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CHAPTER 1.0 - INTRODUCTION AND GENERAL DESCRIPTION OF PLANT

1.1 INTRODUCTION

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

This UFSAR is submitted by Exelon Generation Company for nuclear power plants at Byron, Illinois and at Braidwood, Illinois (Drawings M-1 and M-2) in accordance with the requirements of 10 CFR 50.71(e). Each power plant consists of two units having nearly identical nuclear steam supply systems (NSSS) and turbine generators. The main exception is that the original Unit 1 steam generators were replaced by steam generators of a different design. The power plants at the two sites are as nearly identical as site characteristics permit. The bulk of this UFSAR applies to the standardized, non-site-related aspects of the power plants. Sections which describe features specific to the sites are repeated for each site and the applicable station name appears at the top of these pages. Every effort has been made in the preparation of this document to conform to the Nuclear Regulatory Commission (NRC) Regulatory Guide 1.70, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants", Revision 2, September 1975. The guidance provided in Nuclear Energy Institute (NEI) 98-03, “Guidelines for Updating Final Safety Analysis Reports,” Revision 1, June 1999, as endorsed by NRC Regulatory Guide 1.181, “Content of the Updated Final Safety Analysis Report in Accordance with 10CFR50.71(e),” Revision 0, September 1999, is used to comply with the provisions of 10CFR50.71(e).

Each nuclear power plant consists of two nearly identical generating units, and two pressurized water reactor (PWR) (NSSS) and turbine-generator furnished by Westinghouse Electric Corporation (Westinghouse) similar in design concept to several projects recently licensed or currently under review by the NRC (see Section 1.3). Unit 1 contains steam generators supplied by B&W and Unit 2 contains steam generators supplied by Westinghouse. Westinghouse Electric Corporation, Sargent & Lundy, and the Commonwealth Edison Company jointly participated in the original design and construction of each unit. The plant is operated by Exelon Generation Company. Sargent & Lundy (S&L) is the architect-engineer for both stations.

Each nuclear steam supply system is designed for a power output of 3600.6 MWt which is the license application rating. The equivalent warranted gross and approximate net electrical outputs of each unit are 1242 MWe and 1210 MWe, respectively. The nuclear steam supply system is evaluated for safety analyses at 3658 MWt.

Specifically, the containment and engineered safety features (ESF) are designed and evaluated for operation at the power rating of 3658 MWt. Accidents (such as loss-of-coolant, steamline break, and other postulated accidents having offsite dose consequences) are also analyzed at a power rating of 3658 MWt.

The reactor containments are of post-tensioned concrete construction with a carbon steel liner. Sufficient free volume is provided to contain the energy released in a major accident without need for "pressure suppression" devices. Sargent & Lundy is responsible for containment design.

Byron Station is located in north central Illinois, near the town of Byron and near the Rock River (Drawing M-1). Cooling for the plant is provided by two natural draft cooling towers for non-essential service cooling, and by mechanical draft cooling towers for essential cooling. The fuel loading dates for the two units were November 1984 and November 1986 for Units 1 and 2, respectively. The corresponding dates for commercial operation were September 1985 and August 1987.

The Braidwood Station is located in northeastern Illinois, near the town of Braidwood and near the Kankakee River (Drawing M-1). Cooling for the plant is provided by a large man-made cooling pond of approximately 2500 acres constructed over a previously strip-mined area. Essential service cooling is provided by a 99-acre auxiliary cooling pond which is integral with the main pond. The fuel loading dates for the two units were October 1986 and December 1987 for Units 1 and 2, respectively. The corresponding dates for commercial operation were July 1988 and October 1988.

The standard symbols used on piping and instrument diagrams and other figures in this UFSAR are shown in Drawing M-34.

1.2 REFERENCES

1. NRC letter, "Braidwood, Byron, Dresden, LaSalle, Quad Cities, and Zion - Orders Approving Transfer of Licenses From Commonwealth Edison Company To Exelon Generation Company, LLC, and Approving Conforming Amendments," dated August 3, 2000

1.2 GENERAL PLANT DESCRIPTION

1.2.1 Site and Environment

The characteristics of the sites and their environs have been investigated to establish bases for determining criteria for storm, flood, and earthquake protection and to evaluate the validity of calculational techniques for the control of routine and accidental releases of radioactive liquids and gases to the environment. Field programs to investigate geology and seismology are completed. Preoperational meteorological programs to provide onsite observations of wind speed and direction have continued since the spring of 1973 at Byron and since the fall of 1973 for Braidwood. Radiological studies of the site environs were initiated at least 18 months prior to commercial operation, with the objective of establishing background radiation levels. The geography, demography, meteorology, hydrology, geology, and seismology of the two plant sites are discussed in detail in Chapter 2.0.

1.2.2 Nuclear Steam Supply System

The nuclear steam supply system (NSSS) consists of a Westinghouse pressurized water reactor and supporting auxiliary systems.

Performance at warranted steam flow of the NSSS based on zero percent makeup is as follows:

a. thermal output of NSSS (MWt) - 3600.6;

b. thermal output of reactor core (MWt) –3586.6;

c. steam flow from NSSS (lb/hr) – 16,026,608 for Unit 1/15,958,134 for Unit 2;

d. steam pressure at a steam generator outlet - maximum pressure of 1035 psia at hot leg temperature of 618°F for Unit 1 and a pressure of 910 psia at hot leg temperature of 612.7°F for Unit 2;

e. maximum moisture content (%) - 0.10 for Unit 1 and 0.25 for Unit 2; and

f. assumed feedwater temperature at steam generator inlet (°F) - 446.6.

The NSSS consists of a reactor and closed reactor coolant loops connected in parallel to the reactor vessel, each loop containing a reactor coolant pump and a steam generator. The NSSS also contains an electrically heated pressurizer and certain auxiliary systems.

High pressure reactor coolant circulates through the reactor core to remove the heat generated by the nuclear reaction. The heated reactor coolant flows from the reactor vessel to the steam generators (via reactor coolant loop piping). The coolant gives up its heat to the feedwater in the steam generator to generate steam for the turbine generator. The cycle is completed when the reactor coolant is pumped back to the reactor vessel. The entire reactor coolant system is composed of leaktight components to contain the reactor coolant to the system.

The core is a multiregion type. All fuel assemblies are mechanically identical, although the fuel enrichment is not the same in all assemblies. In a typical initial core loading, three fuel enrichments are used in mechanically identical assemblies. Fuel assemblies with the highest enrichments are placed in the core periphery, or outer region, and the two groups of lower enrichment fuel assemblies are arranged in a selected pattern in the central region. In subsequent refuelings, one third of the fuel is discharged and fresh fuel is loaded into the outer region of the core. The remaining fuel is arranged in the central two-thirds of the core in such a manner as to achieve optimum power distribution.

Rod cluster control assemblies are used for reactor control and consist of clusters of cylindrical absorber rods. The absorber rods move within guide tubes in certain fuel assemblies. Above the core, each cluster of absorber rods is attached to a spider connector and drive shaft, which is raised and lowered by a drive mechanism mounted on the reactor vessel head. The insertion of the rod cluster control assembly for a reactor trip is by gravity.

The reactor coolant pumps are Westinghouse vertical, single-stage, centrifugal pumps of the shaft-seal type.

The steam generators are B&W vertical U-tube units for Unit 1 and Westinghouse vertical U-tube units for Unit 2. All steam generators contain Inconel tubes. Integral moisture separation equipment reduces the moisture content of the steam.

The reactor coolant piping and all of the pressure-containing surfaces in contact with reactor water are stainless steel. The steam generator tubes and fuel cladding are Inconel and Zircaloy/ZIRLO, respectively. Reactor core internals, including control rod drive shafts, are primarily stainless steel.

An electrically heated pressurizer connected to one reactor coolant loop maintains reactor coolant system pressure during normal operation, limits pressure variations during plant load transients, and keeps system pressure within design limits during abnormal conditions.

Auxiliary system components are provided to charge makeup water into the reactor coolant system, purify reactor coolant, provide chemicals for corrosion inhibition and reactivity control, cool system components, remove decay heat, and provide for emergency safety injection.

1.2.3 Engineered Safety Features

The engineered safety features provided for this plant have sufficient redundancy of components and power sources such that under the conditions of a loss-of-coolant accident they can maintain the containment integrity and limit personnel exposure to less than 10 CFR 50.67 limits. The engineered safety features incorporated in the design of this plant and the functions they serve are summarized in the following.

a. The emergency core cooling system injects borated water into the reactor coolant system if coolant is lost. This system limits damage to the core and limits the fission product contamination released into the containment following a postulated loss-of-coolant accident (LOCA).

b. A steel lined, concrete containment vessel consists of a post-tensioned concrete cylindrical wall and shallow dome, and a conventionally reinforced concrete base. The containment forms a virtually leaktight barrier to prevent the escape of radioactivity.

c. Reactor containment fan coolers reduce containment temperature and pressure following a postulated loss-of-coolant accident.

d. A containment spray system is used to reduce containment pressure and to remove iodine and particulate fission products from the containment atmosphere in the event of a loss-of-coolant accident.

e. The auxiliary feedwater system provides for heat removal from the reactor coolant system by providing makeup water to the steam generator under a variety of postulated conditions.

f. A combustible gas control system is provided to ensure that the containment atmosphere is mixed following a loss-of-coolant accident. A mixed containment atmosphere prevents local accumulation of combustible or detonable gases that could threaten containment integrity or equipment operating in a local compartment.

1.2.4 Emergency Core Cooling System

The emergency core cooling system (ECCS), with passive and active subsystems, is designed to inject borated water into the reactor coolant system (RCS) following a LOCA. This will provide cooling to limit core damage, metal-water reactions, and fission-product release. The ECCS provides long-term postaccident cooling of the core by drawing borated water from the containment sump.

1.2.5 Control and Instrumentation

The reactor is controlled by a variety of reactivity coefficients (temperature, pressure, doppler) by control rod cluster motion which is required for load follow transients and for startup and shutdown, and by a soluble neutron absorber, i.e., boron in the form of boric acid which is adjusted in concentration to compensate for such effects as fuel consumption and accumulation of fission products.

1.2.6 Electrical System

Each unit's main generator is an 1800-rpm, 3-phase, 60-cycle, hydrogen-innercooled unit with water-cooled stator windings and is rated at 1361 MVA at 75 psig gas pressure. Field excitation is provided by a direct shaft-driven brushless excitation system. Two one-half size main step-up transformers deliver power to the 345-kV switchyard.

The station's auxiliary power system consists of system and unit auxiliary transformers; 6900-V, 4160-V, and 480-V switchgear; 480-V motor control centers; 120-Vac instrument buses; and 250-Vdc and 125-Vdc buses.

Two diesel generators are provided for each unit and are available as onsite sources of power (in the event of complete loss of normal a-c power) for operating essential safeguard features. Each diesel generator is capable of supplying required electrical loads for a simultaneous LOCA and loss of offsite power to any one unit.

1.2.7 Turbine and Auxiliaries

The turbine for each unit is a four-casing, tandem-compound, six-flow exhaust, 1800-rpm unit with 40-inch last-row blades. There are two combination moisture-separator/steam-reheater assemblies per unit. The turbine-generator for Units 1 have a maximum guaranteed rating of 1242 MWe gross at 15,154,804 lb/hr steam flow with inlet steam conditions of 1004 psia, 0.22% moisture, exhausting at 3.5 in. Hg abs, at zero percent makeup. The turbine-generators for Units 2 have a guaranteed rating of 1210 MWe gross at 15,260,975 lb/hr steam flow with inlet steam conditions of 869 psia, 0.24% moisture, exhausting at 3.5 in. Hg abs, at zero percent makeup. There are seven stages of feedwater heating for all units.

The turbine is equipped with a redundant fault tolerant Westinghouse Ovation based distributed control system. All control algorithms and processes within the turbine control system are redundant and configured to allow unrestricted turbine operation. This system utilizes a fire-resistant hydraulic fluid to control throttle and governor valve positioning.

The condenser is of the single-pass deaerating type. There are three parallel strings of feedwater heaters that utilize extraction steam from the low pressure turbines, two parallel strings of feedwater heaters that utilize extraction and exhaust steam from the high pressure turbine, four one-third-sized feedwater condensate and condensate booster pumps, and three one-half-sized feedwater and heater drain pumps. Heater drains from the three highest-pressure feedwater heaters are pumped into the feedwater system; drains from the four lowest-pressure heaters are cascaded to the condenser.

1.2.8 Fuel Handling System

The reactor is refueled with equipment which handles the spent fuel under water from the entire time from leaving the reactor vessel until it is secured in a cask for shipment. Underwater transfer of spent fuel provides a transparent radiation shield and a reliable coolant for decay heat removal.

Fuel handling is performed in the refueling cavity which is flooded for refueling, and the fuel storage pool which is in the fuel building. The two areas are connected by a fuel transfer system which carries the fuel through an opening in the reactor containment.

Spent fuel is removed from the reactor vessel by a refueling machine, placed on the fuel transfer cart conveyor and transferred to the spent fuel storage pool. The fuel is removed from the transfer cart and placed into storage racks. After a suitable decay period, the fuel may be removed from storage and loaded into a shipping cask for removal from the plant.

Refer to Section 9.1.2.3.11 for a description of spent fuel storage and handling using Dry Cask Storage (DCS) system and the Independent Spent Fuel Storage Installation (ISFSI).

All important pumps, piping, and equipment are replicated and capable of being supplied from one of two independent ESF divisions.

1.2.9 Radioactive Waste Management System

The radioactive waste system provides equipment necessary to collect, process, and prepare for the disposal of radioactive liquid, gaseous, and solid wastes produced as a result of reactor operation or to transfer the wastes to a vendor-supplied radwaste system.

After collection, depending on chemical composition, liquid wastes may be demineralized and/or filtered. The treated water is discharged at concentrations within the limits of 10 CFR 20. Sludges and spent demineralizer resins are processed by a vendor-supplied radwaste system for ultimate disposal in an authorized location.

Gaseous wastes are collected from the waste gas header. Discharge of the gaseous wastes to the environment is controlled to ensure that the offsite dose is as low as reasonably achievable (ALARA).

1.2.10 Features of Special Interest

The fundamental concept for the design and construction of the Byron/Braidwood Stations is one of commonality and duplication to the maximum extent permitted by site characteristics. For those features not dictated specifically by site characteristics, identical designs have been employed for the two stations. The concept has been extended to the point where the limiting (i.e., worst case) parameters of the sites are considered in the common design. An example of this is the use of the most restrictive site's seismic building response spectra for the design of systems and components in both plants.

Common plans, drawings, and specifications have been issued for construction at the two sites. Design and construction management for both sites have been conducted by the same major organizations, using the same quality assurance and project management programs. This approach embraces the concept of standardization in nuclear power plant design and construction.

1.2.11 Structures

The major structures include a separate and independent containment for each reactor, a common auxiliary building, a common turbine building, a common solid radwaste storage, and administration and service building. General layouts of the plant and interior components' arrangements are shown on Drawings M-5 through M-18 and M-20 and M-22 (Byron), and Drawings M-5 through M-20 and M-22 (Braidwood).

For purposes of design and analysis, structures are designated by Safety Category according to their relation to plant safety. The Safety Category definitions are as follows:

a. Safety Category I - Those structures important to safety that must be designed to remain functional in the event of the safe shutdown earthquake (SSE) and other design-basis events (including tornado, probable maximum flood, operating basis earthquake (OBE), missile impact, or accident internal to the plant) are designated as Safety Category I.

b. Safety Category II - Those structures which are not designated as Safety Category I are designated as Safety Category II.

The design criteria and analysis methods for these structures are discussed in Chapter 3.0.

1.3 COMPARISON TABLES

1.3.1 Comparisons with Similar Facility Designs

The design is conceptually similar to Exelon Generation Company's Zion Station. Differences in the design of the two plants have been allowed only (1) when dictated by the site characteristics, (2) when the change would result in significant safety improvement, simplification of construction or operation procedures, or cost savings; or (3) as required to comply with appropriate codes and standards, NRC criteria, regulatory guides, and regulations.

The nuclear steam supply system is similar to that of the Zion Station but has a slightly higher power rating. The reactor containments are of the same materials and size as those at the Zion Station, but each has only three buttresses, rather than six as used at Zion. The number of post-tensioning tendons is reduced, and the number of wires per tendon increased, from that used at Zion. The reduced number of buttresses allows for greater separation of penetration areas for redundant safety-related systems. Several plants on which this buttress design has been used are listed in Table 1.3-1.

The polar cranes in the reactor containment are mounted on the containment wall, rather than on the missile barrier as at Zion. This allows use of a greater area for component laydown in the containment.

Two 100%-capacity containment spray systems are used, rather than the three systems used at Zion. Four containment fan coolers are used, rather than the five used at Zion. The emergency diesel-generator systems for each unit are entirely independent and use two 5500-kW diesel generators per unit. The arrangement of equipment in the common auxiliary building allows greater physical separation of redundant systems and their piping and cables than was possible at Zion.

The Byron Station uses natural draft cooling towers for heat rejection. Zion utilizes once-through cooling. Mechanical draft cooling towers are provided for essential service cooling at Byron.

The Braidwood Station uses a large man-made cooling pond for heat rejection. An auxiliary cooling pond, integral with the main pond, is provided for essential service cooling.

Table 1.3-2 of the FSAR provided the design comparison of the Byron/Braidwood nuclear steam supply system with Comanche Peak, Indian Point 2, South Texas, Sun Desert, W. B. McGuire Nuclear Station, Trojan Nuclear Power Plant, SNUPPS, and the Watts Bar Application. This information was current at the time the Byron Unit 1 operating license was granted and has not been included in the UFSAR.

1.3.2 Comparison of Final and Preliminary Information

The Byron/Braidwood Power Plant design was subject to continuing review throughout the construction of the stations. The experience gained at Zion Station and other PWRs was used to enhance equipment reliability and performance. Current design technology was used to upgrade earlier plant design methods.

No significant design changes have been made to the Byron Station or the Braidwood Station which have not been previously reported by amendment to the PSAR, except for the inclusion of 17 x 17 optimized fuel. Table 1.3-3 of the FSAR listed those significant changes reported since the issuance of the Byron and Braidwood Stations Construction Permits. This information was current at the time the Byron Unit 1 operating license was granted and has not been included in the UFSAR.

Other changes included the removal of the part length control rods (they are not needed to control Xenon induced axial instabilities), the enlargement of spent fuel capacity, the use of more corrosion-resistant materials in the steam generators and moisture steam separators, improved equipment packaging to do a reactor refueling in a shorter time period, an upgraded design for the reactor coolant pump seals, and replacement steam generators for Unit 1. These concepts are described in later chapters.

1.3.3 References

1. Exelon Generation Company, "Byron/Braidwood Stations Fire Protection Report in Response to Appendix A of BTP APCSB 9.5-1," (current amendment).

TABLE 1.3-1

PLANTS USING THREE-BUTTRESS CONTAINMENT DESIGN

PLANT/UTILITY	DATE OF OPERATION
Arkansas Nuclear One Arkansas Power & Light Co.	5-21-74
Millstone-2 Northeast Utilities	8-1-75
Rancho Seco Sacramento Municipal Utility District	8-16-74
Trojan Portland General Electric Co.	11-21-75
J.M. Farley-1 Alabama Power Co.	6-25-77

1.4 IDENTIFICATION OF AGENTS AND CONTRACTORS

1.4.1 Licensee

Exelon Generation Company is the Licensee for the Byron Station, Units 1 and 2, which is located in Rockvale Township, Ogle County, approximately 4 miles south of Byron, Illinois, and for Units 1 and 2 of the Braidwood Station, which is located in Reed Township, Will County, approximately 6 miles southwest of Wilmington, Illinois. The Licensee is responsible for the design, construction, and operation of the nuclear power plants.

Commonwealth Edison supplies electrical service to an area of 13,000 square miles with a population of approximately 8 million persons, located primarily in the northern third of Illinois.

Dresden 1, Commonwealth Edison's first nuclear generating station, went into commercial service during August 1960, and has produced more than 10 billion kWh. Additional nuclear units in service or under construction are listed in Table 1.4-1.

1.4.2 Architect-Engineer

For the work covered by this application, Sargent & Lundy (S&L) has been retained as the design consultants. The Licensee has employed Sargent & Lundy for power plant design work for over 80 years.

Sargent & Lundy is an independent consulting engineering organization founded in Chicago, in 1891. For over three-quarters of a century, the firm has specialized exclusively in the design of generation, transmission, distribution, and utilization of steam and electric power and related facilities. The firm has provided the complete engineering services for more than 600 turbine-generator units with a total capacity of 53,000,000 kW. Of this total, some 9,800,000 kW is in nuclear generating capacity. Table 1.4-2 lists the nuclear plants completed by or currently under design by Sargent & Lundy.

1.4.3 Reactor Designer

Westinghouse has designed, developed, and manufactured nuclear power facilities since the 1950s, beginning with the world's first large central station nuclear power plant (Shippingport), which started producing power in 1957. Completed or contracted commercial nuclear capacity totals were in excess of 98,000 MWe. Westinghouse pioneered new nuclear design concepts, such as chemical shim control of reactivity and the rod cluster control concept, throughout the last two decades. Westinghouse manufacturing facilities include the largest commercial nuclear fuel fabrication facility in the world and the world's most modern heat transfer equipment production facility, as well as other facilities producing nuclear steam supply system (NSSS) components. Table 1.4-3 lists all Westinghouse pressurized water reactor (PWR) plants to date, including those plants under construction or on order at the time of the Byron/Braidwood application.

The U.S. Nuclear Regulatory Commission (NRC) and the Electric Power Research Institute have contracted with Westinghouse for research into NSSS-related activities. Westinghouse experience was also utilized by the NRC and Metropolitan Edison immediately following the Three Mile Island Unit 2 accident and the corporation continues to participate with the Westinghouse Owner's Group of utilities in addressing the NRC action plan and other operations improvements.

1.4.4 Constructor

Construction coordination of all activities at the site was under the supervision of the Commonwealth Edison's Station Construction Department. The department exercises site managerial functions as discussed in Chapter 17.0 of the UFSAR. The Station Construction Department was the constructor for Zion Station. This department has coordinated the construction activities for almost all of Commonwealth Edison's existing power plants. It was also the construction coordinator for La Salle County Station.

1.4.5 Consultants and Service Organization

1.4.5.1 Security System - ETA

The design of the physical security system and the administrative controls was performed by ETA, Inc.

ETA personnel have had varied and in-depth experience in the design, safety analysis, and environmental review of nuclear power plants and related facilities as well as in the management and organization of security systems. They are very familiar with the details of the current generation of light water reactors and, in particular, those critical areas and components of the plants which might be the most vulnerable to sabotage. They are also familiar with the current regulations and guidelines of the NRC that define the required performance and objectives of a security system for licensed activities.

1.4.5.2 Dames & Moore

The independent consulting firm of Dames & Moore was employed to conduct studies relating to the geology, seismology, and groundwater hydrology at both sites. The firm also conducted preconstruction baseline studies, including wildlife surveys as well as soil and vegetation analyses.

Having performed environmental studies for approximately 30 nuclear power plant sites, Dames & Moore is a recognized authority in the field of environmental engineering of nuclear power plants.

1.4.5.3 HARZA Engineering

HARZA was employed in the design of the water treatment facilities at both stations.

HARZA 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 earthfilled dams.

1.4.5.4 Murray and Trettel, Inc.

Murray and Trettel (M&T) is an environmental consulting firm which, since 1960, has provided significant meteorological input to both preoperational and operational phases of meteorological programs for nuclear power stations. M&T has also provided meteorological input to a wide variety of air pollution and environmental problems as well as allied control technique programs.

Murray & Trettel provided meteorological data for both stations by implementation of an onsite measurement program incorporating a tower for elevation measurements.

1.4.5.5 Shirmer Engineering Corporation

Shirmer Engineering is a firm of consulting fire protection engineers. The firm has done work on 17 Department of Energy nuclear fuel production and laboratory facilities, as well as for numerous nuclear power stations for Sargent & Lundy. Shirmer Engineering has also performed services for many fossil units.

Shirmer Engineering provided evaluation of the fire protection systems at both stations and assisted in the preparation of the Byron/Braidwood Fire Protection Report.

1.4.5.6 Hyla S. Napadensky

Ms. Napadensky was retained to help evaluate the probability of an accidental explosion occurring on a train carrying explosives in the vicinity of the Braidwood Station.

Ms. Napadensky is the Manager of Fire Safety Research at the IIT Research Institute of the Illinois Institute of Technology. Ms. Napadensky has directed analytical and experimental research in the areas of explosion effects, hazards and risk analysis, safety of chemical systems, explosives and propellant sensitivity, and initiation mechanisms during her 17 years with IIT Research Institute.

1.4.5.7 NALCO Chemical Company

The NALCO Chemical Company (formerly Industrial Bio-Test, Inc.) consisted of two divisions, Industrial Bio-Test Laboratories, and NALCO Environmental Sciences, which conduct studies relating to toxicology and ecological sciences, respectively. The Environmental Science Division includes seven subdivisions: (1) aquatic biology, (2) fisheries and field operations, (3) water and wastewater chemistry, (4) radiochemistry, (5) air sciences and data processing, (6) land and plant sciences, and (7) environmental physiology.

As a technical consultant on the Braidwood project, the NALCO Chemical Company provided a clam bed mapping survey in the area of the station's intake and discharge structures located on the Kankakee River.

1.4.5.8 Westinghouse Environmental Systems Department (WESD)

WESD, established as a department of the Westinghouse Power Systems Company in 1969, consisted of environmental scientists and engineers experienced in the areas of aquatic and terrestrial biology and ecology; geology; limnology; environmental chemistry and physics; physical oceanography, meteorology and climatology, radiology, public health aspects of pollutant emissions, and systems engineering and integration.

WESD conducts broad environmental surveys, environmental program planning and data interpretation, and provides recommended action programs for meeting federal, state, and local environmental quality regulations. As a technical consultant on the Braidwood project, WESD staff biologists conducted a 2-year baseline study of the Braidwood Station site. Distributions of phytoplankton, zooplankton, periphyton, benthos, fish, fish eggs and larvae, and water chemistry in the Kankakee River in the vicinity of the site were determined, and quantitative data on terrestrial flora and fauna were collected. The impacts of plant construction and operation in the biotic communities of the site were predicted.

1.4.5.9 Illinois Natural History Survey (INHS)

The Illinois Natural History Survey (INHS), which has its beginnings almost 120 years ago, is a division of the State Department of Registration and Education and provides services to farmers, homeowners, sportsmen, and all other citizens of Illinois as well as to industries. INHS cooperates in biological research with the Illinois Department of Agriculture, Conservation, and Public Health; the University of Illinois, Southern Illinois University, and other educational institutions; various research branches of the federal government; and other agricultural, conservation, municipal, and business organizations throughout the state.

INHS aquatic biologists were involved in a 4-year study of the Kankakee River and Horse Creek near Custer Park, Illinois. The purpose of the study is to obtain biological, physical, and chemical data which will be used to evaluate any effects of the construction and operation of the Braidwood Station and its associated cooling lake on the biota and water quality of the Kankakee River and Horse Creek. The station's cooling pond will use the Kankakee River as a source of water for both intake and discharge purposes.

1.4.5.10 NUS Corporation

NUS Corporation is a consulting engineering, research, and testing firm specializing in environmental and energy systems engineering, systems analysis, design engineering, management consulting, and training programs related to these areas. NUS has provided advice and professional guidance to utility, industrial, and government clients throughout the United States and in a number of foreign countries.

As a technical consultant on the Braidwood project, NUS was involved in a study to determine the adequacy of the station's ultimate heat sink.

1.4.5.11 Eberline Instrument Corporation (EIC)

Eberline Instrument Corporation (EIC) has provided radiation measurement equipment, comprehensive radiation protection services, and analytical laboratory services to the nuclear industry since 1953.

As a technical consultant on the Byron/Braidwood projects EIC performed preoperational environmental radiological baseline studies on and around the site.

1.4.5.12 Meteorology Research, Inc. (MRI)

Meteorology Research, Inc. (MRI) is an environmental consulting firm which, since 1951, has provided meteorological and air quality instruments and services to all aspects of industry in the solution of weather-related problems. These range from environmental impact assessments of existing or proposed airports and other major developments to problems of warehousing and marketing seasonal consumer goods. Of particular interest is the influence the local topography has on temperatures and winds. MRI provided meteorological data from 1973 through mid-1975 for Byron and Braidwood Stations by implementation of an onsite meteorological measurement program.

1.4.5.13 Illinois State Museum (ISM)

The Illinois State Museum conducts archaeological investigations throughout the state of Illinois. As a member of the Illinois Archaeological Survey, they have the expertise and services to perform contract archaeological work. Their studies included a pedestrian reconnaissance survey, subsurface testing and excavating, and laboratory analyses of datifacts, pollen, and soils.

As a technical consultant on the Braidwood project, ISM identified and made recommendations which Commonwealth Edison acted upon to aid in preserving the archaeological sites on Braidwood Station and pipeline corridor property.

1.4.5.14 Equitable Environmental Health, Inc. (EEH)

Equitable Environmental Health, Inc. (EEH), successor to Environmental Analysts, Inc./Tabershaw-Cooper Associated, Inc., is a multidisciplinary firm that offers the consulting services of medical professionals, scientists, engineers, economists, and technical support personnel in all areas of environmental health and economics.

EEH staff biologists conducted a 2-year baseline study of the Byron Station site. Distributions of phytoplankton, zooplankton, periphyton, benthos, fish, fish eggs and larvae, and water chemistry in the Rock River in the vicinity of the site were determined and quantitative data on terrestrial flora and fauna were collected. The impacts of plant construction and operation on the biotic communities of the site were predicted, and data were provided for a benefit-cost analysis of the project.

1.4.5.15 Espey, Huston & Associates, Inc. (EH & A)

Espey, Huston & Associates, Inc. (EH & A) is a consulting firm addressing the environmental problems associated with industrial and urban development. EH & A professionals cover a broad range of expertise including civil engineering, environmental engineering, mathematics and computer science, and all phases of aquatic, estuarine, and terrestrial ecology.

As a technical consultant on the Byron project, EH & A conducted the construction phase terrestrial and aquatic monitoring programs.

1.4.5.16 University of Wisconsin-Milwaukee (UWM)

The University of Wisconsin-Milwaukee under Dr. Elizabeth Benchley of the Dept. of Anthropology, conducts archaeological investigations throughout Wisconsin and northern Illinois. As a member of the Illinois Archaeological Survey, they have the expertise and services to perform contract archaeological work. Their studies included a pedestrian reconnaissance survey, subsurface testing, and lab analysis of datifacts, pollen, and soils.

As a technical consultant on the Byron project, UWM identified and made recommendations which Commonwealth Edison acted upon to aid in preserving the archaeological sites on Byron Station and pipeline corridor property. Also, UWM conducted archaeological investigations on the Byron transmission line right-of-ways.

1.4.5.17 Aero-Metric Engineering, Inc. (AME)

Aero-Metric Engineering, Inc., founded in 1969, is based in Sheboygan, Wisconsin. The staff was made up of over 50 technical photogrammetric personnel, many having professional engineer and/or survey registration. AME's capabilities allow for a complete range of precision photogrammetric services, including aerial photography, mapping, and multiple survey skills.

As a technical consultant on the Byron project, AME will be providing annual aerial infra-red photographs.

1.4.5.18 Iowa Institute of Hydraulic Research

The Iowa Institute of Hydraulic Research, formally organized in 1931, is a Division of the University of Iowa's College of Engineering. The Institute staff exceeded 80 in number and was comprised of a professional staff with Ph.Ds in the areas of Civil Engineering, Mechanical Engineering, Physics, Mechanics and Hydraulics, and Aeronautical Engineering, with most of these personnel holding joint academic appointments in the College of Engineering's Division of Energy Engineering. The Institute of Hydraulic Research conducts programs of fundamental research and advanced design and analysis in the areas of environmental pollution, bioengineering, naval hydrodynamics, river mechanics, ice hydraulics, hydrology, water resources, hydraulic structures, fluid mechanics, advanced instrumentation and data-handling techniques for fluids research, and mathematical modeling of watersheds and hydrology.

As a technical consultant on the Braidwood project, the Institute conducted a thermal evaluation to determine the adequacy of the ultimate heat sink.

1.4.5.19 Babcock and Wilcox International (B&W)

B&W is located in Cambridge, Ontario, Canada. B&W has fabricated fossil-fueled boiler components for over 100 years and has fabricated nuclear system components since the late 1950's. B&W has supplied replacement steam generators for Byron Unit 1 and Braidwood Unit 1.

1.4.5.20 Framatome Technologies, Incorporated (FTI)

FTI is located in Lynchburg, Virginia and has been providing services to the electric power industry for over four decades. FTI engineering services include the necessary expertise, experience, and NRC-approved computer codes and methodologies to support the transient analysis of the Unit steam generators.

1.4.5.21 Stone & Webster Engineers and Constructors, Inc. (S&W)

S&W is located in Boston, Massachusetts and has been providing services to the electric power industry for over 100 years. S&W has provided balance-of-plant design-engineering support services in support of the power uprate of the Byron and Braidwood units.

TABLE 1.4-1

EXELON GENERATION COMPANY'S NUCLEAR POWER PLANTS
IN SERVICE OR UNDER CONSTRUCTION

UNIT	NOMINAL GROSS[1] RATING (MWe)	SCHEDULED COMMERCIAL SERVICE DATE
Dresden 1	210	1960
Dresden 2	850	1972
Dresden 3	850	1972
Quad-Cities 1	850	1972
Quad-Cities 2	850	1972
Zion 1	1085	1973
Zion 2	1085	1973
La Salle 1	1122	1978
La Salle 2	1122	1979
Byron 1	1175	1985
Byron 2	1175	1987
Braidwood 1	1175	1988
Braidwood 2	1175	1988

[1]Note that this is a gross rating, not a net rating.

TABLE 1.4-2

NUCLEAR POWER PLANTS COMPLETED OR
CURRENTLY UNDER DESIGN BY SARGENT & LUNDY

UNIT	NOMINAL GROSS[2] RATING (MWe)	YEAR OF POWER OPERATION
EBWR	5	1956
Elk River	22	1962
La Crosse	60	1967
SEFOR	20 (MWt)	1969
Dresden 2	850	1969
Dresden 3	850	1971
Quad-Cities 1	850	1971
Quad-Cities 2	850	1972
Zion 1	1085	1973
Zion 2	1085	1973
Fort St. Vrain, Unit 1	330	1973
Enrico Fermi, Unit 2	1200	1988
La Salle County Station, Unit 1	1122	1979
La Salle County Station, Unit 2	1122	1980
Byron Station, Unit 1	1175	1985
Byron Station, Unit 2	1175	1987
Braidwood Station, Unit 1	1175	1988
Braidwood Station, Unit 2	1175	1988
Clinton Power Station, Unit 1	992	1981
Kaiseraugst	992	1982

[2]Note that this is a gross rating, not a net rating.

TABLE 1.4-3

WESTINGHOUSE PRESSURIZED WATER REACTOR NUCLEAR POWER PLANTS

PLANT	OWNER UTILITY	LOCATION	SCHEDULED COMMERCIAL OPERATION	MWe NET	NUMBER OF LOOPS
Shippingport	Duquesne Light Company; Energy Research & Development Administration	Pennsylvania	1957	90	4
Yankee-Rowe	Yankee Atomic Electric Company	Massachusetts	1961	175	4
Trio Vercellese (Enrico Fermi)	Ente Nazionale per L'Energia Elettrica (ENEL)	Italy	1965	260	4
Chooz (Ardennes)	Societe d'Energie Nucleaire Franco-Belge des Ardennes (SENA)	France	1967	305	4
San Onofre Unit 1	Southern California Edison Co.; San Diego Gas and Electric Co.	California	1968	450	3
Haddam Neck (Connecticut Yankee)	Connecticut Yankee Atomic Power Company	Connecticut	1968	575	4
Jose Cabrera-Zorita	Union Electrica, S.A.	Spain	1969	153	1
Beznau Unit 1	Nordostschweizerische Krafwerke AG (NOK)	Switzerland	1969	350	2
Robert Emmett Ginna	Rochester Gas and Electric Corporation	New York	1970	490	2
Mihama Unit 1	The Kansai Electric Power Company, Inc.	Japan	1970	320	2
Point Beach Unit 1	Wisconsin Electric Power Co.; Wisconsin Michigan Power Co.	Wisconsin	1970	497	2
H. B. Robinson Unit 2	Carolina Power and Light Co.	South Carolina	1971	707	3

TABLE 1.4-3 (Cont'd)

PLANT	OWNER UTILITY	LOCATION	SCHEDULED COMMERCIAL OPERATION	MWe NET	NUMBER OF LOOPS
Beznau Unit 2	Nordostschweizerische Kraftwerke AG (NOK)	Switzerland	1972	350	2
Point Beach Unit 2	Wisconsin Electric Power Co.; Wisconsin Michigan Power Co.	Wisconsin	1972	497	2
Surry Unit 1	Virginia Electric and Power Co.	Virginia	1972	822	3
Turkey Point Unit 3	Florida Power and Light Co.	Florida	1972	745	3
Indian Point Unit 2	Consolidated Edison Company of New York, Inc.	New York	1973	873	4
Prairie Island Unit 1	Northern States Power Company	Minnesota	1973	530	2
Turkey Point Unit 4	Florida Power and Light Co.	Florida	1973	745	3
Surry Unit 2	Virginia Electric and Power Co.	Virginia	1973	822	3
Zion Unit 1	Exelon Generation Company	Illinois	1973	1050	4
Kewaunee	Wisconsin Public Service Corp.; Wisconsin Power and Light Co.; Madison Gas and Electric Co.	Wisconsin	1974	560	2
Prairie Island Unit 2	Northern States Power Company	Minnesota	1974	530	2
Takahama Unit 1	The Kansai Electric Power Company, Inc.	Japan	1974	781	3
Zion Unit 2	Exelon Generation Company	Illinois	1974	1050	4

TABLE 1.4-3 (Cont'd)

PLANT	OWNER UTILITY	LOCATION	SCHEDULED COMMERCIAL OPERATION	MWe NET	NUMBER OF LOOPS
Doel Unit 1	Indivision Doel	Belgium	1975	390	2
Doel Unit 2	Indivision Doel	Belgium	1975	390	2
Donald C. Cook Unit 1	Indiana and Michigan Electric Company (AEP)	Michigan	1975	1060	4
Ringhals Unit 2	Statens Vattenfallsverk (SSPB)	Sweden	1975	822	3
Almaraz Unit 1	Unit Electrica, S.A.; Compania Sevillana de Electricidad, S.A.; Hidroelectrica Espanola, S.A.	Spain	1976	902	3
Beaver Valley Unit 1	Duquesne Light Company; Ohio Edison Company; Pennsylvania Power Company	Pennsylvania	1976	852	3
Diablo Canyon Unit 1	Pacific Gas and Electric Co.	California	1976	1084	4
Indian Point Unit 3	Consolidated Edison Company of New York, Inc.	New York	1976	965	4
Lemoniz Unit 1	Iberduero, S.A.	Spain	1976	902	3
Salem Unit 1	Public Service Electric and Gas Company; Exelon Generation Company; Atlantic City Electric Co.; Delmarva Power and Light Co.	New Jersey	1976	1090	4

TABLE 1.4-3 (Cont'd)

PLANT	OWNER UTILITY	LOCATION	SCHEDULED COMMERCIAL OPERATION	MWe NET	NUMBER OF LOOPS
Trojan	Portland General Electric Co.; Eugene Water and Electric Board; Pacific Power and Light Company	Oregon	1976	1130	4
Almaraz Unit 2	Union Electrica, S.A.; Compania Sevillana de Electricidad, S.A.; Hidroelectrica Espanola, S.A.	Spain	1977	902	3
Asco Unit 1	Fuerzas Electricas de Cataluna, S.A. (FESCA)	Spain	1977	902	3
Diablo Canyon Unit 2	Pacific Gas and Electric Co.	California	1977	1106	4
Joseph M. Farley Unit 1	Alabama Power Company	Alabama	1977	829	3
Ko-Ri Unit 1	Korea Electric Power Co., Ltd.	Korea	1977	564	2
North Anna Unit 1	Virginia Electric and Power Co.	Virginia	1977	898	3
North Anna Unit 2	Virginia Electric and Power Co.	Virginia	1977	898	3
Ohi Unit 1	The Kansai Electric Power Co., Inc.	Japan	1977	1122	4
Ohi Unit 2	The Kansai Electric Power Co., Inc.	Japan	1977	1122	4
Ringhals Unit 3	Statens Vattenfallsvert (SSPB)	Sweden	1977	900	3
Sequoyah Unit 1	Tennessee Valley Authority	Tennessee	1977	1148	4
Angra dos Reis Unit 1	Furnas-Centrais Electricas, S.A.	Brazil	1978	626	2

TABLE 1.4-3 (Cont'd)

PLANT	OWNER UTILITY	LOCATION	SCHEDULED COMMERCIAL OPERATION	MWe NET	NUMBER OF LOOPS
Asco Unit 2	Fuerzas Electricas de Cataluna, S.A. (FESCA); Empresa Nacional Hidroelectrica del Ribagorzana, S.A. (ENHER); Fuerzas Hidroelectricas del Segre, S.A.; Hidroelectrica de Cataluna, S.A.	Spain	1978	902	3
Donald C. Cook Unit 2	Indiana and Michigan Electric Company (AEP)	Michigan	1978	1060	4
Lemoniz Unit 2	Iberduero, S.A.	Spain	1978	902	3
Sequoyah Unit 2	Tennessee Valley Authority	Tennessee	1978	1148	4
Watts Bar Unit 1	Tennessee Valley Authority	Tennessee	1978	1177	4
William B. McGuire Unit 1	Duke Power Company	North Carolina	1978	1180	4
Joseph M. Farley Unit 2	Alabama Power Company	Alabama	1979	829	3
Krsko	Savske Elektrarne, Ljubljana, Slovenia, Elektroprivreda, Zagreb, Croatia	Yugoslavia	1979	615	2
Ringhals Unit 4	Statens Vattenfallsvert (SSPD)	Sweden	1979	900	3

TABLE 1.4-3 (Cont'd)

PLANT	OWNER UTILITY	LOCATION	SCHEDULED COMMERCIAL OPERATION	MWe NET	NUMBER OF LOOPS
Salem Unit 2	Public Service Electric and Gas Company; Exelon Generation Company Atlantic City Electric Co.; Delmarva Power and Light Co.	New Jersey	1979	1115	4
Virgil C. Summer	South Carolina Electric and Gas Company	South Carolina	1979	900	3
Watts Bar Unit 2	Tennessee Valley Authority	Tennessee	1979	1177	4
William B. McGuire Unit 2	Duke Power Company	North Carolina	1979	1180	4
Byron Unit 1	Exelon Generation Company	Illinois	1981	1120	4
Comanche Peak Unit 1	Texas Utilities Generating Co.	Texas	1980	1150	4
Seabrook Unit 1	Public Service Company of New Hampshire; United Illuminating Company	New Hampshire	1980	1200	4
South Texas Project Unit 1	Houston Lighting and Power Co.; Central Power and Light Co.; City Public Service of San Antonio; City of Austin, Texas	Texas	1980	1250	4

TABLE 1.4-3 (Cont'd)

PLANT	OWNER UTILITY	LOCATION	SCHEDULED COMMERCIAL OPERATION	MWe NET	NUMBER OF LOOPS
Beaver Valley Unit 2	Duquesne Light Company; Ohio Edison Company; Pennsylvania Power Co.; Cleveland Electric Illuminating Company; Toledo Edison Company	Pennsylvania	1981	852	3
Braidwood Unit 1	Exelon Generation Company	Illinois	1981	1120	4
Callaway Unit 1	SNUPPS - Union Electric Co.	Missouri	1981	1150	4
Catawba Unit 1	Duke Power Company	South Carolina	1981	1153	4
Jamesport Unit 1	Long Island Lighting Company	New York	1981	1150	4
Ko-Ri Unit 2	Korea Electric Power Co., Ltd.	Korea	1981	605	2
NORCO	Puerto Rico Water Resources Authority	Puerto Rico	-	583	2
Braidwood Unit 2	Exelon Generation Company	Illinois	1982	1120	4
Byron Unit 2	Exelon Generation Company	Illinois	1982	1120	4
Catawba Unit 2	Duke Power Company	South Carolina	1982	1153	4
Comanche Peak Unit 2	Texas Utilities Generating Co.	Texas	1982	1150	4
Marble Hill Unit 1	Public Service Company of Indiana, Inc.; Northern Indiana Public Service Company	Indiana	1982	1150	4

TABLE 1.4-3 (Cont'd)

PLANT	OWNER UTILITY	LOCATION	SCHEDULED COMMERCIAL OPERATION	MWe NET	NUMBER OF LOOPS
Millstone Unit 3	Northeast Nuclear Energy Co.	Connecticut	1982	1156	4
Seabrook Unit 3	Public Service Company of New Hampshire; United Illuminating Company	New Hampshire	1982	1200	4
South Texas Project Unit 2	Houston Lighting and Power Co.; Central Power and Light Co.; City Public Service of San Antonio; City of Austin, Texas	Texas	1982	1250	4
Taiwan Unit 5	Taiwan Power Company	Taiwan	1982	950	3
Wolf Creek Unit 1	SNUPPS - Kansas Gas and Electric Company; Kansas City Power and Light Company	Kansas	1982	1150	4
Alvin W. Vogtle Unit 1	Georgia Power Company	Georgia	1983	1113	4
Callaway Unit 2	SNUPPS - Union Electric Company	Missouri	1983	1150	4
NEP-1	New England Power Company	-	1983	1150	4
Fort Calhoun Unit 2	Omaha Public Power District; Nebraska Public Power District	Nebraska	1983	1150	4
Jamesport Unit 2	Long Island Lighting Company	New York	1983	1150	4
Sears Island	Central Maine Power Company	Maine	-	1200	4

TABLE 1.4-3 (Cont'd)

PLANT	OWNER UTILITY	LOCATION	SCHEDULED COMMERCIAL OPERATION	MWe NET	NUMBER OF LOOPS
Taiwan Unit 6	Taiwan Power Company	Taiwan	1983	950	3
Alvin W. Vogtle Unit 2	Georgia Power Company	Georgia	1984	1113	4
Marble Hill Unit 2	Public Service Company of Indiana, Inc.; Northern Indiana Public Service Company	Indiana	1984	1150	4
Shearon Harris Unit 1	Carolina Power and Light Co.	North Carolina	1984	900	3
Sterling	SNUPPS - Rochester Gas and Electric Corporation; Central Hudson Gas and Electric Corporation; Niagara Mohawk Power Corporation; Orange and Rockland Utilities, Inc.	New York	1984	1150	4
Atlantic Unit 1 (O.P.S.)	Public Service Electric and Gas Company; Atlantic City Electric Co.; Jersey Central Power and Light Company	New Jersey	1985	1150	4
NEP-2	New England Power Company	-	1985	1150	4
South Dade Unit 1	Florida Power and Light Co.	Florida	1985	1150	4
Sundesert Unit 1	San Diego Gas and Electric Co.	California	1985	950	3

TABLE 1.4-3 (Cont'd)

PLANT	OWNER UTILITY	LOCATION	SCHEDULED COMMERCIAL OPERATION	MWe NET	NUMBER OF LOOPS
Tyrone Unit 1	SNUPPS - Northern States Power Company	Wisconsin	1985	1150	4
Shearon Harris Unit 2	Carolina Power and Light Co.	North Carolina	1986	900	3
South Dade Unit 2	Florida Power and Light Co.	Florida	1986	1150	4
Atlantic No. (O.P.S.)	Public Service Electric and Gas Company; Atlantic City Electric Co.; Jersey Central Power and Light Company	New Jersey	1987	1150	4
Shearon Harris Unit 4	Carolina Power and Light Co.	North Carolina	1988	900	3
Sundesert Unit 2	San Diego Gas and Electric Co.	California	1988	950	3
Sayago Unit 1	Iberduero, S.A.	Spain	1980's	1000	3
Sayago Unit 4	Iberduero, S.A.	Spain	1980's	1000	3
Shearon Harris Unit 3	Carolina Power and Light Co.	North Carolina	1990	900	3
Unassigned Unit 1 (O.P.S.)	Public Service Electric and Gas Company; Atlantic City Electric Company	New Jersey	1990	1150	4
Unassigned Unit 2 (O.P.S.)	Public Service Electric and Gas Company; Atlantic City Electric Company	New Jersey	1992	1150	4

1.5 REQUIREMENTS FOR FURTHER TECHNICAL INFORMATION

The design of the Byron/Braidwood units is based upon proven concepts which have been developed and successfully applied to the design of pressurized water reactor systems. There are currently no areas of research and development which are required for operation of this plant.

At the time of issuance of construction permits for the Byron/ Braidwood units, the Preliminary Safety Analysis Report (PSAR) and the standard design report which it referenced, RESAR-3, identified certain research and development programs which were incomplete. These programs, which have been successfully completed, have provided technical information which has been used either to demonstrate the safety of design, more sharply define margins of conservatism, or lead to design improvements. Reference 1 presents descriptions of those safety-related research and development programs which have been carried out for, by, or in conjunction with Westinghouse Nuclear Energy Systems, and which are applicable to Westinghouse pressurized water reactors. The discussion which follows documents the completion of the construction permit stage research programs.

1.5.1 Programs Required for Plant Operation

Two programs were identified as required for plant design and operation in the PSAR:

a. core stability evaluation and

b. fuel rod burst program.

Both programs are complete. The fuel rod burst program was completed at the time of the PSAR. The core stability evaluation program was not. A discussion of the core stability evaluation program follows.

1.5.1.1 Core Stability Evaluation

The program to establish means for the detection and control of potential xenon oscillations and for the shaping of the axial power distribution for improved core performance has been satisfactorily completed. See item 1, Reference 2, for a further discussion of the tests and results.

1.5.2 Other Programs Not Required for Plant Operation

The following programs were not complete at the time of the PSAR but are now satisfactorily complete.

1.5.2.1 Fuel Development Program for Operation at High Power Densities

The program to demonstrate the satisfactory operation of fuel at high burnup and power densities has been satisfactorily completed. See item 8, Reference 2, for a further discussion of the program and its results.

1.5.2.2 Blowdown Forces Program

Westinghouse has completed BLODWN-2, an improved digital computer program for the calculation of local fluid pressures, flows and density transients in the primary coolant systems during a LOCA.

BLODWN-2 is used to evaluate the effects of blowdown forces in this application. Refer to item 15 in Reference 4 for a further discussion of the tests and results.

1.5.2.3 Blowdown Heat Transfer Testing (Formerly Titled Delayed Departure From Nucleate Boiling)

The NRC Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Power Reactors was issued in Section 50.46 of 10 CFR 50 on December 28, 1973. It defines the basis and conservative assumptions to be used in the evaluation of the performance of emergency core cooling systems (ECCS). Westinghouse believes that some of the conservatism of the criteria is associated with the manner in which transient DNB phenomena are treated in the evaluation models. Transient critical heat flux data presented at the 1972 specialists meeting of the Committee on Reactor Safety Technology (CREST) indicated that the time to DNB can be delayed under transient conditions. To demonstrate the conservatism of the ECCS evaluation models, Westinghouse initiated a program to experimentally simulate the blowdown phase of a LOCA. This testing is part of the Electric Power Research Institute (EPRI) sponsored Blowdown Heat Transfer Program, which was started early in 1976. Testing was completed in 1979. A DNB correlation developed by Westinghouse from these test results is used in the ECCS analyses for Byron/Braidwood.

Objective

The objective of the blowdown heat transfer test was to determine the time that DNB occurs under LOCA conditions. This information was used to confirm a new Westinghouse transient DNB correlation. The steady-state DNB data obtained from 15x15 and 17x17 test programs was used to assure that the geometrical differences between the two fuel arrays is correctly treated in the transient correlations.

Program

The program was divided into two phases. The Phase I tests started from steady-state conditions, with sufficient power to maintain nucleate boiling throughout the bundle, and progressed through controlled ramps of decreasing test section pressure or flow initiated DNB. By applying a series of controlled conditions, investigation of the DNB was studied over a range of qualities and flows, and at pressures relevant to a PWR blowdown.

Phase I provided separate-effects data for heat transfer correlation development.

Typical parameters used for Phase I testing are shown in Table 1.5-1.

Phase II simulated PWR behavior during a LOCA to permit definition of the time delay associated with onset of DNB. Tests in this phase covered the large double-ended guillotine cold leg break. All tests in Phase II were also started after establishment of typical steady-state operating conditions. The fluid transient was then initiated, and the rod power decay was programmed in such a manner as to simulate the actual heat input of fuel rods. The test was terminated when the heater rod temperatures reached a predetermined limit.

Typical parameters used for Phase II testing are shown in Table 1.5-2.

Test Description

The experimental program was conducted in the J-Loop at the Westinghouse Forest Hills Facility with a full length 5x5 rod bundle simulating a section of a 15x15 fuel assembly to determine DNB occurrence under LOCA conditions.

The heater rod bundles used in this program were internally-heated rods, capable of a maximum linear power of 18.8 kW/ft, with a total power of 135 kW (for extended periods) over the 12-foot heated length of the rod. Heat was generated internally by means of a varying cross-sectional resistor which approximates a chopped cosine power distribution. Each rod was adequately instrumented with a total of 12 clad thermocouples.

Results

The experiments in the DNB facility resulted in cladding temperature and fluid properties measured as a function of time throughout the blowdown range from 0 to 20 seconds.

Facility modifications and installation of the initial test bundle were completed. A series of shakedown tests in the

J-Loop were performed. These tests provided data for instrumentation calibration and check-out, and provided information regarding facility control and performance. Initial program tests were performed during the first half of 1975. Under the sponsorship of EPRI, testing was reinitiated during 1976 on the same test bundle. The testing was terminated in November 1976 and plans were made for a new test bundle and further testing during 1978-1979. These tests were completed in December of 1979.

1.5.3 References

1. F. T. Eggleston, "Safety-Related Research and Development for Westinghouse Pressurized Water Reactors, Program Summaries," WCAP-8768, October 1978.

2. F. T. Eggleston, "Safety-Related Research and Development for Westinghouse PWRs Program Summaries," WCAP-8768. Spring 1976 Edition.

3. "Safety-Related Research and Development for Westinghouse PWRs Program Summaries," WCAP-8458. Fall 1977 Edition.

4. "Safety-Related Research and Development for Westinghouse PWRs Program Summaries," WCAP-8004. Fall 1972 Edition.

TABLE 1.5-1

BLOWDOWN HEAT TRANSFER PHASE I TEST PARAMETERS

PARAMETERS	NOMINAL VALUE
INITIAL STEADY-STATE CONDITIONS	
Pressure	1250 to 2250 psia
Test section mass velocity	1.12 to 2.5x10^{6} lb/hr-ft^{2}
Core inlet temperature	550° F to 600° F
Maximum heat flux	306,000 to 531,000 Btu/hr-ft^{2}
TRANSIENT RAMP CONDITIONS	
Pressure decrease	0 to 350 psia/sec and subcooled depressurization from 2250 psia
Flow decrease	0 to 100%/sec
Inlet enthalpy	constant

TABLE 1.5-2

BLOWDOWN HEAT TRANSFER PHASE II TEST PARAMETERS

PARAMETERS	NOMINAL VALUE
INITIAL STEADY-STATE CONDITIONS	
Pressure	2250 psia
Test section mass velocity	2.5×10^6 lb/hr-ft^2
Inlet coolant temperature	545° F
Maximum heat flux	531,000 Btu/hr-ft^2
TRANSIENT CONDITIONS	
Simulated break	Double-ended cold leg guillotine breaks

1.6 MATERIAL INCORPORATED BY REFERENCES

Table 1.6-1 lists topical reports which provide information additional to that provided in this UFSAR and which have been filed separately with the Nuclear Regulatory Commission (NRC) in support of this and similar applications.

A legend to the review status code letters follows:

A	- NRC review complete; NRC acceptance letter issued.
AE	- NRC accepted as part of the Westinghouse Emergency Core Cooling System (ECCS) evaluation model only; does not constitute acceptance for any purpose other than for ECCS analyses.
B	- Submitted to the NRC as background information; not undergoing formal NRC review.
O	- On file with NRC; older generation report with current validity; not actively under formal NRC review.
U	- Actively under formal NRC review.

TABLE 1.6-1

TOPICAL REPORTS INCORPORATED BY REFERENCE

REPORT	REFERENCE SECTION(S)	REVIEW STATUS
"The Doppler Effect for a Non-Uniform Temperature Distribution in Reactor Fuel Elements," WCAP-2048, July 1962	4.3	0
"Single Phase Local Boiling and Bulk Boiling Pressure Drop Correlations," WCAP-2850 (Proprietary), April 1966 and WCAP-7916 (Non-Proprietary), June 1972	4.4	0
"In-Pile Measurement of UO_2 Thermal Conductivity," WCAP-2923, 1966	4.4	0
"Hydraulic Tests of the San Onofre Reactor Model," WCAP-3269-8, June 1964	4.4	0
"LEOPARD - A Spectrum Dependent Non-Spatial Depletion Code for the IBM - 7094," WCAP-3269-26, September 1963	4.3, 4.4 15.0, 15.4	0
"Saxton Core II Fuel Performance Evaluation," WCAP-3385-56, Part II, "Evaluation of Mass Spectrometric and Radiochemical Analyses of Irradiated Saxton Plutonium Fuel," July 1970	4.3, 4.4	0
"Xenon-Induced Spatial Instabilities in Large PWRs," WCAP-3680-20, (EURAEC-1974) March 1968	4.3	0
"Control Procedures for Xenon-Induced X-Y Instabilities in Large PWR's," WCAP-3680-21, (EURAEC-2111) February 1969	4.3	0
"Xenon-Induced Spatial Instabilities in Three-Dimensions," WCAP-3680-22, (EURAEC-2116) September 1969	4.3	0
"Pressurized Water Reactor pH – Reactivity Effect Final Report," WCAP-3698-8, (EURAEC-2074) October 1968	4.3	0
"PUO_2 - UO_2 Fueled Critical Experiments," WCAP-3726-I, July 1967	4.3	0

TABLE 1.6-1 (Cont'd)

REPORT	REFERENCE SECTION(S)	REVIEW STATUS
"Melting Point of Irradiated UO_2," WCAP-6065, February 1965	4.2, 4.4	0
"Burnup Physics of Heterogeneous Reactor Lattices," WCAP-6069, June 1965	4.4	0
"LASER - A Depletion Program for Lattice Calculations Based on MUFT and THERMOS," WCAP-6073, April 1966	4.3	0
"Supplementary Report on Evaluation of Mass Spectrometric and Radiochemical Analyses of Yankee Core I Spent Fuel, Including Isotopes of Elements Thorium Through Curium," WCAP-6086, August 1969	4.3	0
"Subchannel Thermal Analysis of Rod Bundle Cores," WCAP-7015, Revision 1, January 1969	4.4	0
"The PANDA Code," WCAP-7048 (Proprietary) and WCAP-7757 (Non-Proprietary), January 1975	4.3	A
"Evaluation of Protective Coatings for Use in Reactor Containment," WCAP-7198-L (Proprietary), April 1969 and WCAP-7825 (Non-Proprietary), December 1971	4.3	0
"Power Distribution Control of Westinghouse Pressurized Water Reactors," WCAP-7208 (Proprietary), September 1968 and WCAP-7811, (Non-Proprietary), December 1971	4.3	-
"The TURTLE 24.0 Diffusion Depletion Code," WCAP-7213 (Proprietary) and WCAP-7758 (Non-Proprietary), January 1975	4.3, 15.0 15.4	A
"Core Power Capability in Westinghouse PWRs," WCAP-7267-L (Proprietary), October 1969 and WCAP-7809 (Non-Proprietary), December 1971	4.3	-
"Reactor Protection System Diversity in Westinghouse Pressurized Water Reactors," WCAP-7306, April 1969	15.4	-
"Evaluation of Nuclear Hot Channel Factor Uncertainties," WCAP-7308, December 1971	4.3	A

TABLE 1.6-1 (Cont'd)

REPORT	REFERENCE SECTION(S)	REVIEW STATUS
"Application of the THINC Program to PWR Design," WCAP-7359-L (Proprietary), August 1969 and WCAP-7838 (Non-Proprietary), January 1972	4.4	O
"Seismic Testing of Electrical and Control Equipment," WCAP-7397-L (Proprietary) and WCAP-7817 (Non-Proprietary), December 1971	3.10	O
"Seismic Testing of Electrical and Control Equipment (WCID Process Control Equipment)," WCAP-7397-L, Supplement 1 (Proprietary) and WCAP-7817, Supplement 1 (Non-Proprietary), December 1971	3.10	O
"Sensitized Stainless Steel in Westinghouse PWR Nuclear Steam Supply Systems," WCAP-7477-L (Proprietary), March 1970 and WCAP-7735 (Non-Proprietary), August 1971	5.2	A
"Radiological Consequences of a Fuel Handling Accident," WCAP-7518-L (Proprietary) and WCAP-7828 (Non-Proprietary), June 1970	15.7	O
"Seismic Vibration Testing with Sine Beats," WCAP-7558, October 1972	3.10	O
"An Evaluation of the Rod Ejection Accident in Westinghouse Pressurized Water Reactors Using Spatial Kinetics Methods," WCAP-7588, Revision 1-A, January 1975	15.4	A
"Dynamic Fracture Toughness Properties of Heavy Section A533 Grade B Class 1 Steel Plate," WCAP-7623, December 1970	5.4	O
"Interchannel Thermal Mixing with Mixing Vane Grids," WCAP-7667-L (Proprietary) and WCAP-7755 (Non-Proprietary), January 1975	4.4	A
"DNB Tests Results for New Mixing Vane Grids (R)," WCAP-7695-L (Proprietary) and WCAP-7958 (Non-Proprietary) and Addendum, January 1975	4.4	A

TABLE 1.6-1 (Cont'd)

REPORT	REFERENCE SECTION(S)	REVIEW STATUS
"An Evaluation of Solid State Logic Reactor Protection in Anticipated Transients," WCAP-7706, February 1973	4.6, 7.1	O
"Electric Hydrogen Recombiner for PWR Containments," WCAP-7709-L, Supplements 1 through 7 (Proprietary) and WCAP-7820, Supplements 1 through 7 (Non-Proprietary), 1971 through 1977	3.11, 6.2	A
"A Comprehensive Space-Time Dependent Analysis of Loss of Coolant (SATAN-IV Digital Code)," WCAP-7750, August 1971	3.6	O
"Overpressure Protection for Westinghouse Pressurized Water Reactors," WCAP-7769, October 1971	15.2	O
"Overpressure Protection for Westinghouse Pressurized Water Reactors," WCAP-7769, Revision 1, June 1972	5.2	O
"Behavior of Austenitic Stainless Steel in Post Hypothetical Loss of Coolant Accident Environment," WCAP-7798-L (Proprietary) and WCAP-7803 (Non-Proprietary), January 1972	6.1	O
"Nuclear Fuel Division Quality Assurance Program Plan," WCAP-7800, Revision 4-A, April 1975	4.2, 17	A
"Nuclear Design of Westinghouse Pressurized Water Reactors with Burnable Poison Rods," WCAP-7806, December 1971	4.3	B
"Power Distribution Control of Westinghouse Pressurized Water Reactors," WCAP-7811, December 1971	4.3	O
"Seismic Testing of Electrical and Control Equipment (Low Seismic Plants)," WCAP-7817, Supplements 1-8, December 1971-March 1974	3.10	O

TABLE 1.6-1 (Cont'd)

REPORT	REFERENCE SECTION(S)	REVIEW STATUS
"Evaluation of Steam Generator Tube, Tubesheet and Divider Plate Under Combined LOCA Plus SSE Conditions," WCAP-7832, December 1973	5.4	A
"Inlet Orificing of Open PWR Cores," WCAP-7836, January 1972	4.4	B
"Neutron Shielding Pads," WCAP-7870, May 1972	3.9	A
"LOFTRAN Code Description," WCAP-7907, June 1972	5.2, 15.0 15.1, 15.2, 15.3, 15.4, 15.5, 15.6	A
"FACTRAN - A FORTRAN-IV Code for Thermal Transients in a UO_2 Fuel Rod," WCAP-7908, June 1972	15.0, 15.2 15.3, 15.4	A
"MARVEL, A Digital Computer Code for Transient Analysis of a Multiloop PWR System," WCAP-7909, June 1972	6.3	O
"Power Peaking Factors," WCAP-7912-L (Proprietary) and WCAP-7912 (Non-Proprietary), January 1975 and Supplement	4.3, 4.4	A
"Damping Values of Nuclear Power Plant Components," WCAP-7921, May 1974	1A, 3.7	A
"Basis for Heatup and Cooldown Limit Curves," WCAP-7924, April 1975	5.3	A
"Effect of Axial Spacing on Interchannel Thermal Mixing with the R Mixing Vane Grid," WCAP-7941-L (Proprietary) and WCAP-7959 (Non-Proprietary), January 1975	4.4	A
"Fuel Assembly Safety Analysis for Combined Seismic and Loss of Coolant Accident, 15x15," WCAP-7950, July 1972	3.7	A
"THINC-IV An Improved Program for Thermal and Hydraulic Analysis of Rod Bundle Cores," WCAP-7956, June 1973	4.4	A

TABLE 1.6-1 (Cont'd)

REPORT	REFERENCE SECTION(S)	REVIEW STATUS
"Axial Xenon Transient Tests at the Rochester Gas and Electric Reactor," WCAP-7964, June 1971	4.3	O
"TWINKLE - A Multi-Dimensional Neutron Kinetics Computer Code," WCAP-7979 (Proprietary) and WCAP-8028 (Non-Proprietary), January 1975	15.0, 15.4	A
"WIT-6 Reactor Transient Analysis Computer Program Description," WCAP-7980, November 1972	15.0, 15.4	A
"Application of Modified Spacer Factor to "L" Grid Typical and Cold Wall Cell DNB," WCAP-7988 (Proprietary) and WCAP-8030 (Non-Proprietary), October 1972	4.4	A
"Application of the THINC-IV Program to PWR Design," WCAP-8054 (Proprietary) and WCAP-8195 (Non-Proprietary), October 1973	4.4	A
"Pipe Breaks for the LOCA Analysis of the Westinghouse Primary Coolant Loop," WCAP-8082 (Proprietary) and WCAP-8172 (Non-Proprietary), January 1975	3.6	A
"Reactor Coolant Pump Integrity in LOCA," WCAP-8163, September 1973	lA, 5.4	O
"Calculational Model for Core Reflooding After a Loss of Coolant Accident (WREFLOOD Code)," WCAP-8170 (Proprietary) and WCAP-8171 (Non-Proprietary), June 1974	15.6	A
"Effect of Local Heat Flux Spikes on DNB in Non-Uniform Heated Rod Bundles," WCAP-8174 (Proprietary) and WCAP-8202, (Non-Proprietary), August 1973	4.4	A
"WFLASH, A FORTRAN-IV Computer Program for Simulation of Transients in a Multi-Loop PWR," WCAP-8200, Revision 2 (Proprietary) and WCAP-8261, Revision 1 (Non-Proprietary), July 1974	15.6	A

TABLE 1.6-1 (Cont'd)

REPORT	REFERENCE SECTION(S)	REVIEW STATUS
"Fuel Densification Experimental Results and Model for Reactor Application," WCAP-8218 (Proprietary) and WCAP-8219 (Non-Proprietary), March 1975	4.1, 4.2, 4.3, 4.4	A
"Safety Analysis of the 17x17 Fuel Assembly for Combined Seismic and Loss of Coolant Accident," WCAP-8236 (Proprietary), December 1973 and WCAP-8288 (Non-Proprietary), January 1974 and Addenda	3.7, 4.2	A
"Safety Analysis of the 8-Grid 17x17 Fuel Assembly for Combined Seismic and Loss of Coolant Accident," WCAP-8236, Addendum 1 (Proprietary), March 1974 and WCAP-8288, Addendum 1 (Non-Proprietary), April 1974	3.7	A
"Documentation of Selected Westinghouse Structural Analysis Computer Codes," WCAP-8252, Revision 1, July 1977	3.6, 3.9	O
"Hydraulic Flow Test of the 17x17 Fuel Assembly," WCAP-8278 (Proprietary) and WCAP-8279 (Non-Proprietary), February 1974	4.2, 4.4	O
"Effect of 17x17 Fuel Assembly Geometry on DNB," WCAP-8296 (Westinghouse Proprietary) and WCAP-8927 (Non-Proprietary), February 1975	4.4	A
"The Effect of 17x17 Fuel Assembly Geometry on Interchannel Thermal Mixing," WCAP-8298 (Proprietary) and WCAP-8299 (Non-Proprietary), January 1975	4.4	A
"LOCTA-IV Program: Loss of Coolant Transient Analysis," WCAP-8301 (Proprietary) and WCAP-8305 (Non-Proprietary), June 1974	15.0, 15.6	AE
SATAN-IV Program: Comprehensive Space-Time Dependent Analysis of Loss of Coolant," WCAP-8302 (Proprietary) and WCAP-8306 (Non-Proprietary), June 1974	15.0, 15.6	AE

TABLE 1.6-1 (Cont'd)

REPORT	REFERENCE SECTION(S)	REVIEW STATUS
"Prediction of the Flow-Induced Vibration of Reactor Internals by Scale Model Tests," WCAP-8303 (Proprietary) and WCAP-8317 (Non-Proprietary), July 1975	3.9	A
"Control of Delta Ferrite in Austenitic Stainless Steel Weldments," WCAP-8324-A, June 1974	1A, 5.2	A
"Containment Pressure Analysis Code (COCO)," WCAP-8327 (Proprietary) and WCAP-8326 (Non-Proprietary), June 1974	15.6	AE
"Westinghouse Anticipated Transients Without Trip Analysis," WCAP-8330, August 1974	4.3, 4.6, 15.1, 15.2, 15.4, 15.8	O
"Westinghouse ECCS Evaluation Model - Summary," WCAP-8339, July 1974	6.2, 15.6	AE
"Westinghouse ECCS - Plant Sensitivity Studies," WCAP-8340 (Proprietary) and WCAP-8356 (Non-Proprietary), July 1974	15.6	AE
"Westinghouse ECCS Evaluation Model Sensitivity Studies," WCAP -8341 (Proprietary) and WCAP-8342 (Non-Proprietary), July 1974	1A(N), 17	A
"Effects of Fuel Densification Power Spikes on Clad Thermal Transients," WCAP-8359, July 1974	4.3	AE
"Westinghouse Nuclear Energy Systems Division Quality Assurance Plan," WCAP-8370, Revision 9A, September 1977	1A, 17	A
"Qualification of Westinghouse Seismic Testing Procedure for Electrical Equipment Tested Prior to May 1974," WCAP-8373, August 1974	3.10	O
"Revised Clad Flattening Model," WCAP-8377 (Proprietary) and WCAP-8381 (Non-Proprietary), July 1974	4.2	A

TABLE 1.6-1 (Cont'd)

REPORT	REFERENCE SECTION(S)	REVIEW STATUS
"Power Distribution Control and Load Following Procedures," WCAP-8385 (Proprietary) and WCAP-8403 (Non-Proprietary), September 1974	4.3, 4.4	A
"An Evaluation of Loss of Flow Accidents Caused by Power System Frequency Transients in Westinghouse PWRs," WCAP-8424, Revision 1, June 1975	15.3	O
"17x17 Drive Line Components Tests - Phase IB, II, III, D-Loop Drop and Deflection," WCAP-8446 (Proprietary) and WCAP-8449 (Non-Proprietary), December 1974	3.9, 15.0	A
"Analysis of Data from the Zion (Unit 1) THINC Verification Test," WCAP-8453-A (Proprietary), May 1976 and WCAP-8454 (Non-Proprietary), January 1975	4.4	A
"Westinghouse ECCS Evaluation Model - Supplementary Information," WCAP-8471 (Proprietary) and WCAP-8472 (Non-Proprietary), April 1974	15.6	AE
"Incore Power Distribution Determination in Westinghouse Pressurized Water Reactors," WCAP-8498, July 1975	4.3	O
"UHI Plant Internals Vibration Measurement Program and Pre and Post Hot Functional Examinations," WCAP-8516-P (Proprietary) and WCAP-8517 (Non-Proprietary), April 1975	3.9	A
"Critical Heat Flux Testing of 17x17 Fuel Assembly Geometry with 22 Inch Spacing," WCAP-8536 (Proprietary) and WCAP-8537 (Non-Proprietary), May 1975	4.4	A
"Westinghouse ECCS - Four Loop Plant (17x17) Sensitivity Studies," WCAP-8565 (Proprietary) and WCAP-8566 (Non-Proprietary), July 1975	15.6	A

TABLE 1.6-1 (Cont'd)

REPORT	REFERENCE SECTION(S)	REVIEW STATUS
"Improved Thermal Design Procedure," WCAP-8567-P, July 1975 (Proprietary) and WCAP-8568, July 1975 (Non-Proprietary)	4.4, 15.0	A
"Augmented Startup and Cycle 1 Physics Program Supplement 1," WCAP-8575, June 1976 (Proprietary) and WCAP-8576, June 1976 (Non-Proprietary) and Supplements.	4.3	O
"The Application of Preheat Temperatures After Welding Pressure Vessel Steels," WCAP-8577, February 1976	1A	A
"Failure Mode and Effects Analysis (FMEA) of the Engineered Safeguard Features Actuation System," WCAP-8584 (Proprietary) and WCAP-8760 (Non-Proprietary), April 1976	4.6	O
"Environmental Qualification of Westinghouse NSSS Class 1E Equipment," WCAP-8587, September 1975	1A, 3.10, 3.11	A
"Westinghouse ECCS Evaluation Model - October 1975 Version," WCAP-8622 (Proprietary) and WCAP-8623 (Non-Proprietary), November 1975	15.6	A
"Experimental Verification of Wet Fuel Storage Criticality Analyses," WCAP-8682 (Proprietary) and WCAP-8683 (Non-Proprietary), December 1975	4.3	B
"Fuel Rod Bowing," WCAP-8691 (Proprietary) and WCAP-8692 (Non-Proprietary), December 1975	4.2	O
"Delta Ferrite in Production Austenitic Stainless Steel Weldments," WCAP-8693, January 1976	1A, 5.2	B
"MULTIFLEX - A FORTRAN-IV Computer Program for Analyzing Thermal-Hydraulic-Structure System Dynamics," WCAP-8708 (Proprietary) and WCAP-8709 (Non-Proprietary), February 1976	3.9	A

TABLE 1.6-1 (Cont'd)

REPORT	REFERENCE SECTION(S)	REVIEW STATUS
"Improved Analytical Models Used in Westinghouse Fuel Rod Design Computations," WCAP-8720 (Proprietary) and WCAP-8785 (Non-Proprietary), October 1976	4.2	A
"New Westinghouse Correlation WRB-1 for Predicting Critical Heat Flux in Rod Bundles with Mixing Vane Grids," WCAP-8762, July 1976 (Proprietary) and WCAP-8763, July 1976 (Non-Proprietary)	4.4	A
"Safety-Related Research and Development for Westinghouse Pressurized Water Reactors, Program Summaries," WCAP-8768, Revision 2, October 1978	1.5, 4.2, 4.3	B
"Verification of Neutron Pad and 17x17 Guide Tube Designs by Preoperational Tests on the Trojan 1 Power Plant," WCAP-8780, May 1976	3.9	B
"Improved Analytical Models Used in Westinghouse Fuel Rod Design Computations," WCAP-8785, October 1976	4.2	-
"Hybrid B_{4C} Absorber Control Rod Evaluation Report," WCAP-8846, October 1977	4.2, 15.0 15.3	A
"Westinghouse ECCS - Four Loop Plant (17x17) Sensitivity Studies with Upper Head Fluid Temperature at T_{hot}," WCAP-8865, May 1977	15.6	A
"7300 Series Process Control System Noise Tests," WCAP-8892-A, April 1977	7.1	A
"Safety Analysis for the Revised Fuel Rod Internal Pressure Design Basis," WCAP-8963 (Proprietary), November 1976 and WCAP-8964 (Non-Proprietary), August 1977	4.2	A
"Westinghouse Emergency Core Cooling System Small Break October 1975 Model," WCAP-8970 (Proprietary) and WCAP-8971 (Non-Proprietary), April 1977	15.6	A

TABLE 1.6-1 (Cont'd)

REPORT	REFERENCE SECTION(S)	REVIEW STATUS
"Failure Mode and Effects Analysis of the Solid State Full Length Rod Control System," WCAP-8976, September 1977	4.6	O
"Nuclear Design of Westinghouse Pressurized Water Reactors with Burnable Poison Rods," WCAP-9000-L, Revision 1 (Proprietary), July 1969 and WCAP-7806 (Non-Proprietary), December 1971.	4.3	
"Axial Power Distribution Monitoring Using Four-Section Ex-Core Detectors," WCAP-9105 (Proprietary) and WCAP-9106 (Non-Proprietary), July 1977	4.3	A
"Westinghouse Emergency Core Cooling System Evaluation Model for Analyzing Large LOCAs During Operation with One Loop Out of Service for Plants Without Loop Isolation Valves," WCAP-9166 (Proprietary) and WCAP-9167 (Non-Proprietary), February 1978	15.6	O
"Westinghouse Emergency Core Cooling System Evaluation Model - Modified October 1975 Version," WCAP-9168 (Proprietary) and WCAP-9150 (Non-Proprietary), September 1977	15.6	O
"Properties of Fuel and Core Component Materials," WCAP-9179 (Proprietary), September 1977 and WCAP-9224 (Non-Proprietary)	4.2	O
"Westinghouse ECCS Evaluation Model, February 1978 Version," WCAP-9220 (Proprietary Version), WCAP-9221 (Non-Proprietary Version), February 1978	15.6	A
"Verification Testing and Analyses of the 17x17 Optimized Fuel Assembly," WCAP-9401 (Proprietary) and WCAP-9402 (Non-Proprietary), March 1979	4.1, 4.2, 4.4	A
"PALADON - Westinghouse Nodal Computer Code," WCAP-9485 (Proprietary) and WCAP-9486 (Non-Proprietary) December 1978	4.3	A

TABLE 1.6-1 (Cont'd)

REPORT	REFERENCE SECTION(S)	REVIEW STATUS
"Reference Core Report 17x17 Optimized Fuel Assembly," WCAP-9500 (Non Proprietary), July 1979	4, 15	A
"RELAP5/MOD2-B&W - An Advanced Computer Code for Light Water Reactor LOCA and non-LOCA Transient Analysis" BAW-10164, Revision 3 (non-proprietary), October 1996	15	A
"CONTEMPT - Computer Program for Predicting Containment Pressure-Temperature Response to a Loss-of-Coolant Accident,", BAW-10095A, Revision 1, April 1978	6	O
“Beacon Core Monitoring and Operations Support System,” WCAP-12472 (Proprietary Class 2), August 1994	4.3, 4.4, 7.7	A
“Relaxation of Constant Axial Offset Control, FQ Surveillance Technical Specification,” WCAP-10216-P-A, Revision 1A (Proprietary Class 2), February 1994	4.3, 4.4	A

1.7 DRAWINGS

The drawings cited in each UFSAR Chapter are included as "General References" only; i.e., refer to the drawings to obtain additional detail or to obtain background information. These drawings are not part of the UFSAR. They are controlled by the Controlled Documents Program. References on the figures contained in the UFSAR to ComEd, CECo, and Commonwealth Edison will be revised to reflect the change in facility ownership to Exelon Generation Company when other changes to that figure are needed.

1.7.1 Electrical, Instrumentation, and Control Drawings

Subsection 1.7.1 of the FSAR provides a list of electrical, instrumentation, and control drawings that were provided to the NRC during the initial licensing phase.

1.7.2 Drawings for Independent Structural Review

Subsection 1.7.2 of the FSAR provides a list of the structural, architectural, mechanical loading and electrical loading drawings that were provided to the NRC to enable them to perform the Project Structural Review and the Independent Structural Review during the licensing phase.

TABLE 1.7-1

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Pages 1.7-3 through 1.7-17 have been intentionally deleted.

Figures 1.1-1 through 1.1-3 have been deleted intentionally.

Figures 1.2-1 through 1.2-17 have been deleted intentionally.