

1.1.2 Identification of Applicant

Clinton Power Station Unit 1 is owned by Exelon Generation Company, LLC (EGC). EGC is a limited liability company formed to acquire and operate nuclear power plants in the United States. The USAR is submitted in support of the Exelon license to own and operate the nuclear generating station designated as Clinton Power Station Unit 1.

1.1.3 Number of Plant Units

The information presented in this USAR relates to Unit 1 of the Clinton Power Station. The CPS Preliminary Safety Analysis Report (PSAR) was docketed on October 30, 1973 (Docket Nos. 50-461 and 50-462 for Units 1 and 2, respectively). Construction permits (CPPR-137 and CPPR-138) were issued on February 24, 1976.

1.1.4 Description of Location

The nuclear power facility is located in Harp Township, DeWitt County approximately six miles east of the city of Clinton in east-central Illinois. The Clinton Power Station with its associated approximately 4900-acre man-made cooling reservoir (Lake Clinton) is an irregular U-shaped site. Condenser cooling is provided by water taken from Lake Clinton which was formed by the construction of a dam downstream from the confluence of Salt Creek and the North Fork of Salt Creek. The ultimate heat sink for emergency core cooling is a submerged pond and intake flume of 590 acre-feet capacity that underlies Lake Clinton.

1.1.5 Type of Nuclear Steam Supply

Unit 1 of the Clinton Power Station has a boiling water reactor nuclear steam supply system (218-inch vessel with 624 fuel assemblies) as designed and supplied by the General Electric Company and designated as a BWR/6 unit.

1.1.6 Type of Containment

The containment system designed by Sargent & Lundy employs the drywell/pressure suppression features of the BWR-Mark III containment concept. The containment is a right cylindrical, reinforced concrete, steel-lined pressure vessel with a hemispherical dome.

1.1.7 Core Thermal Power Levels

The information presented in this USAR pertains to Clinton Power Station, Unit 1, rated at a Licensed power level of 3473 MWt and this power level is used for evaluating radiological consequences of a Loss of Coolant Accident. The station utilizes a single-cycle forced circulation boiling water reactor (BWR) provided by General Electric (GE). The heat balance for rated power is shown in Figure 1.1-1. The unit is designed to operate at a gross electrical power output of approximately 1138.5 MWe.

1.1.8 Scheduled Completion and Operation Dates

The operating license was issued in September 1986 and commercial operation commenced in April 1987.

1.1.9 Organization of Contents

1.1.9.1 Subdivisions

The USAR is organized into 17 chapters and appropriate appendices, each of which consists of a number of sections that are numerically identified by two numerals separated by a decimal (e.g., 3.4 is the fourth section of Chapter 3). Further subdivisions are referred to as subsections.

1.1.9.2 Standard Format

The USAR has been written to comply with the Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants (Regulatory Guide 1.70, Revision 3, November 1978). The USAR uses the same chapter, section, and subsection headings as those used in the standard format except in cases where this format is not applicable to plant design. Where appropriate, the USAR is subdivided beyond the extent of the standard format to isolate all information specifically requested in that document. Where information is presented that is not specifically requested by the standard format, the information is identified numerically (chapter, section, or subsection), and it is given as a sub-division under the appropriate general heading. For example, this Subsection 1.1.9 is not requested in the standard format. Since it apparently belonged in Section 1.1, it was placed after the eight subsections containing information requested by the standard format.

1.1.9.3 References

References to another location in the USAR are made by chapter or section number. References to another document are indicated by the notation (Reference 1). The reference section is located at the end of the applicable section text and before any tables in the section. Material incorporated into the USAR by reference is listed in Section 1.6.

1.1.9.4 Tables and Figures

Tabulations of data are designated as "tables." They are identified by the section number, followed by a number indicating their order of mention in the section (e.g., Table 3.3-5 is the fifth table of Section 3.3). Tables are located at the end of the applicable section and are paginated sequentially with the section text. Drawings, sketches, curves, graphs, and engineering diagrams are all identified as "figures" and are numbered according to the order of mention in the section (Figure 3.4-2 is the second figure of Section 3.4). Figures are located at the end of the applicable section.

1.1.9.5 Page Numbering

Pages are numbered sequentially within each section. Two numerals separated by a decimal correspond to the chapter and section number and are followed by a hyphen and a number representing the page within the section, i.e., the third page in Section 4.1 of Chapter 4 is numbered 4.1-3. When it becomes necessary during revision of this USAR to insert a page(s) between two existing pages within a section, letters may be used. For example, to insert two pages between 3.2-4 and 3.2-5, the following page sequence would appear: 3.2-4, 3.2-4a, 3.2-4b, 3.2-5.

1.1.9.6 Revisions

The following instructions will be used for revisions:

a. When a change is made to the USAR text, those pages affected will be marked with the revision number and date in the lower right-hand corner. The changed or revised portion on each page will be highlighted by a "change indicator" mark consisting a bold vertical line drawn in the right-hand margin next to the material affected. Further revising of previously revised pages will delete the original vertical change bar.

b. Figures will be revised by indicating the revision number and date in the upper right-hand corner. Notation will be made of the revisions made in that revision.

c. Changes to text, tables, and figures in the USAR will be itemized and explained in the amendment transmittal letter or in an attachment thereto.

1.1.9.7 Maintenance of the USAR

CPS will utilize Regulatory Guide 1.181 in conjunction with NEI 98-03, Guidelines for Updating Final Safety Analysis Reports, as guidance for maintaining the USAR in accordance with the requirements of 10 CFR 50.71(e).

1.2 GENERAL PLANT DESCRIPTION

1.2.1 Principal Design Criteria

The principal design criteria are presented in two ways. First, they are classified as either a power generation function or a safety function. Second, they are grouped according to system. Although the distinctions between power generation or safety functions are not always clear cut and are sometimes overlapping, the functional classification facilitates safety analyses, while the grouping by system facilitates the understanding of both the system function and design.

1.2.1.1 General Design Criteria

1.2.1.1.1 Power Generation Design Criteria

a. The station is designed to produce steam for direct use in a turbine-generator unit.

b. Heat removal systems are provided with sufficient capacity and operational adequacy to remove heat generated in the reactor core for the full range of normal operational conditions and abnormal operational transients.

c. 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.

d. The fuel cladding, in conjunction with other plant systems, is designed to retain integrity such that any failures are within acceptable limits throughout the range of normal operational conditions and abnormal operational transients for the design life of the fuel.

e. Control equipment is provided to allow the reactor to respond automatically to load changes and abnormal operational transients.

f. Reactor power level is manually controllable.

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

h. Reactor controls, including alarms, are arranged to allow the operator to rapidly assess the condition of the reactor system and locate system malfunctions.

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

j. The station is designed for routine continuous operation whereby steam activation products, fission products, corrosion products, and coolant dissociation products are processed within acceptable limits.

1.2.1.1.2 Safety Design Criteria

a. The station design conforms to applicable codes and regulations.

b. The station is designed, fabricated, erected, and operated in such a way that the release of radioactive materials 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 for abnormal transients and accidents.

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

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

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

f. The design provides means by which plant operators are alerted when limits on the release of radioactive material are approached.

g. Sufficient indications are provided to allow determination that the reactor is operating within the envelope of conditions considered by plant safety analysis.

h. 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 plant operations.

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

j. Nuclear safety systems and engineered safety features shall function to assure that no damage to the reactor coolant pressure boundary results from internal pressures caused by abnormal operational transients and accidents.

k. 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 plant operations personnel.

l. Essential safety actions are provided by equipment of sufficient redundancy and independence such that no single failure of active components or of passive components in certain cases in the long term will prevent the required actions. For systems or components to which IEEE 279, "Criteria for Protection Systems for Nuclear Power Generating Stations" and/or IEEE 308, "Criteria for Class 1E Electrical Systems for Nuclear Power Generating Stations," applies, single failures of either active or passive electrical components are considered in recognition of the higher anticipated failure rates of passive electrical components relative to passive mechanical components.

m. Provisions are made for control of active components of nuclear safety systems and engineered safety features from the control room.

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

o. The design of nuclear safety systems and engineered safety features includes allowances for natural environmental disturbances such as earthquakes, floods, and storms at the station site.

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

q. 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.

r. A containment is provided that completely encloses the reactor system, drywell, and suppression pool. The containment employs the pressure suppression concept.

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

t. A secondary containment is provided that completely encloses the primary containment. This secondary containment contains a system for controlling the release of radioactive materials from the primary containment.

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

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

w. Piping that penetrates the primary containment and could serve as a path for the uncontrolled release of radioactive material to the environs is automatically isolated whenever such uncontrolled radioactive material release is imminent. Such isolation is performed in time to limit radiological effects to less than the specified acceptable limits.

x. Emergency core cooling systems are provided to limit fuel cladding temperature to less than the limits of 10 CFR 50.46 in the event of a loss-of-coolant accident.

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

z. Operation of the emergency core cooling systems is initiated automatically when required, regardless of the availability of offside power supplies and the normal generating system of the station.

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

bb. In the event the control room is unusable, it is possible tobring the reactor from power range operation to a hot shutdown condition following the control room scram and eventually place it in a cold shutdown condition by utilizing the local controls and equipment that are available outside the control room.

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

dd. Fuel storage facilities, under dry and flooded conditions, and handling equipment are designed to prevent inadvertent criticality and to maintain shielding and cooling of spent fuel.

ee. Systems that have redundant or backup safety functions are physically separated and arranged such that any credible events causing damage to any one region of the reactor island complex have minimum prospect for compromising the functional capability of the designated counterpart system.

1.2.1.2 <u>System Criteria</u>

The principal design criteria for particular systems are listed in the following subsections:

1.2.1.2.1 <u>Nuclear System Criteria</u>

a. The fuel cladding is designed to retain integrity such that any failures are within acceptable limits as a radioactive material barrier throughout the design power range.

b. The fuel cladding, in conjunction with other plant systems, is designed to retain integrity such that any failures are within acceptable limits throughout any abnormal operational transient.

c. 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 during normal operation and following abnormal operational transients and accidents.

d. Heat removal systems are provided in sufficient capacity operational adequacy to remove heat generated in the reactor core for the full range of normal operational transients as well as for abnormal operational transients. The capacity of such systems is adequate to prevent fuel cladding damage.

e. 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. The reactor is capable of being shut down automatically in sufficient time to permit decay heat removal systems to become effective following loss of operation of normal heat removal systems.

f. The reactor core and reactivity control system is 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.

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

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

1.2.1.2.2 Power Conversion Systems Criteria

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

a. 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 the gases and particulate impurities removed.

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

1.2.1.2.3 Electrical Power Systems Criteria

Sufficient normal auxiliary and standby sources of electrical power are provided to attain prompt shutdown and continued maintenance of the station in a safe condition under all credible circumstances. The power sources are adequate to accomplish all required essential safety actions under all postulated accident conditions.

1.2.1.2.4 Radwaste System Criteria

a. The gaseous and liquid radwaste systems are designed to limit the release of radioactive effluents from the station to the environs to the lowest practical values. Such releases, as may be necessary during normal operations, are limited to values that meet the requirements of applicable regulations including 10 CFR 20, 10 CFR 50 and 40 CFR 193.

b. The solid radwaste disposal system is designed so that inplant processing and offsite shipments are in accordance with all applicable regulations, including 10 CFR 20, 10 CFR 71, and 49 CFR 171 through 49 CFR 179 and DOT Regulations, as appropriate.

c. The systems' design provides means by which station operations personnel are alerted whenever specified limits on the release of radioactive material may be approached.

1.2.1.2.5 Auxiliary Systems Criteria

a. Fuel storage facilities, under dry and flooded conditions, and handling equipment are designed to prevent criticality and to maintain adequate shielding and cooling for spent fuel. Provisions are made for maintaining the cleanliness of spent fuel cooling and shielding water.

b. Other auxiliary systems, such as service water, cooling water, fire protection, heating and ventilating, communications, and lighting, are designed to function during normal and/or accident conditions.

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

1.2.1.2.6 Shielding and Access Control Criterion

a. Radiation shielding is provided, and access control patterns are established to allow the operating staff to control radiation doses within the limits of published regulations for normal modes of station operation.

b. The control room is shielded against radiation so that occupancy is possible under accident conditions and whole body doses are less then those set by Criterion 19 of 10 CFR 50, Appendix A.

1.2.1.2.7 Nuclear Safety Systems and Engineered Safety Features Criteria

Principal design criteria for nuclear safety systems and engineered safety features are as follows:

a. These criteria correspond to Criteria j through q, x through z, and bb through cc in Subsection 1.2.1.1.2.

b. Standby electrical power sources have sufficient capacity to power all Class 1E and all engineered safety features requiring electrical power.

c. Standby electrical power sources are provided as necessary for support of all engineered safety feature functions (e.g., decay heat removal) under all circumstances where normal auxiliary power is not available.

d. In the event that the control room is unusable, it is possible to bring the reactor from power range operation to a hot shutdown condition following the control room scram and eventually place it in a cold shutdown condition by utilizing the local controls and equipment that are available outside the control room.

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

1.2.1.2.8 Process Control Systems Criteria

The principal design criteria for the process control systems are discussed in the following subsections:

1.2.1.2.8.1 Nuclear System Process Control Criteria

a. Control equipment is provided to allow the reactor to respond automatically to main load changes within design limits.

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

c. Control of the nuclear system is possible from a central location.

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

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

1.2.1.2.8.2 Power Conversion Systems Process Control Criteria

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

b. The turbine is able to respond automatically to minor changes in load.

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

d. Control of the power conversion equipment is possible from a central location.

e. Interlocks or other automatic equipment are provided in addition to procedural controls to avoid conditions requiring the actuation of engineered safety features.

1.2.1.2.8.3 Electrical Power System Process Control Criteria

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

b. Protective relaying is used to detect and isolate faulted equipment from the system with a minimum of disturbance in the event of equipment failure.

c. 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 to initiate starting of the standby emergency power system diesel generators.

d. The standby emergency power diesel generators are started and loaded automatically to meet the existing emergency condition.

e. Electrically operated breakers are controllable from the control room.

f. Monitoring of essential generators, transformers, and circuits is provided in the main control room.

1.2.2 Station Description

This subsection provides an overview of those station features that are important to safety considerations. Site characteristics and general arrangement of structures and equipment are described. There are no unusual site characteristics.

1.2.2.1 Site Characteristics

1.2.2.1.1 Location

Clinton Power Station with its associated approximately 4900-acre man-made cooling reservoir (Lake Clinton) is an irregular U-shaped site in DeWitt County in east-central Illinois about 6 miles east of the city of Clinton. The site is located between the cities of Bloomington and Decatur to the north and south, respectively, and Lincoln and Champaign-Urbana to the west and east, respectively.

The total area of the Clinton Power Station site is about 13,730 acres. The site includes an area which extends approximately 14 miles along Salt Creek and 8 miles along the North Fork of Salt Creek. The reactors are about 3 miles northeast of the confluence of Salt Creek and the North Fork of Salt Creek. The station facilities occupy about 150 acres of the site property.

1.2.2.1.2 Description of Station Environs

Most of the land on the site as well as in the county is flat; however, the land along the Salt Creek and the North Fork of Salt Creek drainage courses is steeply sloped, hilly, and covered with trees and shrubbery. Typically, flat lands are used as cropland. Some of the timberland is used as pasture.

The station facilities are located on the peninsula of the U-shaped Lake Clinton which was formed by the construction of a dam approximately 3/4 mile downstream from the confluence of the Salt Creek and the North Fork of Salt Creek.

Drawings M01-1102-1 and M01-1103-1 show the site and the orientation of the principal station structures on the site. The grade elevation of the station structures is 736 feet MSL.

1.2.2.1.2.1 Meteorology

The climate at the site is typically continental, with cold winters, warm summers, and frequent short-period fluctuations in temperature, humidity, cloudiness, and wind direction. Maximum rainfall in a 40-year period of record (1937-1976) at the National Weather Service Station in Peoria, Illinois, was 5.52 inches in 24 hours, and the maximum average monthly rainfall for 29 years (1931-1960) was about 3.14 inches. Prevailing winds are southerly. Maximum wind speed recorded at both Springfield, Illinois and Peoria, Illinois, weather stations was 75 mph.

From April 14, 1972, through January 31, 1973, maximum wind speed recorded at the Clinton Power Station site was 32 mph at 60 meters above the ground.

During 53 years of record, there were 33 tornadoes in areas surrounding the site, three of which originated in DeWitt County. An average of ten tornadoes per year occur on an average of 5 days per year in Illinois, based on the above period of record. An onsite meteorological measurement program was initiated on April 30, 1972.

1.2.2.1.2.2 Hydrology

The condenser cooling water is provided from the U-shaped cooling lake that has been formed by construction of a dam just downstream from the confluence of the North Fork of Salt Creek with the Salt Creek. Salt Creek is a principal tributary of the Sangamon River and is located in the central region of Illinois within the Sangamon River Basin which drains into the Illinois River.

The cooling lake (Lake Clinton) for the Clinton Power Station is located in the upper reaches of Salt Creek 28 miles from its source. It has a normal pool elevation of 690 feet MSL. The drainage basin has an area of 296 mi^2 with the highest elevation being 910 feet and the lowest, at the dam site, being 650 feet. The drainage basin consists of farm lands and pasture with trees abounding along the floodplains and adjacent areas.

The station is located between two fingers of the lake with a station grade elevation of 736 feet and plant floor elevation of 737 feet. The ground topography along the station access route is favorably high and the grades have been located well above the probable maximum flood level in the lake. The station site is located in a sector of the morainal system known as the Bloomington Ridged Plain. Elevations on the general drift surface between drainageways in the general region of the site average about 750 feet MSL. The probable maximum flood (PMF) and 100-year flood elevations of the lake in the plant vicinity are at 708.9 and 697 feet MSL, respectively.

Within 5 miles of the station site, the principal aquifer is the Mahomet bedrock valley outwash deposit. This aquifer or its tributaries serves municipal needs, such as at the City of Clinton. Small quantities of water needed for domestic and stock watering use can be developed from the shallower alluvium along stream courses or from small permeable lenses in the upper glacial drift materials. No groundwater is used for CPS. Service water and makeup water come from Lake Clinton and are processed through a station water treatment plant.

1.2.2.1.2.3 Geology

The Clinton Power Station is located in the northern part of the Illinois Basin west of the La Salle Anticlinal Belt. The main plant is located in an area of uplands, consisting of Wisconsinan-age ground moraine, that have been dissected by Salt Creek and the North Fork of Salt Creek. The uplands consist of gently rolling ground moraine, located just east of the Shelbyville end moraine, with local relief of about 10 feet, except near the drainageways. Average elevation of the uplands is approximately 740 feet MSL.

The two perennial streams have eroded through the upland deposits of the Wisconsinan-age Wedron Formation and Robein Silt, the Illinoian-age weathered Glasford Formation, and into the upper part of the Illinoian-age unaltered Glasford Formation in the site area. The elevation of the floodplains of the two streams in the site area is at approximately 660 feet MSL. Total relief is on the order of 80 feet.

The stratigraphy consists of relatively thick overburden deposits, 170 to 330 feet in thickness in the upland areas, overlying Pennsylvanian-age bedrock. The overburden materials, in order of increasing age, consist of stream alluvium, windblown loess, and glacial drift. There were four major periods of glaciation during Pleistocene time in the regional area. In each of these periods (Nebraskan, Kansan, Illinoian, and Wisconsinan) glaciers periodically advanced and retreated across parts of the regional area, depositing complex sequences of glacial or glacially-derived sediments, the youngest being classified as Holocene.

Excavations for plant structures extended down into only the unaltered Glasford Formation or "Illinoian Till". Beneath the Glasford Formation is a complex assemblage of glacial materials consisting of gray-to-brown clay till (which is occasionally sandy), reworked till and outwash, and glaciolacustrine gray silt. In some areas of the site, as beneath the main power block, the complex of probable pre-Illinoian till, outwash, and glaciolacustrine deposits lies in direct contact with bedrock of Pennsylvanian age. The bedrock varies in elevation from 360 to 510 feet MSL.

The deepest borings within the site area penetrated the uppermost bedrock formations of Pennsylvanian age. Nearly 600 feet of Pennsylvanian age strata were deposited in the site area. Uplift on the Wapella East Dome portion of the Downs Anticline near the site occurred during and/or after Pennsylvanian time.

The Downs Anticline trends south to south-southeastward from a point about 10 miles north of Bloomington, Illinois, and passes approximately 4 miles northeast of the site. It is a very asymmetrical, almost monoclinal fold with several domes along its axis. Rocks of Devonian and older age are involved in this fold. Three small domes lie between 5 and 10 miles from the site: The Wapella East Dome to the northwest, the Parnell Dome to the Northeast, and the Deland Dome to the Southeast.

1.2.2.1.2.4 <u>Seismology</u>

Eleven seismogenic regions can be delineated within 200 miles of the Clinton Power Station; however, the site area is not seismically active. Only one earthquake occurring in the Central Stable Region at distances greater than 200 miles from the site has been felt at the site itself (from Anna, Ohio, March 8, 1937, having a felt site Intensity (MM) I-III). There is no record of earthquakes with an Intensity (MM) VIII or greater within 200 miles of the site. The maximum felt intensity experienced at the Clinton site from any earthquake within a 200 mile radius of the site was Intensity (MM) V.

One of the most significant earthquakes in the region was the July 18, 1909, Central Illinois (Havana) earthquake, with epicentral Intensity (MM) VII, which represents the maximum earthquake which could be expected at the station. Therefore, the recommended safe shutdown earthquake was correlated to a maximum horizontal ground acceleration of 0.13 g;

however, an additional conservatism was adopted with a resultant maximum safe shutdown peak horizontal ground surface acceleration of the site of 0.26 g. Design spectra for a safe shutdown earthquake with horizontal acceleration of 0.26 g and for a variety of damping values have been used for analysis of plant structures and equipment.

1.2.2.1.3 Design Bases Dependent on Site Environs

Off-Gas System - A common station HVAC vent approximately 200 feet in height provides for the continuous dispersal of gaseous effluent to the atmosphere. Gaseous releases are as low as is reasonably achievable and less than 10 CFR 20 limits.

Liquid Waste Effluents - Liquid waste is released such that concentrations at the point of discharge are as low as is reasonably achievable and less than 10 CFR 20 limits.

Flooding - The elevations of the station grade and surface of the cooling lake are 736 feet and 690 feet, respectively. The spillway limits the maximum elevation of the cooling lake to 708.9 feet. Flooding of the station is, therefore, extremely unlikely.

Wind Loading and Seismic Design - The structures and components whose failure might cause a loss-of-coolant accident or result in an uncontrolled release of radioactive fission products, are designed to resist wind loads of tornado velocity and earthquake ground motions which could be expected to occur at the site during the service life of the station.

Safety-related structures are designed for a maximum tornado rotational velocity of 290 mph and translational wind load of 70 mph. Note: Revision 1 of RG 1.76 was adopted in September 2007 which lowered the design-basis tornado characteristics. The design wind velocity is 85 mph.

The maximum horizontal ground accelerations at the foundation level were conservatively selected to be 10% of gravity for the operating basis earthquake (OBE) and 25% of gravity for the safe shutdown earthquake (SSE).

1.2.2.2 General Arrangement of Structures and Equipment

The principal structures located on the station site are the following:

a. containment - houses the major portion of the nuclear steam supply system, the drywell, the suppression pool, and the containment pool;

b. auxiliary building - houses the emergency core cooling system pump rooms, the RCIC turbine and pump room, the RHR heat exchanger rooms, the switchgear rooms, and the battery rooms;

c. fuel building - houses the fuel storage and shipping area and the integrated fuel pool cooling and cleanup equipment;

d. turbine building - houses the power conversion equipment;

e. radwaste building - houses the liquid, gaseous, and solid radioactive waste treatment and shipping facilities, machine shop, and storeroom:

f. control building - houses the control room, the computer facilities, cable spreading areas, switchgear, batteries, and other miscellaneous equipment:

g. diesel generator and HVAC building - houses the emergency diesel-generator equipment, HVAC equipment, and standby gas treatment equipment;

h. circulating water screen house - houses the travelling screens, the pumps and strainers for the shutdown service water systems, the circulating water pumps, the fire pumps;

i. service building;

j. makeup water pump house;

k. switchyard;

l. outdoor storage tanks;

m. permanent warehouse; and

n. gatehouse.

The arrangement of the these structures on the station site is shown in Drawings M01-1102-1 and M01-1103-1. The arrangement of the equipment inside the main building is shown in Drawings M01-1105 through M01-1116 and M01-1119.

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 1.1-1.

1.2.2.3.1 Reactor Core and Control Rods

The reactor fuel and core are described in Section 2 of NEDE-24011-P-A, "General Electric Standard Application for Reactor Fuel," and Section 1 of NEDE-24011-P-A-US, "General Electric Standard Application for Reactor Fuel (U.S. Supplement)." Experience has shown that the control rods are not susceptible to distortion and have an average life expectancy many times the residence time of a fuel loading.

1.2.2.3.2 Reactor Vessel and Internals

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

The reactor vessel is designed and fabricated in accordance with applicable codes for a pressure of 1250 psig. The nominal operating pressure in the steam space above the separators is 1040 psia. The vessel is fabricated of low alloy steel and is clad internally with stainless steel (except for the top head, nozzles, and nozzle weld zones which are unclad).

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 containment barrier.

1.2.2.3.3 Reactor Recirculation System

The reactor recirculation system consists of two recirculation pump loops external to the reactor vessel. These loops provide the piping path for the driving flow of water to the reactor vessel jet pumps. Each external loop contains one high capacity motor-driven recirculation pump, two motor-operated maintenance valves, and one hydraulically operated flow control valve. The variable position hydraulic flow control valve operates in conjunction with a low frequency motor-generator set to control reactor power level through the effects of coolant flow rate on moderator void content.

The jet pumps are reactor vessel internals. The 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 inner wall. 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.

1.2.2.3.4 Residual Heat Removal System

The residual heat removal (RHR) system is a system of pumps, heat exchangers, and piping that fulfills the following functions:

a. Removes decay and sensible heat during and after plant shutdown.

b. Injects water into the reactor vessel, following a loss-of-coolant accident, to reflood the core independent of other core cooling systems. This is discussed in Subsection 1.2.2.4.8, "Emergency Core Cooling Systems."

c. Removes heat from the containment, following a loss of-coolant accident, to limit the increase in containment pressure. This is accomplished by cooling and recirculating the suppression pool water (containment cooling) and by spraying the containment air space (containment spray) with suppression pool water.

1.2.2.3.5 Reactor Water Cleanup System

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

1.2.2.3.6 Nuclear Leak Detection System

The nuclear leak detection and monitoring system consists of temperature, pressure, flow, and fission-product sensors with associated instrumentation and alarms. This system detects and annunciates leakage in the following systems:

a. main steamlines,

b. reactor water cleanup (RWCU) system,

c. residual heat removal (RHR) system,

d. reactor core isolation cooling (RCIC) system,

e. feedwater system,

f. ECCS systems,

g. fuel pool cooling and cleanup (FPC&C) system, and

h. instrument lines.

Small leaks generally are detected by monitoring the radiation levels and drain sump fill-up and pump-out rates. Large leaks are also detected by changes in reactor water level and changes in flow rates in process lines.

1.2.2.4 Nuclear Safety Systems and Engineered Safety Features

1.2.2.4.1 Reactor Protection System

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

1.2.2.4.2 Neutron Monitoring System

Those portions of the neutron monitoring system that are part of the reactor protection system qualify as a nuclear safety system. The intermediate range monitors (IRM) and the average power range monitors (APRM), which monitor neutron flux via incore detectors, provide scram logic inputs to the reactor protection system to initiate a scram in time to prevent excessive fuel clad damage as a result of over-power transients. The APRM system also generates a simulated thermal power signal. Both upscale neutron flux and upscale simulated thermal power are conditions which provide scram logic signals.

1.2.2.4.3 Control Rod Drive System

When a scram is initiated by the reactor protection system, the control rod drive system inserts the negative reactivity necessary to shut down the reactor. Each control rod is controlled individually by a hydraulic control unit. When a scram signal is received, high pressure water stored in an accumulator in the hydraulic control unit or reactor pressure forces its control rod into the core.

1.2.2.4.4 Control Rod Drive Housing Supports

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

1.2.2.4.5 Control Rod Velocity Limiter

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

1.2.2.4.6 Nuclear System Pressure Relief System

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

1.2.2.4.7 Reactor Core Isolation Cooling System

The reactor core isolation cooling system (RCIC) provides makeup water to the reactor vessel when the vessel is isolated following events that result in a loss of feedwater flow and for the control rod drop accident. The RCIC system uses a steam-driven turbine-pump unit and operates automatically in time and with sufficient coolant flow to maintain adequate water level in the reactor vessel for events defined in Subsection 5.4.6.1.

1.2.2.4.8 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:

High pressure core spray (HPCS)

Automatic depressurization (ADS)

Low pressure core spray (LPCS)

Low pressure coolant injection (LPCI), an operating mode of the residual heat removal system

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

b. Automatic Depressurization - The automatic depressurization system rapidly reduces reactor vessel pressure in a loss-of-coolant accident (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 uses some of the relief valves that are part of the nuclear system pressure relief system. The automatic relief valves are arranged to open on conditions indicating both that a break in the reactor coolant pressure boundary has occurred and that the HPCS system is not delivering sufficient cooling water to the reactor vessel to maintain the water level above a preselected value. The ADS 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.

c. Low Pressure Core Spray - The LPCS system consists of one independent pump and the valves and piping to deliver cooling water to a spray sparger over the core. The system is actuated by conditions indicating that a breach exists in the reactor coolant pressure boundary but water is delivered to the core only after reactor vessel pressure is reduced. This system provides the capability to cool the fuel by spraying water into each fuel channel. The LPCS loop functioning in conjunction with the ADS or HPCS can provide sufficient fuel cladding cooling following a loss-of-coolant accident.

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

1.2.2.4.9 Containment Systems

1.2.2.4.9.1 Containment Functional Design

The containment design for this plant has been given the name Mark III. This containment design incorporates the drywell/pressure suppression feature of previous BWR containment designs into a dry-containment type of structure.

In fulfilling its design basis as a fission product barrier in case of an accident, the Mark III containment is a low-leakage structure even at the elevated pressures that could follow a main steamline rupture or a recirculation line break.

The containment system consists of the following major components:

a. A drywell enclosing the reactor pressure vessel, the reactor coolant recirculation loops and pumps, and other branch connections of the reactor primary system. The drywell is a cylindrical reinforced concrete structure with a removable steel head.

b. A suppression pool that serves as a heat sink during normal operational transients and accident conditions. It contains a large amount of water used to rapidly condense steam from a reactor vessel blowdown or from a break in a major pipe.

c. A containment upper pool for shielding, refueling operations and makeup to the suppression pool.

d. A leaktight containment vessel completely surrounding the drywell and the suppression pool. The containment building is formed by an upright cylinder, founded on a soil-supported flat concrete slab and covered with a hemispherical dome. This reinforced-concrete pressure vessel is lined with steel plate which serves as a leaktight membrane.

Access and logistics penetrations through the containment cylinder include standard, double-door personnel locks at grade level and at the refueling floor level. In addition, there is a circular hatch at grade level to allow movement of equipment into the containment. The piping and electrical penetrations are sealed to the containment liner plate by welding. The containment building structure provides shielding to minimize direct radiation to operating personnel and/or to the public under normal operating and accident conditions.

Although part of the suppression pool water is inside the drywell (retained by a cylindrical concrete weir wall), the major part is outside the drywell between the drywell wall and the containment wall. A system of horizontal vents in the drywell wall below the suppression pool water level, connects the inner and outer parts of the suppression pool. In the event of a process piping failure within the drywell, the increased pressure inside the drywell will force a mixture of air, steam, and water through the vents to the major volume of the suppression pool where the steam will rapidly condense. The noncondensable gases will escape into the free air volume inside the containment vessel where they will be contained.

Equipment and facilities located inside the containment vessel, but outside the drywell, include the control rod drive modules, major components of the reactor water cleanup system and the standby liquid control system, and the reactor refueling facilities.

1.2.2.4.9.2 Residual Heat Removal System (Suppression Pool Cooling)

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

1.2.2.4.9.3 Residual Heat Removal System (Containment Spray)

The containment spray system will function, by automatic or manual initiation, to condense this steam to prevent over-pressurization of the containment. The containment spray system consists of two redundant subsystems, each with its own full capacity spray header. Each subsystem is supplied from a separate redundant RHR subsystem.

The containment spray system may also serve as an iodine removal method to reduce releases to the environment following a LOCA, although no credit for iodine removal is taken for dose-analysis purposes.

1.2.2.4.9.4 Combustible Gas Control

In the unlikely event of an accident that results in a degraded core, hydrogen and oxygen will be generated in the drywell and containment. The combustible gas control system will ensure that hydrogen concentrations are kept below the limits specified in Regulatory Guide 1.7. The systems used include a hydrogen control system and a backup containment purge system.

In addition to the above systems, the Hydrogen Ignition System is designed to control hydrogen concentrations in the containment and drywell during a degraded core event which results in excessive quantities of hydrogen being generated. The conditions requiring operation of this system are beyond the scope of a design basis accident and are identified in 10 CFR 50.44.

1.2.2.4.10 Containment and Reactor Vessel Isolation Control System

The containment and reactor vessel isolation control system automatically initiates closure of isolation valves to close off all process lines which are potential leakage paths for radioactive material to the environs. This action is taken upon indication of a breach in the reactor coolant pressure boundary.

1.2.2.4.10.1 Main Steamline Isolation Valves

Although all pipelines that both penetrate the containment and offer a potential release path for radioactive material are provided with redundant isolation capabilities, the main 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. These valves fulfill the following objectives:

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

b. Limit the release of radioactive materials by isolating the reactor coolant pressure boundary in case of a gross release of radioactive materials from the fuel to the reactor cooling water and steam.

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

1.2.2.4.10.2 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 containment.

1.2.2.4.11 Process Radiation Monitoring System

1.2.2.4.11.1 Main Steamline Radiation Monitoring System

The main steamline radiation monitoring system consists of two gamma radiation monitors located externally to the main steamlines just outside the containment. The monitors are designed to detect a gross release of fission products from the fuel. On detection of high radiation, the trip signals generated by the monitors are used to trip the mechanical vacuum pump, if running.

1.2.2.4.11.2 Containment Building Ventilation Radiation Monitoring System

The containment building ventilation radiation monitoring system consists of radiation monitors arranged to monitor the activity level of the ventilation exhaust from the fuel transfer area and in the building's main exhaust duct. The detection of high radiation will automatically shut down the containment and fuel building ventilation systems, start the standby gas treatment system, initiate closure of the various purge and exhaust paths to these buildings, and isolate the normal exhaust from the ECCS pump rooms to the fuel building ventilation system.

1.2.2.4.11.3 Fuel Building Ventilation Radiation Monitoring System

The fuel building ventilation radiation monitoring system consists of radiation monitors arranged to monitor the activity level of the ventilation exhaust from the fuel handling area and fuel building. On detection of high radiation, the fuel building is automatically isolated and the standby gas treatment system is started.

1.2.2.4.12 Standby Gas Treatment System

The standby gas treatment system consists of two identical Seismic Category I, 100%-capacity equipment trains for the station. Either train may be considered as an installed spare with the other train capable of processing the required amount of air. Either train will be capable of maintaining the secondary containment at 0.25 inch H_2O negative pressure with respect to the outdoors. The system will process all gaseous effluent before discharging to the atmosphere.

Each equipment train contains a fan, demister, prefilter, electric heater, two high efficiency particulate filter banks (water- and fire-resistant), two charcoal iodine adsorbers (fire resistant), and a device to measure and control flow.

1.2.2.4.13 Safety-Related Electrical Power Systems

Standby a-c power is supplied by three diesel generators. Each engineered safety features (ESF) division is supplied by a separate diesel generator. There are no provisions for transferring ESF division buses between standby a-c power supplies or supplying more than one ESF division from one diesel generator. The one-to-one relationship between diesel generator and ESF division ensures that a failure of one diesel generator can affect only one ESF division.

The unit has four independent Class 1E 125-Vdc systems, one per each of the three ESF divisions of the Class 1E electric power system and a fourth for portions of the emergency core cooling and reactor protection systems.

1.2.2.4.14 Standby Liquid Control System

Although not intended to provide prompt reactor shutdown, as the control rods are, the standby liquid control system provides a redundant, independent, and alternate way to bring the nuclear fission reaction to subcriticality and to maintain subcriticality as the reactor cools. The system makes possible an orderly and safe shutdown in the event that not enough control rods can be inserted into the reactor core to accomplish shutdown in the normal manner. The system is sized to counteract the positive reactivity effect from rated power to the cold shutdown condition. The system also functions to control suppression pool pH after an assumed Loss-of-Coolant Accident (LOCA), as described in Section 15.6.5.6.

1.2.2.4.15 Safe Shutdown from Outside the Control Room

In the event that the control room is unusable, it is possible to bring the reactor from power range operation to a hot shutdown condition following the control room scram and eventually place it in a cold shutdown condition by utilizing the local controls and equipment that are available outside the control room.

1.2.2.4.16 Main Steamline Isolation Valve Leakage Control System

Note: As a result of the re-analysis of the Loss of Coolant Accident (LOCA) using Alternative Source Term (AST) Methodology, it is no longer necessary to credit the Main Steam Isolation Valve Leakage Control System (MSIVLCS) for post-LOCA activity leakage mitigation. The system has been left in place as a passive system and is not required to perform any safety function.

1.2.2.4.17 Suppression Pool Makeup System

The suppression pool makeup system provides water from the upper containment pool to the suppression pool by gravity flow following a LOCA. The quality of water provided is sufficient to maintain required drywell upper-most vent coverage for all postulated accidents.

1.2.2.4.18 Control Room HVAC

The control room HVAC system provides an environment in the control room suitable for the operation of equipment necessary for the safe shutdown of the plant and will function in the event of a LOCA. The system protects the control room operators from the results of any accident which could impair their safety and therefore compromise the safety of the plant.

1.2.2.4.19 Shutdown Service Water System

The shutdown service water system is designed to remove heat from station auxiliaries which either are used to safely shutdown or are required following an assumed LOCA. The shutdown service water system provides a means of flooding the drywell and containment, if ever required, and provides a reliable backup source of makeup water for the spent fuel pool and the upper containment pool. The system consists of three subsystems which correspond to the three ESF divisions. The system is designed to Seismic Category I requirements.

1.2.2.5 Power Conversion System

1.2.2.5.1 Turbine-Generator

The turbine-generator is a General Electric 1800-rpm, tandem compound four-flow, reheat steam turbine with 43-inch last-stage blades. The turbine-generator is equipped with an electro-hydraulic control system and supervisory instruments to monitor performance. The gross electrical output of the turbine generator is 1138.4 MW.

Each generator is a direct-driven, three-phase, 60 Hz, 22,000-V, 1800 rpm, conductor-cooled, synchronous generator rated at approximately 1265 MVA, at 0.90 power factor, 75 psig hydrogen pressure, and 0.63 short-circuit ratio. The generator exciter is a 3607 kW, 545-V, shaft-driven type.

1.2.2.5.2 Main Steam System

The main steam system delivers steam from the reactor to the turbine generator, the reheaters, and the steam jet air ejectors from warmup to full-load operation. The main steam system also provides steam for the reactor feed pump turbines, the steam seal evaporator, and the auxiliary steam system when other steam sources are not available.

1.2.2.5.3 Main Condenser

The main condenser is a single-shell, single-pass, deaerating type condenser with divided water boxes. During plant operation, steam expanding through the low pressure turbines is directed downward into the main condenser and is condensed. The main condenser also serves as a heat sink for the turbine bypass system, emergency and high-level feedwater heater and drain tank dumps, and various other startup drains and relief valve discharges.

1.2.2.5.4 Main Condenser Evacuation System

The main condenser evacuation system removes the noncondensable gases from the main condenser and discharges them to the gaseous radwaste system. This system consists of two 100%-capacity, twin-element, two-stage steam jet air ejectors (SJAE) with intercondensers, for normal station operation, and mechanical vacuum pumps for use during startup.

1.2.2.5.5 Turbine Gland Sealing System

The turbine gland sealing system provides clean steam to the turbine shaft glands and the turbine valve stems. The turbine gland sealing system prevents leakage of air into or radioactive steam out of the turbine shaft and turbine valves. The clean steam is provided by evaporating condensate taken downstream from the condensate demineralizers or from the auxiliary steam system during startup. The steam packing exhauster collects air and steam mixture, condenses the steam, and discharges the air leakage to the atmosphere via the common station HVAC vent by a motor-driven blower.

1.2.2.5.6 Steam Bypass System and Pressure Control System

A turbine bypass system 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 load permitted to pass to the turbine generator. The capacity of the turbine bypass system is 28.8% of the reactor rated steam flow. The pressure regulation system provides main turbine control valve and bypass valve flow demands so as to maintain a nearly constant reactor pressure during normal plant operation.

1.2.2.5.7 Circulating Water System

The circulating water system provides a continuous supply of cooling water to the condenser to remove the heat rejected by the steam cycle. The circulating water system has three motor-driven pumps which take suction from a cooling lake (Lake Clinton) and discharge through the main condenser and back to the cooling lake.

1.2.2.5.8 Condensate and Feedwater System

The condensate and feedwater system provides a dependable supply of high-quality feedwater to the reactor at the required flow, pressure, and temperature. The condensate pumps take the deaerated condensate from the condenser hotwell and deliver it through the steam jet air ejector condenser, the steam packing exhausters condenser, the off-gas condenser, and the condensate demineralizer to the suction of the condensate booster pumps. The condensate booster pumps deliver the condensate through two parallel strings of five low pressure feedwater heaters to the reactor feed pumps' suction. The reactor feed pumps discharge through two parallel high pressure feedwater heaters to the reactor. The drains from the feedwater heaters are cascaded through the low-pressure feedwater heaters to the main condenser.

1.2.2.5.9 Condensate Cleanup System

The unit is served by a 100%-capacity condensate cleanup system, consisting of nine deep-bed demineralizer vessels designed for parallel operation. One demineralizer vessel may be a spare. A condensate filter is placed upstream from deep-bed demineralizers A thru F. The condensate cleanup system with instrumentation and automatic controls is designed to ensure a constant supply of high-quality water to the reactor.

1.2.2.6 Electrical Systems and Instrumentation and Control

1.2.2.6.1 Electrical Power Systems

The following systems provide electrical power to station auxiliaries:

a. 138-kV offsite power system,

b. 345-kV offsite power system,

c. unit auxiliary a-c power system,

d. unit class 1E a-c power system,

e. Nuclear System Protection System Power System,

f. unit auxiliary d-c power system,

g. instrument power system,

h. uninterruptible power system, and

i. unit Class 1E d-c power system.

Each of these systems is described briefly in the following subsections.

1.2.2.6.1.1 138-kV Offsite Power System

The 138-kV offsite power system provides power to the station through one three-terminal transmission line that terminates directly (through a circuit switcher) at the emergency reserve auxiliary transformer, which transforms the electrical power to the 4160-V station bus voltage.

1.2.2.6.1.2 345-kV Offsite Power System

The 345-kV offsite power system provides power to the station through three transmission lines. All three lines terminate at the station switchyard ring bus that feeds reserve auxiliary transformers which, in turn, transforms the electrical power to the 6900-V and 4160-V station bus voltages.

1.2.2.6.1.3 Unit Auxiliary A-C Power System

The unit auxiliary a-c power system supplies power to unit loads that are non-safety-related and uses the main generator as the normal power source with the reserve auxiliary transformers as a backup source. The unit auxiliary transformer steps down the a-c power to the 6900-V and 4160-V station bus voltages.

1.2.2.6.1.4 Unit Class 1E A-C Power System

The unit Class 1E a-c power system supplies power to the unit Class 1E loads. The offsite power sources converge at the system. The system includes diesel generators that serve as standby power sources, independent of any onsite or offsite source. Therefore, the system has three sources. Furthermore, the system is divided into three divisions, each with its own independent distribution network, diesel generator, and redundant load group.

1.2.2.6.1.5 Nuclear System Protection System Power System

Four divisions of the nuclear system protection system (NSPS) power system provide a Class 1E source of 120-Vac single phase control power. The primary power source for the NSPS power system is the Class 1E d-c power system, with an automatic throw-over switch to the Class 1E a-c power system for back-up.

1.2.2.6.1.6 Instrument Power System

The instrument power system supplies 120-Vac single-phase power to instrument and control loads which do not require an uninterruptible power source.

1.2.2.6.1.7 Uninterruptible Power System

The uninterruptible power system (UPS) supplies regulated 120-Vac single-phase power to non-Class 1E instrument and control loads which require an uninterruptible source of power. The power sources for the UPS are similar to those for the NSPS, but are non-Class 1E.

1.2.2.6.1.8 Unit Auxiliary D-C Power System

The unit auxiliary d-c power system supplies power to unit d-c loads that are non-safety-related. The system consists of uninterruptible power supplies, batteries, motor control centers, distribution panels and two regular and one spare battery chargers. The spare battery charger normally is not connected to the system.

1.2.2.6.1.9 Unit Class 1E D-C Power System

The unit Class 1E d-c power system supplies 125-Vdc power to the unit Class 1E loads. Battery chargers are the primary power sources. The system, which includes storage batteries that serve as standby power sources, is divided into four divisions, each with its own independent distribution network, battery, battery charger, and redundant load group. A non safety-related swing battery charger is also part of the system that will be connected to the 125 VDC buses for supplying backup power during periods when the normal battery charger for the division 1, 2 or 4 bus is being maintained.

1.2.2.6.2 Nuclear System Process Control and Instrumentation

1.2.2.6.2.1 Rod Control and Information System

The rod control and information system provides the means by which control rods are positioned from the control room for power control. The system operates valves in each hydraulic control unit to change control rod position. One gang of control rods can be manipulated at a time. The system includes the logic that restricts control rod movement (rod block) under certain conditions as a backup to procedural controls.

1.2.2.6.2.2 Recirculation Flow Control System

During normal power operation a variable position discharge valve is used to control flow. Adjusting this valve changes the coolant flow rate through the core and thereby changes the core power level. During periods of low power operation, such as plant startup and shutdown, the recirculation pump and motor will be powered by the LFMG set and will operate at approximately 25% rated full load speed.

1.2.2.6.2.3 Neutron Monitoring System

The neutron monitoring system is a system of in-core neutron detectors and out-of-core electronic monitoring equipment. The system provides indication of neutron flux, which can be correlated to thermal power level for the entire range of flux conditions that can exist in the core. The source range monitors (SRM's) and the intermediate range monitors (IRM's) provide flux level indications during reactor startup and low power operation. 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. The traversing in-core probe system (TIP) provides a means to calibrate the individual LPRM sensors. The oscillation power range monitor (OPRM) is used to monitor for the onset of thermal hydraulic oscillations. The neutron monitoring system provides inputs to the rod control and information system to initiate rod blocks if preset flux limits are exceeded, and inputs to the reactor protection system to initiate a scram if other limits are exceeded.

1.2.2.6.2.4 Refueling Interlocks

A system of interlocks that restricts movement of refueling equipment and control rods when the reactor is in the refueling and startup modes is provided to prevent an inadvertent criticality during refueling 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.

1.2.2.6.2.5 Reactor Vessel Instrumentation

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

1.2.2.6.2.6 Process Computer System

On-line process computers are provided to monitor and log process variables and to make certain analytical computations.

1.2.2.6.3 Power Conversion Systems Process Control and Instrumentation

1.2.2.6.3.1 Pressure Regulator and Turbine-Generator Control

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

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

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 on loss of the generator electric load.

1.2.2.6.3.2 Feedwater Control System

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

1.2.2.7 Fuel Handling and Storage Systems

1.2.2.7.1 New and Spent Fuel Storage

New and spent fuel storage racks are designed to prevent load buckling and inadvertent criticality under dry and flooded conditions. 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 dry or flooded conditions. This subject is further discussed in Section 9.1.

1.2.2.7.2 Fuel Handling System

The fuel handling equipment includes a 125-ton cask crane, fuel handling platform, fuel inspection stand, fuel preparation machine, fuel assembly transfer mechanism, containment refueling platform, 100-ton containment crane, and other related tools for reactor servicing. All equipment conforms with applicable codes and standards.

The principal function of the cask crane is to handle the spent fuel cask. The fuel handling platform transfers the fuel assemblies between the transfer pool, storage pools, and cask.

Fuel assemblies are transferred through the transfer tube between the containment building and the fuel building. The fuel assemblies inside the containment are handled by the refueling platform.

The handling of the reactor head, removable internals, and drywell head during refueling is accomplished using the containment crane.

All tools and servicing equipment necessary to meet the reactor general servicing requirements are designed for efficiency and safe serviceability.

1.2.2.8 Cooling Water and Auxiliary Systems

1.2.2.8.1 Closed Cooling Water Systems

1.2.2.8.1.1 Component Cooling Water System

The Component Cooling Water System (CCWS) is a closed-loop system which is designed to cool station auxiliary equipment over the full range of normal reactor operation, normal shutdown, and testing conditions. The closed loop provides a barrier between nonessential contaminated systems and the plant service water discharged to the environment. Heat is removed from the closed loop by the plant service water system. Since CCWS may not be available under emergency conditions, it is designed with the capability to transfer fuel pool cooling heat exchangers to the shutdown service water system during loss of offsite a-c power and/or LOCA conditions. The portion of the component cooling water system penetrating the containment is designed to permit containment isolation under all station conditions. A radiation monitor is provided in the CCWS to indicate inleakage into this system from the potentially radioactive systems.

1.2.2.8.1.2 Turbine Building Closed Cooling Water System

The Turbine Building Closed Cooling Water System (TBCCWS) is designed to cool the station auxiliary equipment associated with the power conversion systems over the full range of normal station operation and normal shutdown.

1.2.2.8.2 Fuel Pool Cooling and Cleanup System

The Fuel Pool Cooling and Cleanup (FPC&C) system maintains the water temperature, purity, and radiation level in the spent fuel and upper containment pools within acceptable limits. This system has two 100%-capacity processing trains with each consisting of a transfer pump, filter demineralizer unit, and heat exchanger. Each heat exchanger is designed to provide the required cooling capacity to accommodate expected long term spent fuel storage. Both processing trains may be operated in parallel for cooling larger than expected heat loads. The RHR system is connected to the FPC&C system to provide supplemental cooling during shutdown, if necessary. A filter-demineralizer unit is employed to maintain water purity and control radioactive exposure.

1.2.2.8.3 Plant Service Water System

The Plant Service Water System (PSWS) is designed to cool plant auxiliaries which are not required for safe reactor shutdown and can be efficiently cooled by lake water (Lake Clinton). The system draws water from the cooling lake, pumps the coolant through the heat exchangers, and discharges it into the circulating water discharge, which is directed back to the station cooling lake. A radiation monitor is provided to monitor the discharge.

1.2.2.8.4 Ultimate Heat Sink

The ultimate heat sink (UHS) provides sufficient water volume and cooling capability for the station for at least 30 days with no water makeup. It is capable of withstanding the most severe natural phenomenon and postulated site-related incidents.

The UHS is a submerged pond and intake flume of about 590 acre-feet capacity that underlies Lake Clinton.

1.2.2.8.5 Condensate Storage and Transfer System

The condensate storage and transfer system stores condensate and distributes it to the main condenser, RCIC, and HPCS systems. The system is designed to maintain the water level of condensate in the condenser hotwell and provide condensate quality water to other station systems, as required. The condensate storage system also stores water required for refueling. The system consists of a storage tank, with a capacity of approximately 400,000 gallons, condensate transfer pumps, piping, and instrumentation.

1.2.2.8.6 Raw Water Treatment Plant and Makeup Water Treatment System

The Makeup Water Pump House pretreating system uses up flow filtration and reverse osmosis to provide filtered and potable water for station needs. Major system components are identified in Table 9.2-10, and the simplified flow path is shown in Figures 9.2-4 and 9.2-5. The Makeup Water Pump House also contains a Mixed Bed Polisher System (demineralizer) utilizing off site regenerated mixed bed resin media bottles. The combination of reverse osmosis and mixed bed polishers provides a reliable supply of demineralized water for station equipment and systems. Section 9.2.3, Demineralized Water Makeup System, provides more information on these treatment methods.

1.2.2.8.7 Potable and Sanitary Waste Water System

Potable and sanitary water is supplied from Lake Clinton. The water is treated, as required, to meet Illinois Department of Public Health drinking water standards.

Sanitary waste from the station area is treated to meet the requirements of National Pollutant Discharge Elimination System (NPDES) Permit issued by the Federal or State Permitting Authority.

Sewage treatment consists of primary and secondary aerated lagoon cells. The effluent of the lagoon is normally treated by tertiary sand filtration before release to the circulating water discharge flume. This filtration ensures compliance with the NPDES Permit for Total Suspended Solids. The sewage treatment system includes effluent sampling and flow measurement.

1.2.2.8.8 Plant Chilled Water Systems

The control room HVAC system circulates chilled water to the control room coolers.

The drywell cooling system provides chilled water to a total of six drywell coolers (four original and two supplemental drywell coolers).

The plant chilled water system supplies chilled water to area coolers and fan-coil units in the drywell (2 units), and the containment, turbine, radwaste, fuel, and auxiliary buildings' ventilation systems.

1.2.2.8.9 Process Sampling System

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

1.2.2.8.10 Plant Equipment and Floor Drainage

Drainage from equipment and the floor drain system throughout the containments, auxiliary, fuel, radwaste, control, and turbine buildings is collected in drain tanks and sumps. From these collection points, the drainage is pumped to collector tanks. Then a suitable method of treatment and disposal or recycling is selected.

1.2.2.8.11 Service and Instrument Air Systems

Oil free service and instrument air is supplied by three 50%-capacity centrifugal air compressors. Three 50%-capacity dryer filter packages are provided in the system to filter and dry the air. Air amplifiers provide the higher pressure air required for automatic depressurization system (ADS) valve actuation. Additional reliability is provided by equipping safety-related equipment with individual accumulators. The service and instrument air piping is conservatively sized to provide a source of compressed air in the event air compressor service is lost.

1.2.2.8.12 Diesel Generator Fuel-Oil Storage and Transfer System

The purpose of this system is to supply and store the fuel oil required to operate the diesel-generator units during post-LOCA maximum load demands. The principal design criteria for this system includes a 7-day fuel oil supply for each diesel, seismic Category I design, and missile protection.

1.2.2.8.13 Auxiliary Steam Systems

The auxiliary steam system is designed to be a separate and independent steam supply. Steam is generated by two package electrode boilers and sent to the two reboilers. The reboilers are used to generate steam free of the chemicals required for electrode boiler conductivity control. The steam generated by the reboilers is used primarily during station shutdown when main or extraction steam is not available. Auxiliary steam is used during shutdown for condensate deaeration/heating, reactor feedpump testing, and radwaste evaporator heating. During shutdown and startup, auxiliary steam can be used for main turbine shaft sealing. There is no impact on nuclear safety from the use or non-use of the steam blanketing system.

1.2.2.8.14 Heating Ventilation, and Air Conditioning/Environmental Systems

The station heating, ventilation, and air conditioning systems are designed to maintain proper air quality for personnel comfort and safety and satisfy requirements for equipment operation. In addition, the HVAC system for the control room and the ventilation systems for the standby diesel-generator rooms, essential switchgear rooms, service water pump rooms, and ECCS equipment cubicles are designed to operate and to maintain ambient conditions for equipment protection under postulated accident conditions as well as normal operation. The battery rooms are provided with exhaust fans. Habitability of the control room is provided for all postulated station accident conditions. Air distribution systems are designed so that airflow is directed from areas of lesser potential contamination to areas of progressively greater potential contamination.

The control room HVAC system consists of two Seismic Category I, 100%-capacity air conditioning systems; one of which is normally operating and one which is in standby. It serves the control room, back row panel room, computer room, and associated control room support areas.

The turbine building is supplied with filtered and tempered outside air for ventilation purposes. Fan-coil units supplied with chilled water remove heat lost from piping and equipment.

The radwaste building is ventilated by filtered and tempered outside air, and all exhaust air is filtered to remove contaminated particulates before being exhausted to the atmosphere. Fan-coil units supplied by chilled water is used for heat removal.

The standby diesel-generator rooms are ventilated when the diesel generators are operating. Fans start automatically to supply the rooms with outside air and to exhaust the air through a vent to the outside.

Shutdown service water-cooled fan-coil units for the service water pump rooms operate when the pumps run.

The essential switchgear rooms' heat removal is accomplished by the switchgear heat removal units. Outside air is introduced into the rooms to makeup for the battery room exhaust.

Other miscellaneous systems provide ventilation to the storage areas, circulating water pump house, waste water treatment house, auxiliary building, and service building.

The containment and fuel buildings are provided with filtered and tempered outside air for ventilation purposes. Fan-coil units using chilled water are used for heat removal.

The ECCS equipment cubicles have individual coolers supplied with shutdown service water to remove heat from the operating ECCS equipment.

The drywell is provided with two independent chilled water cooling systems and also plant chilled water. The control room is provided with two independent chilled water cooling systems; one of which is normally operating, while the other is on standby. A separate chilled water cooling system is provided for the service building.

1.2.2.8.15 Lighting Systems

The lighting system is comprised of four separate subsystems designated as normal, standby, emergency, and battery pack emergency. Special attention is given to areas where proper lighting is imperative during normal and emergency operations. The system design does not use mercury vapor fixtures above the containment pool and fuel handling pool areas. Note: The polar crane lights do not follow this restriction, see Section 9.1.4.2.2.1. The normal lighting systems are fed from the normal buses. Standby lighting is supplied from Class 1E power system buses. Emergency lighting is 125-Vdc. Battery pack emergency lighting units are installed on each floor at strategic locations. Normal operation and regular simulated offsite power-loss tests verify system integrity.

1.2.2.8.16 Fire Protection System

The fire protection system is designed to provide an adequate supply of water or other chemicals to points throughout the plant where fire protection may be required. Diversified fire-alarm and fire-suppression type systems are selected to suit the particular areas being protected or the hazards which could be encountered. The fire protection water is drawn from the ultimate heat sink which is sized to include 900,000 gallons of water for fire protection. The fire protection system consists of two 100% capacity diesel-driven fire pumps (primary fire protection system water supply), one connection to the plant service water system, a dedicated pressure maintenance jockey pump, and the associated piping, valves, and hydrants.

Chemical fire-fighting systems, such as CO_2 and Halon 1301, are also provided as additions to, or in lieu of, the water systems.

Appropriate instrumentation and controls are provided for the proper operation of the fire detection, annunciation, and firefighting systems.

1.2.2.8.17 Communication Systems

The station communcation systems are designed to provide reliable intraplant and offsite communications.

The public address system consists of an integrated system of speakers and handset paging units located throughout the station.

The dial telephone system consists of utility owned PBX equipment with several telephone stations located throughout the station.

Four independent radio communication systems are provided. Three systems are used primarily for normal onsite communications while the fourth system is used primarily for offsite emergency communications.

The fiber optics system consists of fiber optic cable and solid-state, battery-operated equipment designed to provide offsite voice communication, and protective relaying of the transmission system between CPS and Illinois Power's Brokaw substation.

The SpectraLink 900MHz Personal Communication Service (PCS) System consists of a computer driven master control unit located in the Service Building El. 722' LAN room and a dedicated telephone PBX for the system. Base station antennas for transmitting and receiving signals from SpectraLink cellular phones are mounted throughout the Clinton Nuclear Power Station.

1.2.2.9 Radioactive Waste Systems

1.2.2.9.1 Gaseous Radwaste System

The purpose of the gaseous radwaste system is to process and control the release of gaseous radioactive wastes to the site environs so the total radiation exposure to persons outside the controlled area does not exceed the maximum limits of the applicable 10 CFR 20 requlations even with some defective fuel rods.

The offgases from the main condenser are the major source of gaseous radioactive waste. The treatment of these gases includes volume reduction through a catalytic hydrogen-oxygen recombiner, water vapor removal through a condenser, decay of short-lived radioisotopes through a holdup line, further condensation and cooling, adsorption of isotopes on activated charcoal beds, further filtration through high efficiency filters, and final releases.

Continuous radiation monitors are provided which indicate radioactive release from the reactor and from the charcoal adsorbers. The radiation monitors are used to isolate the offgas system on high radioactivity in order to prevent releasing gases of unacceptably high activity.

1.2.2.9.2 Liquid Radwaste System

The liquid radwaste system collects, monitors, and treats liquid radioactive wastes for return to the station for reuse insofar as is practicable. The processing equipment is located in the radwaste building. Processed waste volumes discharged to the environs are expected to be small. Any discharge is such that concentrations and quantities of radioactive material and other contaminants are in accord with applicable local, state, and federal regulations.

All potentially radioactive liquid wastes are collected in sumps or drain tanks at various locations in the station. These wastes are transferred to collection tanks in the radwaste facility.

Waste processing is done on a batch basis. Appropriate processing procedures are determined based on known collection sources for the batch and operator observation of inline conductivity instrumentation. Equipment drains and other low-conductivity wastes are treated by filtration and demineralization, are sampled to verify quality, and then transferred to the cycled condensate storage tank for reuse. Floor drains and high-conductivity drainage are treated by evaporation and ion exchange and then are transferred to the cycled condensate storage tank. Chemical wastes and laundry drain wastes are treated by evaporation and demineralization and then are returned to cycled condensate. Protection against inadvertent release of liquid radioactive wastes is provided by design redundancy, instrumentation for the detection and alarm of abnormal conditions, automatic isolation, and procedural controls.

Equipment is selected, arranged, and shielded to permit operation, inspection, and maintenance with minimum radiation exposure to personnel.

1.2.2.9.3 Solid Radwaste System

The solid radwaste system provides for the safe handling packaging, and short-term storage of radioactive solid and concentrated liquid wastes that may be produced. Wet waste processed by this system is transferred to the mobile solidification system where it is solidified and packaged in containers. Dry waste such as rags, paper, and tools, is accumulated at designated storage areas for processing and disposal. Refer to Subsection 11.4.2.5.

1.2.2.10 Radiation Monitoring and Control

1.2.2.10.1 Process Radiation Monitoring

Process 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 selected plant process streams and effluent paths leading to the environment. All effluents from the plant which are potentially radioactive are monitored.

1.2.2.10.2 Area Radiation Monitors

Radiation monitoring devices are provided in areas where radioactive materials may be present, stored, handled, or inadvertently introduced. The devices provide signals for indicating, recording, and alarming abnormal levels of radioactivity.

1.2.2.10.3 Site Environs Radiation Monitors

The important air, aquatic, and terrestrial exposure pathways to man are monitored by measurements, including radiological surveys, passive dosimeters, and samples collected for laboratory analyses. The site radiological monitoring program was fully implemented at least 6 months prior to reactor criticality. The program is designed to document levels of direct radiation and concentrations of radionuclides that exist in aquatic and terrestrial ecosystems before and during station operation. The information gathered by the program will aid in assessing the radiological impact of station operation on the environment.

1.2.2.11 Shielding

Shielding is provided throughout the station, as required, to reduce radiation levels to operating personnel and to the general public within the applicable limits to set forth in 10 CFR 20 and 10 CFR 100. It is also designed to protect certain station components from radiation exposure which might result in component failure.

1.3 COMPARISON TABLES

1.3.1 Comparison with Similar Facility Design

This subsection highlights the principal design features of the Clinton Power Station (CPS) and provides a comparison of its major features with other boiling water reactor facilities. The design of this facility is based on proven technology attained during development, design, construction, and operation of boiling water reactors of similar or identical types. The data, performance characteristics, and other information presented here represent the design at the time the CPS Operating License was issued. There has been no effort to maintain this information current since that time.

Tables 1.3-1 through 1.3-7 compare CPS with River Bend, GESSAR and Grand Gulf, listing the design characteristics of the following:

a. nuclear steam supply and engineered safety features,

b. power conversion systems,

c. instrumentation and electrical systems,

d. containment,

e. structural requirements,

f. standby gas treatment system, and

g. radioactive waste management.

1.3.2 Comparison of Final and Preliminary Information

Table 1.3-8 provides a list of significant differences between the final and preliminary designs of the Clinton Power Station at the time the CPS Operating License was issued. These changes, which occurred following the submission of the CPS-PSAR, were controlled and approved in accordance with administrative procedures and were within the scope of the principal design criteria. In addition to identified changes, much information relating to newly designed equipment was included in the USAR; whereas only conceptualization of functional descriptions were available for the PSAR.

TABLE 1.3-1
COMPARISON OF NUCLEAR SYSTEM DESIGN AND OPERATING CHARACTERISTICS (1)

	CLINTON BWR/6, 218-624	RIVER BEND BWR/6, 218-592	GESSAR BWR/6, 238-748	GRAND GULF BWR/6, 251-800
THERMAL AND HYDRAULIC DESIGN (See Section 4.4)				
Rated power (MWt)	2894	2894	3579	3833
Design power (MWt) (ECCS design basis)	3015	3051	3729	4025
Steam flow rate (x 10^6 lb/hr)	12.453	12.45	15.40	16.491
Core coolant flow rate (x 10^6 lb/hr)	84.5	84.5	104.0	112.5
Feedwater flow rate (x 10^6 lb/hr)	12.428	12.42	15.372	16.455
System pressure, nominal in steam dome (psia)	1040	1040	1040	1040
Average power density (kW/1)	52.4	52.4	54.1	54.1(kW/1)
Minimum critical power ratio	>1.20	>1.18	>1.20	>1.23
Incore Neutron Instrumentation (Chapters 4 and 7)				
Number of incore neutron detectors (fixed)	132	132	164	176
Number of incore detector assemblies	33	33	41	44
Number of LPRM detectors	132	-	164	176

TABLE 1.3-1 (Cont'd)

	CLINTON	RIVER BEND	GESSAR	GRAND GULF
Number of SRM detectors	4	-	4	6
Number of IRM Detectors	8	-	8	8
Number of APRM Channels	4	-	4	8
Number and type of incore neutron sources	5 Sb-Be	5 Sb-Be	7 Sb-Be	7 Sb-Be
REACTOR VESSEL DESIGN (See Section 5.3)				
Material	-----------------Low alloy steel/partially clad-----------------			
Design pressure (psig)	1250	1250	1250	1250
Design temperature (°F)	575	575	575	575
Inside diameter (ft-in.)	18-2	18-2	19-10	20-11
Coolant enthalpy at core inlet (Btu/lb)	527.8	527.8	527.7	527.9
Core maximum exit voids within assemblies	76	76	79	76
Separator design inlet quality (% steam)	15.0	14.6	14.7	14.7
Feedwater temperature (°F)	420	420	420	420
Total peaking factor	2.33	2.23	2.26	2.26
CORE MECHANICAL DESIGN	(See Table A.1.3-1 of NEDE 24011-P-A)			
REACTOR CONTROL SYSTEM	(See Chapters 4 and 7)			

TABLE 1.3-1 (Cont'd)

	CLINTON	RIVER BEND	GESSAR	GRAND GULF
Method of varying reactor power	---Movable control rods; variable forced coolant flow----			
Number of movable control rods	145	145	177	193
Type of control rod drives	-----------------Bottom entry; locking piston-----------			
Inside height (ft-in.)	69-4	69-4	70-4	73-0
Minimum base metal thickness (cylindrical section) (in.)	5.379	5.4	6.0	6.14
Minimum cladding thickness (in.)	1/8	1/8	1/8	1/8
<u>REACTOR COOLANT RECIRCULATION DESIGN</u> (See Chapter 5)				
Number of recirculation loops	2	2	2	2
Design Pressure:				
Inlet leg (psig)	1250	1250	1250	1250
Outlet leg (psig)	1650 (2)	1650 (2)	1650 (2)	1625 (2)
	1550 (3)	1550 (3)	1550 (3)	1525 (3)
Design temperature (°F)	575	575	575	575
Pipe diameter (in.)	20	20	22/24	24
Pipe material (AISI)	304/316	304/316	304/316	304
Recirculation pump flow rate (gpm)	32,500	32,500	42,000	44,900
Number of jet pumps in reactor	20	20	20	24

TABLE 1.3-1 (Cont'd)

	CLINTON	RIVER BEND	GESSAR	GRAND GULF
MAIN STEAMLINES (See Section 5.4)				
Number of steamlines	4	4	4	4
Design pressure (psig)	1250	1250	1250	1250
Design temperature (×F)	575	575	575	575
Pipe diameter (in.)	24	24	26	28
Pipe material	-----------------Carbon steel-----------------			
EMERGENCY CORE COOLING SYSTEMS				
Low Pressure Core Spray Systems	(See Section 6.3)			
Number of loops	1	1	1	1
Flow rate (gpm)	5010 at 119 psid	5010 at 119 psid	6000 at 122 psid	7115 at 128 psid
High Pressure Core Spray System	(See Section 6.3)			
Number of loops	1	1	1	1
Flow rate (gpm)	1400 at 1147 psid 5010 at 200 psid	1400 at 1147 psid 5010 at 200 psid	1550 at 1147 psid 6110 at 200 psid	1650 at 1147 psid 7115 at 200 psid
Automatic Depressurization System	(See Section 6.3)			
Number of relief valves	7	7	8	8

TABLE 1.3-1 (Cont'd)

	CLINTON	RIVER BEND	GESSAR	GRAND GULF
Low Pressure Coolant Injection (4)	(See Section 6.3)			
Number of loops	3	3	3	3
Number of pumps	3	3	3	3
Flow rate (gpm/pump)	5050 at 20 psid	5050 at 20 psid	7100 at 20 psid	7450 at 20 psid
AUXILIARY SYSTEMS				
Residual Heat Removal System (See Section 5.4)				
Reactor shutdown cooling, number of pumps	2	2	2	2
Flow rate (gpm/pump) (5)	5050	5050	7100	7450
Duty (x 10^6 Btu/hr/heat exchanger) (6)	37.8	37.8	46.9	50
Number of heat exchangers	2	2	2	2
Primary containment cooling mode flow rate (gpm)	5050	5050	7100	7450
Shutdown Service Water System (See Section 9.2)				
Flow rate (gpm/heat exchanger)	34,100 (total)	5800	Applicant provided	25,300 (total)
Number of pumps	2 @ 16,500 gpm 1 @ 1,100 gpm	(7)	Applicant provided	2 @ 12,000 1 @ 1,300

TABLE 1.3-1 (Cont'd)

	CLINTON	RIVER BEND	GESSAR	GRAND GULF
Reactor Core Isolation Cooling System	(See Section 5.4)			
Flow rate (gpm)	600 at 1200 psid	600 at 1177 psid	700 at 165-1192 psia reactor pressure	800 at 1120 psid
Fuel Pool Cooling and Cleanup System	(See Section 9.1)			
Heat Removal Capacity ($x10^6$ Btu/hr/heat exchanger)	19.7	9.8	8.0	12.5

TABLE 1.3-1 (Cont'd)

NOTES

1. This table provides an historical comparison of the Clinton Power Station nuclear system design and operating characteristics with other BWR/6 reactor designs, based on the design at the time the Operating License was issued. This table has not been maintained current. See the designated USAR sections for the current CPS design.

2. Pump and discharge piping to and including the discharge block valve.

3. Discharge piping from the discharge block valve to the vessel.

4. A mode of the RHR system.

5. Capacity during reactor flooding mode with two or three pumps running.

6. Heat exchanger duty to provide cooldown in 20 hours.

7. Service water pumps used for cooling water supply.

TABLE 1.3-2
COMPARISON OF POWER CONVERSION SYSTEM DESIGN CHARACTERISTICS [1]

	CLINTON BWR/6, 218-624	RIVER BEND BWR/6, 218-592,	GESSAR BWR/6, 238-748	GRAND GULF BWR/6, 251-800
TURBINE-GENERATOR (See Section 10.2)				
Rated power (MWe) (Gross)	985	991	1269	1306
Generator speed (rpm)	1800	1800	1800	1800
Rated steam flow (lb/hr) (Guar.)	11.340×10^6	12.435×10^6	15.40×10^6	15.542×10^6
Inlet pressure (psig)	965	965	975	965
STEAM BYPASS SYSTEM (See Section 10.4)				
Capacity (% design steam flow)	35	10	35 [2]	35
MAIN CONDENSER (See Section 10.4)				
Heat removal capacity (Btu/hr)	6453×10^6	6860×10^6	7996×10^6	8506×10^6
CIRCULATING WATER SYSTEM (See Section 10.4)				
Number of pumps	3	4	Applicant provided	2
Flow rate (gpm/pump)	189,600	127,890	Applicant provided	285,500
CONDENSATE AND FEEDWATER SYSTEM (See Section 10.4)				
Design flow rate of feedwater to reactor (lb/hr)	12.421×10^6	12.421×10^6	15.372×10^6	15.542×10^6
Number of condensate pumps	4	3	Applicant provided	3

TABLE 1.3-2 (Cont'd)

	CLINTON	RIVER BEND	GESSAR	GRAND GULF
Number of condensate booster pumps	4	0	Applicant provided	3
Number of feedwater pumps	3	3	Applicant provided	2
Number of feedwater booster pumps	0	-	Applicant provided	-
Condensate pump drive	A-C	A-C	Applicant provided	A-C
Booster pump drive	A-C	-	Applicant provided	A-C
Feedwater pump drive	Turbine – 2 A-C – 1	A-C	Applicant provided	Turbine
Feedwater booster pump drive	-	-	Applicant provided	-

Notes:

1. This table provides an historical comparison of the Clinton Power Station power conversion system design characteristics with other BWR/6 reactor designs, based on the design at the time the Operating License was issued. This table has not been maintained current. See the designated USAR sections for the current CPS design.
2. Minimum design value.

TABLE 1.3-3
COMPARISON OF ELECTRICAL SYSTEMS [1]
(See Chapter 8)

SYSTEMS	CLINTON 1 UNIT	RIVER BEND 2 UNITS	GESSAR 1 UNIT	GRAND GULF 2 UNITS
Number of offsite circuits	4	6	NA [2]	First unit - 3 Both units - 4
Number of auxiliary power sources	2 unit aux. trans.;1 reserve aux. trans.; 1 emerg. res.aux. trans.	4 unit aux. trans. 4 reserve aux. trans.	2 unit aux. trans.;2 reserve aux. trans.	3 svce. trans. (1 exclusively for ESF [3])
Number of preferred power circuits to ESF a-c buses	2	2	2 per division	3
Number of a-c ESF buses per unit	3	3	3	3
Number of standby a-c power supplies	3 (1/ESF bus)	6 (1/ESF bus)	3 (1/ESF bus)	6 (1/ESF bus)
Number of 125-Vdc systems supplying a-c ESF buses	3 (1/ESF bus)	6 (1/ESF bus)	3 (1/ESF bus)	6 (1/ESF bus)
Sharing of standby power supplies and Interconnections between safety buses	NONE	NONE	NONE	NONE

Notes:
1. This table provides an historical comparison of the Clinton Power Station electrical system with other BWR/6 reactor designs, based on the design at the time the Operating License was issued. This table has not been maintained current. See the designated USAR sections for the current CPS design.
2. NA is defined as not applicable.
3. ESF is defined as engineered safety features.

TABLE 1.3-4
COMPARISON OF CONTAINMENT DESIGN CHARACTERISTICS (1)
(See Chapter 3)

	CLINTON BWR/6, 218-624	RIVER BEND BWR/6, 218-592	GESSAR BWR/6, 238-748	GRAND GULF BWR/6, 251-800
Type	Mark III; reinforced concrete as for PWR plants, but with pressure suppression; containment encloses drywell and suppression pool	Horizontal pressure suppression; Mark III freestanding steel	Mark III freestanding steel with reinforced concrete shield building	Similar to Clinton
Leak Rate (%/day)	0.65	0.26	1.0	0.35
CONTAINMENT				
Construction	Reinforced concrete cylindrical structure (not pre-stressed) with hemispherical dome; steel lined	Freestanding steel; similar to GESSAR	Cylindrical freestanding steel with ellipsoidal head	Similar to Clinton
Internal design temp. (°F)	185	185	185	185
Design pressure (psig)	15	15	15	15
Free air volume (ft^3)	1,550,800	1.192×10^6	1.140×10^6	1.4×10^6

TABLE 1.3-4 (Cont'd)

	CLINTON	RIVER BEND	GESSAR	GRAND GULF
DRYWELL				
Construction	Reinforced concrete cylinder; steel concrete form plate; steel access head	Reinforced concrete cylinder	Concrete cylinder	Similar to Clinton
Internal design temp. (°F)	330	330	330	330
Design differential pressure (psid)	30	25	25	30
Free (air) volume, total (ft^3)	246,500	251,000	275,000	270,000
SUPPRESSION POOL				
Construction	Steel lined concrete cylinder	Concrete steel	Steel lined concrete cylinder	Steel lined concrete cylinder
Internal design temp. (°F)	185	185	185	185
Design pressure (psig)	15	20	15	15
Water volume (ft^3)	146,400	127,930	129,600	136,000

Note: 1. This table provides an historical comparison of the Clinton Power Station containment design characteristics with other BWR/6 reactor designs, based on the design at the time the Operating License was issued. This table has not been maintained current. See the designated USAR sections for the current CPS design.

TABLE 1.3-5
COMPARISON OF STRUCTURAL DESIGN REQUIREMENTS [1]

	CLINTON BWR/6, 218-624	RIVER BEND BWR/6, 218-592	GESSAR BWR/6, 238-748	GRAND GULF BWR/6, 251-800
SEISMIC DESIGN (See Section 3.2 and 3.7)				
Safe shutdown earthquake [2]				
(horizontal g)	0.25	0.10	0.30	0.15
(vertical g)	0.25	0.10	0.30	0.10
Operating basis earthquake				
(horizontal g)	0.10	-	0.15	0.05
(vertical g)	0.10	-	0.15	0.075
WIND DESIGN (See Section 3.3)				
Maximum sustained wind (mph)	85	100	130	90
Tornadoes (mph)	290 tang. + 70 trans.	290 tang. + 70 trans.	290 tang. + 70 trans.	300 tang. + 60 trans.

Notes:
1. This table provides an historical comparison of the Clinton Power Station structural design requirements with other BWR/6 reactor designs, based on the design at the time the Operating License was issued. This table has not been maintained current. See the designated USAR sections for the current CPS design.
2. Previously called design-basis earthquake.

TABLE 1.3-6
COMPARISON OF STANDBY GAS TREATMENT SYSTEMS [1]
See Sections 6.2 and 6.5)

	CLINTON BWR/6, 218-624	RIVER BEND BWR/6, 218-592	GESSAR BWR/6, 238-748	GRAND GULF BWR/6, 251-800
Charcoal bed design	2, all welded deep bed	2, deep bed	Deep bed	Deep bed
DESIGN EFFICIENCIES				
Elemental iodine (%)	99.9	90	99.0	99.9
Organic iodine (%)	99.9	70	99.0	99.0
0.3-micron particles (%)	99.97	90	99.97	99.97
System flow (cfm)	4000	15,000	6000	2000

Note: 1. This table provides an historical comparison of the Clinton Power Station standby gas treatment system with other BWR/6 reactor designs, based on the design at the time the Operating License was issued. This table has not been maintained current. See the designated USAR sections for the current CPS design.

TABLE 1.3-7
COMPARISON OF RADIOACTIVE WASTE MANAGEMENT [1]

	CLINTON	RIVER BEND	GESSAR	GRAND GULF
GASEOUS RADWASTE (See Section 11.3)				
Design basis – noble gases (μCi/sec)	100,000 at 30 min	100,000 at 30 min	100,000 at 30 min	100,000 at 30 min
Process treatment	Chilled charcoal	Chilled charcoal	Chilled charcoal	Chilled charcoal
Number of beds	2	8	2	8
Design condenser inleakage (cfm)	30	30	30	40
Release point-feet above ground (ft)	199.5	190	Applicant provided	31.5
LIQUID RADWASTE (See Section 11.2)				
Treatment of:				
1. Floor drains - equipment drains [2]	E, or F-D, returned to condensate storage or discharged	Similar to Clinton	Evaporated demineralized, returned to condensate storage	F, E, D, returned to condensate storage
2. Laundry drains [2]	F-E, discharged or recycled	-	Filtered and discharged, or evaporated with discharged vapor	None [3]
3. Chemical drains [2]	E, D,discharged or recycled	Similar to Clinton	Same as floor drains	Neutralized E returned to equipment drain collector tank

TABLE 1.3-7 (Cont'd)

	CLINTON	RIVER BEND	GESSAR	GRAND GULF
4. Equipment drains (2)	F, D, Recycled or discharged	-	E, D, RO	F, E, D and returned to condensate storage
Expected annual average release (mCi)	100	10	100	110

Notes:
1. This table provides an historical comparison of the Clinton Power Station radioactive waste management systems with other BWR/6 reactor designs, based on the design at the time the Operating License was issued. This table has not been maintained current. See the designated USAR sections for the current CPS design.
2. F = filtered, D = demineralized, E = evaporated, and RO = reverse osmosis.
3. Laundry drains will be processed offsite by an authorized contractor.

TABLE 1.3-8
SIGNIFICANT DESIGN CHANGES FROM PSAR TO FSAR [(1)]

ITEM	CHANGE	REASON FOR CHANGE	FSAR SECTION IN WHICH CHANGE IS DISCUSSED
SRV discharge piping	Upgraded from non-Seismic Category I and Quality Group D to Seismic Category I and Quality Group C	To comply with NRC requirements	3.2
Main steam shutoff valve	Added	Regulatory requirement to prevent bypass leakage	3.2
Component cooling water	Quality group changed from C to D	Provides cooling only during normal operation	1.8, 3.2
Feedwater piping	Quality group of piping between the outermost isolation valve and the second isolation valve changed from A to B	To conform to Regulatory Guide 1.26	3.2
Combustible gas control system	1% metal water reaction changed to 0.73%	Change to be in agreement with Regulatory Guide 1.7, Revision 2	6.2.5
	Zr corrosion rate data updated to state-of-the-art	New test data	6.2.5
Radiation monitoring system	Changed from conventional individual analog type monitoring system to a digital radiation monitoring system	Improvement in system capabilities and operating interface	7.6, 7.7, 11.5, 12.3
Spent fuel pool	Storage capacity increased from 155% of full core to 400% of full core	Increase spent fuel storage capacity	9.1.2
Ultimate heat sink	Continuous discharge during normal plant operation	Plant service water is used to cool essential equipment during normal operation and discharged to ultimate heat sink	9.2.5
ADS air supply	Added air amplifiers and compressed air storage for ADS valves	Increased system reliability	9.3.1

TABLE 1.3-8 (Cont'd)

ITEM	CHANGE	REASON FOR CHANGE	FSAR SECTION IN WHICH CHANGE IS DISCUSSED
Control room HVAC system	No longer serves cable spreading room	Cable spreading room is served by switchgear room HVAC	9.4.1, 9.4.5.2
Control room HVAC system	Pressure differential control changes	It was determined to be more reliable to assure maintenance of control room differential pressure by design of structures and systems than by controls proposed in the PSAR	9.4.1
Essential switchgear room HVAC	Divisional switchgear rooms are now served by a nuclear safety-related HVAC system and a nonnuclear safety-related HVAC system	More economical design	9.4.5.2
Containment building HVAC system	Two speed supply fans used	To enhance a 2-unit purge through drywell purge units	9.4.6
Containment building HVAC isolation valves	Added a normally closed bypass valve	Permit post-LOCA purge	9.4.6
Laundry system	Laundry waste now processed by evaporator	Deletion of reverse osmosis system	11.2
Solid radwaste system	Purchased system different than conceptual design	Purchased available equipment	11.4
Nuclear Fuel	The number of water rods in each fuel bundle has been changed from 1 to 2. Five different U-235 enrichments are now used in the fuel assemblies instead of previous four types	Improved fuel performance	4.2
Control rod drive position indication	Changed to 11 wire probe and solid state	Improved reliability and increased frequency of checking actual rod position	7.7

TABLE 1.3-8 (Cont'd)

ITEM	CHANGE	REASON FOR CHANGE	FSAR SECTION IN WHICH CHANGE IS DISCUSSED
Feedwater sparger	The thermal sleeve was changed to provide improved slip fit design of sparger to nozzle	To eliminate failure, leakage, and provide for possible in-service inspection	5.3
Leak detection system	The leak detection system was revised to upgrade the capability and incorporate the requirements of IEEE 279. Added additional monitors to increase adequacy of detection	To meet IEEE 279 and Regulatory Guide 1.45 requirements	7.6
Control rod drive fast scram	Increased system pressure from 1750 psi to 2000 psi, enlarged insert/withdraw draw lines, and increased accumulator volume to provide faster scram time	Provides increased reactivity control, especially at end of fuel cycle. Provides increased thermal margin, and reduces amount of operation of steam relief	3.9, 4.6
Reactor Recirculation pump trip	Pumps tripped on signals from turbine control or stop valves upon generator load rejection or turbine trip	Reduces transient core flow and reactivity. Works with fast scram to provide increased thermal margin	4.6, 5.4, 7.6
Reactor Protection System	Changes for control system instrument testability. Changed from switches to transmitters and added calibration units	Provides improved testability reliability	7.2
Reactor Recirculation Pump Motor Controls	Added motor-generator sets to provide control for reduced flow during startup and shutdown	Provides improved operation	7.7
Reactor recirculation System	Removed pump bypass lines for reduction of region potentially sensitive to stainless steel stress corrosion problems	Design improvement	5.4
Suppression Pool Clean-up System	Added system	To reduce doses inside containment	9.3

TABLE 1.3-8 (Cont'd)

ITEM	CHANGE	REASON FOR CHANGE	FSAR SECTION IN WHICH CHANGE IS DISCUSSED
Containment Leakage	Raised from 0.5%/day to 0.65%/day	Improved meteorological data	6.2
Condensate cleanup System	An ultrasonic resin cleaning system has been included	To save chemical regeneration cost and radwaste disposal costs	10.4
Turbine Bypass Capacity	Increased from 10% to 35%	Improve operability	10.4
Recirculation System	Flow Control Valve Automatic Runback on Circulating Water Pump Trip	Protect turbine	5.4
Recirculation System	Material Change	Improve plant reliability by reducing cracking.	5.4
RWCU Heat Exchangers	Redundant Heat Exchangers Added	Improve system reliability	5.4
CRD System	Return line to RPV deleted	Reduce nozzle cracking problem.	4.6
NSPS	Replacement of switch type process sensors and relay logic with electronic transmitters and solid-state logic	To avoid switch setpoint drift and improved reliability	8.3
RWCU System	Third pump added	Improve system reliability	5.4.8
Drywell	Interior wall lined with steel	Construction expedient-liner used as concrete form	3.8
Ultimate Heat Sink	Increase in size from 650 acre-feet to 1067 acre-feet	Provide margin for silting	9.2

TABLE 1.3-8 (Cont'd)

ITEM	CHANGE	REASON FOR CHANGE	FSAR SECTION IN WHICH CHANGE IS DISCUSSED
Fuel Pool Cooling and Cleanup System	Increased heat exchanger capacity from 15.95 x 10^6 Btu/hr to 19.7 x 10^6 Btu/hr	Accommodate increased heat load	9.1
Supplemental Drywell Cooling System	Addition of four air handling units - total capacity 2.65 x 10^6 Btu/hr	Provide additional margin	9.4.7

Note: 1. This table provides historical information comparing the significant design changes between the Preliminary Safety Analysis Report and the Final Safety Analysis Report. This table has not been maintained current.

1.4 IDENTIFICATION OF AGENTS AND CONTRACTORS

1.4.1 Applicant

Exelon Generation Company, LLC

Exelon Generation Company, LLC (Exelon) merged with AmerGen in January 2009. Exelon is responsible for the safe and reliable operation of CPS.

AmerGen Energy Company, LLC

AmerGen Energy Company, LLC (AmerGen) was formed in 1997 to purchase and operate nuclear plants in the United States. AmerGen purchased Clinton Power Station (CPS) from the Illinois Power Company on December 15, 1999.

Illinois Power Company

Illinois Power Company (IP) was a corporation duly organized and existing under the laws of the State of Illinois, is engaged in, among other things, the generation, distribution, and sale of electricity to the public in various municipalities and places in the State of Illinois.

1.4.2 Architect-Engineer - Sargent & Lundy

For the work covered by this application, Sargent & Lundy was retained as the architect-engineer and design consultant. Illinois Power Company had previously employed Sargent & Lundy for the design of many of its fossil-fueled plants.

Sargent & Lundy is an independent consulting engineering firm founded in Chicago in 1891. For over 87 years, the firm has specialized exclusively in the design of generation, transmission, distribution, and related facilities for the electric power industry as well as for industrial and commercial clients.

The firm has provided a complete range of engineering services for more than 675 authorized turbine-generator units with a total capacity of over 85,000,000 kW. Of this total, over 20,000,000 kW represent nuclear generating capacity, the majority of which are light water reactor types.

1.4.3 General Electric

1.4.3.1 Nuclear Steam Supply System

The General Electric Company (GE) was awarded the contracts to design, fabricate, and deliver the single-cycle, boiling water, nuclear steam supply system; to fabricate the first core of nuclear fuel; and to provide technical direction of installation and startup of this equipment. GE has engaged in the development, design, construction, and operation of boiling water reactors since 1955. Currently, GE has over 80 reactors completed, under construction, or on order. Thus, GE has substantial experience, knowledge, and capability to design, manufacture, and furnish technical assistance for the installation and startup of reactors.

1.4.3.2 Turbine-Generator

IP awarded the contract to General Electric Co. to design, fabricate, and deliver the turbine-generator for the station and to provide technical assistance for the installation and startup of this equipment. General Electric Co. has a long history in the application of turbine-generators in nuclear power stations, dating back to the inception of nuclear facilities for the production of electrical power. General Electric Co. is furnishing the turbine-generator units for most stations equipped with its BWR nuclear steam supply systems. General Electric Co. is also supplying turbine-generator units for various other nuclear power plants.

1.4.4 Constructor

Baldwin Associates, a joint venture general construction contractor, is responsible for all phases of site and station construction.

Baldwin Associates was originally formed as a joint venture in 1967 to construct the initial unit of Baldwin Power Station for Illinois Power Company. The partners to the joint venture were:

a. Power Systems, Inc.;

b. Fruin-Colnon Contracting Company;

c. Kelso-Burnett Electric Co.; and

d. McCartin-McAuliffe Plumbing and Heating Company, Inc. (Note: In 1973 the name of the company was changed to McCartin-McAuliffe Mechanical Contractors, Inc.)

Power Systems, Inc. was the sponsor of the joint venture.

Baldwin Associates holds ASME Certificate of Authorization for the application of "NA" and "NPT" Stamps.

Power Systems, Inc. was organized in 1963 to take over the construction division of Henry Pratt Company, founded in 1903. In 1971, Power Systems, Inc. became a wholly-owned subsidiary of Fischbach and Moore, Inc.

Power Systems, Inc. is a mechanical constructor engaged in the installation of heavy mechanical equipment and piping, chiefly for utility companies. Its experience covers the installation of more than 9,000,000 kW of turbine-generator capacity in units ranging up to 1100 MW in size, and steam generators ranging in capacity up to 4,000,000 lb/hr.

Power Systems, Inc. holds ASME Certificates of Authorization for the application of "A", "PP", and "U" stamps. It was among the earliest of mechanical constructors to secure ASME approval of a Quality Assurance Manual and to obtain an Interim Letter of Authorization leading to the application of an "NA" stamp to the field fabrication of Class 1, 2, and 3 nuclear piping subassemblies, components, and systems.

Fruin-Colnon Corporation is a civil/structural contractor founded in 1873 in St. Louis, Missouri, where the company is headquartered. There are division offices in New Orleans, Louisiana, and San Francisco, California.

The company is made up of seven divisions based on geographic location and specialities; i.e., waste and water treatment, design and construction projects, industrial projects, and electrical power projects.

Fruin-Colnon Corporation is engaged primarily in earth moving, substructure, and superstructure, including structural steel for industrial and heavy engineered projects. In addition, they have performed work on some commercial and institutional projects. Over the past 40 years they have gained substantial experience in the construction of power plants.

Kelso-Burnett Electric Co. was founded in 1908 as an Illinois corporation engaged in electrical contracting.

Kelso-Burnett Electric Co. has specialized in industrial construction throughout the United States, including electrical work for electrical generating stations, substations, industrial plants, and industrial distributions systems; cement, paper, and chemical plants; refineries; locks and dams; commercial buildings; and ordnance plants.

Previous electric generating experience includes all the electrical construction for 13 power stations totalling over 4300 MW electrical capacity.

McCartin-McAuliffe Mechanical Contractors, Inc. is an Illinois corporation established in 1961. They are contractors in the State of Illinois for the parent firm McAuliffe Mechanical Contractors, Inc., an Indiana corporation which has been serving the piping industry for over 60 years.

McCartin-McAuliffe Mechanical Contractors, Inc., is a piping constructor that has been actively involved in the construction and maintenance of piping systems for power stations, oil refineries, edible oil plants, chemical plants, steel mills, and pipeline transmission systems. Its responsibilities to these industries include power piping, process piping, instrument piping, industrial plumbing, heating, air conditioning, ventilation, and associated equipment placement and setting.

McCartin-McAuliffe Mechanical Contractors, Inc., is authorized to fabricate, assemble, and erect (pressure piping) as provided for by the ASME, Boiler and Pressure Vessel Committee; the company holds the Society's Certificate of Authorization for the application of the "PP" stamp.

1.4.5 Technical Consultants

The technical consultants employed by Illinois Power Company in the design and construction of the Clinton Power Station include Dames & Moore; Hazleton Environmental Sciences Corporation; The Research Corporation (TRC) of New England; Harza Engineering; Nuclear Services Corporation (NSC); and Southwest Research Institute.

A brief description of each follows.

1.4.5.1 Dames & Moore

The independent consulting firm of Dames & Moore was employed in the preparation of the data relating to geology and seismology for the Clinton site. Having performed environmental studies for approximately 30 nuclear power plant sites, Dames & Moore is active in the field of environmental engineering related to nuclear power plant construction.

1.4.5.2 Hazleton Environmental Sciences Corporation

The preconstruction ecological monitoring program at the Clinton site was started in May 1974 by the Environmental Division of Industrial Bio-Test (IBT) Laboratory, Inc., a subsidiary of

NALCO Chemical Company. In the fall of 1975 NALCO purchased the Environmental Division of IBT and the firm's name then became NALCO Environmental Sciences (NES). NES was a part of the Industrial Division of NALCO Chemical Company. In November 1978 Hazleton Laboratories Corporation, Vienna, Virginia, purchased NES from NALCO. The firm's name then became Hazleton Environmental Sciences (HEC).

Through April of 1978, this consultant sampled water chemistry and aquatic and terrestrial ecology. In May of 1978, Illinois Power Company began performing the water chemistry and aquatic ecology sampling and analyses. Hazleton Environmental Sciences has since been acquired by Teledyne Isotopes and became Teledyne Isotopes Midwest Laboratory (TIML). TIML continues to support sampling and analysis activities associated with the CPS Radiological Environmental Monitoring Program.

1.4.5.3 The Research Corporation of New England

The Research Corporation (TRC) of New England provides services to industry, commerce, and government in the fields of environmental pollution and waste management. TRC has conducted environmental studies for existing and proposed nuclear generating stations since 1962. These studies have included design, implementation, and operation of onsite field studies and the preparation of the meteorology and hydrology sections for Safety Analysis Reports. As a technical consultant on the CPS project, TRC provided services in the area of meteorology data collection, compilation, and analysis.

1.4.5.4 Harza Engineering

Harza Engineering has been involved with a variety of technical studies for at least ten nuclear power stations. Among these studies have been facility design, review of design and structure, hydrology, and groundwater. In addition, Harza Engineering has designed some of the largest hydroelectric projects in the world, including major concrete structures and earth-filled dams.

As a technical consultant on the CPS project, Harza Engineering was responsible for the review of the cooling lake main dam embankment design.

1.4.5.5 Nuclear Services Corporation

Nuclear Services Corporation provides a wide range of capabilities including assistance in nuclear power plant engineering, nondestructive examination, quality assurance services, prestartup and inservice inspection, nuclear fuel services, and plant startup and operations planning. The organization consists of a broadly based professional staff covering many engineering and quality assurance fields with particular emphasis on quality assurance.

As a technical consultant on the CPS project, NSC has provided services in the areas of startup testing program development, audits of contractor's quality assurance program, quality assurance program development, and containment analysis.

1.4.5.6 Southwest Research Institute

Southwest Research Institute of San Antonio, Texas, provides pre service and inservice inspection (PSI/ISI) services for utilities. These services include the designing for inspection access, establishing technical specifications, and the planning and performing of the examinations.

For the CPS Project, Southwest Research Institute has provided access engineering support; technical consultation on PSI/ISI matters; a program plan for total work scope; and project plans for access engineering and preservice inspection.

1.5 REQUIREMENTS FOR FURTHER TECHNICAL INFORMATION

1.5.1 Current Development Programs

1.5.1.1 Instrumentation for Vibration

Vibration testing for reactor internals is performed on all GE BWR plants. At the time of issue of NRC Regulatory Guide 1.20, test programs for compliance were instituted. The first BWR 6 plant of each size is considered a prototype design and will be instrumented and subjected to both cold and hot, two-phase flow testing to demonstrate that flow-induced vibrations similar to those expected during operation will not cause damage. Subsequent plants which have internals similar to those of the prototypes will be tested in compliance to the requirements of Regulatory Guide 1.20 to confirm the adequacy of the design with respect to vibration. Further discussion is presented in Subsection 3.9.2.

1.5.1.2 Core Spray Distribution

GE has a program in place to predict BWR 6 core spray distributions using a combination of single nozzle steam and air tests, single and multiple nozzle analytical models, and full scale air tests. This methodology has been confirmed by a full scale 30 degrees sector steam test in early 1979. The results have been submitted to the NRC. A favorable response was received.

1.5.1.3 Core Spray and Core Flooding Heat Transfer Effectiveness

Due to the incorporation of an 8 x 8 fuel rod array with unheated "water rods," tests have been conducted to demonstrate the effectiveness of ECCS in the new geometry.

These tests are regarded as confirmatory only, since the geometry change is very slight and the "water rods" provide an additional heat sink in the inside of the bundle which improves heat transfer effectiveness.

There are two distinct programs involving the core spray. Testing of the core spray distribution has been accomplished, and the Licensing Topical Report NED0-10846, "BWR Core Spray Distribution," April 1973, has been submitted. The other program concerns the testing of core spray and core flooding heat transfer effectiveness. The results of testing with stainless steel cladding were reported in the Licensing Topical Report NED0-10801, "Modeling the BWR/6 Loss-of-Coolant Accident: Core Spray and Bottom Flooding Heat Transfer Effectiveness," March 1973. The results of testing using Zircaloy cladding were reported in the Licensing Topical Report, NED0-20231, "Emergency Core Cooling Tests of an Internally Pressurized, Zircaloy Clad, 8 x 8 Simulated BWR Fuel Bundle," December 1973.

1.5.1.4 Verification of Pressure Suppression Design

The General Electric Company has conducted a large scale test program to verify the performance characteristics of the Mark III containment. The purpose of the Mark III Test Program was to confirm the analytical methods used to predict the drywell and containment pressure response following the postulated LOCA. In addition, this Test Program also was used to obtain information on the hydrodynamic loads that are generated in the vicinity of the suppression pool during a LOCA.

The General Electric Mark III containment pressure suppression testing program was initiated in 1971 with a series of small-scale tests. The test apparatus consisted of small-scale simulations of the reactor pressure vessel, drywell, suppression pool, and horizontal vents. A total of 67 blowdown runs were made. The purpose of these tests was to determine the behavior of the horizontal vents and to obtain data for determining the acceleration of the water in the test section vents during initial clearing. This information was used to establish an analytical model for predicting vent system performance in Mark III and the resulting drywell pressure response.

In November 1973, testing in the Mark III Pressure Suppression Test Facility (PSTF) began. The PSTF consists of an electrically heated steam generator connected to a simulated vent system which can be heated to prevent steam condensation within its volume during the simulated blowdowns. The drywell is modeled as a cylindrical vessel having a 10-foot diameter and 26-foot height. A 6-foot diameter vent duct passes from the drywell into the suppression pool and connects to the simulated vent system. Pool baffles are used to simulate a scaled or full scale sector of a Mark III suppression pool. The pool arrangement is such that both vent submergence and pool areas can be varied parametrically.

The full-scale PSTF testing performed between November 1973 and February 1974 obtained data for the confirmation of the analytical model. In March 1974 pool swell tests were performed in the PSTF. These full-scale tests involved air blowdown into the drywell and suppression pool to identify bounding pool swell impact loads and breakthrough elevation, i.e., that elevation at which the water ligament begins to break up and impact loads are significantly reduced. Impact load data was obtained on selected targets located above the pool.

In June of 1974, after the PSTF vent and pool system was converted to 1/3-scale, four series of tests were performed to provide transient data on the interaction of pool swell with flow restrictions above the suppression pool surface. Other areas where data was obtained included vent clearing, drywell pressurization, and jet forces on pool walls.

The next series of 1/3-scale testing began in January 1975 and was directed at obtaining local impact pressures and total loads for typical small structures located over the pressure suppression pool including I-beams, pipes, and grating. Data from this test series expanded the data base from the full-scale air tests. A further series of 1/3-scale tests was added in June 1975 to obtain comparable data on pool swell velocity and breakthrough elevation to the full-scale air tests.

A series of small scale flow visualization tests were performed in October 1976 in order to qualitatively investigate the steam condensation phenomena for the Mark III vent configuration. The visual investigation of steam bubble formation and collapse under various bulk pool temperature and vent steam flux conditions provided information for the placing of instrumentation in the vicinity of the PSTF drywell vents for subsequent tests.

The final three phases of Mark III confirmatory test program began in November 1976 with a series of 1/3-scale tests under various initial suppression pool temperatures and simulated steam and liquid break sizes to obtain data on the localized conditions associated with the steam condensation portion of the LOCA blowdown. In parallel with this data acquisition, other test data was obtained for use in evaluating the loading conditions on submerged structures located in the suppression pool and for evaluating potential vertical thermal stratification of the suppression pool water. The second of the three phases was begun in September 1977. These full-scale tests also provided data on localized steam condensation conditions and thermal stratification.

Phase three consists of a 1/9-scale test series in which a nine vent array is utilized to evaluate multivent effects. In establishing the LOCA related conditions within the suppression pool, all of the vent stations are conservatively assumed to be in phase even though the random nature of the phenomena indicates that some phase separation is expected during the steam condensation process. This final test phase is primarily aimed at confirming that multiple vent loading conditions are not in excess of these identified from single cell tests.

It should be noted that the emphasis in some testing described above was directed at the evaluation of the pool swell phenomena, while in others the steam condensation phenomena was evaluated. Each test run consisted of a simulation of the postulated blowdown transient. Various postulated break sizes up to two times the design-basis accident for the containment were tested. Data was recorded at selected locations around the test facility suppression pool throughout the blowdown so that the hydrodynamic conditions associated with each phase of the blowdown is available for selecting appropriate design loading conditions. General Electric has utilized this data to develop thermal and hydrodynamic loading conditions in the GE Mark III reference plant pressure suppression containment system during the postulated LOCA. Information on thermal and hydrodynamic loading conditions during the anticipated safety relief valve (SRV) discharge and related dynamic events has also been documented. Separate test data has been utilized to establish the SRV air clearing load prediction model. Information on SRV discharge thermal performance is also provided. The GE reference plant report contains information and guidance to assist the containment designer in evaluating the design conditions for the various structures which form the containment system. Table 1.5-1 identifies all of the LOCA related tests conducted by GE which form the basis for hydrodynamic loads used. Table 1.5-2 identifies the documents referenced in Table 1.5-1 plus other reports containing test data used for non-LOCA related hydrodynamic load definitions.

1.5.1.5 Boiling Transition Testing

Since the formulation of the 1966 Hench-Levy Design Limit Lines for use in BWR thermal design, General Electric has continued to perform extensive steady-state and transient boiling transitive test programs. Prior to 1974, over 14,000 data points had been obtained in water and Freon from many test assemblies having various axial heat flux profiles and rod-to-rod power distributions, covering all prototypical aspects of reactor operating conditions. Among those, 2100 data points were full scale simulation of 7 x 7 and 8 x 8 BWR fuel assemblies performed in the ATLAS test facility. A new boiling transition correlation (GEXL) has been developed and applied to GE-BWR thermal design. Detailed information is provided in the approved Licensing Topical Report, NED0-10958-A, "General Electric BWR Thermal Analysis Basis (GETAB): Data, Correlation, and Design Application," January 1977.

Since the implementation of GEXL correlation on design in 1974, General Electric has continued to conduct full scale 8 x 8 assembly boiling transition tests, accumulating over 1600 data points after GETAB introduction, to extend the data base and to assure applicability to new 8 x 8 fuel designs such as the two water-rod design for BWR/2-5 and for BWR/6. It has been shown that the 8 x 8 GEXL correlation with the appropriate R-factors can predict boiling transition critical power data for the two water-rod assemblies with an accuracy typical of the GEXL correlation predictability for other 8 x 8 design as described in NED0-10958-A.

TABLE 1.5-1
SUMMARY OF PSTF TESTS

TEST SERIES	NUMBER OF BLOWDOWNS	VENTURI RANGE (inch)	TOP VENT G_L SUBMERGENCE RANGE (feet)	INITIAL PRESSURE (psia)	BLOWDOWN TYPE	NUMBER OF VENTS	AREA POOL/ VENT SCALING	PRIMARY OBJECTIVES*	REFERENCE REPORT**
5701	21	2-1/8 - 3-5/8	2.0-15.5	1050	Saturated Steam	1	Full	1. Vent Clearing 2. Full-Scale Condensation Demonstration 3. Drywell Pressure	4
5702	17	2-1/8 - 3-5/8	1.93-11.97	1050	Saturated Steam	2	Full	1. Vent Clearing	4
5703	3	2-1/2 - 3-5/8	6.77-11.05	1050	Saturated Steam	3	Full	1. Vent Clearing	4
5705	4	1 - 4-1/4	6.0-8.0	1065	Air	2	Full	1. Pool Swell Scoping	7
5706	7	4-1/4	5.0-10.0	1065	Air	2	Full	1. Pool Swell 2. Impact Loading	7
5707	22	2-1/8 - 3	7.5	1050	Air and Steam	3	Full	1. Chugging	16
5801	19	2-1/8 - 3	5.0-10.0	1050	Saturated Steam	3	1/3	1. 1/3 Scale Demonstration 2. Pool Swell 3. Roof Density and ΔP	11
5802	3	2-1/8 - 3	6.0	1050	Saturated Steam	3	1/3	1. Pool Swell	11
5803	2	2-1/8 - 3	5.0-7.5	1050	Saturated Liquid	3	1/3	1. 1/3 Scale Demonstration 2. Liquid Blowdown	11
5804	5	2-1/8 - 3	5.0	1050	Saturated Steam	3	1/3	1. Roof Density Density and ΔP Repeatability	11
5805	52	1 – 3	5.0-10.0	1050	Saturated Steam	3	1/3	1. Pool Swell Impact	12

TABLE 1.5-1 (Cont'd)

TEST SERIES	NUMBER OF BLOWDOWNS	VENTURI RANGE (inch)	TOP VENT G_L SUBMERGENCE RANGE (feet)	INITIAL PRESSURE (psia)	BLOWDOWN TYPE	NUMBER OF VENTS	AREA POOL/ VENT SCALING	PRIMARY OBJECTIVES*	REFERENCE REPORT**
5806	12	2-1/2 – 4-1/4	5.0-7.5	1065	Air	3	1/3	1. Pool Swell	13
5807	20	1 – 3	7.5	1050	Saturated Steam and Liquid	3	1/3	1. Steam Condensation	15
6002	13	2-1/8 – 3	5.0-10.0	1050	Saturated Steam	9	1/9	1. Multivent Effect on Pool Swell Loads	17
6003	12	2-1/3	7.5	1050	Saturated Steam	9	1/9	1. Multivent Effect on Condensation Loads	18

* In general tests are not direct prototype simulations, but parametric studies to be used in analytic model evaluations.

** See Table 1.5-2 for Reference Reports

TABLE 1.5-2
REFERENCES

1. W. J. Bilanin, "The General Electric Mark III Pressure Suppression Containment System Analytical Model," NED0-20533, June 1974 and Supplement 1, August 1975.

2. "Mark III Confirmatory Test Program Progress Report," Proprietary Report, NEDM-10848, April 1973.

3. "Mark III Analytical Investigation of Small-Scale Tests Progress Report," NED0-10976, August 1973.

4. "Mark III Confirmatory Test Program Phase 1 - Large Scale Demonstration Tests," Proprietary Report, NEDM-13377, October 1974.

5. "Third Quarterly Progress Report: Mark III Confirmatory Test Program," Proprietary Report, NED0-20210, December 1973.

6. "Fourth Quarterly Progress Report: Mark III Confirmatory Test Program," Supplement 1, Proprietary Report, NED0-20345, April 1974.

7. "Fifth Quarterly Progress Report: Mark III Confirmatory Test Program," Supplement 1, Proprietary Report, NED0-20550, July 1974.

8. "Sixth Quarterly Progress Report," Letter Transmittal to NRC Staff, Proprietary Data Attached, October 1974.

9. "Seventh Quarterly Progress Report: Mark III Confirmatory Test Program," Proprietary Report, NED0-20732-P, December 1974.

10. "Eighth Quarterly Progress Report: Mark III Confirmatory Test Program," Proprietary Report, NED0-20853-P, April 1975.

11. "Mark III Confirmatory Test Program 1/3 Scale Three Vent Tests," Proprietary Report, NED0-13407, April 1975.

12. "Mark III Confirmatory Test Program 1/3 Scale Pool Swell Impacts Tests - Test Series 5805," Proprietary Report, NEDE-13426-P, August 1975.

13. "Mark III Confirmatory Test Program 1/3 Scale Three Vent Air Tests - Test Series 5806," Proprietary Report, NEDE-13435-P, November 1975.

14. "Test Results Employed by GE for BWR Containment and Vertical Vent Loads," Proprietary Report, NEDE-21078-P, October 1975.

15. "Mark III Confirmatory Test Program - 1/3 Scale Condensation and Stratification Phenomena - Test Series 5807," Proprietary Report, NEDE-21596-P, March 1977.

16. "Mark III Confirmatory Test Program - Full Scale Condensation and Stratification Phenomena - Test Series 5707," Proprietary Report, NEDE-21853-P, August 1978.

17. "Mark III Confirmatory Test Program - 1/9 Area Scale Multivent Pool Swell Tests - Test Series 6002," Proprietary Report, NEDE-24648-P, September 1979.

18. "Mark III Confirmatory Test Program - 1/9 Area Scale Multicell Condensation and Stratification Phenomena - Test Series 6003," Proprietary Report, NEDE-24720-P, January 1980.

1.6 MATERIAL INCORPORATED BY REFERENCE

Table 1.6-1 is a list of GE topical reports and any other report or document which is incorporated in whole or in part by reference in this USAR and has been filed with the NRC.

Additional documents which are referenced in this USAR are listed at the end of the sections in which they have been referenced.

TABLE 1.6-1
REFERENCED REPORTS

GENERAL ELECTRIC COMPANY REPORTS

REPORT NUMBER	TITLE	REFERENCED IN USAR SECTION
APED-4827	Maximum Two-Phase Blowdown from Pipes (April 1965)	6.2
APED-4986	Consequences of Operating Zircaloy-2 Clad Fuel Rods Above the Critical Heat Flux (October 1965 BWR 6 only)	4.2
APED-5286	Design Basis for Critical Heat Flux Condition in BWR's (September 1966)	1.5
APED-5458	Effectiveness of Core Standby Cooling Systems for General Electric Boiling Water Reactors (March 1968)	5.4
APED-5460	Design and Performance of General Electric BWR Jet Pumps (July 1968)	3.9
APED-5555	Impact Testing on Collet Assembly for Control Rod Drive Mechanism 7RDB144A (November 1967)	4.6
APED-5640	Xenon Considerations in Design of Large Boiling Water Reactors (June 1968)	4.1 4.3
APED-5652	Stability and Dynamic Performance of the General Electric Boiling Water Reactor (April 1969)	4.1
APED-5706	In-Core Neutron Monitoring System for General Electric Boiling Water Reactors (November 1968, Revised April 1969)	7.6 7.7
APED-5736	Guidelines for Determining Safe Test Intervals and Repair Times for Engineered Safeguards (April 1969)	Appendix 15A
APED-5750	Design and Performance of General Electric Boiling Water Reactor Main Steamline Isolation Valves (March 1969)	5.4
APED-5756	Analytical Methods for Evaluating the Radiological Aspects of the General Electric Boiling Reactor (March 1969)	15.4, 15.7

TABLE 1.6-1 (Cont'd)

REPORT NUMBER	TITLE	REFERENCED IN USAR SECTION
GEAP-4616	Two-Phase Pressure Drop in Straight Pipes and Channels; Water-Steam Mixtures at 600 to 1400 psia (May 1964)	4.4
GEAP-10546	Theory Report for Creep-Plast Computer Program (January 1972)	4.1
GEAP-13112	Thermal Response and Cladding Performance of an Internally Pressurized Zircaloy-Clad, Simulated BWR Bundle Cooled by Spray Under Loss-of-Coolant Conditions (April 1971)	4.2
KAPL-2170	Hydrodynamic Stability of a Boiling Channel (October 1961)	4.4
KAPL-2208	Hydrodynamic Stability of a Boiling Channel, Part 2 (April 1962)	4.4
KAPL-2290	Hydrodynamic Stability of a Boiling Channel, Part 3 (June 1963)	4.4
KAPL-3070	Hydrodynamic Stability of a Boiling Channel, Part 4 (August 1964)	4.4
KAPL-3072	Reactivity Stability of a Boiling Reactor, Part 1 (September 1964)	4.4
KAPL-3093	Reactivity Stability of a Boiling Reactor, Part 2 (March 1965)	4.4
NEDC-33666P	GE Hitachi Nuclear Energy, “GNF2 Fuel Design Cycle-Independent Analyses for Exelon Generation Company LLC Clinton Power Station, “NEDC-33666P, Revision 3, April 2015.	15 App. F
NEDC-20944	Peachbottom Atomic Power Station Units 2 and 3, Safety Analysis Report for Plant Modifications to Eliminate Significant In-Core Vibration (December 1975)	4.4
NEDC-33505P	"Safety Analysis Report to Support Introduction of GE 14: Isotope Test Assemblies (ITA’s) in Clinton Power Station”, May 2009 (IP-F-0159)	4.2

TABLE 1.6-1 (Cont'd)

REPORT NUMBER	TITLE	REFERENCED IN USAR SECTION
NEDC-32989	Safety Analysis Report for Clinton Power Station Extended Power Uprate, Class III (GE Proprietary Information), June 2001.	11.3
NEDE-10313	PDA-Pipe Dynamic Analysis Program for Pipe Rupture Movement (Proprietary (Filing)	3.6, 3.9
NEDE-11146	Design Basis for New Gas System (July 1971)(Company Proprietary)	11.3
NEDE-20386	Fuel Channel Deflections	4.2
NEDE-20943	Urania-Gadolinia Nuclear Fuel Physical and Irradiation Characteristics and Material Properties (January 1977)	4.2
NEDE-20944-P	BWR/4 and BWR/5 Fuel Design (October 1976) (only BWR/4&5)	4.2, 4.3, 4.4 and 4.6
NEDE-20944-1P	BWR/4 and BWR/5 Fuel Design (Amendment 1) (January 1977) (only BWR/4&5)	4.2, 4.3, 4.4 and 4.6
NEDE-21156	Supplemental Information for Plant Modification to Eliminate Significant In-Core Vibration (January 1976)	4.4
NEDE-21175-P	BWR/6 Fuel Assembly Evaluation of Combined Safe Shutdown Earthquake (SSE) and Loss-of-Coolant Accident (LOCA) Loadings (November 1976)	3.9
NEDE-21354-P	BWR Fuel Channel Mechanical Design and Deflection (September 1976)	3.9
NEDE-21821-A	Boiling Water Reactor Feedwater Nozzle/Sparger Final Report", (proprietary version), February 1980.	3.9 and 5.3
NEDE-21056-1-P	"N66 SJAE Off-Gas Treatment System - Amendment No. 1," (Supplements Licensing Topical Report), (Proprietary), August 1978.	11.3
NEDE-23014	Hex 01 User's Manual (July 1976)	15.2

TABLE 1.6-1 (Cont'd)

REPORT NUMBER	TITLE	REFERENCED IN USAR SECTION
NEDE-23542-P and NEDO-23542	Fuel Assembly Evaluation of Shipping and Handling Loadings (March 1977) (Proprietary and Non-proprietary Versions)	4.2
NEDE-33284	Supplement 1P–A, Revision 1, March 2012, Marathon-Ultra Control Rod Assembly Licensing Topical Report	4.2
NEDE-24154-P and NEDO-24154	Safety Evaluation for General Electric Topical Report: Qualification of the One-Dimensional Core Transient Model for Boiling Water Reactors	15.0 15.1 Appendix 15B Appendix 15D
NEDM-10735	Densification Considerations in BWR Fuel Design and Performance (December 1972)	4.2
NEDO-10173	Current State of Knowledge, High Performance BWR Zircaloy-Clad UO2 Fuel (May 1973)	4.2 11.1
NEDO-10174	Consequences of a Postulated Fuel Blockage Incident in a Boiling Water Reactor (May 1970)	4.2
NEDO-10722A	Core Flow Distribution in a Modern Boiling Water Reactor as Measured in Monticello (August 1976)	4.4
NEDO-10320	The General Electric Pressure Suppression Containment Analytical Model (April 1971) Supplement 1 (May 1971)	6.2
NEDO-10329	Loss-of-Coolant Accident and Emergency Core Cooling Models for General Electric Boiling Water Reactors (April 1971) Supplement 1 (April 1971) Addenda (May 1971)	4.3
NEDO-10349	Analysis of Anticipated Transients Without Scram (March 1971)	15.8
NEDO-10466A	Power Generation Control Complex Design Criteria and Safety Evaluation (February 1979)	6.4, 9.5

TABLE 1.6-1 (Cont'd)

REPORT NUMBER	TITLE	REFERENCED IN USAR SECTION
NEDO-10505	Experience with BWR Fuel Through September 1971 (May 1972)	4.2, 11.1
NEDO-10527	Rod Drop Accident Analysis for Large Boiling Water Reactors (March 1972) Supplement 1 (July 1972) Supplement 2 (January 1973)	4.3, 15.4
NEDO-10585	Behavior of Iodine in Reactor Water During Plant Shutdown and Startup (August 1972)	15.6
NEDO-10602	Testing of Improved Jet Pumps for the BWR/6 Nuclear System (June 1972)	3.9
NEDO-10734	A General Justification for Classification of Effluent Treatment System Equipment as Group D (February 1973)	11.3
NEDO-10739	Methods for Calculating Safe Test Intervals And Allowable Repair Times for Engineered Safeguard Systems (January 1973)	6.3, 15A
NEDO-10751	Experimental and Operational Confirmation of Off-Gas System Design Parameters (January 1973) (Company Proprietary)	11.3
NEDO-10801	Modeling the BWR/6 Loss-of-Coolant Accident: Core Spray and Bottom Flooding Heat Transfer Effectiveness (March 1973)	1.5
NEDO-10802	Analytical Methods of Plant Transient Evaluations for General Electric Boiling Water Reactor (February 1973)	4.4, 5.2, 15.1
NEDO-10846	BWR Core Spray Distribution (April 1973)	1.5
NEDO-10899	Chloride Control in BWR Coolants (June 1973)	5.2
NEDO-10905	High Pressure Core Spray System Power Supply Unit (May 1973)	8.1
NEDO-10958	General Electric BWR Thermal Analysis Basis (GETAB): Data, Correlation, and Design Application (November 1973)	4.3, 4.2, 15.0

TABLE 1.6-1 (Cont'd)

REPORT NUMBER	TITLE	REFERENCED IN USAR SECTION
NEDO-10958-A	General Electric BWR Thermal Analysis Basis (GETAB): Data, Correlation, and Design Application (January 1977)	1.5, 4.4, 15.4,
NEDO-20231	Emergency Core Cooling Tests of an Internally Pressurized, Zircaloy-Clad, 8 x 8 Simulated BWR Fuel Bundle (December 1973)	1.5
NEDO-20340	Process Computer Performance Evaluation Accuracy (June 1974)	4.3
NEDO-20360	General Electric Boiling Water Reactor Generic Reload Application for 8 x 8 Fuel (May 1975)	4.2, 15.4
NEDO-20360-IP	General Electric Boiling Water Reactor Generic Reload Application for 8 x 8 Fuel (March 1976)	4.2
NEDO-20377	8 x 8 Fuel Bundle Development Support (February 1975)	4.2
NEDO-20533	The General Electric Mark III Pressure Suppression Containment System Analytical Model (June 1974)	1.5
NEDO-20566 and NEDE-20566-P	General Electric Company Model for Loss-of-Coolant Accident Analysis in Accordance with 10 CFR 50, Appendix K (January 1976)	3.9
NEDO-20605 and NEDE-20606	Creep Collapse Analysis of BWR Fuel Using Safe Collapse Model (August 1974); (Nonproprietary and Proprietary Versions)	4.2
NEDO-20626	Studies of BWR Designs for Mitigation of Anticipated Transients Without Scrams (October 1974)	15.8
NEDO-20626-1	Studies of BWR Designs for Mitigation of Anticipated Transients Without Scrams (June 1975)	15.8
NEDO-20626-2	Studies of BWR Designs for Mitigation of Anticipated Transients Without Scrams (July 1975)	15.8

TABLE 1.6-1 (Cont'd)

REPORT NUMBER	TITLE	REFERENCED IN USAR SECTION
NEDO-20631	Mechanical Property Surveillance of Reactor Pressure Vessels for General Electric BWR/6 Plants (March 1975)	5.3
NEDO-20913	Lattice Physics Methods (June 1975)	4.3
NEDO-20922	Experience With BWR Fuel Through September 1974 (June 1975)	4.2, 11.1
NEDO-20939	Lattice Physics Methods Verification (August 1975)	4.3
NEDO-20943	Urania-Gadolinia Nuclear Fuel Physical and Material Properties (January 1977)	4.2
NEDO-20944	BWR/4 and BWR/5 Fuel Design (October 1976) (Nonproprietary Versions)	4.1, 4.3
NEDE-20944	BWR/4 and BWR/5 Fuel Design (October 1976) (Proprietary Version)	4.1, 1.3
NEDO-20946	BWR Simulator Methods Verification (May 1976)	4.3
NEDO-20948-P	BWR/6 Fuel Design (June 1976)	4.2
NEDO-20953	Three-Dimensional Boiling Water Reactor Core Simulator (May 1976)	15.4
NEDO-20964	Generation of Void and Doppler Reactivity Feedback for Application to BWR Plant Transient Analysis (August 1975)	4.3
NEDO-21142	Realistic Accident Analysis for General Electric Boiling Water Reactor - The RELAC Code and User's Guide, (December 1977)	15.6, 15.7
NEDO-21159	Airborne Release from BWR's for Environment Impact Evaluations (March 1976)	11.1
NEDO-21231	Banked Position Withdrawal Sequence (September 1976)	4.3
NEDO-21291	Group Notch Mode of the RSCS for Cooper (June 1976)	15.4

TABLE 1.6-1 (CONT'D)

REPORT NUMBER	TITLE	REFERENCED IN USAR SECTION
NEDO-21506	Stability and Dynamic Performance of the General Electric Boiling Water Reactor (January 1977)	4.4
NEDO-26453	3D BWR Core Simulator (May 1976) Oyster Creek Station, FSAR Amendment 10	4.3, 1.5
NEDO-21660	Experience with BWR Fuel Through December 1976 (July 1977)	4.2
NEDE-24011-P-A	General Electric Standard Application for Reactor Fuel (latest approved revision)	4.2, 4.3, 4.4, Appendix 15D
NEDO-25257	E. W. Bradley, V. D. Nguyen, "Radiation Exposure From Airborne Effluents – The REFAE Code," NEDO-25257, July 1980.	11.3
NEDO-25132A	E. W. Bradley, "Gamma And Beta Dose To Man From Noble Gas Release To The Atmosphere GEMAN Program," April 1980.	11.3
NEDE-30130-P	Steady State Nuclear Methods (April 1985)	Appendix 15D
<u>OTHER REFERENCED REPORTS</u>		
AE-RTL-788	Void Measurements in the Region of Sub-cooled and Low Quality Boiling (April 1966)	4.4
ANL-5621	Boiling Density in Vertical Rectangular Multichannel Sections with Natural Circulation (November 1956)	4.4
ANL-5552	The Effect of Pressure on Boiling Density in Multiple Rectangular Channel (February 1956)	4.4
ANL-6385	Power-to-Void Transfer Functions (July 1961)	4.4
BHR/DER 70-1	Radiological Surveillance Studies at a Boiling Water Nuclear Power Reactor (March 1970)	11.1
BMI-1163	Vapor Formation and Behavior in Boiling Heat Transfer (February 1957)	4.4
CF 59-6-47 (ORNL)	Removal of Fission Product Gases from Reactor Off-Gas Streams by Adsorption	11.3

TABLE 1.6-1 (CONT'D)

REPORT NUMBER	TITLE	REFERENCED IN USAR SECTION
IDO-ITR-105	The Response of Waterlogged UO2 Fuel Rods to Power Bursts (April 1969)	4.2
IN-ITR-111	The Effects of Cladding Material and Heat Treatment on the Response of Water-logged UO2 Fuel Rods to Power Bursts (January 1970)	4.2
RE-S-76-170	Light Water Reactor Fuel Behavior Program Description; RIA Fuel Behavior Experiment Requirements (September 1976)	4.2
STL-372-38	Kinetic Studies of Heterogeneous Water Reactors (April 1966)	4.4
TID-4500	Relap 3 - A Computer Program for Reactor Blowdown Analysis IN-1321 (June 1970)	3.6
TID-7672	ANS Topical Meeting, Nuclear Performance of Power Reactors (September 1976)	4.3
UCRL-50451	Improving Availability and Readiness of Field Equipment Through Periodic Inspection, P. 10 (July 16, 1968)	16.3
WACP-6065	Melting Point of Irradiated Uranium Dioxide (February 1965)	4.2
WAPD-BT-19	A Method of Predicting Steady-State Boiling Vapor Fractions in Reactor Coolant Channels (June 1960)	4.4
WAPD-TM-283	Effects of High Burnup on Zircaloy-Clad Bulk UO2 Plate Fuel Element Samples (September 1962)	4.2
WAPD-TM-416	WIGLE - A Program for the Solution of the Two-Group Space-Time Diffusion in Slab Geometry (1964)	4.3
WAPD-TM-629	Irradiation Behavior of Zircaloy-Clad Fuel Rods Containing Dished End UO2 Pellets (July 1967)	4.2

1.7 DRAWINGS AND OTHER DETAILED INFORMATION

1.7.1 Electrical, Instrumentation, and Control Drawings

Electrical, instrumentation, and control drawings were provided to the NRC during initial licensing activities. These drawings were considered necessary to evaluate the safety-related features in Chapters 7 and 8. These drawings are not required to be included in the USAR per Generic Letter 81-06, Question/Response C.1.

1.7.2 Piping and Instrumentation Diagrams (P&IDs), Mechanical, Physical, Electrical, and Other Drawings Used in the USAR

The plant drawings cited in each Chapter of Appendix of the USAR are listed in the Table of Contents for that Chapter or Appendix.

1.7.3 Other Detailed Information

No specific request for data has been received from the NRC; therefore, no information is supplied for this subsection.

1.7.4 Process Diagrams And Other Figures

Current design basis documents and drawings should be referred to when reviewing and evaluating the design..

1.8 CONFORMANCE TO NRC REGULATORY GUIDES

The purpose of this section is to indicate the conformance of the Clinton Power Station design with Regulatory Guides issued by the NRC.

On the following pages each Regulatory Guide is identified by number, revision and title. The project position toward the Regulatory Guide is also indicated, accompanied by the appropriate USAR reference section to which the Regulatory Guide applies. If the project position indicates compliance, no further discussion is provided. Exceptions taken to the Regulatory Guide are identified and discussed in this section. Clarifications to compliance positions are also provided in this section when necessary.

Conformance to IEEE Standards - Also included in this section as Table 1.8-1 is a list of IEEE Standards which were considered for the design, construction and operation of the Clinton Power Station. These specific revisions of the Standards apply to the CPS-USAR except as otherwise noted in the USAR text.

Table 1.8-1

CONFORMANCE TO IEEE STANDARDS

IEEE Standard 85 (1973)	Test Procedure for Airborne Sound Measurement on Rotating Electrical Machinery
IEEE Standard 112A (1964)	Test Procedure for Polyphase Induction Motors and Generators
IEEE Standard 275 (1981)	Recommended Practice for Thermal Evaluation of Insulation Systems for AC Electric Machinery Employing Form-Wound Pre-Insulated Stator Coils Machines Rated 6900 V and Below
IEEE Standard 279 (1971)	Criteria for Protection Systems for Nuclear Power Generating Stations
IEEE Standard 308 (1974)	Criteria for Class 1E Power Systems for Nuclear Power Generating Stations
IEEE Standard 317 (1976)	Electrical Penetration Assemblies in Containment Structures for Nuclear Power Generating Stations
IEEE Standard 323 (1974)	Qualifying Class 1E Equipment for Nuclear Power Generating Stations
IEEE Standard 334 (1974)	Standard for Type Test of Continuous Duty Class 1E Motors for Nuclear Power Generating Stations
IEEE Standard 336 (1971)	Installation, Inspection, and Testing Requirements for Instrumentation and Electrical Equipment During the Construction of Nuclear Power Generating Stations
IEEE Standard 338 (1977)	Standard Criteria for the Periodic Testing of Nuclear Power Generating Station Class 1E Power and Protection Systems
IEEE Standard 344 (1975)	Recommended Practices for Seismic Qualification of Class 1E Equipment for Nuclear Power Generating Stations
IEEE Standard 379 (1972)	Trial-Use Guide for the Application of the Single-Failure Criterion to Nuclear Power Generating Station Protection Systems
IEEE Standard 381 (1977)	Standard Criteria for Type-Test of Class 1E Modules Used in Nuclear Power Generating Stations
IEEE Standard 382 (1972)	Trial-Use Guide for the Type-Test of Class 1E Electrical Valve Operators for Nuclear Power Generating Stations

Table 1.8-1 (Cont'd)

Conformance to IEEE Standards

IEEE Standard 383 (1974)	Standard for Type Test of Class 1E Electrical Cables, Field Splices, and Connections for Nuclear Power Generating Stations
IEEE Standard 384 (1974)	Trial-Use Standard Criteria for Separation of Class 1E Equipment and Circuits
IEEE Standard 387 (1977)∗	Trial-Use Criteria for Diesel-Generator Units Applied as Standby Power Supplies for Nuclear Power Generating Stations
IEEE Standard 415 (1976)	Planning of Pre-Operational Testing Programs for Class 1E Power Systems for Nuclear Power Generating Stations
IEEE Standard 450 (1995)	IEEE Recommended Practice for Maintenance, Testing, and Replacement of Vented Lead-Acid Batteries for Station Applications
IEEE Standard 484 (1975)	Recommended Practice for Installation Design and Installation of Large Lead Storage Batteries for Generating Stations and Substations
IEEE Standard 622 (1979)	IEEE Recommended Practice for the Design and Installation of Electric Pipe Heating Systems for Nuclear Power Generating Stations
IEEE Standard 627 (1980)	IEEE Standard for Design Qualification of Safety Systems Equipment Used in Trial-Use Criteria for Nuclear Power Generating Stations
IEEE Standard 634 (1978)	IEEE Standard Cable Penetration Fire Stop Qualification Test

∗ - (see also NEDO-10905)

<u>Regulatory Guide 1.1, Rev. 0 (December 1970)</u>

Net Positive Suction Head For Emergency Core Cooling and
Containment Heat Removal System Pumps

<u>Project Position</u> - Comply

<u>USAR Subsection</u> - 6.3.2.2

<u>Regulatory Guide 1.2, Rev. 0 (December 1970)</u>

Thermal Shock to Reactor Pressure Vessels

<u>Project Position</u> - Comply.

<u>USAR Subsection</u> - 5.3.3

<u>Regulatory Guide 1.3, Rev. 2 (June 1974)</u>

Assumptions Used for Evaluating the Potential
Radiological Consequences of a Loss of Coolant
Accident for Boiling Water Reactors

<u>Project Position</u> – Partially Comply; for Alternative Source Terms, Regulatory Guide 1.183 is also utilized.

<u>USAR Subsections</u> - 6.4.2, 7.6.2.12.5, 9.3.7, 15.6, E3.8.1.2

<u>Regulatory Guide 1.4, Rev. 2 (June 1974)</u>

Assumptions Used for Evaluating the Potential
Radiological Consequences of a Loss-of-Coolant Accident
for Pressurized Water Reactors

<u>Project Position</u> - Not applicable to BWRs.

Regulatory Guide 1.5, Rev. 0 (March 1971)

Assumptions Used for Evaluating the Potential Radiological Consequences of a Steam Line Break Accident for Boiling Water Reactors

Project Position – Partially Comply; for Alternative Source Terms, Regulatory Guide 1.183 is also utilized.

USAR Subsection - 15.6.4

Regulatory Guide 1.6, Rev. 0 (March 1971)

Independence Between Redundant Standby (Onsite) Power Sources and Between Their Distribution Systems

Project Position - Comply

USAR Subsections - 7.3.2.1.2.1.1, 7.3.2.3.2.1.1, 7.3.2.20.2.1.1, 8.3.1.2.2, 8.3.2.2.2

Regulatory Guide 1.7, Rev. 2 (November 1978)

Control of Combustible Gas Concentrations in Containment Following a Loss-of-Coolant Accident

Project Position - The project complies with the requirements of Regulatory Guide 1.7 with the following clarification:

> The discussion contained in Regulatory Guide 1.7 states that the NRC has concluded that a lower flammability limit of 4 volume percent hydrogen in air or steam-air mixtures is well established and is adequately conservative. Regulatory Position C.6 states that materials which would yield hydrogen gas due to corrosion from ECCS or containment spray should be identified and their use should be limited as much as practical.
>
> In USAR Section 6.2.5 regarding the Combustible Gas Control System, hydrogen production due to corrosion following a LOCA has been evaluated based upon the analyses performed considering the amount of hydrogen producing materials allowed in containment by design. The specific quantities of hydrogen producing materials used in evaluating CPS conformance to Regulatory Guide 1.7 are listed in Table 6.2-49, and the resulting hydrogen concentrations as a function of time are exhibited in Figures 6.2-130a and 6.2-130b.
>
> CPS procedures and design baseline documentation ensure the limitation of postulated post-LOCA hydrogen concentrations inside containment and drywell in accordance with Regulatory Guide 1.7.

USAR Subsections - 6.2.5, 9.4.7.2

Regulatory Guide 1.8, Proposed Rev. 2 (February 1979)

Personnel Selection and Training

Project Position - The project complies with the requirements of Regulatory Guide 1.8 with the following clarifications and exceptions:

1. Reference: Paragraph C.3.a, C.3.b, and C.3.c - Exception is taken to the requirement of minimum qualifications for individuals that direct or supervise the conduct of individual preoperational tests and startup tests, and individuals who review and approve preoperational and startup test procedures or results. Instead, the following program will be used to qualify individuals:

   In order to ensure that the various Startup Group activities are performed by qualified personnel, three levels of qualifications are established:

   a. Level I

   b. Level II

   c. Level III

   The qualification levels required to perform specific startup organization activities is in accordance with the following table:

NOTE

1. The checkout and initial operation (C&IO) phase is a period during which checkout and testing is completed which is prerequisite to subsequent preoperational or acceptance tests.

2. The preoperational phase is a period during which preoperational test procedures (PTP's) and acceptance test procedures (ATP's) are performed. These tests, in general, are conducted on an integrated system or subsystem basis to verify that systems are capable of operating in a safe and efficient manner compatible with system design bases. By definition, "preoperational tests" are performed on nuclear safety-related systems.

3. The startup phase is that period beginning with preparations for fuel loading and extending through warranty tests.Startup test procedures (STP's) are conducted to verify the performance of equipment under actual operating conditions.

<u>Regulatory Guide 1.8, Proposed Rev. 2 (February 1979)</u> (Cont'd)

	Minimum Qualification Level		
	I	II	III
Approve, checkout and initial operation test procedures (C&IO)			X
Approve acceptance test procedures (ATP's)			X
Approve preoperational test procedures (PTP's)			X
Approve startup test procedures (STP's)			X
*Direct or supervise conduct of C&IO tests	X		
*Direct or supervise conduct of ATP's	X		
*Direct or supervise conduct of PTP's		X	
*Direct or supervise conduct of STP's		X	
Evaluate test results of C&IO tests prerequisite to ATP	X		
Evaluate test results of ATP's		X	
Evaluate test results of PTP's		X	
Evaluate test results of STP's			X
Supervision of test program			X
Certification of personnel			X
Evaluate test results of C&IO test prerequisite to PTP's		X	

The process by which individuals are evaluated and certified to their appropriate qualification level, is as follows:

a. The person being certified completes a reading list consisting of quality and work related documents as specified in the CPS Startup Manual.

b. A resume or equivalent background information concerning education and experience is gathered.

c. Prior to certification, the candidate performs work under the direction of certified personnel. The candidate's work qualities are documented and evaluated by the Level III.

d. A Level III will interview the candidate, discussing selected items from the reading list, past work experience applicabibilities to the present job, and considers the candidate's performance to item c above. The specified Level III will consider the individual for certification to one of three qualification levels.

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* When an inspection or test requires implementation by a team or group, personnel not meeting the minimum qualification may be used for data taking assignments or equipment operation provided they are supervised by a qualified individual.

Regulatory Guide 1.8, Proposed Rev. 2 (February 1979) (Cont'd)

Delineated below are the requirements used when assigning a level of capability to an individual. The specified requirements are not to be treated as absolute. Other factors such as past performance or proficiency testing may be used to provide assurance that a person can competently perform an assigned task. When these requirements are waived, the basis for the waiver will be documented.

Level I

High school graduate, plus one year of commensurate experience in construction, preoperational, startup or operational testing activities.

Level II

(a) Graduate of four year accredited engineering or science college or university, plus two years of commensurate experience in construction, preoperational, startup, and/or operational testing activities. At least one year should be associated with nuclear facilities or if not, the individual should have training sufficient to acquaint him thoroughly with the safety aspects of a nuclear power plant.

(b) High school graduate, plus four years of commensurate experience in construction, preoperational, startup, and/or operational testing activities in fossil or nuclear power plants, heavy industrial, or other similar equipment or facilities. At least one year should be associated with nuclear facilities or if not, the individual should have training sufficient to acquaint him thoroughly with the safety aspects of a nuclear power plant.

Level III

(a) Graduate of a four year accredited engineering or science college or university plus five years of experience in construction, startup, and/or operational testing activities. At least two years of this experience should be associated with preoperational and/or startup testing in nuclear facilities; or if not, the individual should have training sufficient to acquaint him thoroughly with the safety aspects of a nuclear power plant.

(b) High school graduate, plus ten years of experience in testing, maintenance, or operational activities in nuclear or fossil power plants, heavy industrial, or other similar equipment or facilities. Five years of this experience should be associated with construction, preoperational, startup or operational testing and at least two of the five should be in nuclear facilities; or if not, the individual should have training sufficient to acquaint him thoroughly with the safety aspects of a nuclear power plant.

NOTE

The word commensurate as used in the description of Level I and II is defined as:

> The knowledge and skills acquired through past experience corresponds in the same relative proportions to the knowledge and skill requirements of the tasks to

<u>Regulatory Guide 1.8, Proposed Rev. 2 (February 1979)</u> (Cont'd)

be assigned to that test engineer. In other words, a person who had two years experience doing only electrical testing activities would not be assigned tasks of a mechanical or fluid nature. However, an engineer with two years experience would have gathered sufficient knowledge and skills to be assigned to an HVAC system startup. HVAC balancing would require additional experience.

2. Reference: Paragraph C.3.c - Clarification is provided concerning the qualifications of reviewers of preoperational and startup test procedures. The qualifications of reviewers and approvers is differentiated. The A/E or a consultant may be utilized to review portions of procedures. Furthermore, CPS staff personnel, who do not possess overall qualifications, may be utilized to review specific procedures if they are qualified in the particular area covered by the procedure.

3. Reference: Paragraph C.8 - Exception is taken to the requirement that non-licensed operators must have one year of power plant experience, 6 months of which must be at the assigned facility. This requirement would require the facility to carry at least one additional employee in an "in training" status since the auxiliary operator position is intended to be an entry level job. It is also believed that flexibility is lost when job vacancies are created.

   It is believed that there are numerous power plant duties having no safety significance that can be performed by an operator in an entry level position. This individual can receive on-the-job and formal training while contributing to plant productivity.

4. Reference: ANSI/ANS-3.1-1978, Paragraph 4.6.1 - The engineer in charge of technical support may not possess all specified qualifications. However, adequately qualified personnel will be available in the company engineering staff or through consultants to assist the engineer in charge.

5. Reference: ANSI/ANS-3.1-1978, Paragraph 4.7 - Independent review is provided by the Nuclear Safety Review Board (NSRB). The NSRB is a standing committee. Since the NSRB is a standing committee, paragraph 4.7.1 is not applicable to its members.

6. Reference: ANSI/ANS-3.1-1978, Paragraph 4.7-2 - Exception is taken to the qualification requirements for the staff specialists for the review committee. The review committee is required to contain personnel with expertise in all of the appropriate areas. When such expertise is not available on the review committee unit, the expertise will be available in the company or from outside consultants.

7. Reference: ANSI/ANS-3.1-1978, Section 5.2 and Section 5.5 - Exception is taken to Section 5.2 "Training of Personnel to be Licensed by the NRC" and Section 5.5 "Operator Retraining and Replacement Training." The Clinton Power Station Licensed Operator Initial and Continuing Training Programs are accredited by the Institute of Nuclear Power Operations (INPO) National Academy for Nuclear Training. These programs are based on a systems approach to training (SAT) and are accredited by the National Nuclear

Regulatory Guide 1.8, Proposed Rev. 2 (February 1979) (Cont'd)

Accrediting Board. These programs shall meet the requirements of 10CFR55, "Operators' Licenses."

8. Reference: ANSI/ANS-3.1-1978, Section 4, "Qualifications," Paragraph 4.2.2, "Operations Manager' - Exception is taken to the requirement "the Operations Manager shall hold a Senior Reactor Operator's License." The CPS Technical Specifications require "The Operations Manager or at least one Operations Middle Manager shall hold an SRO license for Clinton Power Station." This is consistent with ANSI/ANS-3.1-1978.

USAR Subsection: 12.5.1, 13.1.3.1, 13.2.1, 13.2.1.1, 14.2.1.5, 14.2.2.5, TS 5.3.1, and ORM 6.4.1.

<u>Regulatory Guide 1.9, Rev. 2 (December 1979), and Rev. 3</u>

Selection, Design and Qualification of Diesel-Generator Units Used as Onsite Electric Power Systems at Nuclear Power Plants

<u>Project Position</u> - The project complies with NRC Regulatory Guide 1.9, Rev. 2 (December 1979), and with selected portions of Rev. 3, as described below:

(1) Reference: Paragraphs C.1 and C.2 - Based upon updated diesel generator loading calculations, the loading on the diesel generators could be higher than the continuous rating of the diesel generator. Paragraph C.1 states that at the construction permit stage of design, the sum of the estimated loads needed at any one time is to be less than the continuous rating of the diesel generator. Paragraph C.2 also specifies that during the operating license stage of review, when a more accurate estimate of the safety loads is possible, a somewhat less conservative approach is permitted. Based on this, CPS uses the 2000-hour rating for illustrating the ability of the diesel to handle the maximum coincident loading expected following an accident.

(2) Reference: Paragraph C.7 of Rev. 2 (December 1979)- Isolation valves are not included on instrument sensing lines, so that calibration of instrument sensors cannot be accomplished without disconnecting the sensor from the sensed variable.

(3) Reference: Section C, Paragraphs 2.2.2, 2.2.8, 2.2.9, and 2.2.10 of Reg. Guide 1.9, Rev. 3, establish the diesel-generator acceptance criteria and loading requirements for the load-run test, the full-load rejection test, the endurance and margin test, and the hot restart test. (The above referenced paragraphs of Reg. Guide 1.9, Rev. 3, were approved for use at CPS via Operating License Amendment 118.)

(4) Reference: The General Electric HPCS system power supply unit Licensing Topical Report, NEDO 10905, gives the starting and accelerating charactersitics of the diesel-generator set with the various loads in the proper sequence. Although the voltage and frequency characteristics do not meet NRC Regulatory Guide 1.9, justification for this is given because of the unique requirements of the system. The HPCS diesel generator is unique in that its load is composed predominantly of one large motor whose horse power is approximately the same as the diesel-engine.

<u>USAR Subsection</u> - 8.3.1.1.2, 8.3.1.1.2.1, 8.3.1.2.2

Regulatory Guide 1.10, Rev. 1 (January 1973)

Mechanical (Cadweld) Splices in Reinforcing Bars of Category I Concrete Structures

Project Position - The Project complies with NRC Regulatory Guide 1.10 with the following exceptions:

(1) Reference: Paragraph C.1 - Each operator prepared two qualification splices for each of the positions (e.g., vertical, horizontal, diagonal) using the largest bar size for that position.

(2) Reference: Paragraph C.1 - Operator requalification was necessary if (1) the specified splice position has not been used for a period of three months or more, (2) completed splices consistently failed to pass visual inspection or tensile test requirements, or (3) reason existed to question operator's ability.

(3) Reference: Paragraph C.3.a - Testing complied with ASTM A370-75.

(4) Reference: Paragraph C.3 - Rebar detail drawings are prepared showing the location of all reinforcing bar lapped splices or cadwelds. If during construction the location of the splices or cadwelds differs from the location shown on the detail drawing by more than the specification tolerances, a Field Change Request (FCR) or Non-Conformance Report (NCR) is issued by the Constructor. When approval of the FCR or NCR is obtained, the number of the FCR or NCR is posted against the drawing and becomes part of the permanent plant records.

(5) Reference: Paragraph C.5.a - Procedure for Substandard Tensile Test Results were as follows:

a. If any production or sister splice used for testing fails to meet the strength requirements (125% of minimum yield strength specified in ASTM A615-75) and failure occurs in the bar, the failure shall be reported to the Consulting Engineers.

b. If any production splice used for testing fails to meet the strength requirements (125% of minimum yield strength specified in ASTM A615-75) and failure did not occur in the bar, the adjacent production splices on each side of the failed splice shall be tested. If any sister splice used for testing fails to meet the strength requirements and failure did not occur in the bar, two additional sister splices shall be tested. If either of these retests fail to meet the strength requirements, splicing shall be halted. Splicing shall not be resumed until the cause of failures has been corrected and resolved by the Contractor to the satisfaction of the Consulting Engineer.

USAR Reference - Appendix B

Regulatory Guide 1.11, Rev. 0 (March, 1971)

Instrument Lines Penetrating Primary Reactor Containment

Project Position - Comply with the following exception:

Reference: Paragraph C.1.c(2) - Self-actuating Excess Flow Check Valves (EFCV's) that are installed in low pressure instrument sensing lines (i.e., lines that sense, drywell pressure, containment pressure, suppression pool level, and ventilation system pressure) are not designed to close if the instrument line integrity outside containment is lost during normal reactor operation. During normal reactor operation, there exists a small difference in atmospheric pressure between the drywell or containment buildings and the secondary containment, the building where all EFCV's are located. If an instrument line outside containment ruptures during normal reactor operation, there may be insufficient differential pressure to actuate the EFCV. However, since there is negligible radiological source term available for release from inside the drywell or containment during normal reactor operation, the safety consequence of a low pressure instrument sensing line failure is considered to be insignificant. The offsite radiological exposure from a single failure of an EFCV will remain substantially below the guidelines of 10CFR100 and the integrity and functional performance of the secondary containment and its associated standby gas treatment system (SGTS) will be maintained.

USAR Subsections - 6.2.4, 6.2.6.3, Table 6.2-47, 7.1.2.6.3, 7.3.2.2.2.1.1, 7.4.2.1.2.1.2

Regulatory Guide 1.12, Rev. 1 (April 1974)

Instrumentation for Earthquakes

Project Position - Comply with the following exception:

Reference: Paragraph C.4.b - Frequency range required for mechanical response spectrum recorder is minimum of 1 to 30 Hz. One of the recorders (passive) utilized at CPS is a 2-25.4 Hz recorder.

USAR Subsection - 3.7.4

Regulatory Guide 1.13, Rev. 1 (December 1975)

Spent Fuel Storage Facility Design Basis

Project Position - Comply

USAR Section/Subsections - 3.1.2, 9.1.2, 9.1.3

Regulatory Guide 1.14, Rev. 1 (August 1975)

Reactor Coolant Pump Flywheel Integrity

Project Position - Not applicable to BWRs.

Regulatory Guide 1.15, Rev. 1 (December, 1972)

Testing of Reinforcing Bars for Category I Structures

Project Position - Comply.

USAR Section - Appendix B

Regulatory Guide 1.17, Rev. 1 (June 1973)

Protection of Nuclear Power Plants Against Industrial Sabotage

Project Position - Clinton Power Station complies with Regulatory Guide 1.17 with the following clarifications:

1) Reference: Paragraph C.1.b of the Regulatory Guide and GSA Interim Federal Specifications W-A-00450 B (GSA-FSS) dated February 6, 1973 - See Section 5.2.2.1 of the Physical Security Plan for commitments in alarm systems.

2) Reference: ANSI-N18.17-1973 - The Clinton Power Station Physical Security Plan sets forth the principles, policies, and general requirements for security at CPS. Because of its sensitive contents, any deviations are elaborated in the plan and thus are not provided in this section. It is the intent of the plan to meet the more recent requirements delineated in 10 CFR 73.55 and, due to considerable differences in scope between ANSI-N18.17-1973 and 10 CFR 73.55, the Physical Security Plan incorporates both documents to establish an effective security policy to protect CPS against attempts of radiological sabotage.

USAR Section - 13.6

<u>Regulatory Guide 1.18. Rev.1 (December, 1972)</u>

Structural Acceptance Test for Concrete Primary Reactor Containments

<u>Project Position</u> - The Project complies with NRC Regulatory Guide 1.18 with the following exceptions:

(1) Reference: Paragraph C.1 - Due to the low design pressure of 15 psig for the containment, raising the internal pressure of the containment in three approximately equal pressure increments is considered adequate for the structural acceptance test of the containment.

(2) Reference: Paragraph C.3.- Tangential deflections around the equipment hatch which is the largest opening in the containment are insignificant. Therefore, only the radial deflections will be measured around the equipment hatch.

<u>USAR Subsection</u> - 3.8.1.7.1

<u>Regulatory Guide 1.19, Rev. 1 (August 11, 1972)</u>

Nondestructive Examination of Primary Containment Liner Welds

<u>Project Position</u> - The Project complies with NRC Regulatory Guide 1.19 with the following exception:

(1) Reference: Paragraph C.7.b - Examinations by ultrasonic, magnetic particle and liquid penetrant methods were considered acceptable provided the examinations met the acceptance standards of NE 5330, NE 5340 and NE 5350 respectively of Section III of the 1971 ASME Code (Summer '73 Addenda).

<u>USAR Section</u> - Appendix B.

<u>Regulatory Guide 1.20, Rev. 2 (May 1976)</u>

Comprehensive Vibration Assessment Program for Reactor Internals During Preoperational and Initial Startup Testing

<u>Project Position</u> - Comply

<u>USAR Subsections</u> - 3.9.2.4, 14.2.12.1.37

Regulatory Guide 1.21, Rev. 1 (June 1974)

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

Project Position - The Project complies with NRC Regulatory Guide 1.21 with the following exceptions:

(1) Reference: Paragraph C.6 - Turbulent flow in line 0WE32AA-4 causes thorough mixing of the fluid. A representative sample will still be obtained by sampling from the pipe because of this mixing.

(2) Reference: Paragraph C.1 - A summary report of the meteorological measurements taken during each calendar quarter of the January through June period shall be provided as joint frequency distributions of wind direction and wind speed by atmospheric stability class in the July to December report as part of an annual summary. CPS will retain the January through June summary on site in a file that shall be provided to the NRC upon request.

(3) Reference: Paragraph C.2 - CPS will make measurement of effluent volume, rates of release, and specific radionuclides for gaseous releases from the Station Heating, Ventilating, and Air Conditioning (HVAC) and Standby Gas Treatment System (SGTS) stacks.

(4) Reference: Appendix A, Paragraph A.1.a. - CPS will perform gaseous grab samples on the HVAC stack and analyze for principal gamma emitters and tritium weekly and following reactor shutdown, startup, or a thermal power change exceeding 15% of rated thermal power within a one-hour period. CPS will perform gaseous grab samples on the SGTS stack upon the initiation of gaseous releases via the SGTS stack.

(5) Reference: Appendix A, Paragraph B.1.b - A quarterly sample composited from proportional aliquots from each liquid effluent batch release during the quarter will be analyzed for tritium and gross alpha radioactivity.

(6) Reference: Appendix B, Paragraph A.2 - The effluent concentrations of 10CFR20 are not utilized directly for limiting gaseous effluents. The CPS Offsite Dose Calculation Manual establishes requirements to limit the release rate of effluents. Discharges of gaseous radioactive material will not result in annual average exposure concentrations greater than limits for a member of the public in an unrestricted area (inside or outside the site boundaries).

(7) Reference: Appendix B, Paragraph E.2, 5 and 6 - Doses at CPS are calculated in accordance with the NUREG-0133 maximum exposed individual concept. Dose due to the release of radioactive material in waterborne effluents is calculated for the water related pathways as specified in the CPS Offsite Dose Calculation Manual. Beta and gamma air dose due to the release of noble gas in gaseous effluents is calculated at the CPS site boundary in each of the 16 geographical directions surrounding CPS. Dose due to the release of radioactive iodines and particulates in gaseous effluents is calculated at the critical receptor location in each of the 16 geographical sectors surrounding CPS to a distance of

Regulatory Guide 1.21, Rev. 1 (June 1974)

5 miles. Dose summaries based on these calculations are provided in the Radioactive Effluent Release Report.

(8) Reference: Appendix B, Paragraph A.3 - The CPS Offsite Dose Calculation Manual limits the dose rates due to the release of fission and activation gases to less than or equal to 500 mrem per year to the total body and less than or equal to 3000 mrem per year to the skin. Release rates for fission and activation gases in the gaseous effluents are not determined directly from the average energy $(\bar{E})$ of the radionuclide mixture in the effluent. Therefore, the $(\bar{E})$ value for the gamma and beta energies per disintegration is not reported in Radioactive Effluent Release Reports.

USAR Section/Subsections - 7.1.2.6.4, 7.6.1.2.4, 7.6.1.2.5, 7.6.1.2.6, 7.6.2.2.4, 7.6.2.2.5, 7.7.2, 9.3.2, 11.5

Regulatory Guide 1.22, Rev. 0 (February 1972)

Periodic Testing of Protection System Actuation Functions

Project Position - Comply

USAR Subsections - 7.1.2.6.5, 7.2, 7.3, 7.3.2.2.2.1.2, 7.4, 7.6, 8.1.6.1.3.

Regulatory Guide 1.23, Rev. 1 (Proposed)

Onsite Meteorological Program

Project Position - Meets the requirements of ANS 2.5-1984 proposed as Regulatory Guide 1.23, Revision 1, with the following exceptions:

(1) accuracy of the dewpoint temperature;

(2) precipitation is not recorded on the digital portion of the data acquisition system;

(3) digital accuracies.

USAR Subsection - 2.3.3

Regulatory Guide 1.24, Rev. 0 (March 1972)

Assumptions Used for Evaluating the Potential Radiological Consequences of Pressurized Water Reactor Radioactive Gas Storage Tank Failure

Project Position - Not applicable to BWRs.

USAR Subsection - N/A

Regulatory Guide 1.25, Rev. 0 (March 1972)

Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facility for Boiling and Pressurized Water Reactors

Project Position – Partially Comply; for Alternative Source Terms, Regulatory Guide 1.183 is also utilized.

USAR Section - 15.7.4

Regulatory Guide 1.26, Rev. 3 (February 1976)

Quality Group Classifications and Standards for Water-,
Steam-, and Radioactive- Waste-Containing Components
of Nuclear Power Plants

Project Position - The Project complies with NRC Regulatory Guide 1.26 with the following clarifications:

(1) Reference: Paragraph C.2.a - The component cooling water system (CCW), which cools the fuel pool cooling and cleanup system (FPC&C) heat exchangers during normal operation is not classified as Quality Group C. However, that portion of the CCW system which includes the FPC&C heat exchanger inlet and outlet piping and valves which allow isolation from the remainder of the system is designed to Quality Group C standards. Essential service water is used to cool the FPC&C heat exchangers through this piping following a LOCA or LOEP.

(2) Reference: Paragraph C.2.b - The reactor recirculation pumps are not considered essential and therefore are not provided with Quality Group C cooling water during normal operation and following a LOCA. The recirculation pump seals and motor bearings are, however, provided with cooling water following a loss of offsite electrical power in order to minimize any damage during coastdown.

(3) Reference: Paragraph C.2.b - The cooling coils fitted in the thrust bearing housing of the Division 1 and Division 2 shutdown service water pump motors are provided with Quality Group C cooling water during normal operation and following a LOCA. The motor and its cooling coil are not assigned to quality group classification. For further description of classification see USAR Table 3.2-1.

(4) Reference: Paragraph C.2.b – The portion of Shutdown Service Water System (SSWS) piping and components associated with Standby Gas Treatment System (SGTS) Exhaust HI-Range Radiation Monitor Cooler are not classified as Quality Group C. However that portion is designed to Quality Group C standards.

USAR Section - 3.2, 5.4.7.1.1.6.1, 11.2.1.3, 9.2.1.2.1.1, 9.2.1.2.3

<u>Regulatory Guide 1.27, Rev. 2 (January, 1976)</u>

Ultimate Heat Sink for Nuclear Power Plants

<u>Project Position</u> - The Project complies with NRC Regulatory Guide 1.27 with the following exceptions:

(1) Reference: Paragraph C.1.- The time period for the analysis of the ultimate heat sink was based upon the worst conditions for heat transfer from Clinton Lake. These worst conditions were determined by a computer analysis of the lake which predicted transient lake temperature for the years 1949 through 1971. Although two separate periods were not chosen for analysis, the intent of the Regulatory Guide to assure the adequacy of the ultimate heat sink during periods of high evaporative conditions and during period of unfavorable ambient conditions is met.

(2) Reference: Paragraph C.2.C - The failure of the ultimate heat sink dam is not considered credible.

<u>USAR Subsections</u> - 2.4.11.6, 9.2.5; TS3.7.1, 3.7.2 and 5.5.12

<u>Regulatory Guide 1.28, Rev. 3 (August 1985)</u>

Quality Assurance Program Requirements
(Design and Construction)

<u>Project Position</u> - Comply with the following clarification:

The site QA programs, as committed in the PSAR were in compliance with ANSI N45.2(1971) as endorsed by Regulatory Guide 1.28 Rev. 0 dated June 7, 1972. Later revisions of ANSI N45.2 and the associated Regulatory Guide are incorporated in site QA programs.

<u>USAR Section</u> - 17.1

Regulatory Guide 1.29 Rev. 3 (September 1978)

Seismic Design Classification

Project Position - The Project complies with NRC Regulatory Guide 1.29 with the following clarifications:

1) Reference: Paragraph C.1.e - The main steam system from the outermost containment isolation valve up to and including the shut-off valve in the Seismic Category I Auxiliary Building is classified and designed as Seismic Category I. The portion of the system from the main steam shut-off valve up to the boundary of the Auxiliary Building is not classified as Seismic Category I, but is designed seismically. Seismic interface restraints are provided for each line near the Auxiliary/Turbine Building boundary. The piping from the Auxiliary/Turbine Building boundary up to and including the turbine stop valve is not classified or designed as Seismic Category I. The piping from the shut-off valve to the turbine stop valve is classified as Quality Group D in accordance with Regulatory Guide 1.26.

2) Reference: Paragraph C.1.h - The reactor recirculation pumps are not considered essential and therefore are not provided with Seismic Category I cooling water during normal operation and following a LOCA. The recirculation pump seals and motor bearings are, however, provided with cooling water following a loss of offsite electrical power in order to minimize any damage during coastdown.

3) Reference: Paragraph C.3 - The seismic design requirements for piping and supports beyond the defined Seismic Category I boundaries are described in Subsection 3.7.3.13.

USAR Section/Subsection - 3.2, 3.7.3, 5.4.7.1.1.6.1, 7.1.2.6.6, 7.3.2.3.2.1.3, 7.3.2.20.2.1.2, 7.6.2.5.5

Regulatory Guide 1.30, Rev. 0 (August 1972)

Quality Assurance Requirements for the Installation,
Inspection and Testing of Instrumentation and Electrical Equipment

Project Position - The project complies with the requirements of Regulatory Guide 1.30 with the following exception:

> Reference: ANSI N45.2.4-1972/IEEE 336-1971, Section 6.2.1 It is specified that for installed equipment, "items requiring calibration shall be tagged or labeled on completion indicating date of calibration and identity of person that performed the calibration." Complying with an obligation of this nature would be very restrictive and costly.
>
> It should clearly be noted that compliance is possible, but is neither necessary nor desirable since instrument ID numbers permit traceability to calibration records.
>
> The status of calibration of all permanently installed equipment covered by the Clinton Power Station "CALIBRATION PROGRAM" will be easily attained due to the traceability of records to that item.

USAR Subsections - 6.5.1.5, 7.1.2.6.7, 7.3.2.2.2.1.4, 7.3.2.20.2.1.3, 7.4.2.1.2.1.5, 7.4.2.2.2.1.5, 7.6.2.4.2.1, 7.6.2.7.2, 7.6.2.8.2, 8.1.6.1.5, 8.1.6.2.6.

Regulatory Guide 1.31, Rev. 3 (April 1978)

Control of Ferrite Content in Stainless Steel Weld Metal

Project Position for Balance of Plant Systems Scope of Supply - Comply

Project Position for Nuclear Steam Supply Systems Scope of Supply - Compliance evaluation was based upon a comparison of the work performed for this project against the requirements of Revision 2 of Regulatory Guide 1.31. Compliance with this regulatory guide is based upon an extensive test program which demonstrates that controlling weld filler metal ferrite at 5% minimum produces production welds which meet the regulatory guide requirements.

Reactor internals were fabricated prior to the issuance of Revision 3, however, ferrite measurements were made in accordance with the requirements of the ASME code in effect at that time.

USAR Subsections - 4.5.1.2, 4.5.2.4, 5.2.3.4, 6.1.1, 10.3.6

Regulatory Guide 1.32, Rev. 2 (February, 1977)

Criteria for Safety-Related Electric Power Systems for Nuclear Power Plants

Project Position - Comply, with the following exceptions:

1) Reference: Paragraph C.1.c - Battery performance discharge tests are performed as required in TS 3.8.4, TS 3.8.5, and TS 3.8.6.

   Position C.1.e - with the clarifications and exceptions noted in subsection 8.1.6.1.6.

2) Reference: Paragraph C.1.c - IEEE Standard 450-1995 revision is used in lieu of earlier revisions.

USAR Subsections - 7.1.2.6.8, 7.3.2.1.2.1.6, 7.3.2.20.2.1.4, 7.4.2.1.2.1.6, 8.1.6.1.6, 8.3.1.1.2.1, 8.3.1.2.2, 8.3.2.1.2.1, 8.3.2.2.2.2, TS 3.8.4, 3.8.5, 3.8.6.

Regulatory Guide 1.33, Rev. 2 (February 1978)

Quality Assurance Program Requirements (Operation)

Project Position - The Project complies with this guide with the following clarifications:

1) Reference: Paragraph C.2 - CPS compliance positions to ANSI Standards included and referenced an ANSI N18.7-1976 are addressed under the appropriate Regulatory Guide listed in this section.

2) Reference: Section 5.2.13.2 of ANSI N18.7-1976, Control of Purchased Material, Equipment and Services, 4th paragraph, 1st sentence - The phrase "...prior to installation or use of such items." is considered to be too restrictive to the efficient utilization of the plant. The following alternate course of action will provide the needed controls and ensure that items are not "released for operation" until the documentary evidence is available at CPS or an engineering evaluation has been performed, reviewed and accepted.

   Inservice items that are found to be nonconforming shall be reviewed to determine equipment operability as defined by the Technical Specifications. For items that represent significant conditions adverse to quality or safety, or require a repair or use-as-is disposition, an engineering evaluation shall be performed. The engineering evaluation shall provide support for the initial operability decision and provide the correction or resolution for the identified nonconformance. These items shall be controlled in accordance with approved procedures.

   Installed items not inservice that are nonconforming or become nonconforming as a result of maintenance shall be corrected or resolved prior to operational reliance. These items shall be controlled in accordance with approved procedures.

Regulatory Guide 1.33, Rev. 2 (February 1978) (Cont'd)

A nonconforming item may be conditionally released for fabrication, installation or testing following an engineering evaluation to determine if such a conditional release is not detrimental to other components or systems. Conditional release items are controlled in accordance with approved procedures. The nonconformance for the conditionally released item shall be corrected or resolved prior to operational reliance.

3) Reference: Section 6 of ANSI N18.7-1976, References Subsequent revisions to the American National Standards referred to in ANSI N18.7-1976 will be evaluated to consider the necessity for incorporation of the revision into the CPS Operational Quality Assurance Program.

4) Reference: Paragraph C.1 3rd sentence - The following will be used in place of Appendix A to determine if a procedure is Safety-Related.

   A procedure shall be considered Safety-Related if the procedure operates, performs maintenance on, installs, modifies, or maintains the integrity of the pressure boundary for any system, components, or structure with a Safety Classification listed in Table 3.2-1 of the USAR and the Administrative Procedures listed in Appendix A, Item 1 of this Regulatory Guide.

5) Reference: Paragraphs C.5.i and C.5.j - The formats described in Section 13.5.2.1.3 shall be used in place of the format described in Section 5.3.9 of ANSI N18.7-1976.

6) Reference: Section 5.2.6 of ANSI N18.7-1976, Equipment Control, 5th paragraph - Startup complies with the administrative controls for temporary modifications such as temporary bypass lines, electrical jumpers, lifted electrical leads and temporary trip point settings except that during the Checkout and Initial Operation (C&IO) Test Phase, temporary modifications shall not require independent verification. Documented verification of restoration shall be provided, but may be provided by the individual restoring the modification.

7) With regard to Section 5.2.15 of ANSI N18.7-1976 titled <u>Review Approval and Control of Procedure:</u> Programmatic controls for periodic reviews of procedures will consist of four key elements for the periodic review process:

   - At least every two years Nuclear Oversight shall audit a representative sample of routine plant procedures that are used more frequently than every two years,

   - All applicable plant procedures shall be reviewed following an unusual incident or unexpected transient, operator error, and following a modification,

   - Routine plant procedures that have been used at least biennially receive scrutiny by individuals knowledgeable in the procedures, and are updated as necessary to ensure adequacy during suitable controlled activities, and,

Regulatory Guide 1.33, Rev. 2 (February 1978) (Cont'd)

- Routine plant procedures that have not been used for two years will be reviewed before use to determine if changes are necesary or desirable.

The third element is an acceptable method to review procedures because the procedure is tested through actual use. This satisfies the intent of the review criteria in ANSI 18.7-1976, Section 5.2.15.

8) Reference: Paragraphs C.4.a, C.4.b, C.4.c - The following audit frequencies will be followed:

   a. Audits of the results of actions taken to correct deficiencies occurring in unit equipment, structures, systems, or method of operation that affect nuclear safety - 24 months.

   b. Audits of the conformance of unit operation to provisions contained within the Technical Specifications and applicable license conditions - 24 months.

   c. Audits of the performance, training and qualifications of the unit staff - 24 months.

9) Reference: ANSI N18.7-1976, Section 5.2.2, Procedure Adherence, sentence 4, prescribes that one of the approvers of a temporary change to a procedure shall be the supervisor in charge of the shift and hold a senior operator license on the unit affected. The station was originally licensed with SAR section 13.5 requiring one of the approvers to hold a senior operators license on the unit affected, but did not specify that individual be the supervisor in charge of the shift. A CPS administrative procedure specifies that one of the unit management staff members approving temporary procedure changes shall be from the on-duty shift and hold an SRO license.

USAR subsection - 12.5.2, 13, 17.2, TS 5.4.1a

Regulatory Guide 1.34, Rev. 0 (December 1972)

Control of Electroslag Weld Properties

Project Position - Not applicable, electroslag welding not utilized.

USAR Subsections - 5.2.3.3.2.2

Regulatory Guide 1.35, Rev. 3 (April, 1979)

Inservice Inspection of Ungrouted Tendons in Prestressed
Concrete Containment Structures

Project Position - Not applicable to Clinton Power Station.

Regulatory Guide 1.36, Rev. 0 (February, 1973)

Nonmetallic Thermal Insulation For Austenitic Stainless Steel

Project Position - Comply

USAR Subsections - 5.2.3.2.4, 6.1.1, 10.3.6

Regulatory Guide 1.37, Rev. 0 (March 1973)

Quality Assurance Requirements for Cleaning of Fluid
Systems and Associated Components of Water-Cooled Nuclear Power Plants

Project Position - Comply

USAR Subsections - 6.1.1, 10.3.6, 14.2.7

<u>Regulatory Guide 1.40, Rev. 0 (March 1973)</u>

Qualification Tests of Continuous - Duty Motors Installed
Inside the Containment of Water-Cooled Nuclear Power Plants

<u>Project Position</u> - Comply

<u>USAR Section</u> - 7.1.2.6.9, 8.1

<u>Regulatory Guide 1.41, Rev. 0 (March 1973)</u>

Preoperational Testing of Redundant Onsite Electrical Power
Systems to Verify Proper Load Group Assignments

<u>Project Position</u> - Comply

<u>USAR Chapter</u> - 8.3.1.2.1, 14

Regulatory Guide 1.42 has been withdrawn

<u>Regulatory Guide 1.43, Rev. 0 (May.1973)</u>

Control of Stainless Steel Cladding of Low-Alloy Steel Components

<u>Project Position</u> - Regulatory Guide 1.43 prescribes qualification and production cladding controls for ASME SA 508-2 material made to COARSE GRAIN practice. This material is not used for any of the safety class components. ASME SA 508-2 composition material employed on the reactor pressure vessel for this plant is produced to FINE GRAIN practice. Therefore, this Regulatory Guide is not applicable to the components in this plant.

<u>USAR Subsection</u> - 5.2.3.4, 5.3.1.4.1.3, 6.1.1

<u>Regulatory Guide 1.44, Rev. 0 (May, 1973)</u>

Control of the Use of Sensitized Stainless Steel

<u>Project Position for Balance of Plant Systems Scope of Supply</u> - The requirements of Regulatory Guide 1.44 are met except as follows:

1. Reference: Paragraph C.3. Testing is not performed. Material is procured in the solution annealed conditon.

2. Reference: Paragraph C.6. All welding performed on austenitic stainless steel is with low heat input welding processes. Materials used are in the solution annealed condition and the following additional safeguards are taken:

    a. The preheat and interpass temperature used during the welding of austenitic stainless steel is kept to 350 degrees F maximum.

    b. Postweld heat treatment in the range of 800 degrees to 1500 degrees F is strictly forbidden. Solution annealing heat treatment, after welding, although not required, is permitted.

    Because severe sensitization is avoided by these safeguards, testing to determine susceptibility to intergranular attack is not performed.

<u>Project Position for Nuclear Steam Supply System Scope of Supply</u> - Complies with the intent of the Regulatory Guide 1.44

<u>USAR Subsections</u> - 5.2.3.4, 6.1.1, 10.3.6.

Regulatory Guide 1.45, Rev. 0 (May, 1973)

Reactor Coolant Pressure Boundary Leakage Detection Systems

Project Position - The requirements of the NRC Regulatory Guide 1.45 are met with the following exceptions for those portions of the station under the balance of plant systems scope of design:

1. Reference: Paragraph C.5 - The sensitivity and response time of airborne particulate and gaseous radioactivity monitors is not adequate to detect a leakage rate of 1 GPM in less than one hour. The correlation between flow rate and radioactivity is not valid due to various complex factors discussed in Section 5.2.5.2.2. The monitor will not always alarm for 1 GPM in one hour and, therefore, is considered as qualitative indication of the presence of abnormal leakage. Similarly, because the drywell floor drain sump flow monitoring system calculates the unidentified leakage rate based on drywell floor drain sump pump discharge flow, this system is not sensitive enough to detect a leakage rate of 1 GPM in less than one hour over the entire range of potential leakages. This system is, however, capable of promptly detecting leakage rates and leakage rate increases prior to exceeding Technical Specification operating limits.

   CPS will follow the guidelines of ANSI/ISA S67.03, Standard for Light Water Reactor Coolant Pressure Boundary Leak Detection, October 3, 1982.

2. Reference: Paragraph C.6 - The sump flow monitoring instrumentation which is located at the sumps is seismically qualified to OBE. The calculation devices and instrumentation outside of the drywell have not been seismically qualified. Should these devices fail during a seismic event, they are readily accessible for maintenance and/or replacement to reestablish the functionality of the monitoring equipment.

3. Reference: Paragraph C.3 - Compliance is met through the implementation of NUREG 1434, Standard Technical Specifications for BWR-6 plants, LCO 3.4.7, Reactor Coolant Systems Leakage Detection Instrumentation, which was approved by the NRC in License Amendment 95.

USAR Subsections - 5.2.5, 7.3, 7.4, 7.6, 7.6.1.4, 7.6.2.4, TS 3.4.7

Regulatory Guide 1.46 has been withdrawn.

Regulatory Guide 1.47, Rev. 0 (May 1973)

Bypassed and Inoperable Status Indication for Nuclear
Power Plant Safety Systems

Project Position - Comply

USAR Subsections - 7.1.2.6.11, 7.2, 7.3, 7.3.2.20.2.1.5, 7.4, 7.6, 8.1.6.1.9, 8.3.2.2.2.3, App. D

Regulatory Guide 1.48, Rev. 0 (May 1973)

Design Limits and Load Combinations for Seismic Category
1 Fluid System Components

Project Position - Comply with stress limits for Active Components with the following exception:

> The operability requirements for all active components will be assured by performing a detailed deformation analysis and/or by performing a seismic test. Therefore, the allowable stress limits that shall be used for each category shall be in conformance with the applicable ASME Codes.

USAR Section - 3.9

Regulatory Guide 1.49, Rev. 1 (December 1973)

Power Levels of Nuclear Power Plants

Project Position - Comply

USAR Sections - 15.0, 15.1

<u>Regulatory Guide 1.50, Rev. 0 (May 1973)</u>

Control of Preheat Temperature for Welding of Low-Alloy Steel

<u>Project Position for Balance of Plant Systems Scope of Supply</u> - Low-alloy steels were not used on ASME Class 1, 2 and 3 piping systems during plant construction, therefore, control of preheat temperature for welding as required by Regulatory Guide 1.50 was not applicable to these systems at that time. Monitoring of plant operation has revealed certain sections of piping to be susceptible to Flow Accelerated Corrossion (FAC). Low-alloy steels, such as 2¼ Cr – 1 Mo, may be used as repair/replacement materials is these piping sections. Where low-alloy steel is used the requirements of Regulatory Guide 1.50 for control of welding preheat temperature will be complied with.

<u>Project Position for Nuclear Steam Supply Systems Scope of Supply</u> – During plant construction the use of low-alloy steel was restricted to the reactor pressure vessel.

For fabrication of the reactor pressure vessel welding preheat control complied with Regulatory Guide 1.50. Low-alloy steels were not used on the remainder of NSSS systems during plant construction, therefore, control of preheat temperature for welding as required by Regulatory Guide 1.50 was not applicable to these systems at that time. Monitoring of plant operation has revealed certain sections of piping to be susceptible to Flow Accelerated Corrosion (FAC). Low-alloy steels, such as 2¼ Cr – 1 Mo, may be used as repair/replacement materials is these piping sections. Where low-alloy steel is used the requirements of Regulatory Guide 1.50 for control of welding preheat temperature will be complied with.

<u>USAR Subsections</u> – 5.2.3.3.2.1, 5.3.1.4.1.5, 10.3.6

Regulatory Guide 1.51 has been withdrawn

<u>Regulatory Guide 1.52, Rev. 2 (March 1978)</u>

Design, Testing and Maintenance Criteria for Post Accident
Engineered - Safety - Feature Atmosphere Cleanup System Air
Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants

<u>Project Position</u> - As required by Regulatory Guide 1.70, Revision 3, a detailed discussion of the extent of compliance to the requirements of Regulatory Guide 1.52 is provided in USAR Table 6.5-3. In addition, the CPS "ANSI N509/510 Variance Report", identifies minor deviations to this Regulatory Guide referenced in the ANSI standards. This Variance Report is a controlled document, maintained by the Nuclear Station Engineering Department.

<u>USAR Section</u> - 6.5, 9.5.1.2.2.3, 12.3.3.1, E4.0.D.4.d, Table 7.1-3, Table 14.2-1

Regulatory Guide 1.53, Rev. 0 (June 1973)

Application of the Single Failure Criterion
to Nuclear Power Plant Protection Systems

Project Position - Comply

USAR Subsections - 7.1.2.6.12, 7.3.2.3.2.1.5, 7.3.2.20.2.1.6, 7.6.2.5.5, 8.1.6.1.10

Regulatory Guide 1.54, Rev. 0 (June 1973)

Quality Assurance Requirements for Protective Coatings
Applied to Water Cooled Nuclear Power Plants

Project Position - Comply

USAR Sections - 6.1, 6.2, 9.1.4.2.5.7, 17.1, 17.2

Regulatory Guide 1.55, Rev. 0 (June 1973)

Concrete Placement in Category I Structures

Project Position - Comply

USAR References - Appendix B and Section 17.1

Regulatory Guide 1.56, Rev. 1 (July 1978)

Maintenance of Water Purity in Boiling Water Reactors

Project Position - Comply with the following exceptions:

1. Reference: Paragraph C.4d - The resin samples will be taken every 5th removal from service for resin cleaning.

   Since most units may be removed from service one or more times for resin cleaning with remaining capacity available, they will be returned to service without regeneration. To count such removals from service may mean determining remaining capacity at a time the bed would not normally be considered at "minimum residual capacity". The resin beds will be replaced when the calculated resin capacity approaches 50% of initial capacity.

2. Condensate polishing resin beds may be used beyond 50% of initial capacity while the reactor is shutdown, circ water is isolated and feedwater is not being supplied to the vessel via the condensate polishers.

USAR Subsections - 5.2.3.2, 10.4.6, 12.3

Regulatory Guide 1.57, Rev. 0 (June 1973)

Design Limits and Loading Combinations for Metal Primary
Reactor Containment System Component

Project Position - The design complies with Regulatory Guide 1.57 with the following clarification:

For Class MC penetration assemblies with respect to Regulatory Guide Position C.1.d, the design load combination for faulted loads are:

a. Maximum operating pressures and temperatures, plus loads due to pipe rupture and jet impingement where applicable.

b. Process pipe maximum operating pressure applied in the annulus between the process pipe and the penetration sleeve for MC penetration assemblies only.

USAR Sections - 3.8, 3.9

Regulatory Guide 1.59, Rev. 2 (August, 1977)

Design Basis Floods for Nuclear Power Plants

Project Position - Comply

USAR Subsections - 2.4.2.3, 2.4.3, 2.4.8

Regulatory Guide 1.60, Rev. 1 (December, 1973)

Design Response Spectra for Seismic Design of Nuclear Power Plants

Project Position – Comply with clarification as discussed in 2.5.2.6 and 2.5.2.7.1.

USAR Subsections - 2.5.2, 3.7.1

Regulatory Guide 1.61, Rev. 0 (October 1973)

Damping Values for Seismic Design of Nuclear Power Plants

Project Position - Comply

USAR Subsections - 3.7.1, 3.7.2, 3.7.3

Regulatory Guide 1.62, Rev. 0 (October 1973)

Manual Initiation of Protective Actions

Project Position - Comply

USAR Subsections - 7.1.2.6.14, 8.1.6.1.11.

Regulatory Guide 1.63, Rev. 2 (July, 1978)

Electric Penetration Assemblies in Containment Structures
for Light-Water-Cooled Nuclear Power Plants

Project Position - Comply.

USAR Subsections - 8.1.6.1.12.

Regulatory Guide 1.65, Rev. 0 (October 1973)

Materials and Inspections for Reactor Vessel Closure Studs

Project Position - Comply

USAR Subsection - 5.3.1.7

Regulatory Guide 1.66 has been withdrawn.

Regulatory Guide 1.67, Rev. 0 (October 1973)

Installation of Overpressure Protective Devices

Project Position - The main steam line safety/relief valves relieve to closed discharge systems. The guidelines delineated in Regulatory Guide 1.67 are not applicable to that situation.

The installation of safety/relief valves in other ASME Class 1 and 2 systems complies with the regulatory guide.

USAR Subsections - 3.9.3.3.2, 5.2.2

Regulatory Guide 1.68, Rev. 2 (August 1978)

Initial Test Programs for Water-Cooled Nuclear Power Plants

Project Position - The Project complies with the following exceptions.

1. Reference: Appendix A, Paragraph 1H(10) - There is no practical way of lowering the lake level for testing of the ultimate heat sink. Testing for NPSH to ensure vortexing does not occur will be performed by lowering the water level in the pump pit. Testing will be performed at simulated normal lake level and minimum design ultimate heat sink level.

2. Reference: Appendix A, Paragraphs 1k(2) and (3) - There will be no "preoperational test" of personnel radiation monitoring instruments or laboratory radiation measuring equipment. Equipment of this type will be calibrated to assure proper operation.

3. Reference: Appendix A, Paragraph 5k - This test will not be performed during power operation since this would induce an unnecessary thermal transient cycle to the nozzles and spargers. The high pressure coolant injection systems will be preoperationally tested to verify starting, flow rates and head loss.

4. Reference: Appendix A, Paragraph 5L - RHR System testing in Steam Condensing Mode was not performed during the Startup Test Program. If this mode of operation is ever used it will be evaluated for testing requirements at that time.

5. Reference: Appendix A, Paragraph 5hh - Reactor coolant flow control system calibration and performance in the Automatic Load Following mode was not performed during the Startup Test Program. CPS operational programs were revised to delete this feature of recirc. control. The feature has been subsequently removed.

6. Reference: Appendix A, Paragraph 5x - A test to determine the heat removal capacity was not performed for the ECCS Equipment Cooling HVAC System. Heat removal adequacy was demonstrated for compliance to this paragraph.

USAR Chapter – 14

Regulatory Guide 1.68.1, Rev. 1 (January 1977)

Preoperational and Initial Startup Testing of Feedwater and
Condensate Systems for Boiling Water Reactor Power Plants

Project Position - Comply

USAR Section - 14.2

Regulatory Guide 1.68.2, Rev. 1 (July 1978)

Initial Startup Test Program to Demonstrate Remote Shutdown
Capability for Water-Cooled Nuclear Power Plants

Project Position - Comply

USAR Section - 14.2

Regulatory Guide 1.69, Rev. 0 (December 1973)

Concrete Radiation Shields for Nuclear Power Plants

Project Position - Comply

USAR Subsection - 12.3.2

Regulatory Guide 1.70, Rev. 3 (November, 1978)

Standard Format and Content of Safety Analysis Reports
for Nuclear Power Plants - LWR Edition

Project Position - Comply with the following clarifications and exception:

1. Clarification: The FSAR was written in accordance with the guidance of Regulatory Guide 1.70, Rev. 3. However, CPS will utilize Regulatory Guide 1.181 in conjunction with NEI 98-03, Guidelines for Updating Final Safety Analysis Reports, as guidance for maintaining the USAR in accordance with the requirements of 10 CFR 50.71(e).

2. Clarification: CPS is a single unit nuclear facility, owned and operated by Exelon. As such, Exelon has a corporate based organization directly involved in the operation and in providing technical or operational support to CPS. The description of the organizations providing technical or operational support to CPS including the corporate and site organizations are described in the Exelon Generation Company Quality Assurance Topical Report.

3. Exception: Reference Section 1.7.1. Consistent with Regulatory Guide 1.70. Electrical and Instrumentation and Control drawings were provided to the NRC during initial licensing of the facility. However, per the Questions/Responses enclosed with Generic Letter 81-06 (see Question/Response C.1), these drawings are not included in the USAR.

USAR Subsection - 1.1.9.2, 4.1.2.1.1.(5), 6.2.8.13, 6.3.2.8, 7.1.1.1, 7.1.2.6.17, 7.6.2.4.2.1, 7.6.2.7.2.1, 7.6.2.8.2.2, 7.7.2.9.2, .7.2.10.2, 17.0, Appendix D II.D.1, Appendix D II.K.3.44.

Regulatory Guide 1.71, Rev. 0 (December 1973)

Welder Qualification for Areas of Limited Accessibility

Project Position - For Clinton Power Station construction, requirements or Regulatory Guide 1.71 are met with the following exceptions:

1. Reference: Paragraph C.1 - Performance of welders under simulated access conditions is not necessary to assure acceptable welds. There would be an excessive number of required qualifications. Accept ability of welds will be determined by required examinations.

2. Reference: Paragraph C.2 - Requalification is not necessary under the conditions listed in the regulatory position. This would require an excessive number of requalifications. Acceptability of welds will be determined by required examination.

For Clinton Power Station operation, welder qualification for areas of limited accessibility during the operation stage complies with the intent of Regulatory Guide 1.71 through the implementation of the following program:

> If clearance around the production joint of at least 12 inches (except for minor obstructions) cannot be obtained, and clearance is less than 12 inches where obstructions exist which may prevent the welder from gaining an advantageous position for welding, the weld will be evaluated for limited access considerations by the supervisor mechanical. In this case, the structure to be welded, including its actual access limits, may be simulated.

USAR Subsections - 5.3.1.4, 10.3.6

Regulatory Guide 1.72, Rev. 2 (November, 1978)

Spray Pond Plastic Piping

Project Position - Not applicable; Clinton Power Station does not have spray ponds.

Regulatory Guide 1.73, Rev. 0 (January 1974)

Qualification Tests of Electric Valve Operators Installed
Inside the Containment of Nuclear Power Plants

Project Position - Comply

USAR Section - 8.1

Regulatory Guide 1.75, Rev. 2 (September, 1978)

Physical Independence of Electric Systems

Project Position - The project complies with NRC Regulatory Guide 1.75 with the clarifications and exceptions noted in Subsections 7.1.2.6.19 and 8.1.6.1.14.

USAR Subsections - 7.1.2.6.19, 7.3.2.20.2.1.7, 7.6.2.5.5, 8.1.6.1.14, 8.3.1, E4.0.D.3.c

Regulatory Guide 1.76, Rev. 1 (March, 2007)

Design Basis Tornado and Tornado Missiles for Nuclear Power Plants

Project Position - Comply.

USAR Subsection - 3.3.2, 3.5.1.4

* Note: Reg 1.76, Rev. 1 is effective at CPS from September 2007.

Regulatory Guide 1.77, Rev. 0 (May, 1974)

Assumptions Used for Evaluating a Control Rod Ejection Accident for PWRs

Project Position - Not applicable to BWRs.

Regulatory Guide 1.78, Rev. 0 (June, 1974)

Assumptions for Evaluating the Habitability of a Nuclear Power Plant
Control Room During a Postulated Hazardous Chemical Release

Project Position - Comply.

USAR Sections/Subsection: 2.2, 6.4, 9.4.1, 9.5.8.3

Regulatory Guide 1.79, Rev. 1 (September 1975)

Preoperational Testing of Emergency Core Cooling
Systems for Pressurized Water Reactors

Project Position - Not applicable to BWRs.

Regulatory Guide 1.80, Rev. 0 (June 1974)

Preoperational Testing of Instrument Air System

Project Position - The project complies with the requirements of Regulatory Guide 1.80 with the following exception:

> Reference: C.5 - The Instrument Air System will meet the Class 'B' cleanliness requirements as defined in ANSI N 45.2.1-1973, "Cleaning of Fluid Systems and Associated Components during Construction Phase of Nuclear Power Plants." Particulate contamination criteria will be established for portions of the instrument air system that supply air to selected active safety-related components to ensure operational reliability.
>
> Reference: C.8 - A loss-of-instrument-air supply test will be performed on the branches of the system which serve safety-related equpment, specifically those which supply the Automatic Depressurization System (ADS) and Non-ADS Low Low Set Safety Relief Valves (LLS-SRV) valve operators with operating air. Testing will be conducted on a valve-by-valve basis for those branches which do not serve safety-related equipment.
>
> Reference: C.10 - A loss-of-instrument-air supply test as described in C.10 will not be performed in the Instrument Air Preoperational Test. The testing required by C.8 and C.9 in combination with the Checkout and Initial Operation Testing of each valve will satisfy the position of C.10.

USAR Section/Subsection - 9.3.1.4, 14.2.12.1.46.

Regulatory Guide 1.81, Rev. 1 (January, 1975)

Shared Emergency and Shutdown Electrical Systems for
Multi-Unit Nuclear Power Plants

Project Position - Does not apply since Clinton is a single unit plant.

Regulatory Guide 1.82, Rev. 2 (May 1996)

Water Sources for Long-Term Recirculation Cooling
Following a Loss-of-Coolant Accident

Project Position - Comply.

USAR Subsections - 6.2.2.2, 6.3.2.2

Regulatory Guide 1.83, Rev. 1 (July 1975)

Inservice Inspection of PWR Steam Generator Tubes

Project Position - Not applicable to BWRs.

Regulatory Guide 1.84

Design and Fabrication Code Case Acceptability
ASME Section III Division I

Project Position

1. CPS will apply all applicable approved code cases, not limited to those listed in the latest revision of Regulatory Guide 1.84, as found necessary and proper for the construction activities.

   a. N-237 - Hydrostatic Testing of Internal Piping, Section III, Division 1.

      Exemption from hydrostatic testing of the portion of ASME Class 2 containment spray piping system extending downstream from the containment penetration to the containment spray device

      Reference is made to NRC's letter Docket Nos. 50-461/462 dated August 18, 1980, with expressed comment that this code case requires the examination of the exempted piping in accordance with NB 5200 even though the piping is ASME Class 2.

   b. N-241 - Hydrostatic Testing of Piping, Section III, Division 1.

      Exemption from hydrostatic testing of safety relief valve discharge piping from the main steam, HPCS, LPCS, RHR, and RCIC systems to the suppression pool; including the submerged portions and terminations.

      Reference is made to NRC's letter Docket Nos. 50-461/462 dated August 18, 1980, which clarified the code case applicability only to the submerged portions of the above-cited systems.

   c. N-240 - Hydrostatic Testing of Open Ended Piping, Section III, Division 1.

      Exemption from hydrostatic testing of:

      (1) Shutdown service water discharge to the ultimate heat sink beyond the last isolation valve.

      (2) Various diffusers, drains, and return piping to the several fuel pools in the containment and fuel buildings.

      (3) Suppression pool makeup system piping connecting upper containment pool with suppression pool.

      (4) Diesel oil system pump suction piping from and return piping to the diesel generator fuel oil storage tanks and the fuel oil day tanks.

      (5) Standby liquid control system pump suction piping from storage tank to first isolation valves.

<u>Regulatory Guide 1.84</u> (Cont'd)

(6) Pump suction and return piping to the RCIC storage tank.

(7) HPCS, LPCS, RHR, RCIC, and suppression pool cleanup pump suction piping from suppression pool to first isolation valve outside containment and return lines to the suppression pool.

The approval to use Code Case N-240, "Hydrostatic Testing of Open Ended Piping, Section III, Division 1," was authorized by NRC for construction of above cited components with no limitations by letter Docket Nos. 50-461/462 dated February 25, 1980.

d. N-341 - Certification of Level III NDE Examiner Section III, Divisions 1 and 2.

The approval to use Code Case N-341 without limitations was granted by NRC letter Docket No. 50-461 dated April 11, 1983.

e. N-315 - Repair of Bellows, Section III, Division 1.

Repair of bellows under ASME Section III, Division 1 for ten guard pipe expansion bellows at CPS. The approval for use of Code Case N-315 was authorized by NRC letter Docket No. 50-461 dated September 27, 1983 with the condition that CPS submit the description of the repair as well as justification for repairing the bellows rather than replacing them. Following receipt of NRC's approval for the repair, but prior to making the repair, CPS is to provide NRC the results of the qualification on the full-scale facsimile bellows, including the design requirements to assure that the repair meets the requirements of the design specification.

2. Illinois Power Company requested and received approval for application of the following code cases:

a. N-356 - Certification Period for Level III NDE Personnel Section XI, Divisions 1, 2, and 3.

The approval to use Code Case N-356 was authorized by NRC letter Docket No. 50-461 dated April 11, 1983 with no limitations.

b. N-397 - Alternate Rules to the Spectral Broadening Procedures of N-1226.3 for Classes 1, 2, and 3, Section III, Division 1.

The approval to use Code Case N-397 was authorized by NRC letter Docket No. 50-461 dated April 5, 1985 with no additional requirements other than those specified in the code case.

c. N-411 - Alternate Damping Values for Seismic Analysis of Piping Section III, Division 1, Class 1, 2 and 3 Construction.

The approval of Code Case N-411 was conditionally granted by NRC letters dated July 19, 1985 and April 5, 1985. This code case will be used

<u>Regulatory Guide 1.84</u> (Cont'd)

for piping systems analyzed by response spectrum methods and not those using time-history analysis methods. This code case will be used, as necessary, for any future piping and equipment dynamic analysis or reanalysis. If, as a result of using the damping value in ASME Code Case N-411, piping supports are moved, modified or eliminated, any increased piping displacements due to the greater piping flexibility will be checked to assure that they can be accommodated and that there will be no adverse interaction with adjacent structures, components, or equipment. When the alternative damping values of this code case are used, they will be used in their entirety in a given analysis and not a mixture of Regulatory Guide 1.61 and Code Case N-411.

This code case will be applied for both hydrodynamic loads as well as seismic loads; however, the damping values in Code Case N-411 are limited to a response frequency below 33 Hz.

d. N-413 - Minimum Size of Fillet Welds for Linear Type Supports, Section III, Division 1, Subsection NF.

The approval to use Code Case N-413 was authorized by NRC letter Docket No. 50-461 dated April 30, 1985 with no limitations other than those stated in the text of the code case.

3. Acceptable code cases annulled by action of the ASME Council (or deleted in later revisions to this guide), but specified for procurement or other activities, shall remain valid.

Commitment to meet a specific revision of this guide has little significance since the guide is revised as new code cases are issued by the ASME and approved by the NRC.

<u>USAR Subsection</u> - 5.2.1.2

Regulatory Guide 1.85

Materials Code Case Acceptability
ASME Section III Division I

Project Position

1. CPS will apply all applicable approved code cases, not limited to those listed in the latest revision of the Regulatory Guide 1.85, as found necessary and proper for the construction activities of the CPS.

2. Acceptable code cases annulled by action of the ASME Council (or deleted in later revisions to this guide), but specified for procurement, shall remain valid.

Commitment to meet a specific revision of this guide has little significance since the guide is revised as new code cases are issued by the ASME and approved by the NRC.

USAR Subsection - 5.2.1.2

Regulatory Guide 1.86, Rev. 0 (June 1974)

Termination of Operating Licenses for Nuclear Reactors

Project Position - Comply.

Reference - License Application

Regulatory Guide 1.87, Rev. 1 (June 1975)

Guidance for Construction of Class 1 Components
in Elevated-Temperature Reactors

Project Position - Not applicable to BWRs.

Regulatory Guide 1.89, Rev. 0 (November 1974)

Qualification of Class 1E Equipment for Nuclear Power Plants

Project Position - Comply

USAR Section/Subsections - 3.11, 7.3.2.20.2.1.8, 7.6.2.5.5, 8.1

Regulatory Guide 1.90, Rev. 1 (August 1977)

Inservice Inspection of Prestressed Concrete
Containment Structures with Grouted Tendons

Project Position - Not applicable; Clinton Power Station has reinforced concrete containments.

Regulatory Guide 1.91, Rev. 1 (February, 1978)

Evaluations of Explosions Postulated to Occur on
Transportation Routes Near Nuclear Power Plants

Project Position - Comply.

USAR Subsection - 2.2.3

Regulatory Guide 1.92, Rev. 1 (February 1976)

Combining Modal Responses and Spatial Components in
Seismic Response Analysis

Project Position - The project complies with NRC Regulatory Guide 1.92 with the following clarification:

> In combining the modal response using double sum method, a corrected damping factor β_k' is used for computing the damped frequency ω_k' of a system or a subsystem with closely spaced modes, while an uncorrected damping factor β_k is used per the regulatory guide. The justification for using β_k' instead of β_k in the calculation of ω_k' is presented in the attached "Supplement to the Position on Regulatory Guide 1.92".

USAR Section/Subsection - 3.7.2.7, 3.7.2.12, 3.7.3.7, 3.7.3.8

Supplement to Position on Regulatory Guide 1.92

In the double sum method of modal combination, a modified damping factor, instead of an uncorrected value as in Equation (10) of Regulatory Guide 1.92, should be used for the damped frequency to evaluate the correlation coefficient of the closely space modes.

Equations (8), (9), (10) and (11) of the regulatory guide are based on a study by Rosenblueth and Elorduy (Reference 1) Referring to that paper, for a single-degree-of-freedom system governed by the equation of motion,

$$\ddot{q}(t) + 2\xi_1\omega_1 \dot{q}(t) + \omega_1^2 q(t) = \ddot{x}(t) \qquad (1)$$

The correction factor for damping can be expressed as (Equation (4) of Reference 1)

$$\frac{E(Q)}{E(Q)} = (1 + \xi_1 \omega_1 s / 2)^{1/2} \qquad (2)$$

where s is the duration of a segment white noise excitation. $E(Q)$ and $E(Q_0)$ are the expected value of the damped and the undamped systems.

The maximum response of a system to a transient disturbance of form $\ddot{x}(t)$ = f(t) W(t) can be expressed (Equation (8) of Reference 1)

$$Q^2 \;\mu \int_0^\infty \Psi \;\; q^2 \; dt \qquad (3)$$

The transfer function q(t) for the deformation of the system expressed by Equation (1) is (Equation 10.3 of Reference 2)

$$y\; q(t) = \frac{-1}{\omega_1'} \;\; \exp\;(-\xi_1\, \omega_1 t)\;\; \sin\;\; \omega_1' t \qquad (4)$$

where

$$\omega_1' = \omega_1 \sqrt{1 - \xi_1^{\,2}} \qquad (5)$$

When q is the pseudo-velocity of a single-degree system, the second member in Equation 4 gives 1/2 $\xi_1\ \omega_1$.

In order to adjust the percentage of damping to coincide with the expected response, Rosenblueth suggested the use of a modified damping factor (Equation (9) of Reference 1)

$$\xi_1' = \xi_1 + 2/\omega_1 s \qquad (6)$$

in the system's natural mode of vibration.

In other words, the uncoupled equation of motion of a multi-degree-of-freedom system should be adjusted as

$$\ddot{q}_i(t) + 2\omega_i\xi'_i \dot{q}_i(t) + \omega_i^2 q_i(t) = a_i \ddot{x}(t) \tag{7}$$

and the transfer function is given by (Equation 10.3 of Reference 2)

$$\Psi q(t) = -\sum_i \frac{a_i}{\omega_i'} \exp(-\xi'_i \omega_i t) \sin \omega_i' t \tag{8}$$

where

$$\omega_i' = \omega_i \sqrt{1-\xi_i'^2} \tag{9}$$

Note here the modified damping factor ξ_i' and not the uncorrected damping value ξ_i is used in Equation (9) for the damped frequency of the adjusted system.

The final solution is then obtained based on the transfer function of Equation (8) but not Equation (4), as

$$Q^2 = \sum_i Q_i^2 + \sum_{i \neq j} \sum Q_i Q_j / (1+\varepsilon_{ij}^2) \tag{10}$$

where

$$\varepsilon_{ij} = \left|\omega_i' - \omega_j'\right| / (\xi_i' \omega_i + \xi_j' \omega_j) \tag{11}$$

$$\omega_i' = \omega_i \sqrt{1-\xi_i'^2} \tag{12}$$

Thus, a modified damping factor for the damped frequency should be used. Amin and Gungor (Reference 3) and Singh, Chu and Singh (Reference 4) also used the modified damping factor for the damped frequency in their computation of the correlation coefficent of closely spaced modes.

For a lightly damped system and an earthquake duration of 10 seconds as in the Clinton design basis, the damped frequency based on a modified damping factor and the damped frequency based on a uncorrected damping factor are approximately the same. However, on a theoretical basis, the modified damping factor should be used for the damped frequency in the evaluation of the correlation of the closely spaced mode response. For a 10 Hz system with 2% damping and 10 seconds earthquake duration, the damped frequency using modified and uncorrected damping factors is 9.9973 Hz and 9.9980 Hz respectively. Thus the use of either modified or uncorrected damping factor does not affect the results.

Supplement to Position on Regulatory Guide 1.92 (Cont'd)

References

1. E. Rosenblueth and J. Elorduy, "Response of Linear Systems to Certain Transient Disturbances" Proceedings, Fourth World Conference on Earthquake Engineering, Vol. 1 Santiago, Chile, 1969.

2. N. M. Newmark and E. Rosenblueth, "Fundamentals of Earthquake Engineering", Prentice Hall Inc., 1971.

3. M. Amin and I. Gungor, "Random Vibration in Seismic Analysis - an Evaluation", ASCE National Structural Engineering Meeting, Baltimore, Maryland, April 1971.

4. A. K. Singh, S. L. Chu and S. Singh, "Influence of Closely Spaced Modes in Response Spectrum Method of Analysis", Proceedings of the Special Conference on Structural Design of Nuclear Plant Facilities, Chicago, IL, December 1973.

Regulatory Guide 1.93, Rev. 0 (December 1974)

Available Electric Power Sources

Project Position - The project complies with the requirements of Regulatory Guide 1.93 with the following exception:

Reference: Paragraphs C.1 and C.5 - The requirements provided in these paragraphs concerning the time limit for inoperability is not followed for:

a) the Division 3 diesel generator and the Division 3 and 4 batteries. This exception is based on the facts that the only load on Division 3 is the high pressure core spray system (HPCS), and the Division 4 battery support for HPCS initiation. The CPS Technical Specifications require HPCS power source availability similar to the requirements contained in NRC Standard Technical Specification (NUREG 1434).

b) the Division 1 and 2 diesel generators have allowed outage time (AOT) of 14 days rather than 72 hours and this time may be used for preventive maintenance. The basis for this change is risk-informed Techical Specification Amendment 141.

USAR Subsection - 8.1.6.1.17; TS3.8.1, 3.8.4

Regulatory Guide 1.96, Rev. 1 (June 1976)

Design of Main Steam Isolation Valve Leakage Control
Systems for Boiling Water Reactor Nuclear Power Plants

Project Position - Partially Comply. As a result of the re-analysis of the Loss-of-Coolant Accident (LOCA) using Alternative Source Term (AST) Methodology, it is no longer necessary to credit the Main Steam Isolation Valve Leakage Control System (MSIVLCS) for post-LOCA activity leakage mitigation. The system has been left in place as a passive system and is not required to perform any safety function.

USAR Subsection - 5.4.7.1.1.6.1, 6.7.1.2

Regulatory Guide 1.97, Rev. 3 (May, 1983)

Instrumentation for Light-Water-Cooled Nuclear Power Plants
to Assess Plant Conditions During and Following an Accident

Project Position - The project complies with the requirements of Regulatory Guide 1.97 with the clarifications and exceptions itemized in Table 7.1-13. In accordance with the SER to TS Amendment 164 Containment Hydrogen Monitoring is changed from a Category 1 variable to a Category 3 variable.

USAR Tables - 7.1-13, 7.1-14, 7.5-1

USAR Sections - 7.1.2.6.23, 7.3.2.20.2.1.9, 7.6.2.12.4, 7.6.2.12.5, 7.7.1.26.3.6, 9.3.7.1.2, 12.5.2, Appendix D

Regulatory Guide 1.98, Rev. 0 (March 1976)

Assumptions Used for Evaluating the Potential Radiological Consequences
of a Radioactive Offgas System Failure in a Boiling Water Reactor

Project Position - Comply

USAR Subsection - 15.7.1

Regulatory Guide 1.99, Rev. 2 (May 1988)

Effects of Residual Elements on Predicted
Radiation Damage to Reactor Vessel Materials

Project Position - Comply

USAR Subsections - 4.1.4.5, 4.3.2.8, 5.3.1.4, 5.3.1.5, 5.3.1.6, 5.3.2.1, and 5.3.2.2

Regulatory Guide 1.100, Rev. 1 (August 1977)

Seismic Qualification of Electric Equipment for Nuclear Power Plants

Project Position - Comply

USAR Section - 3.10, 6.2.5.2.3, 7.3.2.20.2.1.10, 7.6.2.5.5, 8.1.6.1.18, 8.3.1

Regulatory Guide 1.101, Rev. 2 (October 1981)

Emergency Planning for Nuclear Power Plants

Project Position - Comply

USAR Section - 13.3, E4.0.B.5

Regulatory Guide 1.102, Rev. 1 (September, 1976)

Flood Protection for Nuclear Power Plants

Project Position - Comply.

USAR Subsection - 3.4.1

Regulatory Guide 1.103, Rev. 1 (October 1976)

Post-Tensioned Prestressing Systems for Concrete
Reactor Vessels and Containments

Project Position - Since Clinton Power Station has reinforced concrete containments and steel reactor vessels, this regulatory guide does not apply.

Regulatory Guide 1.104 has been withdrawn

Regulatory Guide 1.105, Rev. 1 (November 1976)

Instrument Setpoints

Project Position - The Project complies with NRC Regulatory Guide 1.105 with the following clarification:

> Reference: Paragraph C.5 - CPS shall comply with the regulatory positions established in this regulatory guide. However, some equipment setpoint adjustments are not mechanical and, therefore, do not have mechanical securing devices. The equivalent of mechanical securing devices are provided in the equipment design.

USAR Section - 7.1.2.6.25, 7.3.2.20.2.1.11, 7.6.2.5.5, TS3.3, ORM Att. 2

Regulatory Guide 1.106, Rev. 1 (March 1977)

Thermal Overload Protection for Electric Motors on Motor-Operated Valves

Project Position - Comply with the following clarification:

> Reference: Paragraph c.1(a) - CPS complies with the intent of this regulatory guide in that thermal overload protection is continuously bypassed with the exception that it may be placed into service for short periods of time during valve maintenance, testing, and repositioning during normal plant operation. This action is limited by the requirements of ORM 2.5.2. Therefore, automatic actuation devices to bypass the thermal overloads during accident conditions are not required. As such, the CPS program for control of MOV thermal overload protection devices complies with the requirements of Regulatory Guide 1.106 to prevent overload protection circuitry from inhibiting the ability of MOVs to perform their safety function.

USAR Section - 8.1.6.1.19; ORM 2.5.2, 3.5.2, 4.5.2.1, 4.5.2.2, and 5.5.2.

Regulatory Guide 1.107, Rev. 1 (February 1977)

Qualifications for Cement Grouting for Prestressing
Tendons in Containment Structures

Project Position - Not applicable since Clinton Power Station does not have prestressed tendons in its containment structures.

Regulatory Guide 1.108, Rev. 1 (August 1977)

Periodic Testing of Diesel Generator Units Used as Onsite Electric Power Systems at Nuclear Power Plants

Project Position - Comply with the following clarifications:

1) Reference: Section C.1.b(3) - the Division 3 diesel generator design includes override capability to ensure automatic switchover from the test mode to ready-to-load operation in response to a loss-of-coolant-accident (LOCA) initiation signal. However, during testing with a non-zero droop setting in effect (to support paralleling the diesel generator with the offsite power source), in the event of a LOCA initiation signal concurrent with a loss of the offsite power source to the bus, operator action may be required (in addition to the automatic actions) to reset the governor and thus ensure bus frequency is within required limits when the diesel generator alone is subsequently supplying power to the Division 3 bus.

2) Reference: Section C.2 - Periodic testing of diesel generator units will be as required by the Technical Specifications. Reporting of diesel generator unit test failures will be as required by the Operational Requirements Manual.

3) Reference: Section C.2.c - Tests, such as the largest load rejection test, full load rejection test, 24-hour run, and the test mode override test, may be performed during normal plant operations, as well as during plant shutdown. (Reference License Amendment 132).

USAR Section/Subsection - 8.3.1; TS 3.8.1, 3.8.2; ORM 6.9.2.1

Regulatory Guide 1.109, Rev. 1 (October 1977)

Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I

Project Position - Comply with the following clarification:

Reference: Paragraph C.2 - The effect of a finite cloud and elevated plume will be considered for all releases meeting the criteria of Regulatory Guide 1.111, Revision 1, Paragraph C.2.b. This effect will not be limited to stacks more than 80 meters high.

USAR Sections - 11.2, 11.3, 15.6.5.5.1, TS 5.6.3, and ODCM

Regulatory Guide 1.110, Rev. 0 (March 1976)

Cost-Benefit Analysis for Radwaste Systems for
Light-Water-Cooled Nuclear Power Reactors

Project Position - Clinton Power Station has opted to comply with Annex to 10 CFR 50, Appendix I. Hence, this regulatory guide is not applicable.

Regulatory Guide 1.111, Rev. 1 (July 1977)

Methods for Estimating Atmospheric Transports and Dispersion of Gaseous
Effluents in Routine Releases from Light-Water-Cooled Reactors

Project Position - Comply with the following clarification:

> Reference: Paragraph C.3.a - Radio-decay will be considered individually for each nuclide. "Conservative" estimates considered here are unnecessary.

USAR Sections - 2.3, 11.3, ODCM

Regulatory Guide 1.112, Rev. O-R (May 1977)

Calculation of Releases of Radioactive Materials in Gaseous and
Liquid Effluents from Light-Water-Cooled Power Reactors

Project Position - Comply

USAR Subsections - 11.2.3, 11.3.3

Regulatory Guide 1.113, Rev. 1 (April 1977)

Estimating Aquatic Dispersion of Effluents from Accidental and Routine
Reactor Releases for the Purpose of Implementing Appendix I

Project Position - Comply

USAR Subsections - 2.4.12, 11.2.3

Regulatory Guide 1.114, Rev. 2 (May 1989)

Guidance on Being Operator at the Controls of a Nuclear Power Plant

Project Position - Comply with the following clarification:

Reference: Footnote 2 - This footnote defines operational control panels as those that enable the operator at the controls to perform required manual safety functions and equipment surveillance and to monitor plant conditions under normal and accident conditions. Unobstructed view of and access to these panels is required.

The nature of the CPS design is such that certain surveillance and monitoring actions not requiring prompt corrective action will be conducted in back row panels.

USAR Section - 13.5

Regulatory Guide 1.115, Rev. 1 (July, 1977)

Protection Against Low-Trajectory Turbine Missiles

Project Position - Comply.

USAR Subsection - 3.5.1.3

Regulatory Guide 1.116, Rev. O-R (June 1976)

Quality Assurance Requirements for Installation, Inspection
and Testing of Mechanical Equipment and Systems

Project Position - Comply with the following clarification:

As committed to in the PSAR, Installation, Inspection and Testing of Mechanical Equipment and Systems during the design and construction of CPS was in compliance with ANSI N45.2.8 (Draft 3, Rev. 3 April, 1974), as endorsed by WASH 1309, UC-80. The draft standard was replaced by ANSI N45.2.8 - 1975 of the subject Regulatory Guide.

Superceded by ASME, NQA-1 (1994) Subpart 2.8.

USAR Chapters - 14, 17

Regulatory Guide 1.117, Rev. 1 (April, 1978)

Tornado Design Classification

Project Position - The project complies with the requirements of Regulatory Guide 1.117 with the following clarifications:

The discussion contained in Regulatory Guide 1.117 states that protection of designated structures, systems, and components may generally be accomplished by designing protective barriers to preclude tornado damage, and if protective barriers are not installed, the structures and components themselves should designed to withstand the effects of the tornado, including tornado missile strikes.

Important systems and components (as discussed in Regulatory Guide 1.117) are generally protected. The limited amount of unprotected portions of important systems and components are analyzed using a probabilistic missile strike analysis consistent with the acceptance criteria in Standard Review Plan 3.5.1.4, Missiles Generated By Natural Phenomena.

USAR Subsection - 3.5.1.4, 3.5.2.4, and 3.5.2.5

Regulatory Guide 1.118, Rev. 2 (June 1978)

Periodic Testing of Electric Power and Protection Systems

Project Position - The Project complies with NRC Regulatory Guide 1.118 with the following clarifications:

1) Reference: Paragraph C.6 - Trip of an associated protective channel or actuation of an associated Class 1E Load Group is required on removal of fuses or opening of a breaker only for the purpose of deactivating instrumentation or control circuitry.

2) Reference: Paragraph C.8.a - Appropriate state-of-the art technology will be implemented to periodically assure that the sensor (beginning at the sensor input), trip unit, logic, and actuator response times have not deteriorated so as to compromise the respective system design requirements.

3) Reference: Paragraph C.8.b - Test intervals, both initial and revised, should be such that significant changes in failure rates can be detected.

The Nuclear System Protection System has been designed to support the requirements of this guide.

USAR Chapters - 7 and 8

Regulatory Guide 1.119 has been withdrawn

<u>Regulatory Guide 1.120, Rev. 1 (November 1977)</u>

Fire Protection Guidelines for Nuclear Power Plants

<u>Project Position</u> - The fire protection guidelines for the Clinton Power Station were taken from the Branch Technical Position APCSB 9.5-1 Appendix A, "Guidelines for Fire Protection for Nuclear Power Plants Docketed prior to July 1, 1976". A complete evaluation of the projects' compliance with this Branch Technical Position is contained in Section 4.0 of the Clinton Power Station Fire Protection Evaluation Report.

<u>Reference</u> - Appendix E

<u>Regulatory Guide 1.121, Rev. 0 (August 1976)</u>

Bases for Plugging Degraded PWR Steam Generator Tubes

<u>Project Position</u> - Not applicable to BWRs.

<u>Regulatory Guide 1.122, Rev. 1 (February 1978)</u>

Development of Floor Design Response Spectra for Seismic
Design of Floor-Supported Equipment or Components

<u>Project Position</u> - This Regulatory Guide was first issued in September, 1976 while the date of construction permit for Clinton is February, 1976. However, the project complies with the intent of the Regulatory Guide 1.122 with the following clarification:

> Floor response spectra were generated at fifty periods lying between .02 to 2.0 seconds interval. These periods compare well with the recommended values of periods in Table 1 of the Regulatory Guide 1.122 and therefore the selected periods in Clinton project for floor response spectra generation are satisfying the intent of the Regulatory Guide 1.122.

<u>USAR Subsections</u> - 3.7.2.5, 3.7.2.5.1, 3.7.2.5.2, 3.7.2.5.3

Regulatory Guide 1.124, Rev. 1 (January, 1978)

Service Limits and Loading Combinations for Class 1
Linear-Type Component Supports

Project Position - Comply.

USAR Subsection - 3.9.3

Regulatory Guide 1.125, Rev. 1 (October, 1978)

Physical Models for Design and Operation of Hydraulic
Structures and Systems for Nuclear Power Plants

Project Position - Physical hydraulic models were not used for hydraulic design of Clinton Power Station structures, therefore, Regulatory Guide 1.125 is not applicable.

Regulatory Guide 1.126, Rev. 1 (March 1978)

An Acceptable Model and Related Statistical Methods
for the Analysis of Fuel Densification

Project Position - GE meets the requirements of this regulatory guide with the exception of the method of derivation of the densification values for resintering tests.

GE believes that the NRC method in the regulatory guide represents a significant departure from previously approved GE methods.

Specifically, this departure requires an increase in the level of statistical confidence level from 50 percent to 95 percent employed to infer population parameters from the sampling results.

This departure from the GE approach previously approved by the Staff is interpreted as the introduction of additional conservatism in the GE densification analyses, and such a Staff position is considered inappropriate in recognition of the GE/NRC agreed upon conservatisms incorporated into the individual densification models during their formulation.

USAR Section - 4.2

Regulatory Guide 1.127, Rev. 1 (March 1978)

Inspection of Water-Control Structures Associated with Nuclear Power Plants

Project Position - The requirements of Regulatory Guide 1.127 are not specifically applicable to the Lake Clinton Dam since the dam is not required for the emergency cooling water system or flood protection of the Clinton Power Station. The requirements of Regulatory Guide 1.127 are, however, applicable to the submerged dam and baffle dike within the lake that forms the Ultimate Heat Sink. CPS complies with the Regulatory Guide 1.127 in respect to the submerged dam and baffle dike.

USAR Subsections - 2.4.11.6 and 2.5.6.8

Regulatory Guide 1.128, Rev. 1 (October, 1978)

Installation Design and Installation of Large
Lead Storage Batteries for Nuclear Power Plants

Project Position - The project complies with the NRC Regulatory Guide 1.128 with the following exceptions and clarification:

Exceptions:

1) Reference: Paragraph C.1 - CPS complies with the IEEE 484 requirement of limiting the hydrogen accumulation to less than two percent of the total battery area volume. The regulatory position of limiting the concentration to less than two percent at any location within the battery area would be impossible to verify.

2) Reference: Paragraphs C.4 & C.6.h. IEEE Standard 450-1995 revision is used in lieu of earlier revisions.

Clarification:

Reference: Paragraph C.2 - Compliance to Regulatory Guide 1.120 is addressed in USAR Subsection 9.5.1 and the Fire Protection Evaluation Report for Clinton Power Station.

USAR Section/Subsection - 8.1, 9.5.1, Appendix E.

Regulatory Guide 1.129, Rev. 1 (February 1978)

Maintenance, Testing, and Replacement of Large Lead
Storage Batteries for Nuclear Power Plants

Project Position - Conformance with this Regulatory Guide is provided by the maintenance and testing program described in the CPS Technical Specifications:

1) Reference: Paragraphs B,C, & C.2. IEEE Standard 450-1995, IEEE Recommended Practice for Maintenance, Testing, and Replacement of Vented Lead-Acid Batteries for Stationary Applications, is used in lieu of earlier revisions.

2) Reference: Paragraph C.1. Battery performance discharge tests are performed as required in CPS Technical Specifications 3.8.

USAR Section - TS 3.8.4, TS 3.8.5, TS 3.8.6, TS 5.5.14, USAR 8.3.2

Regulatory Guide 1.130, Rev. 1 (October 1978)

Service Limits and Loading Combinations for Class 1
Plate - and - Shell-Type Component Supports

Project Position - Comply

USAR Subsections - 3.8.3.5.5, 3.9.3

Regulatory Guide 1.131, Rev. 0 (August 1977)

Qualification Tests of Electric Cables, Field Splices and
Connections for Light-Water-Cooled Nuclear Power Plants

Project Position - Comply with respect to testing of cables. See section 8.1.6.24 for statements concerning qualification testing of UCI splicing tape.

USAR Subsections - 8.1.6

Regulatory Guide 1.132 Rev. 1 (March, 1979)

Site Investigations for Foundation of Nuclear Power Plants

Project Position - Most of the geotechnical site investigation work for Clinton Power Station was done prior to the original issuance of Reg. Guide 1.132 in September, 1977, therefore compliance was not possible. However, the work done complies with Reg. Guide 1.132 with the following exceptions:

(1) Reference: Paragraph C.2 - The coordinates for the borings are not shown on the boring logs. However, drawings showing the locations of these borings are provided and have the state plane coordinate system identified on them.

(2) Reference: Paragraph C.3 - Piezometers were not used to monitor the effects of dewatering for the main plant excavation. The principle reasons for not establishing a monitoring system were:

   1. The surrounding tills were impermeable and very little seepage was expected and obtained.

   2. The major dewatering problem would be the removal of rain water.

   3. There were no ground water users in the area of the excavation that would be affected.

(3) Reference: Paragraph C.5 - The boring programs for the Main Plant complex, the Emergency Core Cooling System (ECCS) pipeline, the Ultimate Heat Sink, and the Main Dam do not fully comply with the requirements as provided in Appendix C of this regulatory guide. However, the borings for the Main Plant Ultimate Heat Sink, and Main Dam were located and drilled to provide an adequate geologic cross-section of the respective areas. It was also felt that the borings located near the Main Plant provided an adequate description of the subsurface materials to be encountered along the ECCS pipeline. Therefore, additional borings along the pipeline route were not drilled.

(4) Reference: Paragraph C.6 - Continuous sampling of the soils encountered in the borings was performed in a few borings, but not to the extent required by this regulatory guide. As previously stated, the boring and sampling programs were established to provide adequate geologic information to design the structures.

USAR Subsections - 2.5.4, 2.5.5, 2.5.6

Regulatory Guide 1.133, Rev. 1 (May 1981)

Loose Part Detection Program for the Primary
System of Light-Water-Cooled Reactors

Project Position - This Regulaotry Guide is no longer a requirement for BWRs as accepted by NRC SER contained within NEDC-32975P-A February 2001.

USAR Subsection - N/A

Regulatory Guide 1.134, Rev. 2 (April 1987)

Medical Certification and Monitoring of Personnel Requiring Operating License

Project Position - Comply

USAR Section/Subsection - 13.1

Regulatory Guide 1.135 Rev. 0 (September, 1977)

Normal Water Level and Discharge at Nuclear Power Plants

Project Position - Comply.

USAR Section - 2.4

Regulatory Guide 1.136, Rev. 1 (October, 1978)

Material for Concrete Containments (Article CC-2000 of
the "Code for Concrete Reactor Vessels and Containments")

Project Position - This Regulatory Guide and the referenced Code, 1977 Edition, did not exist at the time the Construction Permit was obtained. Nevertheless, the project is not in conflict with the regulatory positions C.1 and C.2. Regulatory positions C.3 and C.4 do not apply.

USAR Reference - Appendix B

Regulatory Guide 1.137, Rev. 0 (January 1978)

Fuel Oil Systems for Standby Diesel Generators

Project Position - The Project complies with NRC Regulatory Guide 1.137 with the following clarifications and exceptions:

(1) Reference: ANSI N195-1976 Section 4, Paragraph (3) - The Diesel Fuel Oil System is located in the plant vital area except for the last portion of the fill and vent lines which extend outside the building.

(2) Reference: ANSI N195-1976 Section 5.4 – Stored fuel requirements for emergency diesel generators (EDGs) at CPS are determined using post-LOCA maximum electrical load demands for each EDG for 7 days versus the continuous rating or time-dependent diesel loading of each EDG with 10% margin.

(3) Reference: ANSI N195-1976 Section 6.3 and Appendix A - A duplex strainer is not provided in the transfer pump suction line to preclude a postulated loss of pump suction by strainer plugging. A Y-type strainer is provided in a recirculation line. The tank outlet nozzle is six inches above the bottom of the tank to prevent sludge from entering the line.

(4) Reference: ANSI N195-1976 Section 7.5 Paragraph 1 - A non-nuclear safety related strainer is provided in each fill line. There is a locked closed valve between the strainer and the storage tank to preclude entry of deleterious material.

(5) Reference: ANSI N195-1976 Section 8, Paragraph 1d - A high level alarm is not provided for the supply tanks. A high level alarm for the storage tank could be useful only when filling the tank. Since there are level indications which will be monitored during the filling, a high level alarm would not enhance the safety of the system.

(6) The fuel oil will be sampled and analyzed periodically to verify that its quality meets the diesel manufacturer's recommendations, the requirements of ASTM-D975-06b and the CPS Technical Specifications.

The methods and tests specified in ASTM-D975 will be followed when there are differences between ASTM-D975 and the diesel manufacturer’s recommendations. However, all the diesel manufacturer’s recommended limits will be met when they are relevant to the test methods specified in ASTM-D975.

(7) Reference: Paragraph C.1.g The Diesel Fuel Oil fill lines have a protective coating and an impressed type cathodic protection system, although the cathodic protection system may not meet the requirements of NACE Standard RP-01-69. This is adequate because the fill lines are not required for proper operation of the diesel generator units during post-LOCA maximum load demands. All other equipment in the Diesel Generator Fuel-Oil Storage and Transfer System do meet NACE Standard RP-01-69 requirements.

(8) Fuel oil will be sampled in accordance with ASTM D4057-95

(9) Reference: Paragraph C.2.f. The current 25% interval extension that is allowed by Technical Specification Surveillance Requirement (SR) 3.0.2 for the 10-year tank cleaning SR 3.8.3.6 removed by License Amendment 186 is transferred to this Section.

USAR Subsection - 9.5.4; TS 3.8.3 Q&R 040.33, Q&R 040.34

Regulatory Guide 1.138, Rev. 0 (April, 1978)

Laboratory Investigations of Soils for Engineering
Analysis and Design of Nuclear Plants

Project Position - Comply.

USAR Subsections - 2.5.4, 2.5.5, 2.5.6

Regulatory Guide 1.139, For Comment (May 1978)

Guidance for Residual Heat Removal

The Clinton plant is designed so that the RHR system complies with the intent of this guide except as follows:

1. Regulatory Position C.2.a, second sentence suggests alarms which the RHR system does not have.

2. Regulatory Position C.2.a, third and fifth sentences suggest independent and diverse interlocks. The RHR interlocks are not diverse.

3. Regulatory Position C.2.a, fourth sentence suggests that failure of a power supply should not cause any valve to change position. Failure of the Reactor Protection System instrument power bus in one division will cause automatic isolation of the RHR shutdown suction valve in that division.

The Clinton design complies with Regulatory Guide 1.139, by using the following alternate approaches to Regulatory Guide 1.139 position C-2a and C-5.

1. Position C.2.a, second sentence - The design provides for valve position indication in the control room. The addition of alarms to alert the operator that an RHR valve is open is not considered necessary because of the number of procedural and administrative controls that must be exercised in taking the reactor from a power mode to the RHR mode and because the reactor high pressure signals are permissively interlocked into the valve opening logic to disallow valve opening on reactor high pressure.

2. Position C.2.a, third and fifth sentences - The design complies with these suggestions by the use of independent interlocks and series valves powered by redundant power sources.

3. Position C.2.a, fourth sentence - The design complies with the intent of the Regulatory Guide (i.e., protect the low pressure piping). Upon failure of logic the RPS power supply, valves are signaled to close. However, division power would need to be available to implement the change in valve position to the close position.

USAR Subsection - 5.4.7

Regulatory Guide 1.140, Rev. 0 (March 1978)

Design, Testing, and Maintenance Criteria for Normal
Ventilation Exhaust System Air Filtration and Adsorption
Units of Light-Water-Cooled Nuclear Power Plants

Project Position - All of the equipment of the filter systems were specified and purchased prior to the issuance of Regulatory Guide 1.140. However, the design of the non-safety-related filter systems meet the intent of the requirements of this guide, except as noted below:

(1) Reference: C.1.a and C.1.b - The equipment and components (excluding charcoal and filter pads) are designed to withstand a maximum of 40 year integrated radiation dose and worst-case anticipated continuous service, rather than 40 years of continuous service.

(2) Reference: C.2.a - All of the exhaust systems contain ducts, dampers, fans, related instrumentation, prefilters and HEPA filters. Charcoal adsorbers are only used when iodine is anticipated to be present. Heaters are only used upstream of charcoal adsorbers when the potential exist for the air stream relative humidity to exceed 70%.

(3) Reference: C.2.d - For discussion of conformance to Regulatory Guide 8.8, see discussion under Regulatory Guide 8.8 in this section.

(4) Reference: C.2.3 - The exhaust systems are not directly connected to outdoors because they take their suction from inplant areas. Therefore, this section is not applicable.

(5) Reference: C.2.f - All filter housings are of welded construction. All the filter housings and associated ductwork are designed to meet the requirements of Section 4.12 of ANSI N509-1976, but were not leak tested.

(6) Reference: C.3.b - The HEPA filters are designed and constructed in accordance with ANSI N509-1976, Section 5.1. The filter banks are field tested in accordance with ANSI N510-1980. Further procurement of HEPA filters after January 1, 1986 shall be in accordance with ANSI N509-1980.

(7) Reference: C.3.e - Even though prefilter and HEPA filter bank in 0VW06SA/SB is higher than three HEPA filters and no permanent gallery is provided, the access areas are adequate to support servicing and maintenance of filters and lighting.

(8) Reference: C.3.f - Ductwork associated with non-safety related filter systems is not designed to post-LOCA pipe break loadings, or to exhaust wind conditions such as tornadoes. However, all ductwork in Seismic Category I buildings is seismically supported.

(9) Reference: C.3.h - The drywell purge charcoal adsorber shall be field leak tested in accordance with ANSI N510-1980. Viewports are provided to observe adequate fill of the charcoal adsorber.

Regulatory Guide 1.140, Rev. 0 (March 1978) (Cont'd)

(10) Reference: C.3.i - The system fans and motors, mounting, and ductwork connections are designed, constructed and tested in accordance with the intent of ANSI N509-1976.

(11) Reference: C.3.l - The dampers were designed in accordance with the intent of ANSI N509-1976.

(12) Reference: C.4.b - A clearance of approximately two feet, instead of three feet, has been provided between the upstream edges of the HEPA filter mounting frames and the adjacent upstream components of filter units 0VQ01SA/SB/SC. This is acceptable since a minimum 4 feet 7 inch clearance has been provided for maintenance of these filters from the downstream side.

(13) Reference: C.6.a - The activated carbon is in accordance with Regulatory Guide 1.140, except that it is tested to the requirements of ANSI N509-1980, Table 5-1.

(14) In addition, the CPS "ANSI N509/510 Variance Report," HVAC-02-CP identifies minor deviations to this Regulatory Guide referenced ANSI standards. This Variance Report is a controlled document, maintained by the Nuclear Station Engineering Department.

USAR Subsections - 9.4.7.2, 9.4.9, 9.4.11, 9.4.13

Regulatory Guide 1.141, Rev. 0 (April 1978)

Containment Isolation Provisions for Fluid Systems

Project Position - Comply with the following clarification:

1. Reference ANSI N271-1976 Paragraph 4.2.3 - A short circuit in the valve indication circuit would cause the fuse at the motor control center to open, thus rendering the valve electrically inoperable.

   This is an acceptable condition since only a single failure in one Electrical Division need be considered. Inboard and outboard isolation valves have different Divisional assignments so that containment isolation can be accomplished even considering the loss of one of the two valves.

2. Reference Regulatory Guide Position C.2 - Valve Numbers IE12-F008, 1CC071, 1CC072, ICC073, 1CC074, 1CY020, 1CY021, 1FP051, 1FP054, 1FP078 and 1FP079 will have power removed by locking the circuit breaker in the open position. During the time that power is removed from the valve it will be considered as a manual valve and will not have position indication in the Main Control Room. During times that power is applied and the valve is required to be open, valve position indication will be provided.

USAR Section - 6.2.6, TS3.6.1.3

Regulatory Guide 1.142, Rev. 0 (April, 1978)

Safety Related Concrete Structures for Nuclear Power Plants
(Other than Reactor Vessels and Containments)

Project Position – Comply with the following clarifications:

1. R.G. 1.142 generally endorses ACI 349-76. However, the Regulatory Guide does not endorse the provisions of ACI 349-76 as adequate for drywell structures. CPS design utilizes the provisions of Sections 10.6.3 of ACI 349-80 in place of the 1976 version of the standard.

2. In some cases, CPS utilized the bend test requirements of ASTM A615 in place of ACI 349. Additional detail on this clarification is provided in USAR Section 3.8.3.2.2.

USAR Subsections - 3.8.3, 3.8.4, 3.8.5.

Regulatory Guide 1.143, Rev. 0 (July 1978)

Design Guidance for Radioactive Waste Management Systems,
Structures, and Components Installed in Light-Water
Cooled Nuclear Power Plants

Project Position - The Project complies with this guide with the following exceptions and clarifications:

1. Reference: Section B, Page 2, First Paragraph, First Sentence - It is clarified that the radwaste systems for CPS are considered to begin with and include the interface valves.

2. Reference: Section B, Page 2, First Paragraph, Second Sentence - It is clarified that the radwaste system for CPS terminates at the end of the pipe containing the last isolation valve before the point of controlled discharge to the service water discharge line, or at the end of the pipe containing the last isolation valve before the cycle condensate storage tank, or at the point of storage of packaged solid waste prior to shipment offsite to licensed burial ground.

3. Reference: Paragraphs C.1.1.2, C.2.1.2, C.3.1.2 Materials for pressure-retaining components conform to the requirements of ASTM Specifications.

4. Reference: Paragraph 4.3 - It is clarified that the scope of radwaste system pressure testing includes all pressure-retaining components, appurtenances, and completed systems. Bolts, studs, washers, gaskets, and possible localized instances of pump and valve packing are exempted from the pressure test. This is consistent with ASME Section III NB6000 and ANSI B31.1 (1983 edition). The Off-Gas System will be pneumatically tested at a minimum of 75 psig for no less than 30 minutes.

Portions of the radioactive waste management system contain polypropylene lined steel pipe and valves. These pipes and valves are in the demineralizer subsystem because of the superior corrosion resistance of polypropylene to chemicals. This portion of the subsystem will only be inservice leak tested at normal operating pressure.

5. Non-Category 1 equipment is evaluated per DC-ME-17-CP, Revision 1.

6. Plastic spacers are used on an as needed basis at flanged joints of plastic lined pipe. The spacers are provided by the manufacturer of the lined piping to adjust for small fit-up differences.

7. Reference: Paragraph C.1.2.1 - There are no local alarms to alert people in the area of potential overflow conditions. In lieu of this, the operator annunciator response procedure in the Radwaste Operations Center (ROC) has a required action to make a plant wide announcement regarding the potential overflow and alerting individuals in the area of potentially changing radiological conditions.

8. Reference: C.3.1 – Materials for non-pressure retaining attachments and appurtenances to solid waste tanks are not required to be constructed of materials conforming to the requirements of Section II of the ASME Boiler and Pressure Vessel Code. Examples of this type of equipment are nozzles, piping and welds in atmospheric tanks which do not carry process fluid and whose elevation is above the tank overflow level. In addition, these non-pressure retaining attachments and appurtenances are exempted from the design, construction and testing criteria set forth in this section, section C.4 and Table 1 to this regulatory guide. Failure of these components would not result in a radioactive release.

USAR Sections - 11.2.1, 11.3.2.2.1, 11.4.2, 11.5.1.1.2

Regulatory Guide 1.147

Inservice Inspection Code Case Acceptability ASME Section XI Division 1

Project Position - Comply with the following understandings:

1. Application to the Commission for acceptance of selected code cases issued after the latest revision of the guide may be made.

2. Acceptable code cases annulled by action of the ASME council (or deleted in later revisions to this guide), but specified in the Clinton Power Station Preservice, or Inservice, Inspection Program, as applicable, shall remain valid.

Commitment to meet a specific revision of this guide has little significance since the guide is revised as new code cases are issued or deleted by the ASME and its actions approved by the NRC.

USAR Subsection - 5.2.4, 6.6.

Regulatory Guide 1.149 Revision 4 (April 2011)

Nuclear Power Plant Simulation Facilities
For Use In Operator Training, License Examinations, and Applicant Experience Requirements

Project Position - Comply

USAR Subsection - 13.2.1

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<u>Regulatory Guide 1.150, Rev. 1 (February 1983)</u>

Ultrasonic Testing of Reactor Vessel Welds During
Preservice and Inservice Examinations

<u>Project Position</u> - Comply with alternative methods as described in Appendix A of this Guide.

<u>USAR Subsection</u> - 5.2.4

<u>Regulatory Guide, 1.155 (August 1988)</u>

Station Blackout

<u>Project Position</u> - Illinois Power submitted its compliance to the Station Blackout Rule, 10CFR50.63, "Loss of All Alternating Current Power," in letters to the NRC dated April 16, 1989; March 30, 1990; May 17, 1990; July 6, 1992; October 29, 1992; and December 22, 1992.

<u>USAR Subsection</u> - Table 8.1-3

<u>Option B of NEI 94-01 Rev. 2-A</u>

Performance-Based Containment Leak-Test Program

<u>Project Position</u> - Comply with the following clarification:

> Bechtel Topical Report BN-TOP-1 is also an acceptable option for performance of Type A tests.

<u>USAR Subsection</u> - 6.2.6

<u>Regulatory Guide 1.181, Rev. 0, (September 1999)</u>

Content of the Update Final Safety Analysis Report
in Accordance with 10 CFR 50.71(e)

<u>Project Position</u> - Comply

<u>USAR Subsection</u> - 1.1.9.2, 1.1.9.7

<u>Regulatory Guide 1.182, Rev. 0 (May 2000)</u>

Assessing and Managing Risks Before

Maintenance Activities at Nuclear Power Plants

<u>Project Position</u> - Comply

<u>USAR Subsection</u> - 16

<u>Regulatory Guide 1.183, Rev. 0 (July 2000)</u>

Alternative Radiological Source Terms for Evaluating Design Basis Accidents
At Nuclear Power Reactors

<u>Project Position</u> – Comply

<u>USAR Subsection</u> – 15.4.9, 15.6.4, 15.6.5, 15.7.4

Regulatory Guide 1.196, Rev. 0 (May 2003)

Control Room Habitability At Light-Water Nuclear Power Reactors

Project Position – The project only complies with Section C. 2.7.3, Degraded and Nonconforming Conditions of Regulatory Guide 1.196 Revision 0.

The Control Room Envelope Habitability Program is governed by Technical Specification 5.5.15 approved under License Amendment No. 178. Technical Specification Program 5.5.15, "Control Room Envelope Habitability Program" is a result of a commitment of the response to NRC Generic Letter 2003-01 by implementing the requirements of the Consolidated Line Item Improvement Process for implementation of TSTF-448 Revision 3. The Control Room Envelope Habitability Program was created as a result of findings at facilities that existing Technical Specifications may not be adequate to ensure the requirements of 10CFR50 Appendix A GDC 19 are met as described in Generic Letter 2003-01.

Survey of chemical sources is to be performed at least once per 6 years as part of the Periodic Assessment of the Control Room Envelope Habitability Program.

UFSAR Section – 6.4 and 9.4.1

Regulatory Guide 1.197, Rev. 0 (May 2003)

Demonstrating Control Room Envelope Integrity At Nuclear Power Reactors

Project Position – The project has only committed to the requirements for determining the unfiltered air inleakage past the Control Room Envelope boundary into the Control Room Envelope is in accordance with the testing methods and at the frequencies specified in Section C.1 and C.2 of Regulatory Guide 1.197, Revision 0 as described by Technical Specification 5.5.15, Control Room Envelope Habitability Program.

USAR Section – 6.4 and 9.4.1

Regulatory Guide 4.1, Rev. 1 (April 1975)

Programs for Monitoring Radioactivity in the Environs of Nuclear Power Plants

Project Position - Comply

USAR Subsection - 12.5.2

Regulatory Guide 4.13, Rev. 1 (July 1977)

Performance, Testing and Procedural Specifications for Thermoluminescent Dosimetry: Environmental applications

Project Position - Comply

ER Subsection: 6.2.5,

USAR Subsection: 12.5.2

Regulatory Guide 4.15, Rev. 1 (February 1979)

Quality Assurance for Radiological Monitoring Programs (Normal Operations) - Effluent Streams and the Environment

Project Position - Comply

USAR Subsection - 12.5.2, ORM 6.5.2.8

Regulatory Guide 5.66, Rev 0 (June 1991)

Access Authorization Program for Nuclear Power Plants

Project Postion - Comply

USAR Subsection – 13.1.1.1.3, 13.6

Regulatory Guide 8.1, Rev. 0 (February 1973)

Radiation Symbol

Project Position - Comply

USAR Subsection - 12.5.2

Regulatory Guide 8.2, Rev. 0 (February 1973)

Guide for Administrative Practices in Radiation Monitoring

Project Position - Comply with the following exception:

Licensee and contractor personnel will be processed to become radiation workers at Clinton Power Station without a radiation worker physical.

All Exelon personnel will receive a pre-employment physical equivalent to a Radiation Worker physical.

The industrial risks are not any higher than those experienced at fossil plants where contractor physicals are not provided. Additionally, a proficiently managed ALARA program should minimize the risk of potential litigation resulting from routine exposure to ionizing radiation.

Reference: ANSI N13.2-1969, Paragraph 4.7.1

USAR Subsections - 12.3.4.4.1, 12.5.2, 13.2.3

Regulatory Guide 8.3, Rev. 0 (February 1973)

Film Badge Performance Criteria

Project Position - This Regulatory Guide is not applicable at Clinton Power Station. Clinton Power Station uses Thermoluminescent Dosimeters in accordance with 10CFR20.1501(c).

Regulatory Guide 8.4, Rev. 0 (February 1973)

Direct Reading and Indirect - Reading Pocket Dosimeters

Project Position - Comply with the following exception:

1. Reference: Paragraph C.3 - In mixed radiation fields, more suitable and accurate dosimeters have been placed in service to determine neutron and gamma exposures; therefore, pocket dosimeters will not be used for this purpose.

USAR Subsection - 12.5.2

Regulatory Guide 8.5, Rev. 1 (March 1981)

Criticality and Other Interior Evacuation Signals

Project Position - The project complies with the following exception:

1) The NRC granted an exemption to the licensee of Clinton Power Station concerning 10CFR70.24 and the requirement to have a criticality alarm system in the fuel storage area (see pages 5 and 6 of Operating License NPF-62).

USAR Subsections - 9.5.2, 12.3.4, 12.5.2

Regulatory Guide 8.6, Rev. 0 (May 1973)

Standard Test Procedure for Geiger - Mueller Counters

Project Position - The project complies with Regulatory Guide 8.6 with the following clarification:

Operational testing and calibration of Geiger - Mueller type detectors is performed in accordance with American National Standards N323-1978 "Radiation Protection Instrumentation Test and Calibration," Sections 4.1, 4.2.2.1, 4.5, and 5.1. Source check frequency requirements shall be established by procedure. Source check acceptance ranges shall be consistent with applicable portions of ANSI N323-1978 or ANSI N320-1979, "Performance Specifications for Reactor Emergency Radiological Monitoring Instrumentation," depending on the intended application of the instrument.

USAR Subsections - 12.5.2, 12.5.3

Regulatory Guide 8.7, Rev. 1 (June 1992)

Instructions for Recording and Reporting Occupational Radiation Exposure Data

Project Position - Comply

USAR Subsection - 12.5.2

Regulatory Guide 8.8, Rev. 3 (June 1978)

Information Relevant to insuring that Occupational Radiation Exposures at Nuclear Power Stations will be as Low as Reasonably Achievable

Project Position - The project complies with Regulatory Guide 8.8 with the following clarifications and exceptions:

1) Reference: C.1.b(3) - NRC staff is considering certification by peer qualifications for RPM candidates. Should this be adopted, Clinton Power Station will take exceptions to this portion of regulatory guide.

2) Reference: C.2.a - Clinton Power Station's request concerning an exception to 10CFR 20.1601 to raise the radiation levels requiring locking to 1000 mrem/hr as opposed to the presently stated level of 100 mrem/hr was approved. Administrative controls will provide the effective control over ingress to areas greater than 100 mrem/hr.

3) Reference: C.2.c.(4) - The Clinton Project complies with this position in so far as practical. Wherever pressure gauges rather than transmitters are used, they are back flushed with clean water so as to reduce the potential for exposure at the gauge readout locations.

4) Reference: C.2.d.(3) - Temporary openings to exhaust ducts for local control of airborne contaminants during equipment maintenance are not provided, since use of these may result in imbalance of the exhaust system. Imbalance of the exhaust system may result in loss of airborne contamination control in adjacent areas.

5) Reference: C.2.g (2) (Ref. 10) - ANS/HPS 56.8 "Location and Design Criteria for Area Radiation Monitoring Systems for LWR" (draft) is identified by the Regulatory Guide as the reference for placement of detectors for optimum coverage of areas, however that document was never issued. Clinton Power Station will use ANSI/ANS-HPSSC 6.8.1-1981 "Location and Design Criteria for Area Radiation Monitoring Systems for Light Water Nuclear Reactors" with the following exception. Table 1 lists examples of locations for area radiation monitors; Clinton Power Station’s placement and function is described in Table 12.3-2.

6) Reference: C.2.h.(5) - Where use of tees cannot be avoided, tees are oriented in the branch horizontally or above the run as allowed by physical constraints.

7) Reference: C.2.(h) - ANS N197 and ANS 55.1 were not specifically considered in the Clinton Power Station Design.

In addition, CPS has committed to and will comply with the requirements of Regulatory Guide 8.8 (Revision 4) C.4.d(1) and (2). (Q&R 471:17)

USAR Section/Subsection - 12.1.2, 12.3 , 12.5.2, 13.2.3

Regulatory Guide 8.9, Rev. 1 (July 1993)

Acceptable Concepts, Models, Equations,
and Assumptions for a Bioassay Program

Project Position - Comply

USAR Subsections - 12.3.1, 12.5.2

Regulatory Guide 8.10, Rev. I-R (September 1975)

Operating Philosophy for Maintaining Occupational
Radiation Exposures As Low As Is Reasonably Achievable

Project Position - Comply

USAR Subsections - 12.1.3, 12.5.2, 13.2.3

Regulatory Guide 8.11, Rev. 0 (June 1974)

Applications of Bioassay for Uranium

Project Position - This guide does not apply to Clinton Power Station.

Regulatory Guide 8.12, Rev. 0 (December 1974)

Criticality Accident Alarm Systems

Project Position - Comply, with the following clarification:

> Criticality accident alarm systems are not provided in the areas of the spent fuel pools in the containment and the fuel buildings. Spent fuel is stored under water and in storage racks designed to maintain geometric spacing such that criticality is precluded. Also a criticality alarm system is not provided in the area of the new fuel storage vault. Design features and administrative controls preclude the possibility of an accidental criticality.

USAR Subsections - 12.3.4, 12.5.2

Regulatory Guide 8.13, Rev. 1 (November 1975)

Instruction Concerning Prenatal Radiation Exposure

Project Position - Comply

USAR Subsection - 12.5.2

Regulatory Guide 8.14, Rev. 1 (August 1977)

Personnel Neutron Dosimeters

Project Position - Not applicable to Clinton Power Station. Clinton Power Station uses Dosimeters of Legal Record (DLRs) and neutron sensitive radiation measuring instruments to measure neutron exposure to personnel. DLRs are certified for accuracy and sensitivity to neutron radiation in accordance with the National Voluntary Laboratory Accreditation Program (NVLAP).

Regulatory Guide 8.15, Rev. 1 (October 1999)

Acceptable Programs for Respiratory Protection

Project Position - Comply

USAR Subsection - 12.5.2

Regulatory Guide 8.19, Rev. 1 (June 1979)

Occupational Dose Assessment in Light-Water
Reactor Plants Design Stage Man-Rem Estimates

Project Position - Comply, with the exceptions and clarifications discussed in Subsection 12.4.4.

USAR Subsection - 12.4.4

Regulatory Guide 8.20, Rev. 1 (September 1979)

Applications of Bioassay For I-125 and I-131

Project Position - Not applicable at Clinton Power Station. Regulatory Guide 8.9 supersedes Regulatory Guide 8.20.

Regulatory Guide 8.25, Rev. 1 (June 1992)

Air Sampling in the Work Place

Project Position - Not applicable at Clinton Power Station.

Regulatory Guide 8.26, Rev. 0 (September 1980)

Applications of Bioassay for Fission and Activation Products

Project Position - Not applicable at Clinton Power Station. Regulatory Guide 8.9 supersedes Regulatory Guide 8.26.

Regulatory Guide 8.27, Rev. 0 (March 1981)

Radiation Protection Training For Personnel
at Light-Water Cooled Nuclear Power Plants.

Project Position – Comply with the following exception:

> Paragraph C.2.2 – The training program periodic refresher training frequency is determined and performed in accordance with plant procedures.

USAR Subsection - 12.5.2, 12.5.3.5

Regulatory Guide 8.28, Rev. 0 (August 1981)

Audible Alarm Dosimeters

Project Position - Comply with the following exception:

> The noted exception to Regulatory Guide 8.28, Paragraph C.2.c, involves the use of self performance checks for determining if electronic dosimetry is properly operating rather than use of a radiation source.

USAR Subsection - 12.5.2

Regulatory Guide 8.29, Rev. 0 (July 1981)

Instructions Concerning Risks from Occupational Radiation Exposures

Project Position - Comply

USAR Subsection - 12.5.2, 12.5.3.5

1.9 SYMBOLS USED IN ENGINEERING DRAWINGS

The symbols used in engineering drawings are shown in Drawings M05-1000, M05-1001, 197R567, 209A4756, 209A7367, and 921D280, and in K-2999 Standards STD-EC-110 and STD-EC-111.

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1.10 ACRONYMS

The following is a list of the acronyms used in the Clinton Power Station Updated Safety Analysis Report:

ABS	Absolute Sum (Method)
A-C(a-c)	Alternating Current
ACI	American Concrete Institute
ACRS	Advisory Committee for Reactor Safeguards
ADS	Automatic Depressurization System
AE	Architect Engineer
AISC	American Institute of Steel Construction
AISI	American Iron and Steel Institute
ALARA	As Low As Is Reasonably Achievable
ANS	American Nuclear Society
ANSI	American National Standards Institute
APED	Atomic Power Equipment Department (GE)
API	American Petroleum Institute
API	Automatic Priority Interrupt
APLHGR	Average Planar Linear Heat Generation Rate
APRM	Average Power Range Monitor
ARM	Area Radiation Monitor
AR/PR	Area Radiation/Process Radiation
ASCE	American Society of Civil Engineers
ASLAB	Atomic Safety & Licensing Appeals Board
ASLB	Atomic Safety & Licensing Board
ASME	American Society of Mechanical Engineers
ASTM	American Society for Testing Materials
AT	Current Transducer
ATS	Alarm Trip Setting (LPRM Channels)
ATWS	Anticipated Transients Without Scram
AWS	American Welding Society
B&PV	Boiler & Pressure Vessel Code
BA	Baldwin Associates
BOL	Beginning of Life (fuel cycle)
BOP	Balance of Plant
BPWS	Banked Position Withdrawal Sequence
BTP	Branch Technical Position (NRC)
BWR	Boiling Water Reactor
C&I	Controls & Instrumentation
CCW	Component Cooling Water
CFR	Code of Federal Regulations
CGCB	Containment Gas Control Boundary
CGCS	Combustible Gas Control System
CIT	Conductivity Indicator Transmitter
CMAA	Crane Manufacturing Association of America
CMFA	Common Mode Failure Analysis
COC	Certificate of Compliance
CP	Construction Permit
CPM	Critical Path Method
CPR	Critical Power Ratio
CPS	Clinton Power Station

CRD	Control Rod Drive
CRDA	Control Rod Drop Accident
CRDHS	Control Rod Drive Hydraulic System
CRO	Control Room Operator
CRPI	Control Rod Position Indication
CRT	Cathode Ray Tube
CRS	Conductivity Recording Switch
CRSI	Concrete Reinforcing Steel Institute
DAC	Derived Air Concentration
DAP	Data Acquisition Processor
DB	Design Basis
DBA	Design-Basis Accident
D-C (d-c)	Direct Current
DCP	Display Control Processor
DCS	Display Control System (PPCS Subsystem)
DDR	Deviation Disposition Request
DELS	Diesel Engine Lubrication System
DG	Diesel Generator (Diesel Engine-Generator)
DGSS	Diesel Generator Starting System
DLR	Dosimeter of Legal Record
DNB	Departure from Nucleate Boiling
DOP	Dioctyl Phthalate
DOT	U.S. Department of Transportation
DPF	Design Project Flood
DTS	Differential Temperature Switch
ECA	Engineering Change Authorization
ECCS	Emergency Core Cooling System(s)
ECN	Engineering Change Notice
EDS	Engineering Data Systems
EDT	Engineering Data Transmittal
EEP	Engineered Equipment Procurement
EFCV	Excess Flow Check Valve
EFDS	Equipment and Floor Drainage System
EHC	Electrohydraulic Control
EI	Engineering Instructions
EIC	Eberline Instrument Corporation
EIS	Engineering Information System
EOL	End of Life (fuel cycle)
E/P	Converter (Voltage/Pneumatic)
EP&C	Engineering Production and Control
EP&P	Engineering Practices & Procedures
ER	Environmental Report
ERAT	Emergency Reserve Auxiliary Transformer
ERDA	Energy Research and Development Administration
ESF	Engineered Safety Feature
EWA	Engineering Work Authorization
FA	Full Arc (mode of TCV operation)
FAI	Fail As Is
FAP	Fatigue Analysis Program
FC	Closes on Loss of Air or Electrical Power
FCD	Functional Control Diagram
FCF	First Called For

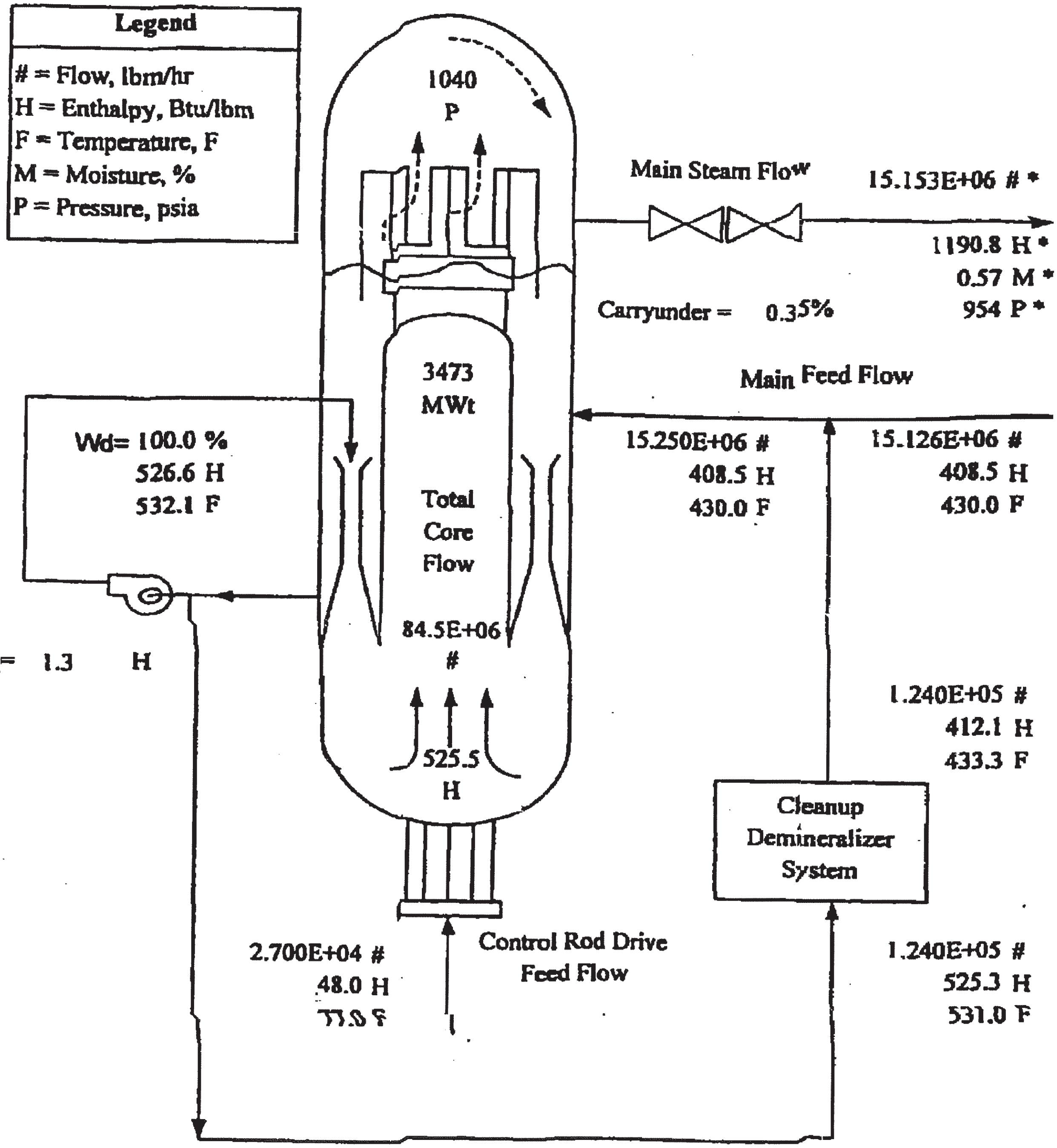
FCR	Field Change Request
FCV	Flow Control Valve
F/D	Filter Demineralizer
FDDR	Field Deviation Disposition Request
FDI	Field Disposition Instruction
FHA	Fuel Handling Accident
FIT	Flow Indicator Transmitter
FLECHT	Full-Length Emergency Cooling Heat Transfer
FM	Frequency Meter
FMEA	Failure Modes and Effects Analysis
FO	Opens on Loss of Air or Electrical Power
FPCC	Fuel Pool Cooling and Cleanup
FPS	Fire Protection System
FRCS	Flow Recording Controller Switch
FSAR	Final Safety Analysis Report
GDC	NRC General Design Criteria
GE	General Electric Company
GESSAR	General Electric Standard Safety Analysis Report
GETAB	General Electric Thermal Analysis Basis
GETSCO	General Electric Technical Services Company
H&V	Heating and Ventilating
HCU	Hydraulic Control Unit
HEPA	High-Efficiency Particulate Air/Absolute (referring to filters)
HHH	High-High-High
HPCS	High-Pressure Core Spray
HVAC	Heating, Ventilating, and Air-Conditioning
HX	Heat Exchanger
I&C	Instrumentation and Control
IAC	Interim Acceptance Criteria (NRC)
IB	Inboard
IBA	Intermediate Break Accident
ID	Inside Diameter
IDS	Instrument Data Sheet
IED	Instrument Electrical Diagram
IED	Instrument Engineering Diagram
IEEE	Institute of Electrical and Electronic Engineers
ILRT	Integrated Leak Rate Test
IO	Input/Output
IP or IPC	Illinois Power Company
IRM	Intermediate Range Monitor
ISI	Inservice Inspection
KEFF	Effective Neutron Multiplication Factor
LC	Lock Closed
LCD	Local Climatological Data
LCO	Limiting Condition of Operation
LCR	Logarithm of Count Rate
L/DRS	Level and Density Recorder Switch
LDS	Leak-Detection System
LFMG	Low Frequency Motor Generator
LFMGS	Low Frequency Motor-Generator Set
LHGR	Linear Heat Generation Rate

LIM SW	Limit Switch
LIRS	Level Indicator Recording Switch
LO	Lock Open
LOCA	Loss-of-Coolant Accident
LOEP	Loss of Electric Power
LOOP	Loss of Off-site Power
LPAP	Low Pressure Alarm Point
LPCI	Low Pressure Coolant Injection
LPCS	Low Pressure Core Spray
LPRM	Local Power Range Monitor
LPSP	Low Pressure Set Point
LPZ	Low Population Zone
LRCP	Liquid Radwaste Control Panel
LRS	Level Recording Switch
LSSS	Limiting Safety System Setting
MAPLHGR	Maximum Average Planar Linear Heat Generation Rate
MBA	Misplaced Bundle Accident
MCC	Motor Control Center
MCPR	Minimum Critical Power Ratio
MCR	Main Control Room
MDRFP	Motor-Driven Reactor Feed Pump
MEOD	Maximum Extended Operating Domain
MG	Motor-Generator Set
MM	Modified Mercalli
MOV	Motor Operated Valve
MREM	Millirem
MREM/YR	Millirem Per Year
MSIV	Main Steam Isolation Valve
MSIV-LCS	Main Steam Isolation Valve Leakage Control System
MSL	Mean Sea Level
MSL	Main Steamline
MTBE	Mean Time Between Event
MV/I	Millivolt to Current Converter
MVP	Mechanical Vacuum Pump
MWD/T	Megawatt-Days of Energy Production Per Ton of UO_2
MWe	Megawatts Electrical
MWt	Megawatts Thermal
NB	Nuclear Boiler
NBR	Nuclear Boiler Rated (power)
NC	Normally Closed
NCR	Nonconformance Report
ND	Normally De-energized
NDT	Nondestructive Testing
NDTT	Nil Ductility Transition Temperature
NE	Normally Energized
NEC	National Electric Code
NED	Nuclear Energy Division (GE)
NELPIA	Nuclear Energy Liability Property Insurance Association
NEPA	National Environmental Policy Act
NFPA	National Fire Protection Association
NMS	Neutron-Monitoring System

NO	Normally Open
NOAA	National Oceanic and Atmospheric Administration
NPSH	Net Positive Suction Head
NSRB	Nuclear Safety Review Board
NRC	U.S. Nuclear Regulatory Commission
NS	Nuclear System
NSOA	Nuclear Safety Operational Analysis
NSPS	Nuclear System Protection System
NSSS	Nuclear Steam Supply System
NSSSS (NS^4)	Nuclear Steam Supply System Shutoff
OB	Outboard
OBE	Operating Basis Earthquake
OD	Outside Diameter
OFS	Orificed Fuel Support
OL	Operating License
OPRM	Oscillation Power Range Monitoring
ORE	Occupational Radiation Exposures
ORM	Operational Requirements Manual
OT	Operational Transient
OTB	Onset of Transition Boiling
P&ID	Piping & Instrumentation Diagram
PA	Public Address (System)
PCA	Primary Coolant Activity
PCIOMR	Pre-Conditioning Interim Operating Management Recommendations
PCMS	Performance Calculation and Monitoring System
PCT	Peak Cladding Temperature
PD	Process Diagram
PFD	Process Flow Diagram
PGCC	Power Generation Control Complex
PMF	Probable Maximum Flood
PMP	Probable Maximum Precipitation
PMS	Performance Monitoring System (PPCS Subsystem)
PORC	Plant Operations Review Committee
PPCS	Plant Process Computer System
PPD	Process Piping Diagram
PQC	Product Quality Certification
PQL	Product Quality Checklist
PRM	Power Range Monitor
PRM	Process Radiation Monitoring
PRMS	Process Radiation Monitoring System
PRT	Prompt Relief Trip
PSAR	Preliminary Safety Analysis Report
PSTF	Pressure Suppression Test Facility
PSW	Plant Service Water
PWR	Pressurized Water Reactor
QA	Quality Assurance
QAP	Quality Assurance Program
QC	Quality Control
RAT	Reserve Auxiliary Transformer
RAU	Remote Analog Unit
RCIC	Reactor Core Isolation Cooling
RCIS	Rod Control and Information System

RCPB	Reactor Coolant Pressure Boundary
RDU	Remote Digital Unit
RECHAR	Recombination and Low Temperature Charcoal Adsorption
RFP	Reactor Feed Pump
RFPT	Reactor Feed Pump Turbine
RG	Regulatory Guide (NRC)
RHR	Residual Heat Removal
RHRS	Residual Heat Removal System
RM	Remote Manual
RMC	Remote Manual Control
RMCS	Reactor Manual Control System
RMS	Remote Manual Switch
RO	Reactor Operator (licensed)
ROC	Radwaste Operations Center
RPCS	Rod Pattern Control System
RPIS	Rod Position Information System
RPM	Radiation Protection Manager
RPS	Reactor Protection System
RPT	Recirculation Pump Trip
RPV	Reactor Pressure Vessel
RWCU	Reactor Water Cleanup
RWE	Rod Withdrawal Error
RWL	Rod Withdrawal Limiter
S&L	Sargent & Lundy
SACF	Single Active Component Failure
SAR	Safety Analysis Report
SBA	Small Break Accident
SDS	System Design Specifications
SEF	Single Equipment Failure
SER	Safety Evaluation Report
SGTS	Standby Gas Treatment System
SGTSET	Standby Gas Treatment System Equipment Train
SIP	Standby Information Panel
SIT	Structural Integrity Test
SJAE	Steam Jet Air Ejector
SLC	Standby Liquid Control
SOE	Single Operator Error
SOF	Single Operator Failure
SPCU	Suppression Pool Cleanup
SPF	Standard Project Flood
SPMU	Suppression Pool Makeup
SPS	Standard Project Storm
SQRTOR	Square Root Convertor
SRM	Source Range Monitor
SRO	Senior Reactor Operator (licensed)
SRP	Standard Review Plan (NRC)
SRSS	Square Root of the Sum of Squares
SRV	Safety/Relief Valve
SS	Safe Shutdown
SS	Selector Switch
SSA	Selective Switch Automatic

SSE	Safe Shutdown Earthquake
SSW	Shutdown Service Water
SW	Service Water
SWS	Service Water System
SWU	Separative Work Unit
TBCCW	Turbine Building Closed Cooling Water
TBCCWS	Turbine Building Closed Cooling Water System
TCV	Turbine Control Valve
TDRFP	Turbine-Driven Reactor Feed Pump
TDS	Total Dissolved Solids
TDS	Time Delay Switch
TG	Turbine-Generator
TIP	Traversing Incore Probe
TPM	Thermal Power Monitor
TQOS	Torque Overload Switch
TQRS	Torque Recorder Switch
TRS	Temperature Recorder Switch
TRU	Test and Reconfigure Unit
TSI	Turbine Supervisor Instruments
TSV	Turbine Stop Valve
UBC	Uniform Building Code
UHS	Ultimate Heat Sink
URC	Ultrasonic Resin Cleaner
USAR	Updated Safety Analysis Report
USGS	United States Geological Survey
UT	Ultrasonic Testing
Vac	Volts-Alternating Current
VARM	VAR Meter
VART	VAR Transducer
VBWR	Vallecitos Boiling Water Reactor
Vdc	Volts-Direct Current
VM	Voltmeter
VRF	Velocity Range Factor
VWO	Valves Wide Open
WHM	Watt-hour meter
WT	Watt Transducer
XFMR	Transformer



* Conditions at upstream side of TSV

Core Thermal Power	3473.0
Pump Heating	9.1
Cleanup Losses	-4.1
Other System Losses	-1.1
Turbine Cycle Use	3476.9 MWt

Figure 1.1-1. Heat Balance at Rated Power