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1.0 INTRODUCTION AND GENERAL DESCRIPTION OF PLANT

1.1 INTRODUCTION

This Updated Final Safety Analysis Report (UFSAR) for Dresden Station is an updated version of the Final Safety Analysis Report (FSAR) and follows a different format from the FSAR. The FSAR was written prior to issuance of Regulatory Guide 1.70, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants." In an effort to provide consistency with the industry and other CECo stations, the UFSAR was rebaselined and reformatted in 1993 using Regulatory Guide 1.70, Revision 3, November 1978, as guidance. The guidance in NEI 98-03 Revision 1, as endorsed by Regulatory Guide 1.181, Revision 0, is also used as applicable.

This UFSAR contains a description of Dresden Units 2 and 3. Unit 1 was retired on August 31, 1984, but its major structures are still present and intact. Discussion of Unit 1 structures, systems, and components is limited to the physical or analytical interfaces with Units 2 and 3.

The Nuclear Regulatory Commission approved the transfer of the facility licenses from Commonwealth Edison (ComEd) Company to Exelon Generation Company, LLC (EGC) on January 12, 2001. References in the UFSAR to ComEd, CECo, and Commonwealth Edison have been retained, as appropriate, instead of being changed to EGC to properly preserve the historical content.

1.1.1 Background

The original FSAR was submitted in support of the application of Commonwealth Edison Company (CECo) for facility licenses for Units 2 and 3 at its Dresden Nuclear Power Station, under Section 104(B) of the Atomic Energy Act of 1954, as amended, and the regulations of the Nuclear Regulatory Commission (NRC) set forth in Part 50 of Title 10 of the Code of Federal Regulations (10 CFR 50).

Construction of Units 2 and 3 was authorized by the NRC by issuance of a construction permit for Unit 2, CPPR-18, on January 10, 1966, in NRC Docket 50-237 and a construction permit for Unit 3, on October 14, 1966, in NRC Docket 50-249.

Units 2 and 3 were completed and went into commercial service in June 1970 and November 1971, respectively. The full-term operating licenses expire on December 22, 2009, for Unit 2 and January 12, 2011, for Unit 3. The renewed operating license for Unit 2 expires on December 22, 2029 and the renewed operating license for Unit 3 expires on January 12, 2031.

This Safety Analysis Report analyzes the design of each unit for operation at a thermal output of 2957 MWt. The Plant Design and Analysis Reports (PDARs) previously filed in NRC Dockets 50-237 and 50-249 in which Units 2 and 3 were each analyzed for a "reference design" thermal output of 2255 MWt, equivalent to a net electrical output of 715 MWe. As stated in the PDARs, each of the units was designed to permit ultimate operation at power levels of about 2600 MWt. Analyses and modifications performed as part of the extended power uprate support operation of the units at 2957 MWt.

1.1.2 Purpose and Scope of the Safety Analysis Report

The purpose of this Safety Analysis Report is to provide the technical information required by 10 CFR 50.34 in order to establish a basis for evaluation of Units 2 and 3 with respect to the operating license for each unit.

Units 2 and 3 are similar in virtually all respects, e.g., design concepts and criteria, capacity, and components. Significant differences in the design of Units 2 and 3 are discussed throughout the UFSAR.

1.1.3 Organization of the Report

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

Tables and figures that are referenced in the text appear at the end of the section in which they are referenced: first tables, then figures. Tables and figures are numbered with the chapter and first level section numbers followed by a hyphen and a sequential number within the section, e.g., the second table in Section 2.4 of Chapter 2 is numbered Table 2.4-2. Key acronyms and initialisms used in the UFSAR are shown on Table 1.1-1.

An alphanumeric value is used to represent an appendix. For example, the second appendix of Chapter 3 would be labeled Appendix 3B. Appendices contain such information as data on site meteorology, conformance to design criteria, primary containment fabrication reports, reactor pressure vessel reports, etc. The Technical Specifications are in Volume III of the original FSAR. They have since been made into a separate document and are referenced in Chapter 16 of the UFSAR.

1.1.4 Update and Revision of the Original FSAR

The UFSAR is separate and distinct from the original FSAR.

The original FSAR and the associated docket files (No. 50-237 and 50-249) are the basis for the licensing of the plant. The bases for the Technical Specifications may reference the UFSAR.

The UFSAR is designed to serve as a reference document, reflecting the current configuration of the plant, including information and analyses required by and submitted to the NRC since submission of the original FSAR.

Revisions are submitted to the NRC on a replacement page basis. Replacement pages include a page change identification (revision number and/or date) and a change indicator (a bold line drawn vertically in the right-hand margin adjacent to the portion actually changed).

1.1.4.1 Information for FSAR Controlled Copy Recipient

Dresden Station reviews the UFSAR pursuant to 10 CFR 50.71(e) for revisions, corrections, and material information additions. Revisions are made in compliance with the 10 CFR 50.71(e) requirement to identify changes and with the requirements defined in 10 CFR 50.59. The 50.59 report (issued to the NRC) refers to changes in the facility as described in the UFSAR, changes in procedures described in the UFSAR, and tests or experiments not described in the UFSAR.

All changes are reviewed against the 50.59 criteria and when the determination is made that such changes do not constitute any unreviewed safety question, the changes are implemented and the necessary revisions to the UFSAR are submitted to the NRC. UFSAR revisions are submitted no later than 24 calendar months from the date of the previous UFSAR revision.

1.1-1 Tracking No. 111365, Letter from T.K. Schuster (CECo) to E.E. Murley (NRC), June 14, 1991, Revision 6 to the Dresden Station GSEP Annex.

1.1-2 FSAR Section 1.1.1.

1.1-3 UFSAR Section 1.1.1; Tracking No. 111276, Letter from B.A. Bolger (NRC) to T.J. Kovach (CECo), February 20, 1991, Issuance of Full-Term Operating License, DPR-19; Tracking No. 250339, Letter from P.L. Eng (NRC) to T.J. Kovach (CECo), April 24, 1990, Issuance of Amendment to Facility Operating License; 10 CFR 50, Appendix A, GDC 12.

1.1-4 FSAR Section 1.1.1.

1.1-5 UFSAR Section 1.1.4.1.

1.1-6 UFSAR Section 1.1.4.1; FSAR Section 1.1.3.

1.1-7 Tracking No. 258161, UFSAR, Revision 5, 1987; 10 CFR 50.71(e); Tracking No. 302430, Procedure DAP 02-06, Revision 5, August 19, 1991.

1.1-8 UFSAR Section 1.1.2.2.

1.1-9 Tracking No 258161, UFSAR, Revision 5, 1987; 10 CFR 50.71(e); Tracking No. 302430, Procedure DAP 02-06, Revision 5, August 19, 1991; Tracking No. 303291, Letter from J.A. Bauer (CECo) to T.E. Murley (NRC), May 4, 1993, "UFSAR Revision Submittal Frequency for CECo Nuclear Stations"; Tracking No. 303269, Letter from J.B. Hickman (NRC) to D.L. Farrar (CECo), June 15, 1993, "CECo Proposal for UFSAR Revision Submittal Frequency."

Table 1.1-1

ACRONYMS AND INITIALISMS

ac	alternating current
ADS	automatic depressurization system
ALARA	as low as reasonably achievable
ANF	Advanced Nuclear Fuels (Exxon Nuclear Company [ENC] prior to January 1, 1987)
APRM	average power range monitor
ASF	Automatic Suppression Function
AST	Alternative Source Term
ASME	American Society of Mechanical Engineers
BTP	Branch Technical Position
Btu	British thermal unit
BWR	boiling water reactor
BWROG	Boiling Water Reactor Owners Group
CECo	Commonwealth Edison Company
CFR	Code of Federal Regulations
cps	counts per second
CSE	containment systems experiments
CCST	contaminated condensate storage tank
CVTR	Carolina Virginia Tube Reactor
DBA	design basis accident
DBE	design basis event
DG	diesel generator
DIB	Digital Isolation Block
DLR	Dosimeter of Legal Record
ECCS	emergency core cooling system
EGC	Exelon Generation Company, LLC
EHC	electrohydraulic control
EOF	emergency operations facility
ESF	engineered safety features
FSAR	Final Safety Analysis Report
FTOL	full-term operating license
GDC	General Design Criterion(a)
GE	General Electric Company
GNF	Global Nuclear Fuel
HELB	high energy line break
HEPA	high efficiency particulate air

Table 1.1-1 (Continued)

ACRONYMS AND INITIALISMS

hp	horsepower
HPCI	high pressure coolant injection
HVAC	heating, ventilating, and air conditioning
IE	Office of Inspection and Enforcement
IEEE	Institute of Electrical and Electronics Engineers
IPSAR	Integrated Plant Safety Assessment Report
IREP	Integrated Reliability Evaluation Program
IRM	intermediate range monitor
ISI	inservice inspection
IST	inservice testing
LCO	limiting condition for operation
LER	licensee event report
LOCA	loss-of-coolant accident
LPCI	low pressure coolant injection
LPRM	Local Power Range Monitor
LWR	light water reactor
MCC	motor control center
MCPR	minimum critical power ratio
MELB	moderate energy line break
MOV	motor-operated valve
mph	mile(s) per hour
MSIV	main steam isolation valve
MWD/MTU	megawatt-days per metric ton of uranium
MWe	megawatt-electric
MWt	megawatt-thermal
NRC	U.S. Nuclear Regulatory Commission
OPRM	Oscillating Power Range Monitor
OSC	Operational Support Center
ORNL	Oak Ridge National Laboratory
PDAR	Plant Design and Analysis Report
pH	hydrogen-ion concentration
PMF	probable maximum flood
PMP	probable maximum precipitation

Table 1.1-1 (Continued)

ACRONYMS AND INITIALISMS

POL	provisional operating license
ppm	parts per million
PRNMS	Power Range Neutron Monitoring System
psi	pounds per square inch
psia	pounds per square inch, absolute
psid	pounds per square inch, differential
psig	pounds per square inch, gauge
PWR	pressurized water reactor
RBCCW	reactor building closed cooling water
RBM	Rod Block Monitor
RCPB	reactor coolant pressure boundary
RETS	Radiological Effluent Technical Specifications
RHRS	residual heat removal system
RPS	reactor protection system
RPV	reactor pressure vessel
RTP	Reactor Thermal Power
RWCU	reactor water cleanup
RVWLIS	reactor vessel water level instrumentation system
SAR	safety analysis report
SBGTS	standby gas treatment system
SEP	systematic evaluation program
SER	safety evaluation report
SLC	standby liquid control
SNP	Siemens Nuclear Power (formerly ANF)
SPC	Siemens Power Corporation (formerly SNP)
SRP	Standard Review Plan
SRV	safety relief valve
SWS	service water system
TBCCW	turbine building closed cooling water
TEDE	Total Effective Dose Equivalent
TIP	traversing incore probe
TMI	Three Mile Island
TSC	technical support center
UHS	ultimate heat sink
USI	unresolved safety issue
WEC	Westinghouse Electric Company
ZIP	zinc injection process

1.2 GENERAL PLANT DESCRIPTION

1.2.1 Principal Design Criteria

The principal criteria for design and construction of Units 2 and 3 are summarized below. Specific design criteria and design features are detailed in later sections.

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

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

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

1.2.1.1 Reactor Core

A. The reactor core is designed as part of a BWR to produce steam for direct use in a turbine-generator.

B. The reactor core is designed and operated to prevent, or to detect and suppress, the occurrence of uncontrolled power oscillations during any mode of operation.

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

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

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

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

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

H. Thermal characteristics of the reactor core preclude fuel clad surface heat flux or fuel enthalpy and temperature which could cause sudden fuel cladding ruptures.

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

1.2.1.2 Reactor Core Cooling Systems

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

B. Heat removal systems are provided to remove decay heat generated in the reactor core if the normal operational heat removal systems are inoperative. The capacities of such systems are adequate to prevent fuel clad damage.

C. Redundant heat removal systems are provided to prevent any fuel clad melting as a result of various postulated but improbable loss-of-coolant accidents (LOCAs).

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

1.2.1.3 Containment

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

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

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

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

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

1.2.1.4 Control and Instrumentation

A. The station is provided with a control room having adequate shielding and air conditioning facilities to permit occupancy during and after all design basis accident situations.

B. Interlocks or other protective devices are provided in addition to procedural controls to prevent serious accidents.

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

1.2.1.5 Electrical Power

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

1.2.1.6 Radioactive Waste Disposal

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

B. Process and discharge streams are appropriately monitored and such features incorporated as may be necessary to maintain releases below the permissible limits of 10 CFR 20. Automatic off-gas monitors located downstream of the main condenser air ejector, at the inlet to the holdup line, are installed and are subject to manual override.

1.2.1.7 Shielding and Access Control

Radiation shielding and station access control are such that the personnel doses are less than the limits in 10 CFR 20.

1.2.1.8 Fuel Handling and Storage

Appropriate fuel handling and storage facilities are provided to preclude accidental criticality and to provide cooling for spent fuel.

1.2.2 Summary Design Description and Safety Analysis

1.2.2.1 Design Bases Dependent On Site and Environmental Characteristics

Information relating to the Dresden site and environment is summarized in Chapter 2 and was used in the design of Dresden Units 2 and 3.

1.2.2.1.1 Gaseous Waste Effluents

The off-gas systems for Units 2 and 3 are designed to use a 310-foot chimney for release of the treated, radioactive, gaseous effluents. The radioactivity release rate limits are as described in the Offsite Dose Calculation Manual (ODCM).

1.2.2.1.2 Liquid Waste Effluents

Units 2 and 3 use common intake and discharge canals, adjacent to the Unit 1 intake and discharge canals. Radioactive liquid releases are made on a batch basis (not continuously) and comply with 10 CFR 20.

1.2.2.1.3 Wind Loading Design

All structures are designed to withstand the maximum potential loadings resulting from a wind velocity of 110 mph. The design is in accordance with standard codes and normal engineering practice.

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

1.2.2.1.4 Geology

The geology of the area indicates that bedrock loading capability ranges from 2000 to 15,000 psi. These values are well above normal high-load footing design values. Consequently, no problems or restrictions beyond normal design practice are anticipated.

1.2.2.1.5 Seismic Design

The following design criteria apply only to seismic Class I items. Class I items are defined as structures (building and equipment) which are vital to the safe shutdown of the unit and the removal of decay heat.

The seismic design for Class I structures and equipment for Dresden Station are based on dynamic analyses using acceleration or velocity response spectrum curves which are based on a horizontal ground motion of 0.1 g; a vertical acceleration equal to two-thirds of the horizontal, or 0.067 g, was assumed to occur simultaneously.

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

Earthquake design is based on ordinary allowable stress as set forth in the applicable codes, but is more conservative because the usual one-third increase in allowable working stresses due to earthquake loadings is not used. As an additional requirement, the design is such that a safe shutdown can be made during a horizontal ground motion of 0.2 g with a simultaneous vertical acceleration of 0.133 g.

1.2.2.1.6 Conclusions with Respect to the Site and Environment

The Dresden site meets the reactor site criteria described in 10 CFR 100 for the following reasons:

A. EGC’s ownership of the large, 953-acre tract provides the requisite exclusion area for power reactors such as Units 2 and 3.

B. There are no residences on the site or within a radius of 0.5 miles of the units.

C. Units 2 and 3 are independent of each other to the extent that an accident in one would not initiate an accident in the other. The simultaneous operation of both units does not result in total radioactive effluent releases beyond allowable limits.

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

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

F. There are numerous access roads, including Interstate Routes 55 and 80, within the LPZ permitting rapid evacuation.

G. The population density and use characteristics of the site environment in the LPZ are compatible with the combined operation of both units.

H. As discussed in Chapter 2, the geological, hydrological, meteorological, and seismological characteristics of the site and environment are suitable for the location of Units 2 and 3.

1.2.2.2 Station Arrangements

The arrangement of buildings at Dresden Station is shown in Drawing M-1.

A single turbine building completely encloses turbine-generators for both units and their control room. The building is a reinforced concrete structure from its foundation at elevation 513'-6" to the main floor at elevation 561'-6". A steel-framed superstructure is used from the main floor to the roof. The roof is of precast concrete deck units overlaid with vapor barrier. 1-inch thick (minimum) loose-laid insulation covered with single ply elastomeric membrane fabric, and ballasted with paver blocks. The sidewalls are of insulated metal construction. The frame also supports a runway for the 175-ton and 125-ton traveling bridge cranes. The turbine building is connected to the reactor building by its main floor at elevation 561'-6" and its steel-framed roof at elevation 622'-6".

The Unit 2 turbine-generator, exciter, condenser, feedwater heaters, feedwater and condensate pumps, demineralizer system, condenser circulating system and electrical switch gear are located in the east half of the turbine building. Duplicate equipment and systems for Unit 3 are located in the west half of the building.

Access to the turbine building is through the access control building and connecting corridors of Unit 1. The turbine building has a common supply and exhaust ventilation system for Units 2 and 3.

The main generators are supported centrally at the top of the concrete portion of the building with the control room at one end of the building on the next lower level. The equipment arrangement and principal dimensions are shown in Drawings M-2 through M-9.

A common reactor building for Units 2 and 3 is constructed abutting the south wall of the turbine building. The east half of the reactor building houses the Unit 2 reactor vessel, recirculation system, primary containment, reactor auxiliary systems, refueling equipment and spent fuel storage, as well as the common fuel storage vault used for both Units 2 and 3. Except as noted, duplicate equipment for Unit 3 is located in the west half of the reactor building. The reactor building consists of monolithic, reinforced concrete floors and walls enclosing the reactor, primary containment, and reactor auxiliaries. The reactor building superstructures consists of sealed panel walls and a precast concrete roof.

Units 2 and 3 use the same radioactive waste building, centrally located adjacent to the north side of the turbine building. The function of the radwaste building is to house the tanks, equipment, and drums used to collect, treat, package, and store the solid and liquid radioactive wastes obtained from various parts of the plant. The building also provides a system of leakproof trenches and sumps which collects all leaks, spills, and overflows and returns them to the storage and treatment system. Equipment is arranged from the standpoint of ease of operation, inspection, and maintenance with minimum personnel exposure.

The radwaste building is a one-story structure with a basement. It is attached to the north side of the turbine building and is constructed of monolithic, reinforced concrete on a solid rock foundation. Floor and roof plans, exterior elevations, sections showing interior walls, and architectural details of the building are shown in Drawings M-2 through M-9. Major items of equipment are also shown on these drawings.

The common control room for Units 1, 2, and 3 is located at the juncture of the Unit 1 and the common turbine buildings. The original administration building located to the south of the Unit 1 turbine building is used for Units 2 and 3 also.

The new administration building is constructed to the south of the plant in close proximity to the main gatehouse. This building lies within the protected area.

The Unit 1 high pressure coolant injection (HPCI) building is constructed east of the Unit 1 sphere and was constructed to backfit Unit 1 with an additional emergency core cooling system (ECCS). This system was not completed, since Unit 1 was retired in August of 1984. A prototype chemical cleaning facility to decontaminate Unit 1 is located east of the Unit 1 HPCI building.

The crib house is a structure that contains the screens, trash racks, motors, and pumps for the condenser water and service water supply. The underground portions of the structure serve as channels for incoming water. The upper portions have the function of protecting the motors and controls from the elements. A common crib house serves Units 2 and 3 with separate channels, screens, and pumps for each. General arrangement of this building is shown on Drawing M-10.

The pump suctions are amply submerged below the lowest low-water surface-elevation of the water in the forebay, which is the elevation of the pool surface adjusted for the friction and velocity drops in the supply channels.

Drawings M-2A, M-10A, M-10B, M-10C Sheets 1 and 2, M-10D Sheets 1 through 3, and M-10E Sheet 1 provide further detail concerning the off-gas, radwaste, and make-up demineralizer facilities.

1.2.2.3 Reactor System

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

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

The reactor pressure vessel contains the reactor core and structure, steam separators and dryers, jet pumps, control rod guide tubes, feedwater spargers, core spray spargers, standby liquid control (SBLC) spargers, and other components as shown in Figure 3.9-4. The inside diameter of the vessel is approximately 21 feet and the inside height between heads is approximately 68 feet. The main connections to the reactor vessel include the steam lines, jet pump lines, feedwater lines, and control rod drive (CRD) thimbles. Other connections are provided for the isolation condenser (IC) system, SBLC system, ECCS, and instrumentation systems.

The fuel for the reactor core consists of uranium dioxide pellets contained in sealed Zircaloy-2 tubes. These fuel rods are assembled into square arrays in individual assemblies. The original assemblies were of a 7x7 configuration; later designs introduced in subsequent fuel cycles were of 8x8, 9x9, and 10x10-configurations. The fuel enrichment is varied from rod to rod within an assembly to achieve desired neutron flux characteristics. Some water rods or a water box may be included, and gadolinium is used in some rods as a burnable poison, in the form of Gd_2O_3-UO_2. Each fuel assembly is surrounded by a Zircaloy-2, Zircaloy-4 or ZIRLO flow channel. The cycle specific reload reports provide the material used for reload fuel. Water serves as both the moderator and coolant for the core.

The original equipment control rods consisted of sealed, stainless steel tubes measuring 3/16-inch in diameter filled with compacted boron carbide (B_4C) powder and held in a cruciform array by a stainless steel sheath with a 1/16-inch wall thickness fitted with castings at each end. The design (except for the additional length required for the longer fuel assemblies used in Dresden Units 2 and 3) is similar to that used in Unit 1 for more than 6 years. Recent control rod designs supplied by the original equipment vendor (GE) employ high-purity stainless steel to minimize stress corrosion cracking. In addition, hafnium may be substituted for B_4C in areas of the blade that experience high neutron flux to extend control rod lifetime. Both Dresden units also have control rods from Westinghouse ATOM AB (formerly ASEA-ATOM and ABB ATOM) which differ from the GE design in that the B_4C tubes are replaced by stainless steel wings bored with a series of horizontal holes. These holes may be filled with either B_4C or hafnium metal. All designs mentioned above are bottom-entry-type and are moved vertically within the core by individual, hydraulically operated, locking piston control rod drives.

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

The systems for reactivity control are of the same design as those used in the Oyster Creek and Nine Mile Point plants, including two features which provide improved plant safeguards.

First, the lower casting of each stainless steel control rod assembly is provided with a rod velocity limiter designed to limit the free-fall velocity of the control rod to less than 5 ft/s in the improbable event of a control rod drop accident. Current control rod designs limit the free-fall velocity to 3.11 ft/sec. Second, the CRD housings have been provided with a support structure designed to prevent significant movement of the CRD housing and drive mechanism in the unlikely event of a drive housing structural failure.

Temporary control curtains fabricated of boron stainless steel were fixed between fuel channels during early life of the initial core to supplement the reactivity control of the control rods. These curtains are not utilized in the current design of the plant.

Reactor coolant enters the bottom of the core and flows upward through the fuel assemblies where boiling produces steam. The steam-water mixture is separated by steam separators and dryers located within the reactor vessel. The steam passes through steam lines to the turbine-generator which produces electricity. The separated water mixes with the incoming feedwater and is returned to the core inlet through jet pumps located within the reactor vessel. The motive force for the jet pumps is supplied by the water from the two reactor recirculation loops. Each loop has a variable speed, centrifugal pump with mechanical seals, motor-operated gate valves (for isolation of pumps for maintenance), and instrumentation for recirculation flow measurement. (See Reference 1 for additional information about Unit 3).

The Dresden Unit 2 and 3 reactor designs are the first to use the jet pump feature for coolant recirculation. This feature also provides capability for reflooding the core in event of the postulated design basis LOCA.

An isolation condenser system provides reactor core cooling if the reactor becomes isolated from the main condenser because of closure of the main steam isolation valves. The isolation condenser operates by natural circulation. During operation of the isolation condenser system, steam flows from the reactor, condenses in the tubes of the isolation condenser, and flows back to the reactor by gravity.

1.2.2.4 Containment Systems

The primary containment consists of a drywell, pressure suppression chamber, and interconnecting vent pipes. It provides the first containment barrier for the reactor pressure vessel and recirculation system. Any leakage from the primary containment is to the secondary containment which consists of the reactor building, standby gas treatment system (SBGTS), and the 310-foot chimney. The integrated containment systems and their associated engineered safety features (ESFs) are designed so that offsite doses resulting from postulated accidents are well below the reference values stated in 10 CFR 100.

1.2.2.4.1 Primary Containment System

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

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

Two features are included in the primary containment design to aid in maintaining the integrity of the primary containment system indefinitely in the event of a LOCA. Two independent, full capacity containment cooling systems are included for the removal of heat within the drywell and the pressure suppression chamber. Capability is provided in the containment structure design to inert or control the composition of the containment atmosphere during operation.

Following construction of the drywell and suppression chamber, the penetrations were sealed with welded end caps and each vessel tested to 1.15 times the design pressure of 62 psig. Following the strength test each vessel was tested for leakage at design pressure and met the criteria of less than 0.5% leakage per day. The torus was half-filled with water during this test to simulate operating conditions.

After complete installation of all penetrations in the drywell and suppression chamber, these vessels were pressurized to the calculated maximum peak accident pressure of 48 psig and measurements taken to verify that the integrated leakage rate from the vessels did not exceed 0.5% per day of the combined volumes. An integrated leakage rate test is performed periodically on the primary containment at 48 psig.

Electrical penetrations are also provided with double seals and are separately testable at 48 psig. The test taps and the seals are located so that the tests can be conducted without entering or pressurizing the drywell or suppression chamber.

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

All containment closures which are fitted with resilient seals or gaskets, except for the personnel access lock, are separately testable at up to the full design pressure of 62 psig to verify leaktightness. The covers on flanged closures, such as the equipment access hatch cover, the drywell head, and access manholes are provided with double seals and with a test tap, which allows pressurizing the space between the seals without pressurizing the entire containment system. The space between the airlock doors is pressurized to 48 psig for local leak rate testing through the use of hold-down bars on the inner door. This door is not designed (opens inward) to withstand a high pressure differential in the inward direction.

1.2.2.4.2 Secondary Containment System

The primary safety functions of the secondary containment are to minimize ground level release of airborne radioactive materials and to provide for a controlled, filtered, elevated release of the building atmosphere under accident conditions. The reactor building provides secondary containment when the primary containment is in service, and primary containment during periods when the primary containment is open. For these reasons, the reactor building is designed as a controlled leakage structure. Units 2 and 3 are designed to use the same reactor building. The reactor building is constructed to provide a single operating floor without separation barriers above that level. Beneath the operating floor the reactor building is provided with a common wall separating Unit 3 operating and equipment areas from those of Unit 2. Access doors between the separate areas are provided to assure ventilation control.

Two redundant SBGTS trains are provided to filter the reactor building ventilation exhaust and discharge it to the 310-foot chimney during containment isolation conditions.

1.2.2.5 Shutdown Cooling System, Isolation Condenser, Standby Coolant Supplies, and ECCS

In addition to the turbine-generator and main condenser system, the following independent systems are provided for the purpose of cooling the reactor and primary containment system under various normal and abnormal conditions:

A. A shutdown cooling system is provided to remove reactor decay heat during shutdown.

B. An isolation condenser is provided for removal of decay heat from the core when the reactor is isolated from the main condenser.

C. A low pressure coolant injection (LPCI)/containment cooling system is provided. It serves three functions:

1. The LPCI system injects water into the reactor vessel subsequent to a postulated LOCA rapidly enough to reflood the core and prevent fuel clad melting;

2. The containment cooling system removes heat from the water in the suppression chamber; and

3. The containment cooling system sprays water into the drywell and torus as an augmented means of removing energy from the containment as required.

D. Two core spray trains are provided. Each train is designed to pump water from the pressure suppression chamber pool directly to the reactor core through separate spray headers or spargers mounted in the reactor vessel above the core.

E. HPCI system is provided for removal of decay heat and to provide coolant inventory control and heat dissipation from the core to the suppression chamber under postulated slow-depressurization accidents. If the HPCI system should fail to operate, automatic depressurization by blowdown is employed through automatic opening of relief valves which vent steam to the suppression pool. This blowdown depressurizes the vessel in sufficient time to allow the core spray or the LPCI function of the ECCS to adequately cool the core and prevent any clad melting.

F. A standby coolant supply system is provided by a crosstie between the service water system and the feedwater system, which makes available an inexhaustible supply of cooling water from the river to the reactor core and containment, independent of all other cooling water sources.

The core cooling provisions itemized above are designed to prevent fuel clad melting for the full range of primary system pipe size breaks which may be postulated to occur.

1.2.2.6 Unit Control and Instrumentation

1.2.2.6.1 Unit Control

Reactor power is controlled by movement of control rods and by regulation of the recirculation flowrate. Control rods are used to shape the core power distribution. The control rods have sufficient negative worth when inserted to make the reactor subcritical with the core at the most reactive time of the fuel cycle and the highest worth single control rod stuck full out. Load-following adjustments in reactor power level are accomplished with recirculation flow control. Procedural controls backed up by protective devices are used so that thermal performance does not exceed established limits.

Reactor pressure is automatically controlled by the pressure regulator by varying steam flow to the turbine to maintain constant pressure in the reactor. As a result, the turbine power output follows the reactor power output.

A bypass system having a capacity of approximately 33% steam flow at rated load is supplied with the turbine to restrict overpressure transients resulting from sudden turbine control valve or stop valve closure.

The bypass valves are operated on an overpressure signal from the initial pressure regulator. Rapid partial load rejection can be accommodated with the bypass system.

The reactor protection system overrides the above controls to initiate any required safety action. A standby liquid control system is provided to inject a borated solution into the reactor and shut it down in the remote event that the control rod system becomes inoperative.

1.2.2.6.2 Reactor Protection System

A reactor protection system is provided which automatically initiates appropriate action whenever the plant conditions monitored by the system approach pre-established limits. The reactor protection system acts to shut down the reactor.

The reactor protection system consists of two buses of relay contacts that are actuated by sensors from the parameters being monitored. The buses are energized during normal operation, and deenergization of both buses in the reactor scram circuit results in the opening of the scram valves in the control rod hydraulic system causing rapid insertion (scram) of the control rods. Each bus has at least two independent devices for each measured variable which initiates a scram, but only one device must operate to trip the bus in which it is connected. Both buses must be deenergized to produce a scram. The reactor protection system initiates a scram on loss of power to the protection system.

Components of the reactor protection system can be removed from service for testing and maintenance without interrupting plant operations and without negating the ability of the protection system to perform its protective functions upon receipt of appropriate signals.

1.2.2.7 Radiation Monitoring Systems

Instrumentation is provided for continuous monitoring of the radioactivity of certain processes. Processes significantly high in radioactivity are monitored for variation from normal. Certain nonradioactive processes are monitored to provide alarm in the event of contamination.

1.2.2.8 Fuel Handling and Storage

The refueling procedure is generally referred to as "wet" refueling since irradiated fuel is always kept under water. The facility's design allows visual control of operations at all times. This feature is instrumental in producing a safe, efficient refueling sequence.

The steam dryer and separator assemblies are transferred to a special storage pit. Water is added to the storage pit prior to transferring the assemblies to provide shielding from the parts of the separator which have been adjacent to the top of the core and which have been the most heavily irradiated.

Spent fuel discharged from the reactor is transferred under water into racks provided in the storage pool. The storage pool is designed to accommodate the channel stripping operation and the many other fuel maintenance operations that are required. Storage space is also provided in the pool for irradiated fuel assembly channels and control rods and for small internal components of the reactor.

New fuel is brought in through the equipment entrance of the reactor building and hoisted to the upper floor utilizing the reactor building crane. The new fuel for both Units 2 and 3 is stored in the new fuel vault located adjacent to the Unit 2 refueling pool area within the reactor building.

Refer to Section 9.1.2.2.4 for description of spent fuel storage and handling of Dry Cask Storage (DCS) systems and the Independent Spent Fuel Storage Installation (ISFSI).

1.2.2.9 Turbine System

The saturated steam leaving the reactor vessel flows through four carbon steel steam lines to the turbine located in the turbine building. After passing through the turbine, the low pressure steam is condensed, the noncondensible gases are removed, and the condensate is filtered and demineralized before being returned to the reactor through the feedwater heaters.

1.2.2.10 Electrical System

The electrical output of the units is fed into a 345-kV switchyard and from the yard to CECo's network grid system via nine 345-kV transmission lines and six 138-kV transmission lines. The 138-kV transmission lines receive power from the Dresden units through 345-kV to 138-kV transformers via a 138-kV switchyard. Auxiliary power is supplied from the respective units themselves, from the 345-kV switchyard. A diesel-generator (DG) system provides emergency power. An additional diesel-generator system is provided for power in the event of a station black-out.

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

1.2.2.11 Shielding, Access Control, and Radiation Protection Procedures

Control of radiation exposure of plant personnel and people external to the plant is accomplished by a combination of radiation shielding, control of access into certain areas, plant ventilation systems, and administrative procedures. The requirements of 10 CFR 20 are used for establishing the basic criteria and objectives.

EGC's policy is to maintain a radioactive exposure as low as is reasonably achievable (ALARA).

Shielding is used to reduce radiation dose rates in various parts of the plant to acceptable limits consistent with operational and maintenance requirements. Access control and administrative procedures are used to limit the integrated dose received by plant personnel to that set forth in 10 CFR 20. Access control and procedures are also used to limit the potential spread of contamination from various areas, particularly areas where maintenance occurs. Table 12.3-1 summarizes the design bases for shielding to assure that radiation levels in various areas of the plant are consistent with operational requirements.

The design bases summarized in Table 12.3-1 are at the shield wall. Generally, areas away from the shield wall have lower dose rates and this, plus occupancy factors, reduces the integrated dose received. Personnel involved in all phases of operation and maintenance normally receive far less than the permissible dose.

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

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

1.2.2.12 Radioactive Waste Control

A gaseous radioactive waste control system is provided which monitors and records the radiation level in the off-gas, recombines the radiolytically produced hydrogen and oxygen, removes moisture, provides a holdup time, and filters the noncondensible gases. The off-gas is then diluted by a large volume of ventilation air, and the radiation level in the effluent is monitored and recorded before release through the 310-foot chimney during normal and abnormal plant operation.

A liquid radioactive waste control system is provided for collection, treatment, temporary storage, and discharge of liquid wastes from both Units 2 and 3. Wastes are collected in pumps and drain tanks and transferred to the radwaste facility for further treatment, temporary storage, or discharge.

In the radwaste facility, liquid wastes to be discharged from the system are handled on a batch basis. The batches are either solidified and stored until they can be disposed of offsite, or they are released to the Illinois River after dilution in the discharge canal.

Solid radioactive wastes are processed, stored, packaged and shipped offsite.

1.2.2.13 Summary Evaluation of Safety

1.2.2.13.1 General

The general safeguard objectives of the design of Dresden Station are to protect the equipment and to limit radiation exposures to a small fraction of established limits, for any person on or off the station premises, either during normal operation or under accident conditions.

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

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

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

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

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

1.2.2.13.2 Summary of Offsite Doses

1.2.2.13.2.1 Normal Operations

The radioactive waste control systems for the combined normal operation of Dresden Units 2 and 3 are designed to limit the radiation exposure of the neighboring population to levels significantly below those doses set forth in 10 CFR 20.

1.2.2.13.2.2 Abnormal Operations

A variety of postulated equipment and component malfunctions, operator errors, and system accidents have been analyzed to evaluate the maximum extent of potential offsite dose consequences. The estimated maximum doses at the boundary of the exclusion area 2 hours after the postulated accidents which have the greatest potential for release of radioactive materials to the environment are contained in Chapter 15.

1.2.3 Summary of Technical Data

Design features and data appropriate to achieve a reactor thermal output of 2957 MW are summarized in Table 1.2-1. Since some of the parameters in Table 1.2-1 are cycle-specific (depending on fuel type and core configuration), typical values have been provided.

1.2.4 Interaction of Units 1, 2, and 3

The objective followed in designing Units 2 and 3 is that each unit would operate independently of the other. A malfunction of equipment or operator error in either of the two units would not affect the continued operation of the remaining unit. A high degree of station reliability is accrued from the standpoint of continuity of power for the operation of standby equipment through the operation of a multiunit generating station.

1.2.4.1 Gaseous Waste Effluents

The 310-foot chimney is used to discharge Units 2 and 3 off-gas, radwaste building ventilation air, turbine building ventilation air, and the effluent from the standby gas treatment system. A stack on top of the reactor building is used to discharge the Units 2 and 3 reactor building ventilation air, except when the standby gas treatment system is used. Fixed release limits are not imposed on the individual chimneys or stack; rather, the aggregate release from both chimneys and the reactor building vent stack is limited.

Continuous monitoring of steam jet air ejector (SJAE) off-gas is provided for both units. Each unit's monitoring system has indication, recording capability, and annunciation in the control room upon receipt of a high-radiation signal. A high-high radiation signal closes off-gas system chimney isolation valve.

The chimney and the reactor building vent stack have release level indication, recording, and high-level annunciation in the control room. The monitors provide high-range monitoring (i.e., are designed to measure the release level in a design basis accident) for particulates, iodine, and noble gases. In addition to conventional recorders, the computerized monitors also have a sliding 21-day memory and can be programmed to sum the releases from the various points or present the data in various ways.

1.2.4.2 Liquid Waste Effluents

Units 2 and 3 share an intake structure and canal for river water to their respective turbine condensers. However, Units 2 and 3 each have their own circulating water systems. Both the Unit 1 and Unit 2/3 intake canals obtain their water from the Kankakee River.

Units 2 and 3 use a single discharge canal separate from that of Unit 1; however, this canal is immediately adjacent to the Unit 1 canal at the point of discharge to the Illinois River.

Here, as in the case of the intake canals, operation of one unit has no effect on the other. Discharges are no longer made from the Unit 1 radioactive waste treatment facility. Discharges from the Unit 2/3 radwaste treatment facility are made infrequently, on a batch basis. Monitoring and sampling of plant discharges are addressed in UFSAR Chapter 11 and controlled in the Offsite Dose Calculation Manual.

1.2.4.3 Unit Auxiliary Power Supplies

The auxiliary power supply for Unit 2 is split between the UAT, which is connected to its generator leads, and the RAT, which is connected to the 345-kV bus at Dresden through a 345-138 kV auto transformer. Either transformer has sufficient capacity to carry the total auxiliary power requirements of Unit 2. The auxiliary power supply for Unit 3 is split between the UAT, which is connected to the generator leads, and the RAT, which is connected to the 345-kV bus at Dresden. Either transformer has sufficient capacity to carry the total auxiliary power requirements of Unit 3.

1.2.4.4 Common Auxiliary Systems

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

1.2.4.4.1 Fire Protection Systems

The Unit 1 and Unit 2/3 fire protection systems are interconnected. Through the use of crosstie valving, the protection afforded to each unit is increased.

1.2.4.4.2 Service and Instrument Air Systems

The service air systems of all three units are interconnected. Through the use of crosstie valving, redundancy of equipment is provided. Units 1, 2 and 3 each have one service air compressor. Unit 2 has two instrument air compressors, and Unit 3 has three instrument air compressors.

The crossties provide operating flexibility between the three units with regard to maintenance of the service air compressors on any of the units.

The only reactor system equipment operated by this air system are valves, which are fail-safe. Loss of instrument and service air causes safe plant shutdown.

1.2.4.4.3 Service Water System

The service water system is a Unit 2 and 3 combined facility that is provided with a common 50% capacity backup.

1.2.4.4.4 Reactor Building Closed Cooling Water System

The reactor building closed cooling water systems for Units 2 and 3 are intertied. The operating flexibility of both cooling systems is enhanced by the use of the interties.

1.2.4.4.5 Turbine Building

The Unit 2 and 3 turbines are housed in a single turbine building. The turbine building supply and exhaust ventilation systems are operated as a combined system.

1.2.4.4.6 Reactor Containment

Units 2 and 3 have separate primary containments and pressure suppression systems but share a common secondary containment (reactor building). Units 2 and 3 also share the same SBGTS and ventilation systems, each having sufficient capacity to accommodate the combined secondary containment volume.

1.2.4.4.7 Demineralized Water Makeup System

Makeup water is obtained from the 200,000-gallon well water tank, demineralized, and discharged either to two 200,000-gallon tanks on Unit 1 or to two 250,000-gallon tanks which serve both Units 2 and 3.

1.2.4.4.8 Control Rooms

The control rooms for Units 1, 2, and 3 are adjacent and open to each other. Since Unit 1 is no longer in service only those instruments and controls of common/interconnected systems required for monitoring, maintenance, or operation are in use. The equipment and panels are arranged and spaced so that each control room occupies a definite and separate area.

1.2.4.4.9 Radioactive Waste Systems

Units 2 and 3 share a common radioactive waste system which is designed to collect, process, control, and dispose of potentially radioactive waste in a safe manner without limiting unit or station operations or availability. The systems utilize tanks from Units 1, 2, and 3 that are crosstied.

1.2.4.4.10 Process Computer

Units 2 and 3 have separate process computers. This system is discussed in Section 7.5.

1.2.4.4.11 Miscellaneous Common Facilities

Several facilities common to Units 1, 2, and 3 or to Units 2 and 3, which are necessary, but not critical, to the safe startup, operation, and shutdown of the plant, are listed below:

A. Administration building;

B. Old administration building and access control building;

C. Machine shop;

D. Laundry;

E. Gatehouse and security fencing;

F. New fuel storage (Units 2 and 3);

G. Technical Support Center (TSC);

H. Operational Support Center (OSC);

I. Wastewater Treatment Facility;

J. Warehouses;

K. Storeroom;

L. Decontamination Building; and

M. New Storage Building for the Old Steam Dryers / Transportation Container Assemblies.

1.2.4.5 Inter-Plant Effects of Accidents

An accident in either of the units, up to and including the maximum postulated accident, will not prohibit control room access or prevent safe operation or shutdown of the other.

1.2.5 New Features

The design of Units 2 and 3 includes certain features which were developed by GE for use in the corresponding generation of nuclear power plants but are not found on previously constructed GE BWRs. These features are summarized below, with further detailed discussion presented in other sections of this report.

1.2.5.1 Features Which Reduce the Probability and Magnitude of Potential Reactivity Insertion Accidents

The design of Dresden Station includes features to limit the maximum control rod worth and to prevent rapid insertion of reactivity, thereby limiting the probability of occurrence and magnitude of postulated reactivity excursion accidents.

These features include the following:

A. Control rod worth minimizer,

B. Rod velocity limiter, and

C. Control rod drive housing support.

The control rod worth minimizer is a device which limits control rod withdrawal sequences and patterns to preselected programs. The original design target for the rod drop velocity limiter was to restrict the free-fall velocity of a control rod to a maximum of 5 ft/s. Current control rod designs limit the free-fall velocity to 3.11 ft/sec. The CRD housing support prevents the ejection of a control rod if the control rod drive housing were to fail.

1.2.5.2 Features Which Mitigate Effects of Postulated LOCAs

The reactor vessel internal components, the ECCS, and the main steam piping have been designed to assure continuity of cooling to the core and containment during and following postulated LOCAs.

The following components or design features are included in this category:

A. Flow restrictors in the main steam lines;

B. Jet pumps and arrangement of reactor vessel internal structure; and

C. Emergency core cooling system, including two core spray systems, a LPCI/containment cooling system, a HPCI, and an automatic depressurization system (ADS).

The flow restrictors are venturi nozzles which are installed in the main steam lines to limit the maximum steam flowrate in the line if the line were to break.

The jet pumps are an improved mechanism for providing reactor coolant flow. The jet pumps and attendant reactor vessel internal configuration allow reflooding of the core following a LOCA. The ECCS, which provides the cooling water necessary to cool and reflood the core following LOCA, includes the HPCI system, core spray system, LPCI system, and ADS system.

1.2.5.3 Features Which Improve Operability of the Units

The recirculation flow control system and the incore neutron monitoring system contribute to operational control.

The recirculation flow control system provides a method for adjusting the output of the units over a power range of approximately 30%. The incore monitors provide operational input data for core performance evaluation and for signals in the reactor protection system.

1.2.6 Drywell Post-Accident Recovery Provisions

Accidents which could occur within the drywell are normally thought to be LOCAs. A LOCA could be a break in any line connected to the primary system from a small line break to a line break equivalent to the double-ended break of a recirculation line. Each break of a given size can be postulated to occur in any one of the primary system pumps or valves or along any point in the varied primary system pipes. In addition to the numerous combinations of break sizes and locations, an almost unlimited set of various conditions within the primary containment, secondary containment, and site area can be postulated.

The primary concerns during an accident situation are identification of the accident, automatic and manual protection following the accident, available information related to the accident, and the ability to take corrective action based on the data. Analyses presented to date have emphasized the automatic protective features provided in the plant design. These automatic features limit and terminate the transient condition associated with the accident situation and enable the plant to be maintained in a safe condition thereafter. Those plant design features which could provide information following the accident and the ability to take action which is deemed proper based on this information are discussed below:

A. Reactor control system: The general status of the neutron flux is available. The control rod positions are indicated by position lights and by rod-notch position. The standby liquid control system is monitored for pump outlet flow and pressure, as well as solution volume and temperature.

B. Primary system parameters: The reactor vessel is monitored for pressure, water level, and temperature.

C. Primary containment: The primary containment is monitored for drywell pressure, drywell temperature, torus pressure, torus water level, and torus vacuum.

D. Emergency core cooling system flow: The status of the flow of the various subsystems is important information. Flow is monitored on each of the core spray systems. The LPCI containment cooling has its flow monitored as a total subsystem and also at the discharge into the recirculation header. The HPCI pump is monitored for flow as is each of the containment cooling service water loops.

E. Emergency core cooling system pump pressure: Each pump in the ECCS is equipped with a pressure indicator. Each pump is also equipped to indicate current flow to the pump motor and to indicate the status of the pump.

F. Emergency core cooling system valve positions: Each valve (except manually operated locked-open valves) in the flow path for each subsystem has its position indicated.

G. Automatic depressurization system: The valve position is monitored and indication is given upon initiation of either the 2-minute timer or the 8½-minute timer prior to blowdown.

H. Diesel-generator electrical power: The diesel-generator status is monitored by indication of the generator volts and amps, the emergency buses' (23-1, 24-1, 33-1, and 34-1) volts and amps, and the fuel oil supply.

I. Standby gas treatment system: The SBGTS flow is monitored, each major valve position is indicated, and various filters have differential pressure and temperature indicators.

In addition to the above listed items, there are many other items which assist in assessing the accident situation. These other items are annunciated, indicated, and/or recorded. Such items are as follows: feedwater flow and pressure; hotwell level; condensate storage tank level; recirculation pump flow and differential pressure; isolation valve positions; isolation condenser water level, radiation at vent, and valve positions; offsite power status essential breaker positions and battery charger voltage and current; area radiation monitors, ventilation monitors, and stack gas monitors.

In a LOCA situation, plant recovery may take several months. The reflooding capability of the jet pump will always assure the core to be two-thirds covered as long as one ECCS pump is available. The decision as how to proceed with such a recovery would be the subject of an extensive safety evaluation. Time would be available for such an evaluation since the core would be maintained in a safe condition with the ECCS subsystems which have redundant components. When time permits, the excess LPCI pumps would be shut down under a carefully controlled procedure. Also, it would be possible to rotate the redundant components in service from time to time, if believed desirable. A few days after the accident it would be possible to inspect LPCI equipment remotely and after a few weeks it would be possible to make close up visual inspections of this equipment.

After the ECCS have been initiated to mitigate the accident, the control room operator's primary function would be to monitor the information available in the control room to see that the core continues to be maintained in a safe condition. Some of the more important things that the operators must do during and after the accident are as follows:

A. Initiate the Emergency Plan (E-Plan);

B. Start up high-priority auxiliary systems as required, consistent with availability of power supply;

C. Secure the unnecessary safety equipment;

D. Ascertain that both primary and secondary containment have functioned properly; and

E. Start investigation to ascertain location and size of leak.

Most of the systems or components would continue to function or would automatically shut down. For example, the reactor feedwater pumps would trip on loss of suction pressure as soon as the condenser hotwell supply were exhausted.

Operating procedures call for the operator to make no changes to the primary coolant loops following an accident.

The recirculation pumps which are shutdown when the accident occurs would not be started up. The shutoff valves on the suction and discharge sides of the recirculation pumps are to be left open if they have not automatically closed. It should be possible, by systematic closing of the loop valves, to determine the approximate location of the break. Closing of these valves will be done only after thorough safety reviews. The closing of these valves will be supervised by one or more of the station senior operators. If it is found that the break is on the pump side of the discharge valve, it will be possible to close this valve in the loop affected and then flood the reactor vessel.

If a break has occurred which prevents complete refilling of the reactor vessel, it may be required to flood the drywell to accomplish eventual fuel removal. Provisions are provided to direct water into the drywell via the standby coolant supply system or via any of several alternative systems. Monitoring of the water level in the drywell is accomplished using the 0-100 foot drywell level indicator in the control room. Again, the flooding of the drywell and removal of the fuel would be accomplished only under direct supervisory control after careful consideration of the situation as it exists.

1.2.7 General Conclusions

Based on favorable site characteristics; on the design of Dresden Station Units 2 and 3 herein analyzed; on the criteria, principles, and design requirements of all major systems related to safety; on the calculated potential consequences of routine and accidental release of radioactive materials to the environs; on the scope of the testing programs which have been conducted; and on the technical competence of the applicant and its contractors, there is reasonable assurance that Dresden Station Units 2 and 3 can be operated at the site without endangering the health and safety of the public.

1.2.8 References

1. “Dresden Station Unit 3 Recirculation Replacement (RPR) Project Completion Report”.

Table 1.2-1

PRINCIPAL FEATURES OF PLANT DESIGN

Westinghouse SVEA-96 Optima2 methods and fuel are only applicable to Unit 2.

Site	
Location	Dresden Site, County of Grundy, State of Illinois
Size of Site	953 acres plus a 1275-acre cooling lake
Site and Plant Ownership	Exelon Generation Company
Plant	
U2 Net Electrical Output	957 MWe*
U2 Gross Electrical Output	1003 MWe
U3 Net Electrical Output	957 MWe*
U3 Gross Electrical Output	1003 MWe
Net Heat Rate	10, 640 BTU/kWh
Feedwater Temperature	355.6 °F

* Based upon assumed house loads of 46 MWe

Thermal and Hydraulic Design	
Design Thermal Output	2957 MWt
Reactor Pressure (dome)	1020 psia
Steam Flowrate	11.713E+06 lb/hr
Recirculation Flowrate	98 x 10^6 lb/hr
Fraction of Power Appearing as Heat Flux	0.971
Core Subcooling (typical)	24.1 BTU/lb
Core Average Void Fraction, Active Coolant	0.364
Core Average Exit Steam Moisture Content	12.0%

Conditions shown are for 100% power operation. The values given for feedwater temperature and steam flowrate serve as nominal reference values, but are not limits

Table 1.2-1 (continued)

PRINCIPAL FEATURES OF PLANT DESIGN

Westinghouse SVEA-96 Optima2 methods and fuel are only applicable to Unit 2.

Deleted

Table 1.2-1 (continued)

PRINCIPAL FEATURES OF PLANT DESIGN

Westinghouse SVEA-96 Optima2 methods and fuel are only applicable to Unit 2.

Approximate Coefficients	Cold	Hot (no voids)	Operating
Moderator Temperature Coefficient [(Δk/k)/°F]	-4×10^{-5}	-17.0×10^{-5}	
Moderator Void Coefficient [(Δk/k)/%Void]	Less than -0.6×10^{-3}	-1.0×10^{-3}	-1.4×10^{-3}
Fuel Temperature (Doppler) Coefficient [(Δk/k)/°F]	-1.2×10^{-5}	-1.2×10^{-5}	-1.2×10^{-5}
Typical Excursion Parameters			
Prompt Neutron Lifetime (l*)	38 μs		
Effective Delayed Neutron Fraction ()			
– at 0 MWd/t	0.0072		
– at 11,000 MWd/t	0.0061		

Core

Equivalent Core Diameter	182.2 in.
Circumscribed Core	189.7 in.
Diameter Core Lattice Pitch	12 in. (4 assemblies per unit cell)
Number of Fuel Assemblies	724

Table 1.2-1 (continued)

PRINCIPAL FEATURES OF PLANT DESIGN

Westinghouse SVEA-96 Optima2 methods and fuel are only applicable to Unit 2.

Fuel Assembly*(**)	Westinghouse SVEA-96 OPTIMA2	AREVA ATRIUM 10XM***
Fuel Rod Array	10x10	10x10
Fuel Rod Pitch (in.)	*	**
Approximate Weight of UO_{2} per Fuel Assembly (lbs)	*	**
Channel Material	Zircaloy-2 or ZIRLO	Zircaloy-4
Approximate Fuel Assembly Weight	*	**
Offset Advanced Channel and Channel Fastener (lbs)	*	**
Fuel Rods	96	91
Water Rods	*	1 square water channel

Fuel Rod, Cold*(**)	Westinghouse SVEA-96 OPTIMA2	AREVA ATRIUM 10XM
Fuel Pellet Diameter (in.)	*	**
Cladding Thickness (in.)	*	**
Cladding OD (in.)	*	**
Active Fuel Length (in.)	145.28	**
Length of Gas Plenum (in.)	*	**
Fuel Material	UO_2	UO_2
Cladding Material	Zircaloy-2	Zircaloy-2
Fill Gas	He	**
Fill Gas Pressure	*	**

* Values are Westinghouse proprietary and can be found in WCAP-15942-P-A for SVEA-96 Optima2 fuel.

** ATRIUM 10XM information for this table is available in ANP-3305P Rev. 4A Exelon Calculation and is subject to proprietary marking.

*** ATRIUM 10XM and Optima2 reside in both Units 2 and 3.

Table 1.2-1 (continued)

PRINCIPAL FEATURES OF PLANT DESIGN

Westinghouse SVEA-96 Optima2 methods and fuel are only applicable to Unit 2.

Moveable Control Rods	
Number	177
Shape	Cruciform
Pitch	12.0 in.
Stroke	144 in.
Width	9.8 in. (nominal)
Control Length	143 in. (nominal)
Control Material	B_4C granules and hafnium metal (a combination of these materials may be used depending on blade type)
Burnable Neutron Absorber	
Control Material	Gd_2O_3
Location	Mixed with UO_2 in several fuel rods per fuel assembly
Concentration	Location and reload dependent
Reactor Vessel	
Inside Diameter	20 ft. 11 in.
Overall Length Inside	68 ft, 7^5
Design Pressure	1250 psig
Coolant Recirculation Loops	
Location of Recirculation Loops	Containment drywell
Number of Recirculation Loops	2
Pipe Size	28 in.
Pump Capacity	45,000 gal/min each
Number of Jet Pumps	20
Location of Jet Pumps	Inside reactor vessel

Table 1.2-1 (continued)

PRINCIPAL FEATURES OF PLANT DESIGN

Westinghouse SVEA-96 Optima2 methods and fuel are only applicable to Unit 2.

Primary Containment	
Type	Pressure suppression
Design Pressure of Drywell Vessel	62 psig
Design Pressure of Suppression Chamber Vessel	62 psig
Design Leakage Rate	0.5% free volume per day at calculated maximum peak accident pressure
Secondary Containment	
Type	Reinforced concrete and steel superstructure with metal siding
Internal Design Pressure	0.25 psig
Inleakage Rate	100% free volume per day at 0.25 in.H_2O negative pressure
Structural Design	
Seismic Resistance	0.1 g horizontal plus 0.067 g vertical
Sustained Wind Loading	110 mph
Control Room Shielding	Dose not to exceed 500 mrem in 8 hours under design basis accident
Unit Electrical Systems	
Number of Incoming Power Sources	Six 345-kV lines Six 345-kV lines
Separate Power Sources Provided	Four auxiliary transformers Three standby diesel generators Two station battery systems

Table 1.2-1 (Continued)
PRINCIPLE FEATURES OF PLANT DESIGN

Westinghouse SVEA-96 Optima2 methods and fuel are only applicable to Unit 2.

Reactor Instrumentation System	
Location of Neutron Monitor System	Incore
Ranges of Nuclear Instrumentation	
Startup Range	Source to 0.01% rated power
Intermediate Range	0.0001% to 10% rated power
Power Range	1% to 125% rated power
Reactor Protection System	
Number of Channels in Reactor Protection System	2
Number of Channels Required to Scram or Effect Other Protective Functions	2
Number of Sensors per Monitored Variable in Each Channel	2
Method to Prevent Unwarranted Withdrawal of Control Rods	Automatic interlocks

Radioactive Waste Control Systems

Liquid, gaseous, and solid radioactive wastes are disposed of in accordance with the requirements of 10 CFR 20.

Other Engineered Safeguards - Summary of Systems and Functions	
ECCS	The multiplicity of subsystems provides core cooling continuity over the entire range of operating conditions and postulated loss-of-coolant accidents to prevent fuel damage.

1.3 COMPARISON TABLES

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

1.4 IDENTIFICATION OF AGENTS AND CONTRACTORS

As owner, CECo engaged, or approved the engagement of, the contractors identified below in the construction of both units. However, irrespective of the explanation of contractual arrangements offered below, CECo was the sole applicant for the construction permit and operating license for both units, and as owner and applicant, is responsible for the design, construction, and operation of them.

Dresden Units 2 and 3 were designed and built by GE as prime contractor for CECo. General Electric Company engaged the architect-engineering services of Sargent and Lundy, Incorporated (S&L), Chicago, Illinois, to provide the design of the nonnuclear portions of the units and to prepare specifications for the purchase and construction thereof. Commonwealth Edison Company reviewed the designs and construction and purchase specifications prepared by S&L and GE to assure that the general plant arrangements, equipment, and operating provisions were satisfactory to it. The units were constructed under the general direction of GE through a construction management organization at the site, United Engineers and Constructors, Inc., utilizing appropriate construction, erection, and equipment subcontracts.

Preoperational testing of equipment and systems and initial operation were performed by CECo personnel under the technical direction of GE. Personnel provided by CECo for operation were drawn from the experienced operating staff of Dresden Unit 1, trained and qualified in the startup of this boiling water reactor, and had several years of operational experience. Startup testing is described in Chapter 14.

The units were turned over to CECo after a demonstration of unit operational capability at a specified output. CECo then assumed responsibility for their subsequent operation.

ENDNOTES

1.4-1 FSAR Section 1.7.

1.4-2 FSAR Section 1.7; UFSAR Section 1.7.

1.5 REQUIREMENTS FOR FURTHER TECHNICAL INFORMATION

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

1.6 MATERIAL INCORPORATED BY REFERENCE

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

A. APED 5286 - Design Basis for Critical Heat Flux in Boiling Water Reactors (September 1966)

B. APED 5446 - Control Rod Velocity Limiter (March 1967)

C. APED 5449 - Control Rod Worth Minimizer (March 1967)

D. APED 5450 - Design Provisions for In-Service Inspection (April 1967)

E. APED 5453 - Vibration Analysis and Testing of Reactor Internals (April 1967)

F. APED 5555 - Impact Testing on Collet Assembly for Control Rod Drive Mechanism 7 RDB144A (November 1967)

G. TR67SL211 An Analysis of Turbine Missiles Resulting from Last Stage Wheel Failure (October 1967)

H. APED 5608 - General Electric Company Analytical and Experimental Program for Resolution of ACRS Safety Concerns (April 1968)

I. APED 5455 - The Mechanical Effects of Reactivity Transients (January 1968)

J. APED 5528 - Nuclear Excursion Technology (August 1967)

K. APED 5448 - Analysis Methods of Hypothetical Super-Prompt Critical Reactivity Transients in Large Power Reactors (April 1968)

L. APED 5458 - Effectiveness of Core Standby Cooling Systems for General Electric Boiling Water Reactors (March 1968)

M. APED 5640 - Xenon Considerations in Design of Large Boiling Water Reactors (June 1968)

N. APED 5454 - Metal Water Reactions - Effects on Core Cooling and Containment (March 1968)

O. APED 5460 - Design and Performance of General Electric Boiling Water Reactor Jet Pumps (September 1968)

1.7 DRAWINGS AND OTHER DETAILED INFORMATION

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

Applicable drawings, pictures, plot and building plans, sketches, electrical diagrams and piping diagrams are included at the end of the sections in which they are referenced or at the end of the related sections in the case of duplicate drawing references. An equipment symbol chart which provides an explanation of the symbols used on the station piping and instrumentation drawings (P&IDs) is shown on Drawing M-11, Sheet 2. A complete P&ID index is provided in Drawing M-11, Sheet 1.

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 EGC when other changes to that figure are needed.

REGULATORY GUIDE REFERENCE SECTIONS

1.8 CONFORMANCE TO NRC REGULATORY GUIDES

Dresden was designed and partially constructed before the issuance of the first Regulatory Guides in 1970. During this time frame the NRC issued Safety Guides for utility guidance. Therefore, Dresden was not designed specifically to conform to Regulatory Guides. Conformance to the provisions of Regulatory Guides is generally indicated under two general categories, full compliance or compliance with intent or objectives of the Regulatory Guide via an alternate approach. Full compliance indicates that the provisions of the Regulatory Guides are met by direct conformance or by the assessed capability of the design.

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

Table 1.8-1

REGULATORY GUIDE REFERENCE SECTIONS

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

Regulatory Guide	Title	UFSAR Section(s)
1.3	Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss-of-Coolant Accident for Boiling Water Reactors	15.6
1.7	Control of Combustible Gas Concentrations in Containment Following a Loss-of-Coolant Accident	6.2
1.8 (Safety Guide 8, March 1971)	Qualification and Training of Personnel for Nuclear Power Plants	T.S. 5.3.1 (1)
1.21	Measuring, Evaluating, and Reporting Radioactivity in Solid Wastes, Releases of Radioactive Materials in Liquid, and Gaseous Effluents from Light-Water-Cooled Nuclear Power Plants	11.2
1.23	Onsite Meteorological Programs	2.3
1.26	Quality Group Classifications and Standards for Water, Steam, and Radioactive Waste Containing Components of Nuclear Power Plants (for Comment)	5.2, 6.6
1.28, Rev. 3, August 1985	Quality Assurance Program Requirements — Design and Construction	(1)

Table 1.8-1 (Continued)

REGULATORY GUIDE REFERENCE SECTIONS

Regulatory Guide	Title	UFSAR Section(s)
1.30 (Safety Guide 30, August 1972)	Quality Assurance Program Requirements for the Installation, Inspection, and Testing of Instrumentation and Electrical Equipment	(1)
1.33 (Safety Guide 33, November 1972)	Quality Assurance Program Requirements — Operation	13.5
1.34	Control of Electroslag Weld Properties	5.2, 5.3
1.36	Nonmetallic Thermal Insulation for Austenitic Stainless Steel	6.1
1.37, March 1973	Quality Assurance Requirements for Cleaning Fluid Systems and Associated Components of Water-Cooled Nuclear Power Plants	(1)
1.38, March 1973	Quality Assurance Requirements for Packaging, Shipping, Receiving, Storage, and Handling of Items for Water-Cooled Nuclear Power Plants	(1)
1.39, March 1973	Housekeeping Requirements for Water-Cooled Nuclear Power Plants	(1)
1.44	Control of the Use of Sensitized Stainless Steel	5.3
1.45	Reactor Coolant Pressure Boundary Leakage Detection Systems	5.2
1.49	Power Levels of Nuclear Power Plants	T.S. 1.1/2.1 bases

Table 1.8-1 (Continued)

REGULATORY GUIDE REFERENCE SECTIONS

Regulatory Guide	Title	UFSAR Section(s)
1.50	Control of Preheat Temperature for Welding of Low-Allow Steel	5.3
1.52	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	6.5 6.4
1.54, June 1973	Quality Assurance Requirements for Protective Coatings Applied to Water-Cooled Nuclear Power Plants	(1)
1.61	Damping Values for Seismic Design of Nuclear Power Plants	3.9, 3.7
1.70, Rev 3, November 1978	Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants, LWR Edition	1.1, 5.3, 12.2
1.75	Physical Independence of Electric Systems	7.5
1.77, May 1974	Assumptions Used for Evaluating a Control Rod Ejection Accident for Pressurized Water Reactors	3.2, 4.3
1.78	Assumptions for Evaluating the Habitability of a Nuclear Power Plant Control Room During a Postulated Hazardous Chemical Release	6.4, 2.2
1.91	Evaluation of Explosions Postulated to Occur on Transportation Routes Near Nuclear Power Plants	2.2
1.97	Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident	7.1, 7.5, 9.1, 3.11

Table 1.8-1 (Continued)

REGULATORY GUIDE REFERENCE SECTIONS

Regulatory Guide	Title	UFSAR Section(s)
1.99, Rev. 2, May 1988	Effects of Residual Elements on Predicted Radiation Damage to Reactor Vessel Materials	5.2, 5.3
1.100	Seismic Qualification of Electric Equipment for Nuclear Power Plants	3.10
1.101, Rev. 2, October 1981	Emergency Planning and Preparedness for Nuclear Power Reactors	13.3
1.109	Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I	11.3 ODCM
1.111	Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine Releases from Light-Water-Cooled Reactors	11.3 ODCM
1.113	Estimating Aquatic Dispersion of Effluent from Accidental and Routine Reactor Releases for the Purpose of Implementing, Appendix I	ODCM
1.181	Content of the Updated Final Safety Analysis Report in Accordance with 10 CFR 50.71(e)	1.1
1.183	Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors	15.4.10, 15.6.4, 15.6.5, 15.7.3
1.190	Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence (Drafts were DG-1053 and DG-1025 (9/93))	5.3
4.8 Table 1, December 1975	Environmental Technical Specifications for Nuclear Power Plants	T.S. 5.5

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

1.9 UNIT 2 SYSTEMATIC EVALUATION PROGRAM

1.9.1 Summary

The Systematic Evaluation Program (SEP) was initiated by the NRC to review the designs of older operating nuclear reactor plants to reconfirm and document their safety. The review provided an assessment of the significance of differences between current technical positions on safety issues and those that existed when a particular plant was licensed, a basis for deciding on how these differences should be resolved in an integrated plant review, and a documented evaluation of plant safety.

The results of the initial review were published as NUREG-0823, entitled, "Integrated Plant Safety Assessment Systematic Evaluation Program for Dresden Nuclear Power Station, Unit 2." This report was issued in February of 1983, and Supplement 1 to NUREG-0823 was issued in October of 1989.

The review compared the as-built design with current review criteria in 137 different areas defined as "topics." The "definition" and other information for each of these topics appear in Appendix A of NUREG-0823. During the review, 49 of the topics were deleted from consideration by the SEP because a review was being made under other programs (Unresolved Safety Issue [USI] or Three Mile Island [TMI] Action Plan Tasks) or the topic was not applicable to the plant; that is, the topic was applicable to pressurized water reactors rather than to BWRs. The topics deleted because they were being reviewed under either the USI or TMI programs are listed in Appendix B of NUREG-0823, and the topics deleted because they did not apply to the plant are listed in Appendix C of NUREG-0823. The status of the USI or TMI tasks are addressed in a provisional operating license conversion safety evaluation report, NUREG-1403. That report was issued following completion of the SEP Integrated Plant Safety Assessment Report (IPSAR) and together with the IPSAR was considered during the conversion of the Dresden Unit 2 provisional operating license to a full-term operating license.

Of the original 137 topics, 88 were, therefore, reviewed for Dresden Unit 2; of those, 54 met current criteria or were acceptable on another defined basis. No modifications were made by CECo during topic review. References for correspondence pertaining to safety evaluation reports (SERs) for each of the 88 topics appear in Appendix E of NUREG-0823.

The review of the remaining 34 topics found that certain aspects of plant design differed from current criteria. The topics that differed from current licensing criteria consisted of 73 individual issues. These issues were considered in the integrated assessment of the plant, which consisted of evaluating the safety significance and other factors of the identified differences from current design criteria to arrive at decisions on whether backfitting was necessary from an overall plant safety viewpoint. To arrive at these decisions, engineering judgement was used as well as the results of a limited probabilistic risk assessment study. This study and staff comments are in Appendix D of NUREG-0823.

Table 4.1 of NUREG-0823 summarizes the staff's backfitting positions reached in the integrated assessment. In general, backfit requirements fell into one or more of the following categories:

A. Equipment modification or addition;

B. Procedure development or Technical Specification changes;

C. Refined engineering analysis or continuation of ongoing evaluation; and

D. No backfit modifications necessary.

Eight issues required primarily equipment modification or addition, 17 issues required primarily procedure development or changes, and 23 issues required primarily refined engineering analysis or continuation of an ongoing evaluation. Twenty-five issues did not require any backfitting.

Safety improvements are being planned as a result of the integrated assessment and are listed below. Some safety improvements have already been implemented by the licensee. The following descriptions summarize the backfit actions addressed by the integrated assessment. The NUREG-0823 sections relating to the issue are given in parentheses.

1.9.2 Safety Improvements Agreed To and To Be Implemented by the Licensee As a Result of SEP

The safety improvements identified by SEP fall into three categories. The first category comprises hardware modifications or additions that CECo agreed to make and that are required by the NRC. The second category comprises procedural or Technical Specification changes that become part of the operating license. The third category comprises additional engineering analysis followed by corrective measures where required. These three categories are listed below, and the issues are discussed in the NUREG-0823 sections given in parentheses.

1.9.2.1 Category 1, Equipment Modifications or Additions Required by NRC

A. Modify roof parapets to ensure ponded water is within roof load capacity (4.1.3);

B. Provide locking devices for manual isolation valves (4.18.3);

C. Provide second isolation valve on containment penetration branch lines (4.18.6);

D. Modify existing dc power system monitoring for breaker or fuse position and battery availability (4.23.3 and 4.28);

E. Install Class 1E protection at interface of reactor protection system and its power supply (4.24.3);

F. Modify diesel-generator annunciators (4.26.1); and

G. Provide for bypassing the diesel-generator underfrequency protective trip during accident conditions (4.26.2).

1.9.2.2 Category 2, Technical Specification Changes and Procedure Development

A. Modify existing flood emergency plan to provide ability to cope with design basis flood (4.1.2 and 4.1.4);

B. Modify the water control structures inspection program to ensure it is overseen by qualified personnel and that special inspections are conducted following extreme events (4.4.3);

C. Develop procedures for achieving cold shutdown from outside the control room (4.15 and 4.25.1);

D. Provide procedures for testing the shutdown cooling system temperature interlocks (4.17 and 4.25.4);

E. Provide mechanical locking devices and administrative procedures to ensure valve closure (4.18.1);

F. Modify procedures for post-accident engineered safety features leakage (4.18.2);

G. Provide procedures to ensure disconnect links between redundant electrical divisions are open (4.21.2);

H. Provide assurance that tie breakers are not used during power operations (4.21.3);

I. Limit allowable time for obtaining DG 2/3 control power from Unit 3 (4.21.4);

J. Prohibit paralleling of shared dc systems during power operations (4.23.1);

K. Prohibit placing DG 2/3 switch in "bypass" during normal operation (4.23.2);

L. Revise procedures to achieve cold shutdown using safety-grade systems (4.25.2); and

M. Modify plant Technical Specification limits for primary coolant and iodine activity (4.31 and 4.32).

1.9.2.3 Category 3, Additional Engineering Evaluation

A. Identify radiography requirements of vessels and pump casing (4.2.1);

B. Demonstrate fracture toughness for various components or that failure consequence is acceptable (4.2.2);

C. Ensure failure of ventilation stack does not affect safe shutdown (4.3.2);

D. Identify and ensure components outside qualified structures can withstand tornado loading or that their loss does not affect safe shutdown (4.3.3);

E. Demonstrate failure of roof decks does not affect plant safety (4.3.4);

F. Demonstrate structural capability of plant to withstand load combinations (4.3.5 and 4.10);

G. Ensure operability of DG 2 and DG 2/3 following loss of ventilation systems resulting from tornado missiles (4.5.3);

H. Ensure capability to achieve safe shutdown using tornado-missile-protected systems (4.5.4);

I. Provide schedule and basis for reinspection of low-pressure turbines (4.6);

J. Address effects of jet impingement on target pipe (4.7.1);

K. Demonstrate deformation of pipe associated with global strain does not affect functionability (4.7.2);

L. Ensure detectability for through-wall cracks in high-energy fluid systems piping (4.7.3);

M. Provide criteria and results of pipe whip load formulation and ensure pipe whip and jet impingement do not affect the containment liner (4.7.4);

N. Determine seismic capability of mechanical equipment (4.9.2);

O. Provide analysis of structural integrity of cable trays (4.9.3);

P. Ensure adequate setpoints for thermal overload protection of motor-operated valves or bypass thermal overloads (4.12.1);

Q. Provide leakage detection capability in conjunction with pipe breaks inside containment (4.13.1);

R. Provide seismically qualified leakage detection system (4.13.2);

S. Ensure adequacy of protective relaying (4.21.1);

T. Demonstrate adequate isolation of Class 1E sources from non-Class 1E loads (4.21.5);

U. Ensure common-mode electrical faults do not disable the neutron flux monitoring systems (4.24.1); and

V. Ensure the reactor protection system is protected from faults generated in process computer (4.24.2).