NUREG-1002

Safety Evaluation Report
related to the operation of Braidwood Station, Units 1 and 2

Docket Nos. 50-456 and 50-457

Commonwealth Edison Company

U.S. Nuclear Regulatory Commission

Office of Nuclear Reactor Regulation

November 1983

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ABSTRACT

The Safety Evaluation Report for the application filed by the Commonwealth Edison Company, as applicant and owner, for a license to operate Braidwood Station, Units 1 and 2 (Docket Nos. STN 50-456 and STN 50-457), has been prepared by the Office of Nuclear Reactor Regulation of the U.S. Nuclear Regulatory Commission. The facility is located in Reed Township, Will County, Illinois. Subject to favorable resolution of the items discussed in this report, the staff concludes that the facility can be operated by the applicant without endangering the health and safety of the public.

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1 INTRODUCTION AND GENERAL DESCRIPTION OF FACILITY

1.1 Introduction

The Commonwealth Edison Company (hereinafter referred to as the applicant) filed an application dated September 20, 1973 for licenses to construct and operate the proposed Byron Station, Units 1 and 2, and Braidwood Station, Units 1 and 2. Byron Station is located in north central Illinois, 2.5 mi east of the Rock River, 3 mi south-southwest of the town of Byron, and 17 mi southwest of Rockford, Illinois. The station is within Rockvale Township, Ogle County, Illinois. Braidwood Station is located in northeastern Illinois, 3 mi southwest of the Kankakee River, 20 mi south-southwest of the town of Joliet, and 60 mi southwest of Chicago, Illinois. The station is within Reed Township, Will County, Illinois.

The application for the Byron and Braidwood Stations was submitted and accepted for review under the Commission's standardization policy statement of March 5, 1973. The duplicate plant option was used as described in Appendix N to the Commission's regulations in Part 50 of Title 10 of the Code of Federal Regulations (10 CFR 50), "Domestic Licensing of Production and Utilization Facilities." This policy option allows for a simultaneous review of the duplicate structures, systems, and components important to the radiological health and safety and the common defense and security of a limited number of duplicate plants that are to be constructed within a limited time span at multiple sites. The application for the Byron and Braidwood Stations was for two duplicate units at each of two sites; therefore, the staff's review of the duplicate design portions of these four units was conducted simultaneously.

The Atomic Energy Commission (now the Nuclear Regulatory Commission (NRC)), reported the results of its review of these four units prior to construction in a Safety Evaluation Report (SER) dated April 4, 1975, in Supplement 1 to the SER dated August 1975 and Supplement 2 dated October 1975 (NUREG-75/023). Following a public hearing before an Atomic Safety and Licensing Board in Rockford, Illinois, and Bethesda, Maryland, held between September 4, 1974, and November 18, 1975, Construction Permit Nos. CPPR-130 and CPPR-131 were issued for Byron Units 1 and 2 on December 31, 1975. Construction Permit Nos. CPPR-132 and CPPR-133 were issued simultaneously for Braidwood Units 1 and 2.

On June 27, 1978, the Commonwealth Edison Company submitted an application requesting the issuance of operating licenses (OLs) for Byron Units 1 and 2 and Braidwood Units 1 and 2. Upon completion of the NRC acceptance review, the Final Safety Analysis Report (FSAR) was docketed on November 30, 1978, for both stations.

FSAR sections that describe features specific to one site or the other, are repeated and are presented on differently colored pages (pale blue for the Byron Station and ivory for the Braidwood Station). The staff reported the results of its review of the Byron facility in the Byron SER (NUREG-0876) dated February 1982. Since significant portions of the review in the Byron SER (including Supplements 1, 2, and 3) apply equally to all four units, considerable reference to the Byron report is made in this SER. Further discussion is provided in Section 1.11, Statement on Standardization.

Before issuing an OL for a nuclear power plant, the NRC staff is required to conduct a review of the effects of the plant on public health and safety. The staff review of Braidwood Station, Units 1 and 2, has been based on the FSAR that accompanied the OL application and 43 Amendments thereto. These documents are available to the public for review at the NRC Public Document Room located at 1717 H Street, N.W., Washington, D.C., at the Rockford Public Library, Rockford, Illinois, and at the Wilmington Township Public Library, Wilmington, Illinois.

During the course of its review, the staff held a number of meetings with representatives of the applicant to discuss the design, construction, and proposed operation of Braidwood. As a consequence, additional information was requested that the applicant provided in amendments to the FSAR. A chronology of the principal actions related to the processing of the application is included as Appendix A to this SER.

Review and evaluation of compliance by the applicant to the licensing requirements established in NUREG-0660, "NRC Action Plan Developed as a Result of the TMI-2 Accident," and NUREG-0737, "Clarification of TMI Action Plan Requirements," have been incorporated into reviews summarized throughout this report. Table 1.1 provides a cross reference of the applicable TMI Action Plan items and the SER section where the item is discussed.

This SER summarizes the results of the staff's radiological safety review of Braidwood and delineates the scope of the technical details considered in evaluating the radiological safety aspects of its proposed operation. The design of the station was reviewed against the Federal regulations, Construction Permit (CP) criteria, and the NRC Standard Review Plan (SRP), NUREG-0800, dated July 1981. The SRP is written to cover a variety of site conditions and plant designs. Each section is written to provide the complete procedure and all acceptance criteria for all of the areas of review pertinent to the section. However, for any given application, the staff reviewers may select and emphasize particular aspects of each SRP section as is appropriate for the application. In some cases, the major portion of the review of a plant feature may be done on a generic basis with the designer of that feature rather than in the context of reviews of particular applications from utilities. In other cases a plant feature may be sufficiently similar to that of a previous plant so that a de novo review of the feature is not needed. For these and other similar reasons, the staff may not carry out in detail all of the review steps listed in each SRP section in the review of every application.

Sections 2 through 18 of the SER address the staff's review and evaluation of radiological safety issues that have been considered during the review of the OL application. Section 19 is reserved for the report of the Advisory Committee on Reactor Safeguards. Section 20 considers whether the operation of the facility will be inimical to the common defense and security. Section 21 presents the staff review and evaluation of the financial qualifications of the applicant. Section 22 describes the financial protection and indemnity requirements for preoperational storage of nuclear fuel and operation of the facility. The conclusions of this report are given in Section 23.

Table 1.1 Cross-reference table for TMI Action Plan Items (NUREG-0737 and SER)

NUREG-0737 Item	Shortened Title	SER Section
I.A.1.1	Shift technical advisor	13.1.2.2, 13.2.2.2
I.A.1.2	Shift supervisor responsibilities	13.5.1.2
I.A.1.3	Shift manning	13.1.2.2
I.A.2.1	Immediate upgrade of RO and SRO training and qualifications	13.2.1.1
I.A.2.2	Training and qualification of operating personnel	13.2.2.2
I.A.2.3	Administration of training programs	13.2.1.1
I.A.3.1	Revise scope and criteria for licensing exams	13.2.1.1
I.B.1.2	Evaluation of organization and management	13.4.6
I.C.1	Short-term accident and procedure review	13.5.2.3
I.C.2	Shift and relief turnover procedures	13.5.1.2
I.C.3	Shift supervisor responsibility	13.5.1.2
I.C.4	Control room access	13.5.1.2
I.C.5	Feedback of operating experience	13.5.1.2
I.C.6	Verify correct performance of operating activities	13.5.1.2
I.C.7	NSSS vendor review of procedures	13.5.2.1
I.C.8	Pilot monitoring of selected emergency procedures for NTOLs	13.5.2.3
I.D.1	Control room design reviews	18.0
I.D.2	Plant safety parameter display console	13.3
I.G.1	Training during low-power testing	14
II.B.1	Reactor coolant system vents	5.4.5
II.B.2	Plant shielding	12.3.2

Table 1.1 (Continued)

NUREG-0737 Item	Shortened Title	SER Section
II.B.3	Postaccident sampling	9.3.2
II.B.4	Training for mitigating core damage	13.2.1.2
II.D.1	Relief and safety valve test requirements	5.2.2, 3.9.3.3
II.D.3	Valve position indication	7.5.2.3
II.E.1.1	Auxiliary feedwater system evaluation	10.4.9
II.E.1.2	Auxiliary feedwater system initiation and flow	7.3.2.8
II.E.3.1	Emergency power for pressurizer heaters	8.4.6
II.E.4.1	Dedicated hydrogen penetrations	6.2.5
II.E.4.2	Containment isolation dependability	6.2.4
II.F.1.1	Noble gas monitor	11.5.2
II.F.1.2	Iodine/particulate sampling	11.5.2
II.F.1.3	Containment high-range monitor	12.3.4.1
II.F.1.4	Containment pressure	6.2.1.1, 7.5.2.4
II.F.1.5	Containment water level	6.2.1, 7.5.2.4
II.F.1.6	Containment hydrogen	6.2.5, 7.5.2.4
II.F.2	Instrumentation for detection of inadequate core cooling	4.4.7
II.G.1	Emergency power for pressurizer equipment	8.4.7
II.K.2.13	Thermal mechanical report	15.5
II.K.2.17	Voiding in RCS	15.5
II.K.2.19	Benchmark analysis sequential AWF flow	15.5
II.K.3.1	Auto PORV isolation	7.6.2.7, 15.5
II.K.3.5	Auto trip of RCPs	15.5
II.K.3.9	PID controller	7.7.2.5

Table 1.1 (Continued)

NUREG-0737 Item	Shortened Title	SER Section
II.K.3.10	Applicant's proposed anticipatory trip at high power	7.2.2.8, 15.5
II.K.3.12	Confirm anticipatory trip	7.2.2.9
II.K.3.25	Power on pump seals	15.5
II.K.3.30	SBLOCA methods	15.5
II.K.3.31	Plant-specific analysis	15.5
III.A.1.1	Emergency preparedness, short term	13.3
III.A.1.2	Upgrade emergency support facilities	13.3
III.A.2	Emergency preparedness	13.3
III.D.1.1	Primary coolant outside containment	9.3.5
III.D.3.3	Inplant I_2 radiation monitoring	12.5.2
III.D.3.4	Control-room habitability	6.4

Appendix A is a chronology of the principal staff actions related to the review of the application. Appendix B is a bibliography of references used during the course of the staff review, including a section listing industry codes and standards. Appendix C is a discussion of how the "Unresolved Safety Issues" relate to the Braidwood application. An emergency preparedness evaluation report will be included as Appendix D in a future supplement to the SER pending submittal of an onsite and offsite emergency plan by the applicant. The Lawrence Livermore National Laboratory has been requested by the staff to perform a probabilistic hazard analysis on the seismological aspects of Braidwood Station. This report will be included as Appendix E in a future supplement to the SER. Appendix F lists the staff and consultants who have contributed to this report. Appendix G is a copy of the Final Duplicate Design Approval (FDDA) for the Byron Station design.

As a part of the staff's review of the Braidwood Station for compliance with the Commission's regulations, the staff has requested the applicant to verify that the Braidwood Station meets the pertinent regulatory requirements in 10 CFR 20, 50, and 100. The applicant has agreed to submit a response to this request.

In accordance with the provisions of the National Environmental Policy Act of 1969, a Final Environmental Statement (FES), which sets forth the considerations related to the proposed construction and operation of Braidwood was prepared by the staff and was issued prior to the issuance of the CP in July 1974.

The Draft Environmental Statement for an operating license is scheduled to be issued in January 1984 and the OL-FES is scheduled to be issued in June 1984.

In addition to the staff review, the Advisory Committee on Reactor Safeguards will review the application and will meet with both the applicant and the staff to discuss the final design and proposed operation of the plant. The Committee's report to the Commission will be included in a supplement to this SER.

Copies of this SER and other documents related to this review are available for inspection at the NRC Public Document Room, 1717 H Street, N.W., Washington, D.C., and at the Wilmington Township Public Library, 201 South Kankakee Street, Wilmington, Illinois 60481. Single copies of the report may be purchased from the sources indicated on the inside front cover.

The review and evaluation of Braidwood Station for OLs is only one stage in the continuing review by the staff of the design, construction, and operating features of the facilities. The proposed design of the facilities was reviewed as part of the CP review. Construction of the facilities has been monitored in accordance with the inspection program of the staff. During the OL review stage, the staff has reviewed the final design to determine that Commission safety requirements have been met. If OLs are granted, Braidwood Units 1 and 2 must be operated in accordance with the terms of the OLs and Commission regulations, and will be subject to the staff's continuing inspection program.

The NRC Project Manager assigned to the OL application for Braidwood Units 1 and 2 is Ms. Janice A. Stevens. Ms. Stevens may be contacted by calling 301-492-7144 or writing:

Janice A. Stevens
Division of Licensing
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

1.2 General Description of Facility

1.2.1 Duplicate Plant Portion

Units 1 and 2 of both the Byron and Braidwood (B/Br) Stations utilize two essentially identical generating units and two pressurized water reactor (PWR) nuclear steam supply systems (NSSS) and turbine generators supplied by Westinghouse Electric Corporation.

Westinghouse, Sargent & Lundy, and Commonwealth Edison Company jointly participate in the design and construction of each unit. The plants are operated by the Commonwealth Edison Company. Sargent & Lundy is the architect-engineer for both units.

Each NSSS incorporates a pressurized water reactor and a four-loop reactor coolant system (RCS). Each loop contains a reactor coolant pump (RCP) and a steam generator. The NSSS also contains one electrically heated pressurizer and certain auxiliary systems. Each NSSS is designed for a power output of 3,425 MWt, with a gross electrical output of 1,175 MWe and net electric output of 1,120 MW.

The reactor in each unit includes a low alloy steel reactor vessel with interior stainless steel cladding. All of the pressure-containing surfaces that come in contact with the reactor coolant water, including the reactor coolant piping, are stainless steel, which resists the corrosive action of the water.

The reactor vessel contains the core, core support structures, control rods, and other parts directly associated with the core. The core is composed of fuel rods made of slightly enriched uranium dioxide pellets contained in Zircaloy tubes with welded end plugs. The fuel rods are grouped and supported into fuel assemblies. The fuel assemblies are loaded into three regions within the core, each utilizing fuel of a different enrichment of U^{235}, with new fuel being introduced into the core periphery and the remaining fuel arranged in the central two-thirds of the core in order to achieve optimum power distribution. During subsequent refuelings, one-half of the fuel from the inner region is removed and transferred to fuel storage and the new fuel is added.

The reactor is controlled during operation by control rod movement and regulation of boric acid concentration in the reactor coolant. Mechanical rod cluster control assemblies consist of stainless-steel-clad hafnium absorber rods that are inserted in Zircaloy guide thimbles located within the fuel assemblies. The rod cluster control assemblies are attached to stainless steel drive shafts which will be raised and lowered within the core by individual drive mechanisms mounted on the reactor vessel head.

Water will serve as both the moderator and the coolant and will be circulated through the reactor vessel and core by four vertical, single-stage, motor-driven, centrifugal pumps of the shaft-seal type. One RCP is located in the cold leg of each loop. The reactor coolant will be heated by the core and circulated through four steam generators where heat will be transferred to the secondary system to produce saturated steam. The coolant is then pumped back to the reactor to repeat the cycle.

An electrically heated pressurizer connected to the hot-leg piping of one of the loops will maintain RCS pressure during normal operation as well as during plant load transients. The pressurizer provides a surge chamber and a water reserve to accommodate changes in reactor coolant volume during operation.

The steam generators are vertical U-tube units with Inconel tubes. The steam produced in the steam generators will be utilized to drive a tandem-compound six-flow exhaust turbine generator and will be condensed in a single-pass, multizone, deaerating-type condenser with four inlet and outlet water boxes. Cooling water supplied by a closed-cycle cooling system, with natural draft cooling towers at Byron and a man-made cooling pond at Braidwood, will be pumped through the tubes of the condenser to remove heat from, and thus condense, the steam after it has passed through the turbine. Makeup water for the circulating water system is drawn from the Rock River at Byron and the Kankakee River at Braidwood.

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

A reactor protection system is provided that automatically initiates appropriate action whenever a condition monitored by the system approaches preestablished limits. This reactor protection system will act to shut down the reactor, close isolation valves, and initiate operation of the engineered safety features should any or all of these actions be required.

Supervision and control of both the NSSS and the steam and power conversion system for each unit will be accomplished from separate facilities within a shared control room in the auxiliary building.

The emergency core cooling systems (ECCS) for each unit are designed to cool the reactor core and provide shutdown capability by injecting borated water during certain accident conditions. The ECCS provide long-term postaccident cooling of the core by drawing borated water from the containment sump. The ECCS for each unit consists of the centrifugal charging, safety injection and residual heat removal (RHR) pumps, accumulators, RHR heat exchangers, and the refueling water storage tank, along with the associated piping, valves, instrumentation, and other related equipment. The active components of the ECCS are powered from separate buses that are energized from offsite power sources. In addition, redundant sources of auxiliary onsite power are available through the use of the emergency diesel generators in the event of a loss of offsite power.

The NSSS for each unit is housed in a separate and independent containment structure. Each containment structure is of prestressed, post-tensioned concrete construction with a carbon steel liner. The structure ensures that leakage of radioactive material to the environment does not exceed an acceptable limit in accordance with 10 CFR 100 even if a loss-of-coolant accident (LOCA) were to occur. Sufficient free volume is provided to contain the energy released in the event of a major accident without the need for "pressure suppression" devices. The Sargent & Lundy Company is responsible for containment design.

The containment heat removal systems for each unit consist of the reactor containment fan cooler system and the containment spray system. The containment fan cooler system provides heat removal by ventilation; the containment spray system uses sodium hydroxide and boric acid to remove iodine and other radionuclides from the containment atmosphere during accident conditions.

Other important structures at the B/Br facilities include an auxiliary building shared by Units 1 and 2, a common turbine building, a common fuel-handling building, a common solid radwaste storage building, and an administration and service building. The auxiliary building is located between the containment structure and the turbine building. The auxiliary building contains the control room, electrical equipment room, switchgear room, and battery and computer rooms. It will also house the diesel generators, radwaste drumming facilities, HVAC laboratories, and filter rooms. The fuel-handling building is located between the containment structure and the turbine building and contains the spent fuel pool and new fuel storage vault.

The station is supplied with electrical power by independent transmission lines from offsite power sources. It also has independent and redundant onsite emergency power supplies capable of supplying power to shut down the plant safely or to operate the engineered safety features in the event of an accident.

1.2.2 Portions of the Plant Outside the Scope of the Duplicate Plant Design

The portions of the plant outside the scope of the duplicate plant design include offsite power systems and water systems (service water, demineralized water makeup, potable and sanitary water, and ultimate heat sink). These water systems differ from the Byron design only in the source of water. The pumphouse ventilation system and the pumphouse diesel generator fuel oil system are not applicable to Braidwood because of the different sources of cooling water at the two sites.

1.3 Comparison With Similar Facilities

The principal design features of the B/Br units are similar to those evaluated and approved previously by the staff for other nuclear power plants now under construction or in operation. In particular, the design of the facilities is conceptually similar to Commonwealth Edison Company's Zion Station. The NSSS are similar, except that the B/Br units have slightly higher power ratings. The reactor containments are of the same materials and size. Additionally, within the NSSS significant similarities exist between the B/Br units and Indian Point 2, Comanche Peak, South Texas, W.B. McGuire, Trojan, SNUPPS, Callaway, Wolf Creek and Watts Bar.

The reactor internals, including the lower core support structure, the upper core support structure, and the incore instrumentation support structure are similar to Indian Point 2 with the exception that B/Br employs a 17 x 17 fuel array and has neutron shield pads similar to Trojan. The RCS components including the reactor vessel, RCPs, steam generators, piping, RHR system, pressurizer, and valves are similar to Comanche Peak, with the exception of improved tube support material in the B/Br steam generators and the addition of loop-stop valves on each loop hot leg and cold leg in the B/Br RCS. The ECCSs in the B/Br units are similar to the Trojan system. The B/Br instrumentation and control systems functions are similar to the SNUPPS design. These systems include the reactor trip system, engineered safety features systems, systems required for safe shutdown, and safety-related display instrumentation. B/Br auxiliary systems include a control rod drive mechanism one-lift ventilation and cooling system similar to South Texas although the B/Br internals are not lifted with the reactor vessel head package. Additionally, the gas-waste management systems are similar for the B/Br units and Watts Bar, although Watts Bar has double the number of gas decay tanks. Source terms for the B/Br units have been determined to be similar to the W.B. McGuire Station, although some differences exist because of new ANSI N237 standards.

To the extent feasible and appropriate the staff has used its earlier reviews of these plants for those features of the B/Br units that were shown to be substantially the same as features previously considered. Where this has been done, the appropriate sections of this SER identify the other facilities involved. The SERs for these other facilities have been published and are available for public inspection at the NRC Public Document Room located at 1717 H Street, N.W., Washington, D.C.

1.4 Identification of Agents and Contractors

The applicant is responsible for the design, construction, and operation of Braidwood Station, Units 1 and 2. The applicant has retained Sargent & Lundy as architect/engineer and design consultant for the project. The NSSS, initial core, and turbine generators are supplied by Westinghouse Electric Corporation. Construction coordination of all activities at the site is under the supervision of Commonwealth Edison's Station Construction Department. Westinghouse also provides some piping design and technical assistance during NSSS erection, core loading, startup, and preoperational testing.

The applicant has also utilized consultants, as required, in specialized areas. These include ETA, Inc. for design of the physical security system; Dames & Moore for conducting environmental studies; HARZA Engineering for water treatment facility design; Murray & Trettel and Meteorology Research, Inc. for onsite meteorological measurement programs; Shirmer Engineering Corporation for evaluation of the fire protection system; Eberline Instrument Corporation to perform preoperational and environmental radiological baseline studies on and around the site; Westinghouse and the Illinois Department of Natural History for terrestrial and aquatic monitoring programs for the site vicinity; Illinois State Museum Society for archaeological investigations and recommendations; and Aero-Metric Engineering, Inc. to provide annual infrared photographs.

1.5 Summary of Principal Review Matters

The staff technical review and evaluation of the information submitted by the applicant considered, or will consider, the principal matters summarized below.

(1) The population density and land-use characteristics of the site environs and the physical characteristics of the site (including seismology, meteorology, geology, and hydrology) to establish (a) that these characteristics have been determined adequately and have been given appropriate consideration in the plant design, and (b) that the site characteristics are in accordance with the Commission siting criteria in 10 CFR 100, taking into consideration the design of the facilities, including the engineered safety features provided.

(2) The design, fabrication, construction, and testing criteria, and the expected performance characteristics of the plant structures, systems, and components important to safety to determine (a) that they are in accord with Commission General Design Criteria, Quality Assurance Criteria, Regulatory Guides, and other appropriate rules, codes, and standards, and (b) that any departures from these criteria, codes, and standards have been identified and justified.

(3) The expected response of the facility to various anticipated operating transients and to a broad spectrum of postulated accidents. Based on this evaluation, the staff determined that the potential consequences of a few highly unlikely postulated accidents (design-basis accidents) would exceed those of all other accidents considered. The staff performed conservative analyses of these design-basis accidents to determine that the calculated potential offsite radiation doses that might result--in the very unlikely event of their occurrence--would not exceed the Commission guidelines for site acceptability given in 10 CFR 100.

(4) The applicant's engineering and construction organization, plans for the conduct of plant operations (including the organizational structure and the general qualifications of operating and technical support personnel),

the plans for industrial security, and the plans for emergency actions to be taken in the unlikely event of an accident that might affect the general public to determine that the applicant is technically qualified to operate the facility safely.

(5) The design of the systems provided for control of radiological effluents from the facility to determine (a) that these systems are capable of controlling the release of radioactive wastes from the facility within the limits of the Commission regulations in 10 CFR 20, and (b) that the applicant is capable of operating the equipment provided so that radioactive releases are reduced to levels that are as low as is reasonably achievable (ALARA) within the context of the Commission regulations in 10 CFR 50 and to meet the dose design objectives of Appendix I to 10 CFR 50.

(6) The applicant's quality assurance program for the operation of the facilities to ensure (a) that the program complies with the Commission regulations in 10 CFR 50, and (b) that the applicant will have proper controls over facility operations so that there is reasonable assurance that the facility can be operated safely and reliably.

1.6 Modifications to the Facility During the Course of the Staff Review

During the review, the staff met a number of times (see Appendix A to this report) with the applicant's representatives, contractors, and consultants to discuss various technical matters related to the facility. Also, the staff made a number of visits to the site to assess specific safety matters related to the station.

The applicant made a number of changes to the facility design as a result of the staff review. These design changes, which were also reviewed by the staff, are applicable to both Byron and Braidwood. Specific details concerning these changes are included in FSAR revisions and in appropriate subsections of this report.

1.7 Summary of Outstanding Items

As a result of the staff review of the safety aspects of the Braidwood application, a number of items remain outstanding at the time this report is issued. Because the staff has not completed its review and reached its final positions in these areas, the staff considers these items to be open. The staff review of these items will be completed before a decision to issue an Operating License is made and will be reported in a supplement to this report. The outstanding items, with appropriate references to subsections of this report, are listed below. The outstanding items are listed in two parts: Part A lists the site-specific items for Braidwood; Part B lists the duplicate plant items. Note that all Part B and some Part A items are in the Byron SER. All items that include both site-specific-related information and duplicate plant design features are highlighted with an asterisk. These items, with appropriate references to subsections of this report, are listed below.

Part A Items	Section
(1) Pump and valve operability	3.9.3.2*

(2) Seismic and dynamic qualification of equipment	3.10*
(3) Environmental qualification of electrical and mechanical equipment	3.11*
(4) Containment pressure boundary components	6.2.7
(5) Organizational structure	13.1, 13.4
(6) Emergency preparedness plans and facilities	13.3*
(7) Procedures generation package (PGP)	13.5.2
(8) Control room human factors review	18.0*

Part B Items	Section
(1) Turbine missile evaluation	3.5.1.3
(2) Improved thermal design procedures	4.4.1
(3) TMI Action Item II.F.2: Inadequate Core Cooling Instrumentation	4.4.7
(4) Steam generator flow-induced vibrations	5.4.2
(5) Conformance of ESF filter system to RG 1.52	6.5.1
(6) Fire protection program	9.5.1
(7) Volume reduction system	11.1, 11.4.2
(8) Control room human factors review	18.0

1.8 Confirmatory Issues

At this point in the staff review, there are a few items which have essentially been resolved to the staff's satisfaction, but for which certain confirmatory information has not yet been provided by the applicant. In these instances, the applicant has committed to provide the confirmatory information. The staff is awaiting confirmation of the applicant's commitment to comply with these positions and/or receipt of the appropriate confirmatory information. The outstanding issues are listed in two parts: Part A lists the site-specific issues for Braidwood; Part B lists the duplicate plant issues. Note that all Part B and some Part A issues are in the Byron SER. All issues which include both site-specific-related information and duplicate plant design features are highlighted with an asterisk. These issues, with appropriate references to subsections of this report, are listed below.

Part A Items	Section
(1) Applicant compliance with the Commission's regulations	1.1, 3.1*
(2) Site drainage	2.4.3.3

(3) Piping vibration test program	3.9.2.1*
(4) Preservice Inspection Program	5.2.4, 6.6*
(5) Reactor vessel materials	5.3
(6) Electrical distribution system voltage verification	8.2.4*
(7) Independence of redundant electrical safety equipment	8.4.4
(8) RPM qualifications	12.5
(9) Revision to Physical Security Plan	13.6

Part B Items	Section
(1) Inservice testing of pumps and valves	3.9.6
(2) Steam generator tube surveillance	5.4.2.2
(3) Charging pump deadheading	6.3.2, 7.3.2
(4) Minimum containment pressure analysis for performance capabilities of ECCS	6.2.1.5
(5) Containment sump screen	6.2.2
(6) Containment leakage testing vent and drain provisions	6.2.6
(7) Confirmatory test for sump design	6.3.4.1
(8) IE Bulletin 80-06	7.3.2.2
(9) Remote shutdown capability	7.4.2.2
(10) TMI Action Plan Item II.D.1 TMI Action Plan Item II.K.3.1 TMI Action Plan Item III.D.1.1	3.9.3.3, 5.2.2 7.6.2.7 9.3.5
(11) SWS process control program	11.4.2
(12) Noble gas monitor	11.5.2
(13) RCP rotor seizure and shaft break	15.3.6
(14) Anticipated transients without scram (ATWS)	15.6

1.9 License Conditions

There are several issues for which a license condition may be desirable to ensure that staff requirements are met during plant operation. The license condition may be in the form of a condition in the body of the operating licenses, or a limiting condition for operation in the Technical Specifications appended to the licenses.

The outstanding license conditions are listed in two parts: Part A lists the site-specific license conditions for Braidwood; Part B lists the duplicate plant license conditions. Note that all Part B and some Part A license conditions are in the Byron SER. All license conditions that include both site-specific-related information and duplicate plant design features are highlighted with an asterisk. These license conditions, with appropriate references to subsections of this report, are listed below.

Part A Items	Section
(1) Inservice inspection program	5.2.4, 6.6*
(2) Natural circulation testing	5.4.3*
(3) Response time testing	7.2.2.5*
(4) Steam valve inservice inspection	10.2*
(5) Implementation of secondary water chemistry monitoring and control program as proposed by the Byron/Braidwood FSAR	10.3.3*

Part B Items	Section
(1) Masonry walls	3.8.3
(2) TMI Item II.B.3 postaccident sampling	9.3.2
(3) Fire Protection Program	9.5.1

1.10 Unresolved Safety Issues

Section 210 of the Energy Reorganization Act of 1974, as amended, reads as follows:

> Unresolved Safety Issues Plan
>
> Section 210. The Commission shall develop a plan for providing for specification and analysis of unresolved safety issues relating to nuclear reactors and shall take such action as may be necessary to implement corrective measures with respect to such issues. Such plan shall be submitted to the Congress on or before January 1, 1978 and progress reports shall be included in the annual report to the Congress thereafter.

In response to this reporting requirement, the NRC provided a report to the Congress, NUREG-0410, in January 1978, which described the generic issues program of the Office of Nuclear Reactor Regulation (NRR) that had been implemented early in 1977. The NRR program described in NUREG-0410 provides for the identification of generic issues, the assignment of priorities, the development of detailed task action plans to resolve the issues, the projections of dollar and personnel costs, continuing high-level management oversight of task progress, and public dissemination of information related to the tasks as they progress.

Since the issuance of NUREG-0410, each annual report has described NRC progress in resolving these issues.

The staff continuously evaluates the safety requirements used in its review against new information as it becomes available. In some cases, the staff takes immediate action or interim measures to ensure safety. In most cases, however, the initial staff assessment indicates that immediate licensing actions or changes in licensing criteria are not necessary. In any event, further study may be deemed appropriate to make judgments as to whether existing staff requirements should be modified. These issues being studied are sometimes called generic safety issues because they are related to a particular class or type of nuclear facility. A discussion of these matters and the NRC program for the resolution of these generic issues is provided in Appendix C to this report. The appendix includes references to sections of this report for more specific discussions concerning this facility.

1.11 Statement on Standardization

The Byron and Braidwood Stations use a duplicate plant design in accordance with the Commission's standardization policy of March 5, 1973, and August 31, 1978. The policy option for duplicate plants allows for a simultaneous review of the duplicate portions of a limited number of plants that are to be constructed within a limited time span at multiple sites. The staff's review of the duplicate design portions of these two plants has been documented in the Byron SER (NUREG-0876), including Supplements 1, 2, and 3. When reading those sections in the Byron SER (Sections 2 through 18) that are referenced by this report, the reader should substitute "Braidwood" for "Byron." The Braidwood SER only addresses those topics which are outside the scope of the duplicate design as delineated in the staff's Final Duplicate Design Approval (FDDA), dated June 16, 1982. A copy of the FDDA is included in Appendix G of this report. Future supplements to the Braidwood SER will address all unresolved issues concerning Braidwood, including both site-specific-related information and duplicate design features.

2 SITE CHARACTERISTICS

Chapter 2, "Site Characteristics," for the Braidwood Station, Units 1 and 2, has been reviewed in accordance with the July 1981 edition of the "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants" (SRP), NUREG-0800.

2.1 Geography and Demography

2.1.1 Site Location and Description

The Braidwood Station is located on a 4,454-acre site in Will County in northeastern Illinois, about 60 mi southwest of Chicago. The site is located approximately 3 mi southwest of the Kankakee River and about 20 mi from Joliet, Illinois (1980 population of 77,956) in a south-southwest direction. Kankakee, Illinois, with a 1980 population of 30,141, is located about 20 mi east-southeast of the site. Figure 2.1 shows the general region of the Braidwood site.

The topography of the site is shown in Figure 2.2. The roughly rectangular site occupies approximately 4,454 acres; 2,537 acres of it comprising the main cooling pond. The pond will have an elevation of 595 ft above mean sea level (MSL) when it is filled to capacity. The dike, surrounding the pond, has an elevation of 600 ft MSL. The site is in an area of flat agricultural farmland that has been scarred from coal strip mining, and the site itself is located principally on terrain that was strip mined. The area of principal plant structures has been graded to an elevation of 600 ft MSL.

The coordinates of Braidwood Unit 1 are 41° 14' 38" north latitude and 88° 13' 42" west longitude. The Universal Transverse Mercator coordinates are 4,565,300 m north and 397,000 m east.

2.1.2 Exclusion Area Authority and Control

The applicant has defined the Braidwood exclusion area as the property located within the site boundary, as shown in Figure 2.3. The minimum exclusion area boundary distance is 1,500 ft (457 m), measured from the outer containment wall. The general location of the plant is shown in Figure 2.3. All of the land within the exclusion area, including the mineral rights is owned by Commonwealth Edison Company. No railroads, highways, or waterways traverse the exclusion area. A railroad spur line, however, is used to transport equipment to the nuclear facility. There are no plant-unrelated activities within the exclusion area. The applicant has, however, made arrangements for a limited number of people to search for fossils in the cooling pond area, which falls outside of the exclusion area boundary (see Figures 2.2 and 2.3).

By virtue of the ownership of both the land and the mineral rights within the exclusion area, the staff concludes that the applicant has the authority to determine all activities within the exclusion area, as required by 10 CFR 100.

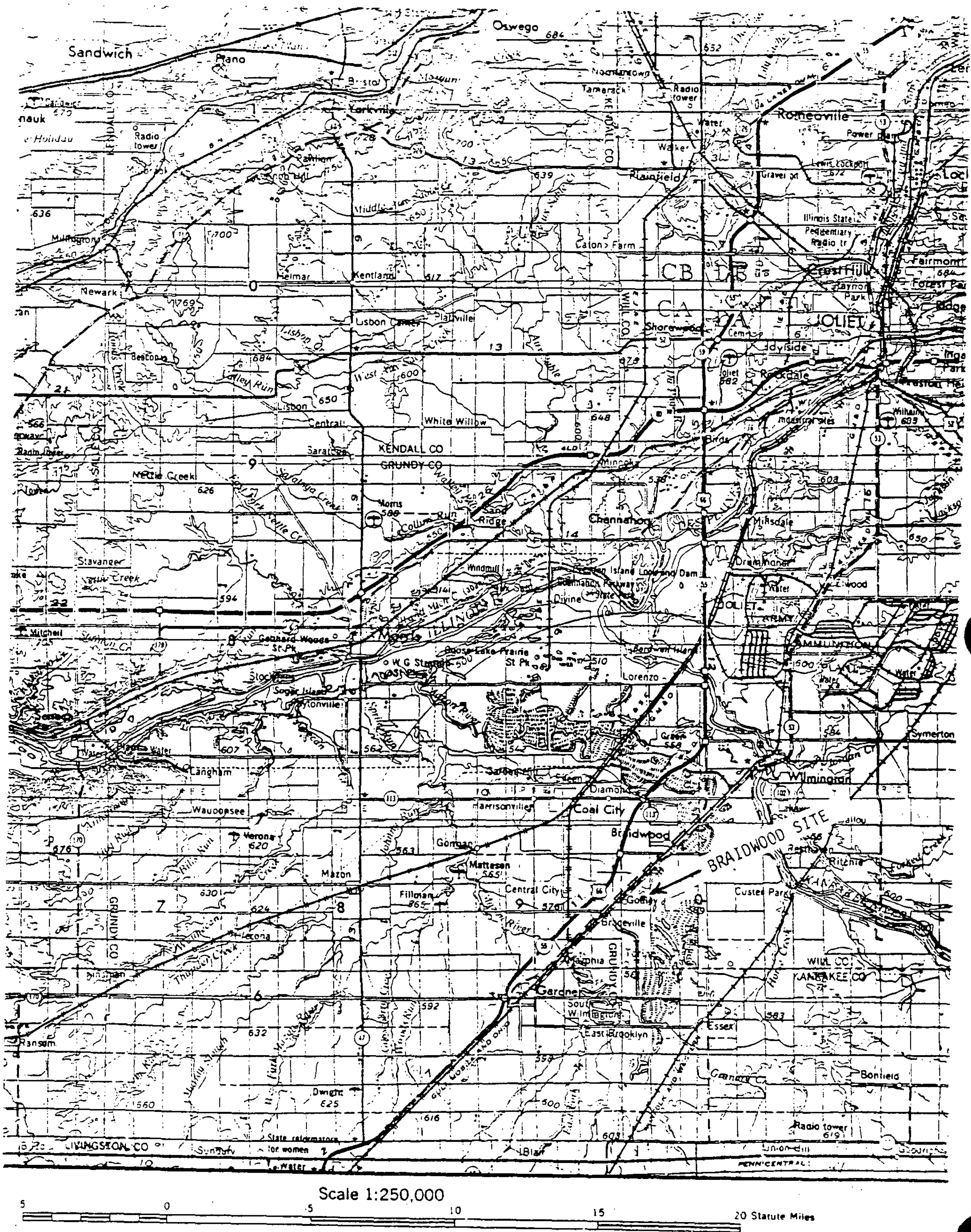

Figure 2.1 Location of site

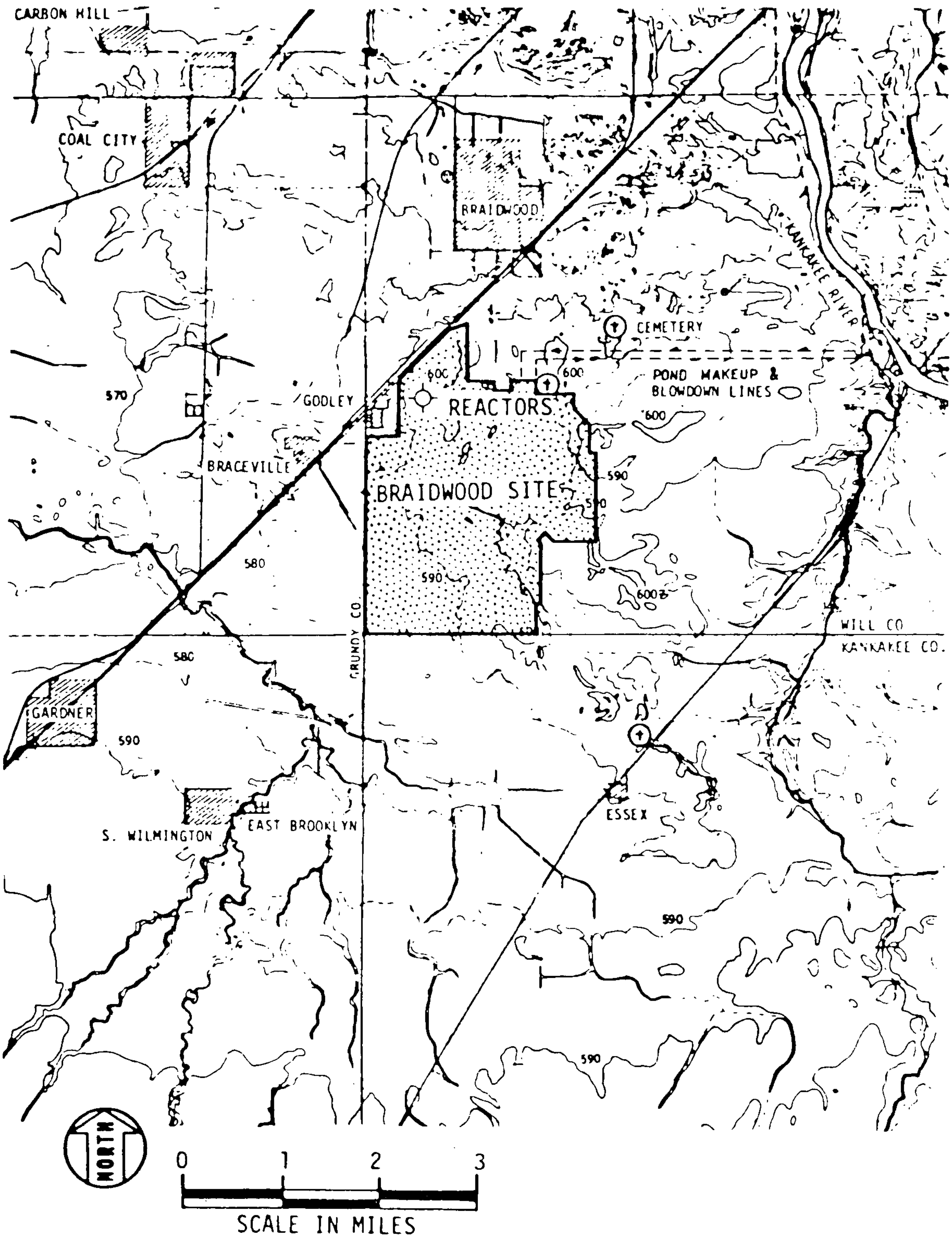

Figure 2.2 Site with respect to Kankakee River and county boundaries
Source: FSAR Figure 2.1-2

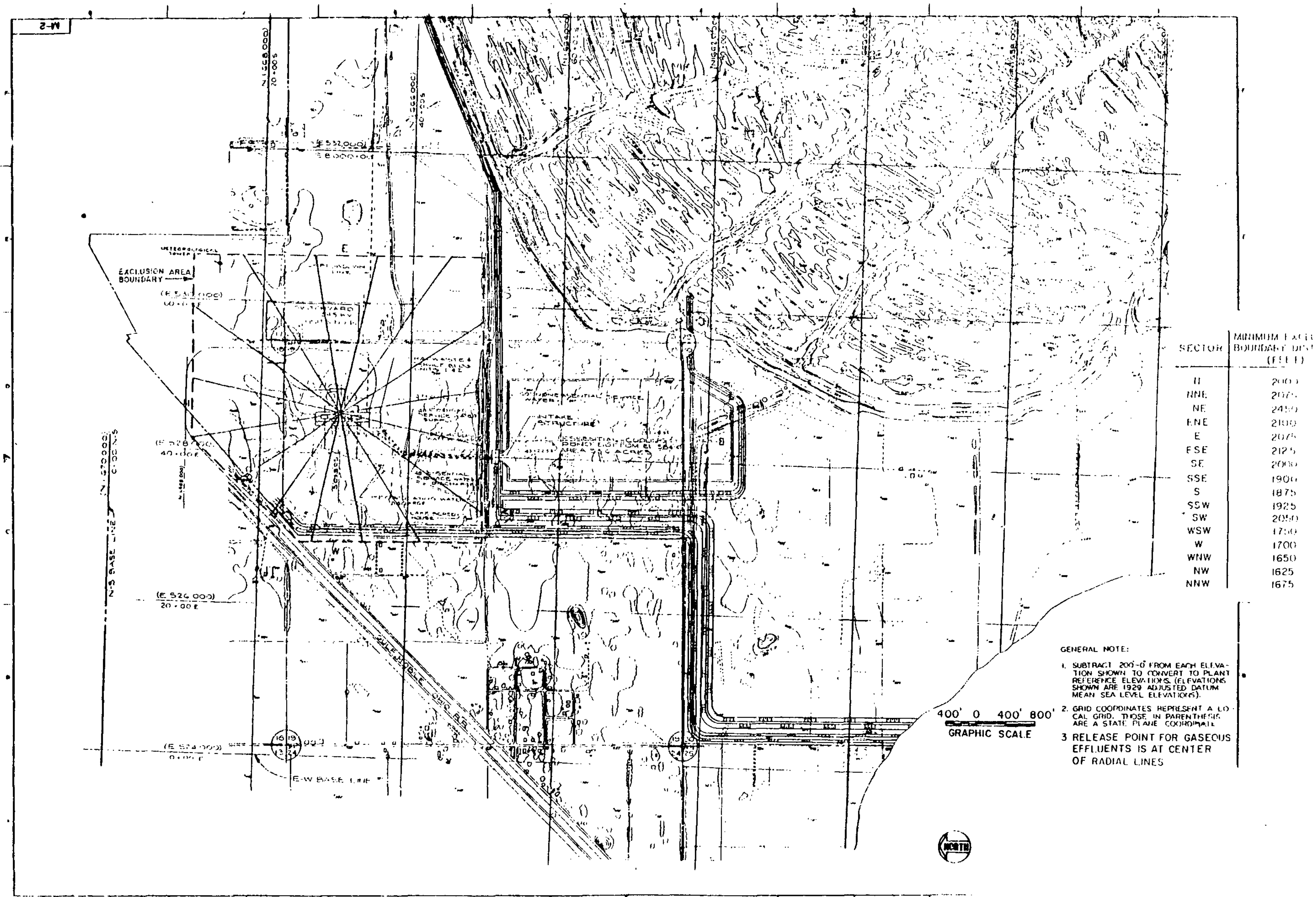

Figure 2.3 Exclusion area gaseous release point orientation and minimum exclusion boundary distance
Source: FSAR Figure

2.1.3 Population Distribution

The resident population in the vicinity of the Braidwood site is shown as a function of distance in Table 2.1. The nearest community with a population of more than 1,000 is Braidwood, Illinois, which had a 1980 population of 3,429 and is about 1-1/4 mi north of the plant site. The 1980 population within 10 mi of the site was 27,482 persons. The largest community near the site is Joliet, Illinois, located 20 mi north-northeast of the site, which had a 1980 population of 77,956, a decrease from 78,827 in 1970.

Table 2.1 Population in the vicinity of the Braidwood site

	Distance from the plant (miles)					
Year	0-1	0-2	0-3	0-4	0-5	0-10
1980	615	3084	5231	8945	12,472	27,482
1990	749	3778	6406	10,767	14,936	31,768
2020	842	4256	7219	12,094	16,760	35,411

The applicant has chosen a low population zone (LPZ) radius of 1-1/8 mi. The 1980 population (including 500 transients) was 1,205 persons and is projected to reach 1,456 (including 500 transients) in the year 2020.

The applicant has indicated that the nearest densely populated center, as defined in 10 CFR 100, of about 25,000 or more persons, is the town of Joliet, Illinois, which is at a distance that is at least one and one-third times the LPZ radius.

The staff made an independent estimate of the 1980 population data within a 50-mi radius of the Braidwood site based on the Bureau of Census data. The staff value is approximately 5% lower than that indicated by the applicant. The applicant projects that this population will increase to about 5,124,734 persons by the year 2020. This represents a growth of about 3% per decade for the period 1980 to 2020. The staff has calculated a 4% per decade growth rate using the Bureau of Economic Analysis (BEA) projections for those BEA areas within a 50-mi radius of the Braidwood site.

2.1.4 Conclusion

On the basis of (1) the 10 CFR 100 definitions of the exclusion area, the LPZ, and the population center distance; (2) the staff's analysis of the onsite meteorological data from which the relative concentration factors (χ/Q) were calculated (see Section 2.3.4 of this report); and (3) the calculated potential radiological dose consequences of design-basis accidents (see Section 15 of this SER), the staff has concluded that the exclusion area, LPZ, and the population center distance meet the criteria of 10 CFR 100 and are acceptable.

2.2 Nearby Industrial, Transportation, and Military Facilities

2.2.1 Transportation Routes

The nearest roads are secondary Illinois State Routes 53 and 129, which pass parallel to the north western boundary of the reactor site (see Figure 2.4). Interstate Route 55 runs parallel to the first two roads at approximately 1-1/10 mi from the reactor buildings.

The nearest railroad, as shown in Figure 2.4, runs between State Routes 53 and 129 at approximately 1,550 ft from plant safety-related structures. The Illinois Central Gulf Railroad also has a line located about 2.5 mi west of the site. In addition, the Norfolk and Western Railroad is located 4.5 mi southeast of the site, and the Atchison, Topeka, and Sante Fe Railroad passes approximately 4 mi northwest of the site.

There are no airports within 5 mi of the site. There are three airports with turf runways located between 5 and 10 mi of the plant that have a few operations daily. The nearest airport with a paved runway is located approximately 13 mi southwest of the site near Dwight, Illinois.

There are two low-altitude Federal airways within 10 mi of the site (see Figure 2.5) that have a maximum altitude of 18,000 ft. On the basis of past reviews of plant sites that meet the staff's criteria and that have equivalent traffic in equal or closer proximity, the staff concludes that the probability of an aircraft causing radiological consequences in excess of the exposure guidelines of 10 CFR 100 is within the acceptance criteria of SRP Section 2.2.3 and is, therefore, acceptable.

The pipelines within 5 mi of the Braidwood site are shown in Figure 2.6. Based on past reviews of similar pipelines carrying petroleum and natural gas products, the staff concludes that these pipelines are sufficiently distant from the plant and thus do not pose a threat to the Braidwood Station.

There is no potential for a barge or ship impact to the Braidwood water intake structure on the Kankakee River because of the location of fixed dams (no navigation locks) upstream and downstream of the intake structure.

The plant control room is designed to protect operating personnel from the effects of a ruptured chlorine railroad tank car in the vicinity of the site. A further description of the control room protection can be found in Section 6.4 of this SER.

2.2.2 Nearby Facilities

The Joliet Arsenal, the nearest military facility to the Braidwood site, is located 8 mi northeast of the site. This military facility ships and receives high explosive materials on Illinois State Highway Routes 53 and 129 and on the Illinois Central Gulf Railroad, which borders the northwest portion of the reactor exclusion area. During the construction permit stage review of this plant, the applicant provided a comprehensive analysis (by the Illinois Institute of Technology Research) of munition shipments past the plant site. This document was attached as Appendix A to Chapter 2 of the Braidwood Station Preliminary Safety Analysis Report.

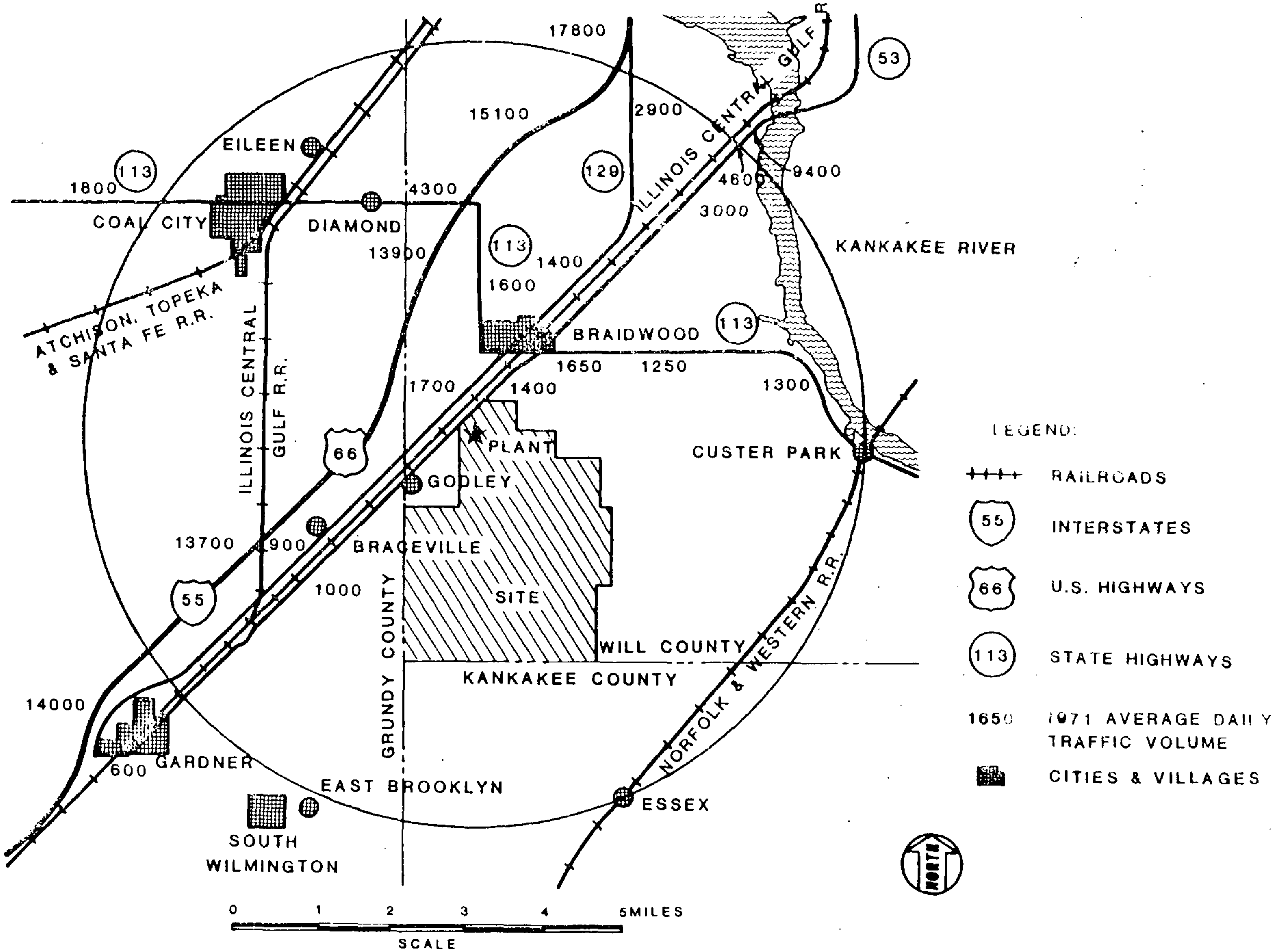

Figure 2.4 Transportation networks within 5 mi of the site

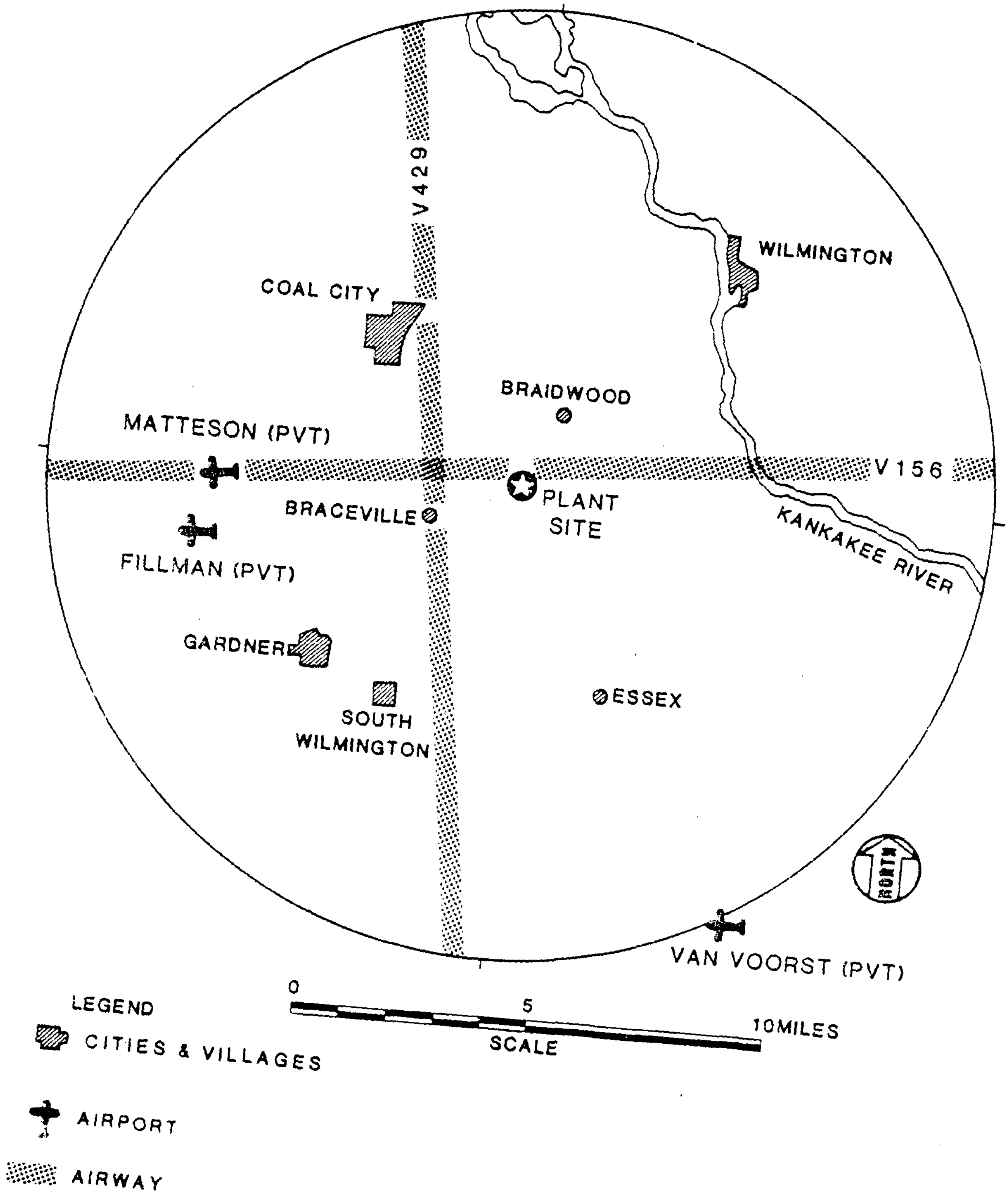

Figure 2.5 Airports and low altitude Federal airways within 10 mi of the site

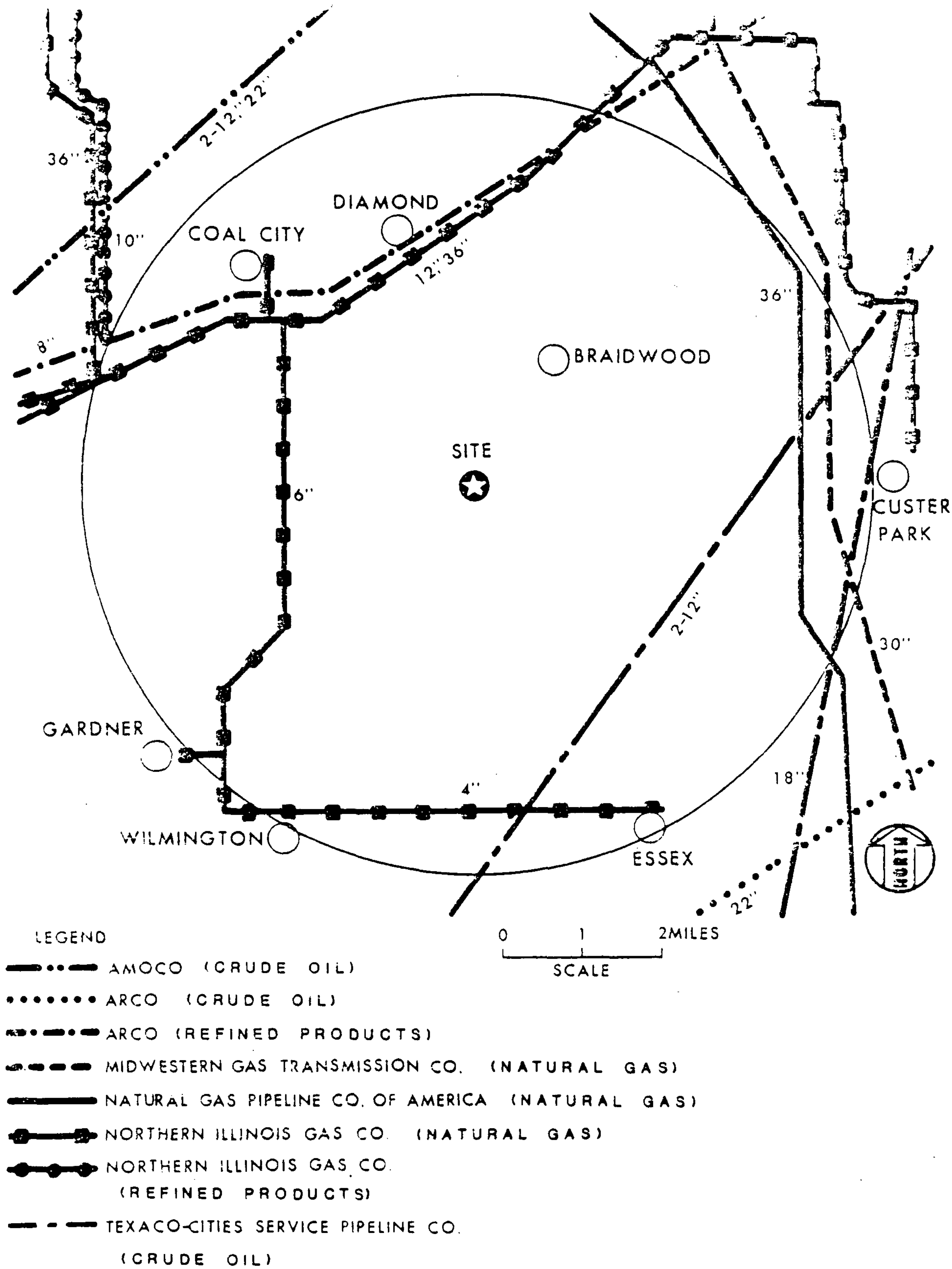

Figure 2.6 Pipelines within 5 mi of the site
Source: FSAR Figure 2.6

The U.S. Atomic Energy Commission (AEC) staff, with the assistance of consultants from the Naval Surface Weapons Center at White Oak, Silver Spring, Maryland, reviewed this analysis and published the January 9, 1975, report entitled, "Partial Safety Evaluation Report on Site Characteristics," related to the application of Commonwealth Edison Company to construct the Byron Station, Units 1 and 2, and Braidwood Station, Units 1 and 2. Section 2.2.1 of that report, discussing industrial, transportation, and military facilities near Braidwood, is attached as Appendix 2A to this SER.

During the operating license review of Braidwood, the staff reviewed the additional information that the applicant provided in Section 2.2 of the Final Safety Analysis Report. One of the major changes indicated by this report is the reduction in shipments of explosive materials by rail past the Braidwood site. The shipments during the period 1974 through 1977 are reported to be less than what was shown for a 6-month period in 1974. At present, the arsenal is inactive and there are no shipments of Class A, B, and C explosives, TNT, or lead azide over the Illinois Central Gulf Railroad. The only ammunition shipped over these tracks is small caliber and very few in number.

The Braidwood facility is designed to withstand the effects of a postulated explosion of a single boxcar load of TNT (132,000 lb) at the closest approach to the rail line (see Section 3.5 for further details on this analysis).

The applicant was informed at the construction permit stage for Braidwood that the staff will require suitable arrangements to be provided for the inspection and maintenance of the track near the plant, and that the applicant agreed to take whatever action is necessary to ensure that the track does not fall below its level as a high quality track according to the track safety standards of the Federal Railroad Administration. On the basis of the data provided by the applicant at the CP- and OL-review stages, the staff again concludes that the probability of an accidental explosion of two or more boxcars on the rail line adjacent to the Braidwood site is of the order of 10^{-7} per year and that traffic along both road and rail transportation routes will not adversely affect the safe operation of the Braidwood station.

2.2.3 Conclusions

The staff's review has been conducted based on GDC 4 and SRP Section 2.2.3. The staff concludes that the plant is adequately protected and can be operated with an acceptable degree of safety as a result of activities at nearby transportation, industrial, and military facilities.

2.3 Meteorology

Evaluation of regional and local climatological information, including extremes of climate and severe weather occurrences that may affect the design and siting of a nuclear plant, is required to ensure that the plant can be designed and operated within the requirements of Commission regulations. Information concerning atmospheric diffusion characteristics of a nuclear power plant site is required to determine that radioactive effluents from postulated accidental releases, as well as routine operational releases, are within Commission guidelines. Sections 2.3.1 through 2.3.5 have been prepared in accordance with the review procedures described in the Standard Review Plan (NUREG-0800), using information presented in FSAR Section 2.3, responses to requests for additional information, and generally available reference materials, as described in the appropriate sections of the SRP.

2.3.1 Regional Climatology

The Braidwood site in northeast Illinois is in an area with a continental type climate. This type of climate is one with cold winters, warm summers, and precipitation throughout the year.

The variability of the meteorological conditions results from the movement of frontal systems across the area with associated precipitation and wind-flow changes. Annual temperatures range from a mean daily low of 10°-20°F in January to a mean daily maximum of 85°-90°F in July.

Total annual precipitation in the area is about 35 in. with 110-120 days a year having 0.01 in. or more of rain. Total annual average snowfall ranges from 24 to 36 in., and freezing rain may occur from 8 to 12 days a year. Hail associated with thunderstorms occurs an average of 2 days a year, although about 60 thunderstorms a year are observed.

On an annual basis, prevailing winds in the area are generally from the southwest, with an average wind speed of 10 mph, although northwesterly winds predominate in the winter.

Tornadoes within the entire State of Illinois are not uncommon. During the 1953-1971 period, 404 tornadoes were observed throughout the state. In the 1° square containing the plant, tornadoes have been estimated to recur on an average of every 757 years using the statistical approach described in WASH-1300 and the tornado occurrence data from 1954 to 1981.

The plant has been designed to meet the RG 1.76 tornado criteria of wind speed equal to 360 mph comprised of 290-mph rotational and 70-mph translational velocities, and a 3-psi pressure drop at a rate of 2 psi/sec appropriate for the RG 1.76 Region I tornado.

Fastest mile winds of 85 mph were used as the operating-basis winds for seismic Category I buildings. This operating-basis wind, which is the 100-year recurrence fastest mile wind, is conservative with respect to values determined in NUREG/CR-2890 for locations in the general area of the site for the 100-year return period wind. Air pollution episodic occurrences in the area result from stagnation of air masses with a corresponding increase of pollutants in the lower levels of the atmosphere. During the period 1936-1956, about 120 stagnation days were observed. Seasonal variation of atmospheric mixing height ranged from a minimum of 305 m in the morning to a maximum of 1,532 m in the summer season in the afternoon. The mixing height is a measure of the layer thickness in which air pollutants can disperse freely up to the top of the layer.

Design meteorological conditions for the ultimate heat sink (cooling pond) were based on conservative values derived from records at Peoria and Springfield, Illinois, during the period 1948-1974. The parameters reviewed indicated those that would result in minimum heat transfer rates and greatest evaporation from the pond.

Although the 26 years of record is less than the 30-year record recommended by RG 1.27, from a meteorological point of view, the use of the shorter record period is expected to have little or no effect on the design of the ultimate heat sink than if 30 years of record had been used. The staff, therefore, concludes that the 26-year period of record is an acceptable basis for the design of the ultimate heat sink.

As discussed above, the staff has reviewed available information on the regional meteorological conditions that are of importance for the safe design and siting of this plant. On the basis of this review, the staff concludes that the applicant has identified and considered appropriate regional meteorological conditions in the design and siting of this plant and, therefore, meets the requirements of 10 CFR 100.10 and GDC 2. The design-basis tornado characteristics selected by the applicant conform to the position in RG 1.76 and, therefore, meet the requirement of GDC 4 to determine an acceptable design-basis tornado for missile generation.

2.3.2 Local Meteorology

Local sources of meteorological data that are indicative of conditions at the Braidwood site include Argonne National Laboratory, Dresden Nuclear Plant, National Weather Service in Peoria, Illinois, and the onsite meteorological monitoring program.

Annual average temperatures and precipitation are similar at all the locations.

Wind direction is generally south or southwest on an annual basis; however, winter conditions produce an increase of west and northwest winds at Braidwood, as shown in the wind rose from the FSAR (Figure 2.7), and other locations. Atmospheric stability conditions in the area are defined at the Braidwood site by the vertical temperature difference between 9 and 61 m. From January 1979 through December 1982, the frequency distribution of stability classes in percent is given below:

Class	A	B	C	D	E	F	G
Frequency, %	4.0	4.3	7.1	45.0	27.5	8.4	3.8

Stability observations at Peoria, based on time of day, surface observations of cloud cover, and wind speed from 1966 through 1975 are

Class	A	B	C	D	E	F	G
Frequency, %	0.3	4.5	10.5	60.8	10.4	9.5	4.0

At the site and Peoria, the stability classes center about the neutral (D) and slightly stable (E) classes as expected for a continental location.

Snow and ice pellet precipitation are observed in the site area and produce an annual average snowfall of about 24 in. at Peoria; the greatest 24-hour amount was 12.2 in. The 100-year return period winter snow loads in the area have been determined to be about 20 lb/ft^2. The roofs of safety-related structures are designed for a 104 lb/ft^2 snow and ice load, which includes the 100-year antecedent snow pack combined with the winter probable maximum precipitation (PMP).

onsite 1/79 - 12/82

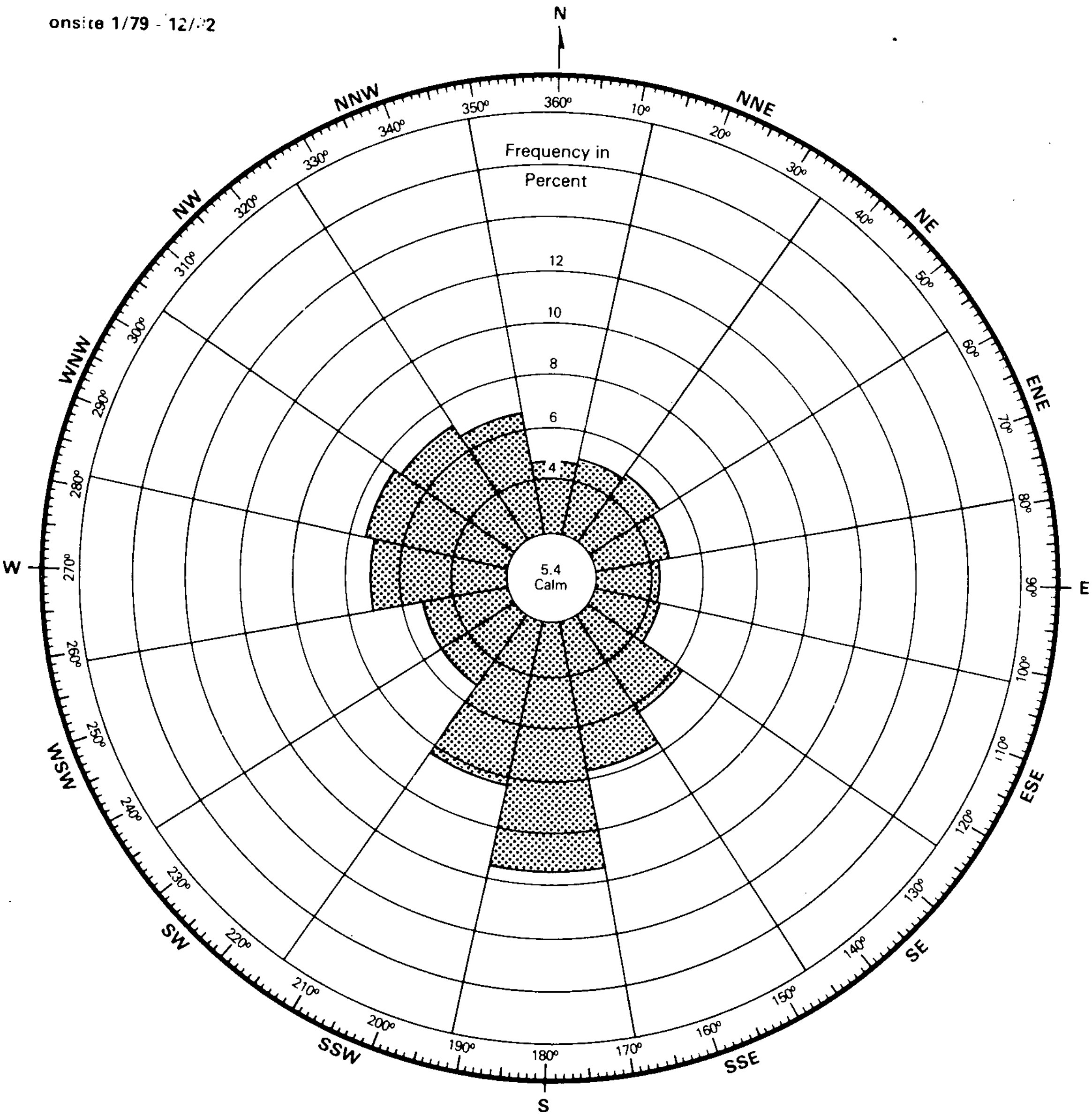

Figure 2.7 9-m wind at Braidwood

Winds in the area are predominantly from the southerly or southwesterly directions; winds from other directions are observed less frequently. An increase in northwest wind frequency is observed during the winter at the site and Argonne, Illinois. At Peoria, the fastest mile wind speed through 1976 was 75 mph, which occurred from the northwest in 1953.

As discussed above, the staff has reviewed available information on local meteorological conditions that are of importance for the safe design and siting of this plant. On the basis of this review, the staff concludes that the applicant has identified and considered appropriate local meteorological conditions in the design and siting of this plant and, therefore, meets the requirements of 10 CFR 100.10 and GDC 2.

2.3.3 Onsite Meteorological Measurements Program

Meteorological data have been collected since November 1973 on a 320-ft tower situated about 0.4 mi northeast of the Unit 1 reactor building. The measurements, following the guidance of RG 1.23 for equipment accuracies and sensitivities, include the parameters in Table 2.2.

Table 2.2 Onsite meteorological measurements

Meteorological parameter	Tower elevation (ft)
Wind speed and wind direction	34 and 203
Temperature difference	34 and 203
Dewpoint temperature	34 and 203
Ambient air temperature	34

Precipitation is measured at ground level near the tower. All measurement signals except those for precipitation are transmitted to the control room.

Data collected from January 1979 through December 1982 yielded over 97% joint frequency data recovery exceeding the 90% recovery rate suggested in RG 1.23. Dispersion estimates for accidental and routine gaseous releases were made with these data. The results of the dispersion calculations are provided in Sections 2.3.4 and 2.3.5.

The meteorological program for emergency response activities in accordance with the guidance of Supplement 1 to NUREG-0737 and RG 1.101 will be reviewed at the time of the emergency preparedness implementation appraisal and evaluated in the appraisal report.

2.3.4 Short-Term (Accident) Diffusion Estimates

Short-term (up to 30 days) accidental releases from buildings and vents were evaluated by the applicant and the staff. The staff analysis used the direction-dependent atmospheric dispersion model described in RG 1.145 and the methods described in NUREG/CR-2858 to evaluate the short-term postulated design-basis accidental gaseous releases.

To evaluate the assumed ground level release, the staff used onsite 9-m wind speed and direction data and the vertical temperature difference between the 9- and 61-m levels as a measure of atmospheric stability. These data, collected from January 1979 through December 1982 in joint frequency form, provided the basis for the determination of relative concentrations at the exclusion area boundary (EAB) and the low population zone (LPZ) distance of 1,810 m, with allowance for building wake effects, effluent recirculation, and plume meander during light-wind and stable atmospheric conditions. The maximum 0- to 2-hour relative concentration (χ/Q) at the 485-m EAB, which is expected to be exceeded less than 0.5% of the time in the most limiting direction, is 5.6×10^{-4} sec/m^3 west-northwest of the plant.

At the LPZ distance of 1,810 m, the relative concentration (χ/Q) values for the various time periods are given in Table 2.3.

Table 2.3 Braidwood low population zone relative concentrations (χ/Q)

Time	χ/Q sec/m^3
0-8 hours	5.9×10^{-5}
8-24 hours	4.4×10^{-5}
1-4 days	2.3×10^{-5}
4-30 days	9.4×10^{-6}

The χ/Q values in Table 2.3, although not as conservative as the applicant's, have been used by the staff to evaluate the plant.

The applicant's evaluation did not follow the guidance of RG 1.145 and yielded more conservative relative concentration values because only the building wake effect was taken into account using 5% of the sector χ/Q values. During the 1974-1976 period used by the applicant, there was poor data recovery that did not provide the 90% or better recovery rate suggested in RG 1.23. The staff, therefore, used the 1979-1982 data, which had 97% data recovery and which provide greater assurance that representative meteorological conditions have been considered in the evaluation.

On the basis of the above evaluation, the staff concludes that the applicant has considered conservative atmospheric dispersion estimates for assessments of the consequences of radioactive releases for design-basis accidents in accordance with the requirements of 10 CFR 100.11. The staff has prepared the atmospheric dispersion estimates provided in this section and has used these estimates in an independent assessment of the consequences of radioactive releases for design-basis accidents.

2.3.5 Long-Term (Routine) Diffusion Estimates

To audit the applicant's estimates, the NRC staff has performed an independent calculation of annual average relative concentration (χ/Q) and relative deposition (D/Q) at nearby receptor locations. The staff made similar calculations for the population dose assessment out to 50 miles from the plant in each of the 16 cardinal point directions around the plant. The meteorological data collected on site (61.7-m wind speed and direction and temperature difference between 9 and 61 m) (described in Section 2.3.3) were used for this assessment. A straight line gaussian dispersion model corrected for effluent recirculation, as described in RG 1.111, was used by the staff. The methods described in NUREG/CR-2919 were utilized to determine the annual average χ/Q and D/Q around the facility.

A continuous mixed-mode release from the plant vent was evaluated with the maximum annual average χ/Q of 1.9×10^{-6} sec/m^3 and D/Q of 2.6×10^{-8} sec/m^2 at the 610-m site boundary north of the plant. The applicant performed a similar analysis except effluent recirculation was not included in the assessment and data collected during the period January 1974 through December 1976 were used.

The staff values are slightly more conservative than those calculated by the applicant. This difference largely results from the use of effluent recirculation factors by the staff to account for spatial and temporal variation in the effluent dispersal. These staff values for the annual average χ/Q and D/Q were used because the higher data recovery rates provide greater confidence in the representativeness of the meteorology data set.

On the basis of the above evaluation, in accordance with SRP Section 2.3.5, the staff concludes that the applicant has considered adequate atmospheric dispersion estimates to demonstrate compliance with the numerical dose guidelines in 10 CFR 50, Appendix I. The slightly more conservative dispersion estimates developed by the staff have been used by the staff to assess the impact of normal operational releases.

2.4 Hydrology

2.4.1 Introduction

The staff has reviewed the hydrologic engineering aspects of the applicant's design, design criteria and design basis of safety-related facilities for the Braidwood Station. The acceptance criteria used as a basis for the staff's evaluations are set forth in SRP Sections 2.4-1 through 2.4-14 (NUREG-0800). These acceptance criteria include the applicable General Design Criteria (10 CFR 50, Appendix A), Reactor Site Criteria (10 CFR 100), and Standards for Protection Against Radiation (10 CFR 20, Appendix B, Table II). Guidelines for implementation of the requirements of the acceptance criteria are provided in regulatory guides, ANSI standards, and branch technical positions identified in Sections 2.4-1 through 2.4-14 of the SRP. Conformance to the acceptance criteria provides the bases for concluding that the site and facilities meet the requirements of 10 CFR 20, 50, and 100 with respect to hydrologic engineering.

2.4.2 Hydrologic Description

The site for the Braidwood Station is located in the Kankakee River Basin about 3 mi southwest of the Kankakee River near the town of Custer Park. Plant grade is 600 ft above mean sea level (MSL) with entrance levels to plant structures at 601 ft MSL. Figure 2.8 shows the site in relation to some of the streams in the area. The Mazon River flows about 4 mi southwest of the site in a northwesterly direction. Crane and Granary Creeks, tributaries to the Mazon River, are several miles from the main plant area, but are within 1 or 2 mi of the southern dike for the cooling pond. The primary plant structures (Figure 2.9) are located on the north side of this 2,537-acre cooling pond. The pond has a normal operating level of 595 ft MSL with makeup from and blowdown to the Kankakee River. The pond provides cooling water for normal operation. The pond was created by constructing a series of long levees or dikes around a strip mine area. The levees incorporate a slurry trench to reduce seepage. There is an excavated area within the cooling pond that is called the essential cooling pond (also the ultimate heat sink) which provides the emergency safety-related water supply for the plant.

The drainage area of the Kankakee River upstream of the site is about 5,000 mi^2. The river has a slope of about 2 ft per mile in the vicinity of the site. Average river flow at the site is 3,952 ft^3/sec.

There are two dams on the Kankakee River, one at Wilmington, about 4 mi downstream from the plant river intake, and the other at Kankakee, about 15 mi upstream. The Wilmington Dam is 11 ft high and forms a pool 2 mi long. The Kankakee Dam is 12 ft high and forms a pool 6 mi long.

The main plant area is drained by both a storm drainage system and surface flow controlled by weir flow over perimeter roads and railroads.

Quaternary Age eolian sand, lacustrine sand, and till overlie the bedrock in the vicinity of the site. Many domestic water supplies in the area are obtained from the sand aquifer with well points (shallow-driven wells). The Quaternary deposits are underlain by Pennsylvanian bedrock, which may locally yield up to 20 gpm from interbedded sandstones, but is essentially an aquatard. The most important aquifer in the region is the Cambrian-Ordovician Aquifer, made up of all bedrock between the shales of the Maquoketa Shale Group and the Eau Claire Formation. The Eau Claire shales separate the Cambrian-Ordovician Aquifer from the Mt. Simon Aquifer, which includes sandstones in the lower portion of the Eau Claire Formation and the Mt. Simon Sandstone. Few wells in the region extend to the Mt. Simon Aquifer.

There are numerous private and public water supply wells within 10 mi of the site. No public water supplies are taken from the Kankakee, Mazon, or Illinois Rivers within 50 mi downstream from the site.

The NRC staff has reviewed the material in the applicant's FSAR and other drawings and photographs obtained during the site visit and concludes that the material fulfills the acceptance criteria of SRP Section 2.4.1.

2.4.3 Flooding Potential

The applicant has analyzed the following potential flood sources: the Kankakee River; the Mazon River (a tributary to the Illinois River); Crane and Granary

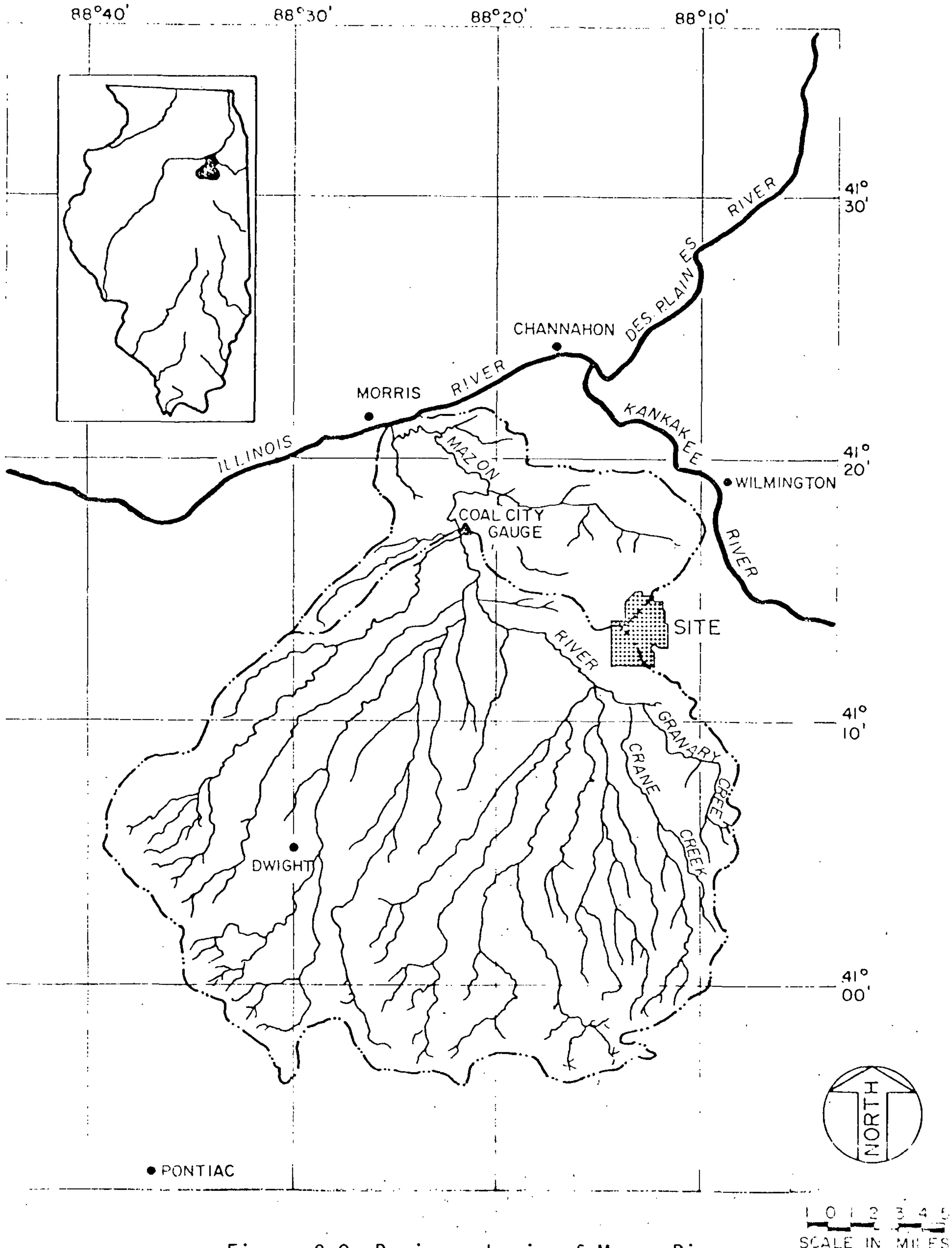

Figure 2.8 Drainage basin of Mazon River
Source: FSAR Figure 2.4-3

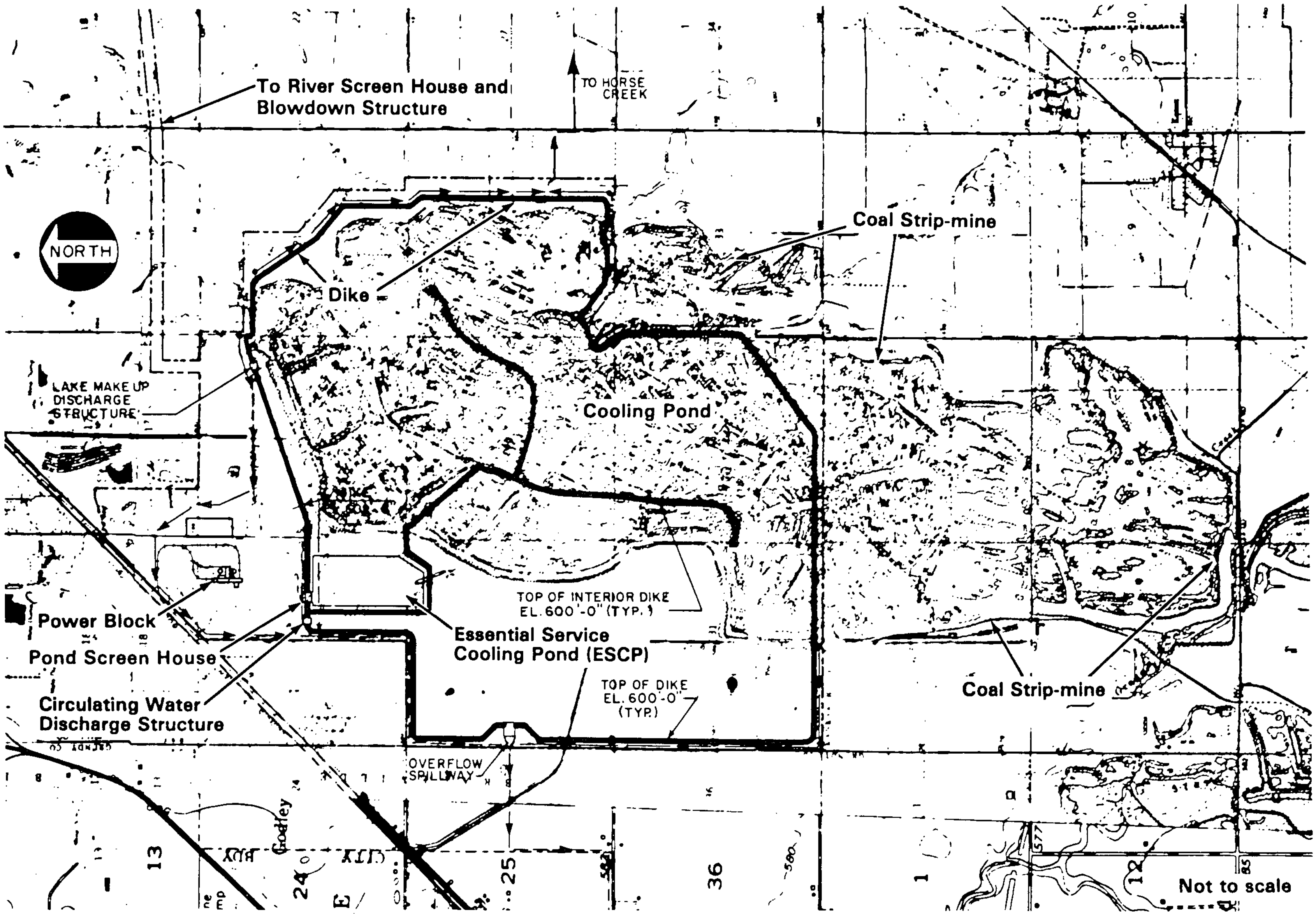

Figure 2.9 Braidwood Station project layout

Creeks (tributaries to the Mazon River); and probable maximum precipitation on the site drainage area, the cooling pond and roofs of safety-related structures. The staff has evaluated the site and surrounding hydrologic features and concludes that these are the only credible potential flood sources for the Braidwood Station.

Further, as discussed below, using the procedures discussed in SRP Sections 2.4.2 through 2.4.7, the NRC staff has been able to conclude that the plant meets the guidelines of RGs 1.59 and 1.102 and, therefore, meets the requirements of GDC 2 with respect to flooding, except as noted in Section 2.4.3.3 on the site drainage system, which is still under review as a confirmatory issue.

2.4.3.1 Kankakee River

The Kankakee River has a drainage area of about 5,000 mi^2 above the river intake facility. The applicant has estimated a probable maximum flood (PMF) peak discharge of 209,000 ft^3/sec with a corresponding water surface elevation of 561.3 ft MSL at that location. Based on a comparison with the PMF calculated for the Dresden site, which is about 12 mi downstream, the staff concludes that the applicant has underestimated the PMF. The Dresden PMF, transposed to the Braidwood site by a simple ratio of the drainage areas, yields a peak discharge of about 350,000 ft^3/sec.

The staff used the results of the detailed PMF developed for the Dresden site and generalized PMF values from RG 1.59 to estimate an upper bound of 400,000 ft^3/sec for the Kankakee River PMF at Braidwood. This discharge would produce a site water surface elevation of about 571.0 ft MSL, which is 29 ft below the plant grade elevation of 600.0 ft MSL. The staff, therefore, concurs with the applicant that the flooding potential from the Kankakee River will be well below the plant grade elevation and thus meets the requirements of GDC 2 of Appendix A to 10 CFR 50 and fulfills the acceptance criteria specified in SRP Section 2.4.3.

2.4.3.2 Mazon River, Crane and Granary Creeks

The Mazon River and its tributaries, Crane and Granary Creeks, have a drainage area of 220 mi^2 near the plant site. The applicant estimated local PMF conditions at locations most critical to the site. The PMF for the Mazon River, at the furthest downstream location (220 mi^2 at Route 66), was estimated by the applicant to have a 112,000 ft^3/sec peak discharge with a corresponding water surface elevation of 582 ft MSL. A PMF peak discharge of 19,500 ft^3/sec, resulting in a water surface elevation of 576 ft MSL, was estimated by the applicant for the combined drainage of Crane and Granary Creeks (52.2 mi^2) at a location about 1 mi south-southwest of the plant and just upstream of the junction with the Mazon River. This indicates that the controlling elevation on Crane and Granary Creeks would result from the Mazon River PMF that produces a higher level downstream.

During the CP-stage review the staff concluded, that although the applicant's PMF estimates appeared to be too low, the plant, at elevation 600 ft MSL, was not threatened by flooding from these creeks. On the basis of its current review, the staff concurs in that assessment.

The staff, therefore, concludes that the plant meets the requirements of GDC 2 of Appendix A to 10 CFR 50 with respect to flood potential from Mazon River and its tributaries.

2.4.3.3 Local Probable Maximum Precipitation in Plant Area

The applicant investigated the effects of flooding as a result of local probable maximum precipitation (PMP) on the plant site drainage system and on the roofs of safety-related structures. For the applicant's analysis of local intense precipitation, the 1-hour PMP on a 1-mi^2 area (17.8 in.) for the site was taken from Hydrometeorological Report (HMR) 52 (1982). This PMP was derived from 6-hour, 10-mi^2 PMP values given in HMR 51 (1978), and is considered a point rainfall value. This 1-hour 1-mi^2 PMP was distributed into values for smaller durations using procedures given in HMR 52. The applicant then used these PMP values along with the rational formula to estimate peak runoff from the various subareas into which the plant site was divided.

The applicant determined maximum water surface elevations adjacent to safety-related buildings by evaluating the weir flow over the peripheral roads and railroads. In the analysis the applicant assumed 100% runoff (i.e., no infiltration) and that the site drainage system was not functioning. The resulting maximum water surface elevation in the power block area was 601.35 ft MSL. The staff has reviewed the applicant's analysis and values and made independent calculations. The staff concurs in the design-basis water level of 601.35 ft MSL.

The design basis flood level of 601.35 ft MSL exceeds the exterior floor entrance levels (601.0 ft MSL) to safety-related buildings. During the CP-stage review, the applicant committed to design the site drainage system so that the local PMF elevation would be below the plant grade floor level.

The staff's acceptance of site drainage in the CP-SER was based on this commitment. RG 1.59 does, however, allow the use of hardened protection at sites that are not flood dry. The applicant has stated that reinforced concrete curbs will be used to protect areas where essential equipment/systems are located. The use of curbs, to elevation 601.4 ft MSL, to keep flood water from safety-related buildings, is acceptable to the staff.

Site drainage will be a confirmatory issue until the staff has reviewed the applicant's provisions for preventing runoff from local intense precipitation from entering safety-related buildings. The applicant needs to provide (1) a tabulation of all accesses to safety-related buildings that are at or below el 601.35 ft MSL and that would allow entrance of floodwater from rainfall runoff and (2) details of the protection to be provided for each opening.

2.4.3.4 Roof Drainage

The applicant has provided data and analyses to show that the roofs of safety-related buildings can support either the all season PMP or the winter PMP combined with the 100-year snowpack. The staff has reviewed the applicant's provisions for roof drainage, using procedures in SRP Section 2.4.2, and finds them to be acceptable and to meet the requirements of GDC 2.

2.4.3.5 Flood Potential From Cooling Water Canals and Reservoirs

There are no canals at the Braidwood Station. Makeup from the Kankakee River is pumped uphill through a 48-in. underground pipeline to the cooling pond. The maximum gross withdrawal rate is 112 ft^3/sec for two units. Blowdown is discharged through a parallel 48-in. pipeline back into the Kankakee River at a maximum rate of 46 ft^3/sec.

The cooling pond has a normal pool elevation of 595 ft MSL with a surface area of 3.96 mi^2. The total drainage area of the pond is 5.3 mi^2. The normal volume of the pond is about 22,300 acre-feet. The pond is contained by dikes having a top elevation of 600 ft MSL, except for that portion of the dike just south of the plant, which has a top elevation of 602.5 ft MSL to protect the plant from the PMF pool level plus coincident wave runup. The dike system is not a seismic Category I structure. The dike or levee system incorporates a slurry trench to reduce seepage from the pond.

The pond was designed to withstand, without overtopping, a PMF calculated from the PMP. The dike system incorporates a 200-ft-wide spillway with an ogee crest that limits the PMF level (including allowance for 1/2 the PMP 3 days before the PMF) to el 598.17 ft MSL with a spillway peak discharge of 2,184 ft^3/sec. The interior slopes of the dikes are protected with 18 to 21 in. of riprap.

The northwest portion of the pond contains an excavated essential cooling pond that provides a safety-related water supply for emergency shutdown of the plant and maintenance thereof for 30 days.

A portion of the dike, in the vicinity of the pond screenhouse, protects the plant area from wave overtopping. The applicant analyzed this condition with a PMF pool (598.17 ft MSL) and a coincident wind of 40 mph, which were the criteria suggested in RG 1.59 at the time of the CP review. This resulted in an estimated maximum wave runup elevation of 602.34 ft MSL. The dike in this area is constructed to el 602.5 ft MSL; thus it was concluded in the CP-SER that it will protect the plant from the maximum pond runup level. Subsequent to the CP review, RG 1.59 was revised to incorporate ANSI Std N170, which calls for using a 2-year extreme wind speed coincident with the PMF. On the basis of its review, using procedures in SRP Sections 2.4.2, 2.4.3, and 2.4.8, the staff concludes that use of a coincident wind speed as determined using the ANSI standard would not differ significantly from that approved at the CP-stage review. Therefore, the staff concludes that the provisions for flood protection from the cooling pond meet the requirements of GDC 2.

Because the portion of the dike that is constructed to el 602.5 ft MSL serves a safety function, it should be included in the inspection program described in RG 1.127.

The staff has reviewed the applicant's information and analyses under the provisions of SRP Section 2.4.8 and finds them acceptable.

2.4.3.6 Flooding Protection Requirements

Safety-related structures, systems, and components are protected from the effects of flooding. The staff concludes that potential flood levels from streams and rivers are well below plant grade level and thus are not a threat to safety-related facilities. The safety-related portion of the cooling pond dike that is built to el 602.5 ft MSL is adequate to protect the plant from wave effects originating on the cooling pond. With the exception of site drainage, the requirements of SRP Section 2.4.10 have been met and provisions for flood protection are acceptable. Review of site drainage provisions is continuing as a confirmatory issue and will be addressed in a future supplement to the SER.

2.4.4 Ice Effects

Ice flooding, which is common on the Kankakee River at the makeup intake structure, could only affect the river intake structure. This would not result in any adverse effects to the plant's safety-related water supply, which is not dependent upon Kankakee River makeup. Ice effects will not cause flood stages greater than the PMF and, thus, are not a design-basis event. The major tributary closest to the plant is the East Fork Mazon River, which lies about 1 mi southwest of the site at its closest point. Because of this distance to the site and the wide floodplain of the river, ice jams would be outflanked at stages less than the PMF and, thus, ice flooding is not a design-basis event.

Safety-related pumps, which draw water from the cooling pond, are protected from ice blockage by means of traveling screens, bar grills, and trash rakes located at the front of the pond screenhouse. Therefore, the staff concludes that Position 2 of RG 1.27 is met with respect to ice blockage of essential water intakes and that the plant design is acceptable and meets the requirements of GDC 2 to 10 CFR 50, Appendix A, and SRP Section 2.4.7 with respect to ice blockage of water intakes.

The staff concurs with the applicant that icing or ice flooding should not adversely affect the plant's safety-related facilities.

2.4.5 Cooling Water Supply

The normal source of cooling water for the plant is the 2,537-acre cooling pond. Cooling water is taken from the pond at the pond screen house by six circulating water pumps. Two 192-in. circulating water pipes carry water to the plant and back again to the pond. A buried pipeline from the plant takes blowdown to the Kankakee River. Makeup water is pumped from the river screen house on the Kankakee River through a buried pipeline to the northeast section of the cooling pond. Should makeup water be eliminated by system failure or extreme low flows, the pond can operate under a closed cycle system. Emergency shutdown water is available from the ultimate heat sink, namely the essential service cooling pond (ESCP).

The ESCP is an excavated area located within the cooling pond designed to provide sufficient volume to permit plant operation for a minimum 30-day period without requiring makeup water in accordance with RG 1.27 (Rev. 2, January 1976). The design basis of the ESCP postulates one unit undergoing a loss-of-coolant accident and the second suffering a loss of external power.

The ESCP is rectangular in shape with a length of about 2,900 ft and a width of about 1,500 ft. Assuming a loss of the main cooling pond, the normal water level in the ESCP is 590.0 ft MSL. The bottom elevation of the ESCP is 584.0 ft MSL, with an area just in front of the pond screen house excavated to about 570 ft MSL. The volume of the ESCP is about 570 acre-ft with a surface area at 590 ft MSL of about 99 acres. The three essential service water pumps are located in the basement of the auxiliary building and the three 48-in.-diameter intake lines are located at the 570-ft MSL of the pond screen house. The 48-in.-diameter discharge lines terminate in a discharge structure at the far (south) end of the ESCP.

The applicant analyzed water loss and maximum pond temperature for the case of a LOCA in one unit with concurrent shutdown of the other unit. The most severe historical meteorological conditions, in conformance with the guidance of RG 1.27, were used in the analysis. The applicant estimated the maximum water loss (resulting from evaporation and seepage) to be about 25% of the initial pond volume and the maximum intake temperature to be 94°F, which is well below the design maximum of 100°F.

The staff independently estimated that less than half of the pond volume would be lost as a result of evaporation and seepage during the worst 30-day meteorological period of record. The staff also reviewed estimates of the maximum plant intake (pond outlet) temperature. The staff compared the Braidwood emergency cooling pond with cooling ponds at other nuclear power plants which had similar heat loading, but smaller surface areas and volumes and more southerly latitudes (NUREG-0954, NUREG-0422, and NUREG-75/034). Without actually performing a mathematical calculation on the Braidwood system, it was possible to unequivocally demonstrate that the ultimate heat sink will provide cooling water at a temperature well below the maximum design-basis temperature of 100°F. The staff, therefore, concludes that the safety-related water supply for the Braidwood Station meets the suggested criteria of RG 1.27 and the hydrothermal aspects of the requirements of GDC 44. This section has been reviewed and accepted under the provisions of SRP Section 2.4.11.

2.4.6 Ground Water

2.4.6.1 Description and Users

The site area is underlain by six hydrogeologic units comprising aquifers and aquitards (confining beds). Characteristics of the units are listed in Table 2.4.

In the vicinity of the site, Quaternary Age eolian sand, lacustrine sand, and till overlie the bedrock. The eolian and lacustrine sands are predominantly fine to medium grained and form a watertable sand aquifer. Many domestic water supplies in the area are obtained from the sand aquifer with well points (shallow-driven wells). The underlying glacial drift ranges from clay to sand and gravel, but is predominantly clayey till. In places, particularly in the northern part of the area, a discontinuous outwash deposit consisting mainly of silty sand and gravel serves as an aquifer within the glacial drift.

The sand aquifer and the aquifer in the glacial drift are thin or absent in the southern part of the area and have a combined average thickness of less than 20 ft in the northern part. Analysis of boreholes on the site indicates that the thickness of the Quaternary deposits ranges from 26 to 62 ft, averaging approximately 42 ft. The saturated thickness of the sand aquifer at the site ranges from 0 to about 30 ft and averages about 14 ft. The saturated thickness

Table 2.4 Stratigraphic units and their hydrogeologic characteristics

SYSTEM	SERIES	GROUP OR FORMATION	HYDROGEOLOGIC UNIT		DESCRIPTION	HYDROGEOLOGIC CHARACTERISTICS
QUATERNARY	Pleistocene	Parkland Sand	Eolian sand	Sand Aquifer	Silty fine sand	Ground water occurs in the sand formations under water table conditions, perched on the underlying till. Ground water also occurs in the outwash layers within the till. The small thickness of the upper sand and the discontinuous nature of the outwash preclude extensive development of the sand aquifer or the aquifer within the till.
		Equality Formation	Lacustrine sand	Sand Aquifer	Fine to medium sand with trace to little silt	
		Wedron Formation	Till	Aquitard	Silty clay, clayey silt, and sandy silt with interspersed sand and gravel, some discontinuous layers of gravelly sand or sandy gravel.	
PENNSYLVANIAN	Desmoinesian	Carbondale Formation	Pennsylvanian siltstone	Aquitard	Principally siltstone, with some interbedded shale, underclay, sandstone, limestone, and coal	Ground water occurs primarily in thin sandstone beds and occasionally in joints in thin limestone beds. Ground water occurs under leaky artesian conditions. The high proportion of siltstone makes the Pennsylvanian strata generally unfavorable as an aquifer. Yields are low and are suitable only for domestic and farm purposes.
		Spoon Formation				
SILURIAN	Alexandrian	Undifferentiated	Silurian dolomites	Shallow Dolomite Aquifer	Dolomite with thin shale partings, and dolomitic siltstone	Ground water occurs primarily in joints in the dolomites and limestones under leaky artesian conditions. The shales are generally not water yielding and act as confining beds between the shallow and deep aquifers.
ORDOVICIAN	Cincinnatian	Maquoketa Shale Group	Maquoketa shale	Aquitard	Silty dolomitic shale at top, silty to pure limestone, siltstone and shale at base.	
	Champlainian	Galena Group	Galena-Platteville dolomites	Cambrian-Ordovician Aquifer	Dolomite and limestone, locally cherty, sandy at base, shale partings	Ground water occurs under leaky artesian conditions in the sandstones and in joints in the dolomites. Yields are variable and depend upon which units are open to the well.
		Platteville Group				
		Ancell Group	Glenwood-St. Peter sandstone		Sandstone, shale at top, little dolomite, locally cherty at base	
	Canadian	Prairie du Chien Group	Prairie du Chien, Eminence, Potosi and Franconia dolomites		Sandy dolomite, dolomitic sandstone, cherty at top, interbedded shale in lower part	In terms of the total yield of a well penetrating the entire thickness of the Cambrian-Ordovician Aquifer, the Glenwood-St. Peter sandstone supplies about 15 percent, the Prairie du Chien, Eminence, Potosi and Franconia dolomites collectively supply about 35 percent, and the Ironton-Galesville sandstone supplies about 50 percent.
CAMBRIAN	Croixan	Eminence Formation				
		Potosi Dolomite				
		Franconia Formation				
		Ironton Sandstone	Ironton-Galesville sandstone		Sandstone, upper part dolomite	
		Galesville Sandstone				
		Eau Claire Formation	Eau Claire shale (upper and middle beds)	Aquitard	Shales, dolomites and shaly dolomitic sandstone	Insignificant amounts of ground water may occur in joints. These beds act as a confining layer between the Cambrian-Ordovician Aquifer and the Mt. Simon Aquifer.
		Mt. Simon Sandstone	Eau Claire and Mt. Simon sandstones	Mt. Simon Aquifer	Sandstone	Ground water occurs under leaky artesian conditions. Ground water in this aquifer is too highl, mineralized for most purposes. Adequate supplies for municipal and industrial use are more easily obta from shallower aquifers

Source: Braidwood FSAR Table 2.4-20.

of the aquifer within the glacial drift ranges from 0 to 35 ft and averages only about 5 ft thick where it is present.

Ground water in the sand aquifer and the aquifer within the glacial drift occurs under water table conditions. These aquifers are recharged by precipitation. Ground water is discharged from these aquifers to surface streams and strip mine pits, to the underlying bedrock, and to pumping wells. Reported well yields are suitable only for domestic or farm purposes, ranging from 2 to 5 gpm.

The Quaternary deposits are underlain by Pennsylvanian bedrock composed of siltstone, shale, sandstone, clay, limestone, and coal. Strip mining has removed the overlying units to the bottom of a coal horizon in the mined-out areas. The Pennsylvanian strata may locally yield up to 20 gpm from interbedded sandstones, but they are essentially aquitards, as are the underlying Maquoketa shales. Silurian dolomite, which lies below the Pennsylvanian strata and forms a shallow dolomite aquifer to the northeast and east of the site, was encountered in only two site borings.

The most important aquifer in the region is the Cambrian-Ordovician Aquifer, made up of all bedrock between the shales of the Maquoketa Shale Group and the Eau Claire Formation. The Cambrian-Ordovician Aquifer is composed of the following strata, in descending order: the Ordovician Aged Galena, Platteville, Ancell (Glenwood - St. Peter Sandstone), and Prairie du Chien Groups, and the Cambrian Aged Eminence Formation, Potosi Dolomite, Franconia Formation, Ironton Sandstone, and Galesville Sandstone.

The shales of the Maquoketa Shale Group act as a confining bed between the overlying shallow dolomite aquifer, where present, and the underlying Cambrian-Ordovician Aquifer. Ground water in the Cambrian-Ordovician Aquifer occurs under artesian pressure. Available data indicate that on a regional basis, the entire sequence of strata, from the top of the Galena-Platteville dolomites to the top of the Eau Claire Shale beds, behaves hydraulically as one aquifer. In places, however, pressure heads between the water-bearing units differ, and the hydraulic connection is imperfect. The Cambrian-Ordovician Aquifer is recharged in northern Illinois.

The Eau Claire Shales separate the Cambrian-Ordovician Aquifer from the Mt. Simon Aquifer. The Mt. Simon Aquifer includes sandstones in the lower portion of the Eau Claire Formation and the Mt. Simon Sandstone. Based on available well logs, the Mt. Simon Sandstone is anticipated at a depth of about 2,400 ft below the surface at the plant site. Few wells in the regional area extend to the Mt. Simon Aquifer, because adequate ground water supplies are more easily obtained from shallower aquifers, and the ground water may be too highly mineralized for most purposes.

Permeability values for the various hydrogeologic units at the site were determined from laboratory tests on soil samples, field permeability tests conducted in the ESCP area, and water pressure tests in the bedrock.

Laboratory permeability test results show the permeability of the sand deposits to range from 3.66×10^{-4} cm/sec to 7.37×10^{-2} cm/sec. The average permeability of the till was found to be 2.6×10^{-6} cm/sec. For discontinuous, well-graded gravel and silts within the glacial drift at a depth of 35.5 to 40.5 ft the permeability was found to average 8.4×10^{-4} cm/sec.

Water pressure tests were performed in the Pennsylvanian Age Carbondale and Spoon Formations and in the underlying Brainard Shale and Fort Atkinson limestone of the Ordovician Age Maquoketa Shale Group. No water losses (indicating no or low permeability) were recorded in 20% and 50% of the tested intervals in the Carbondale and Spoon Formations, respectively, or in 40% of the tested intervals in the Maquoketa Shale Group. In those intervals in which water losses were recorded, permeabilities ranged from 1.93×10^{-6} to 4.92×10^{-4} cm/sec in the Carbondale Formation, 1.76×10^{-6} to 6.20×10^{-4} cm/sec in the Spoon Formation, and 2.33×10^{-6} to 4.58×10^{-5} cm/sec in the Maquoketa Shale Group. These permeability values probably reflect secondary permeability along infrequent joints and fractures within these formations rather than intergranular, primary permeability of the rock mass. In addition, the upper tested intervals of the boreholes generally had higher permeabilities than those at greater depths, probably reflecting the effects of weathering on the strata.

Ground water levels at the time the borings were drilled in the plant area (January 1973 to April 1973) were at approximately 595 ft.

Seepage from the sand aquifer into the power block excavation was limited by a slurry trench installed from approximately 595 ft MSL to 2 ft into the till underlying the sand aquifer. The combined quantities of seepage and precipitation were controlled using a sump pump. Eight observation wells were installed in the glacial drift around the power block excavation and outside the slurry trench in late 1975 to monitor ground water levels during construction. These observation wells were installed in pairs at varying distances away from the slurry trench. During the 3-year period 1976 through 1978, ground water levels in individual wells fluctuated from 10 to 21 ft with upper and lower maximums of 595.5 and 572 ft MSL, respectively. For approximately 77% of the measurements, ground water levels were higher in the outer observation well of each pair, indicating some decline of ground water levels immediately adjacent to the slurry trench as a result of seepage into the excavation. The average difference in ground water levels between pairs of observation wells was 0.7 ft. The slight decline in ground water levels and the small volume of seepage into the excavation indicate that ground water levels in the sand aquifer were affected only in the immediate proximity of the power block excavation.

2.4.6.2 Design Basis for Subsurface Hydrostatic Loading

The applicant's design basis ground water level for hydrostatic and combined loading is 600 ft MSL, which is ground elevation in the main plant area. The staff concurs in this level under the provisions of SRP Section 2.4.12 and it meets the requirements of GDC 2.

2.4.7 Accidental Release of Radioactive Liquid Effluent to Surface and Ground Water

The largest tanks, which are located outside the containment building and contain radioactive effluents, are the boron recycle holdup tanks. Each of these tanks has a capacity of 125,000 gal. The floor and roof elevations of the concrete cells in which these tanks are located are 546.0 and 583.0 ft MSL, respectively. The plant grade elevation is 600.0 ft.

The porous media surrounding the building are comprised of Parkland Sand and Equality Formation from 600.0 ft to 580.0 ft, underlain by about 20 ft of Wedron Formation. The average permeability of the Parkland Sand ranges from 1.2×10^{-5} to 2.4×10^{-3} ft/sec. The average permeability of the Wedron Formation is 8.53×10^{-8} ft/sec.

A cement bentonite slurry trench has been installed around the perimeter of the main plant excavation through the Parkland Sand and the Equality Formation into the silty clay glacial till of the Wedron Formation. This trench would restrict any seepage into or out of the auxiliary building.

The nearest ground water user (Well No. 73, FSAR Figure 2.4-42) is located about 1,850 ft from the auxiliary building.

The design ground water elevation at the plant site is 600.0 ft. A review of ground water withdrawal from the Parkland Sand indicates that this aquifer has been continually supplying water with no evidence of any serious depletion. Therefore, it is reasonable to assume that although the water table fluctuates within the Parkland Sand and Equality Formation it will always be above el 580.0 ft.

To examine the impact of a postulated accidental release of radioactive effluents, it is hypothesized that one of the boron recycle holdup tanks spills its contents into the concrete cell in which it is located. The walls or foundation of this cell is postulated to develop some cracks through which direct communication is established between the interior of the building and the surrounding ground water environment. The maximum elevation of the spilled fluid inside the cell is estimated to be 563.0 ft. As the ambient ground water elevation is 17 to 37 ft higher than the fluid level inside the cell, there would be no hydraulic gradient from the interior of the cell to the outside. Therefore, the effluents will be contained and prevented from moving out of the building and contaminating the surrounding ground water environment.

There are no outside non-Category I tanks that could release radioactive effluent to surface water.

On the basis of its review, using procedures described in SRP Section 2.4.13, the staff concludes that the plant meets the requirements of 10 CFR 100 with respect to potential accidental releases of radioactive effluents.

2.4.8 Technical Specifications and Emergency Operation Requirements

At this time the staff has found no reason to require any Technical Specifications or Emergency Operating Requirements as provided in SRP Section 2.4.14.

2.5 Geology and Seismology

For this SER, the staff has reviewed all available relevant geologic and seismologic information obtained since the issuance of the CP-SER (NUREG-75/023) and supplements to the CP-SER for the construction permit in 1975 in accordance with the SRP.

In the CP-SER the staff concluded that

(1) Geologic and seismologic investigations and information provided by the applicant and required by Appendix A to 10 CFR 100 provide an adequate basis for determining that no capable faults exist at the plant site or within 5 mi.

(2) Earthquakes that have occurred in the region cannot be related directly to any faults in the area.

(3) Ground motion values of 0.20g and 0.09g anchoring RG 1.60 response spectra at the foundation level of Category I structures for the SSE and the operating basis earthquake (OBE), respectively, are adequately conservative.

After careful review of the new information as provided and evaluated by the applicant, the staff concludes that there is no basis for altering its conclusions stated in the CP-SER concerning the safety of the Braidwood site.

The staff has evaluated the FSAR and subsequent documents and information, including excavation mapping, and new determinations by the Illinois Geological Survey on postulated faults. The staff has concluded that the applicant has (1) performed site and regional geologic and geophysical investigations, (2) reviewed all available pertinent literature, and (3) provided the staff with all information necessary to evaluate, assess, and support the applicant's conclusions concerning the safety of the Braidwood site from the geologic and seismologic standpoint. In addition, the staff finds the applicant has satisfied the requirements of and is in compliance with applicable portions of the following:

(1) Appendix A to 10 CFR 50

(2) Appendix A to 10 CFR 100

(3) SRP Sections 2.5.1, 2.5.2, and 2.5.3

(4) RG 1.70 "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants," Rev. 2

(5) those portions of RG 1.132, "Site Investigations for Foundations of Nuclear Power Plants," applicable to the development of geologic and seismologic information relevant to the stratigraphy, lithology, geologic history, and structural geology of the site

(6) RG 4.7, "General Site Suitability Criteria for Nuclear Power Stations"

(7) RG 1.60, "Design Response Spectra for Seismic Design of Nuclear Power Plants"

In the following sections, the staff reviews briefly the geologic and seismologic information and bases for its conclusions.

2.5.1 Geology

2.5.1.1 Summary of Regional and Site Geology

The site is located in the Kankakee Plain subsection of the Till Plains section of the Central Lowland Physiographic province. This region is characterized by undulating low relief topography of Pleistocene (1+ million-15,000 years before the present) loess, glacial drift, and residuum that overlie horizontal or gently dipping strata of the Paleozoic Age (600-250 million years before the present (mybp)). The site is underlain by a thin veneer of loess and glacial drift, ranging in thickness from 26 to 62 ft, which overlies Pennsylvanian Age (330-290 mybp) bedrock. Thickness of the Paleozoic section beneath the site is estimated to be about 5,000 ft. Beneath this lies the granitic Precambrian (800+ mybp) basement.

At the site, the glacial deposits of Wisconsinan stage drift (75,000 to 10,000 ybp) consist mainly of sand, silt, and glacial outwash.

The uppermost rock unit below the glacial and soil overburden is the Pennsylvanian Carbondale Formation. It is characterized by alternating layers of sandstone, siltstone, shale, limestone and coal, which reflect the distinctive cyclic sedimentation of the Paleozoic Illinois Basin in the Pennsylvanian. The Colchester (No. 2 coal) member of the formation, although averaging about 3 ft in thickness, is present throughout the site and served as a marker bed to ascertain that no faults with vertical offset are present at the site or in the site vicinity. The limestone near the surface is thin bedded, silty, and highly fractured, but shows no evidence of solution cavities in borings or excavations.

Below the Carbondale Formation, the Spoon Formation of primarily clastic strata with carbonaceous zones rests with sharp erosional unconformity on Silurian (435-410 mybp) thin discontinuous dolostone and dolomitic siltstone. The average thickness of the Pennsylvanian strata ranges from 15 to 220 ft. Where Silurian rocks have been encountered in borings the total thickness of the strata ranged from 17 to 25 ft.

Underlying the Silurian and Carboniferous strata, with unconformable contact, is the Ordovician succession of Maquoketa shale and subjacent formations, including the Galena and Platteville Groups that make up much of the surface bedrock of northern Illinois. The lithology of these strata indicate solid, stable bedrock below the foundation at the site.

2.5.1.2 Structure

Structurally, the site is located on the northern flank of the Illinois Basin, near the Kankakee Arch, in a region characterized by broad upwarped domes, arches and anticlines, and downwarped basins. These are all considered to be Paleozoic in age, based on stratigraphic evidence. The upwarps are commonly associated with faults or fault zones that parallel them and are related in age.

The major fault closest to the site is the northwest-trending Sandwich Fault, the southern end of which comes to about 10 mi north of the site. Detailed investigation by the Illinois Geological Survey (Illinois State Geological Survey (ISGS) Circular 505) concludes that the fault predates the Pleistocene epoch because the Illinoisan till that overlies the fault is undisturbed in the vicinity of the fault. Knowledge of the regional tectonics supports the conclusion that the Sandwich Fault is Paleozoic in age with later movement probably not after the Cretaceous period (65 mybp). As no seismicity is associated with the fault, and no evidence for surface displacement more recent than 125,000 years (the youngest age of the Illinoisan till) has been observed, the fault is considered noncapable within the meaning of Appendix A to 10 CFR 100.

Structural anomalies in the subsurface in the site region (listed in the CP-SER as postulated faults), such as the Janesville Fault in southern Wisconsin and the Oglesby and Tuscola Faults in northern Illinois, have since been reinterpreted by the Wisconsin and Illinois Geological Surveys as irregular erosion surfaces or minor flexures in subsurface bedrock. They are, therefore, not considered significant in the safety evaluation of the Braidwood Station.

A major fault in northern Illinois, not recognized at the CP stage, was investigated by the Illinois State Geological Survey. This led to the description and characterization of the Plum River Fault Zone (ISGS Circular 491), which trends east-west about 100 mi north of the Braidwood site and has been traced about 60 mi in northwest Illinois and east-central Iowa. Detailed investigation supported the conclusion that the vertical offsets of up to 400 ft were no younger than the overlying Illinoisan till which is undisturbed wherever observed along the fault. This fault, therefore, is also noncapable according to the definition in Appendix A to 10 CFR 100.

On the basis of the information provided by the applicant, the staff agrees that no surface faulting has been identified in the site and site vicinity investigations.

2.5.1.3 Coal Mining

The discovery of coal in the region led to coal mining in the late 19th and early 20th centuries. Both underground and strip-mining activities took place during this time. However, after review of many mining reports, maps and Illinois survey data, and the drilling and excavation programs, the applicant has ascertained that there are no coal mines below the site or close enough to warrant concern for possible subsidence. Coal mining in the area ended in 1974 and is unlikely to be renewed.

2.5.2 Seismology

2.5.2.1 Introduction

In its review the staff has followed the tectonic province approach to determine the vibratory ground motion corresponding to the SSE (Appendix A of 10 CFR 100). Two important considerations in this approach are the earthquakes that can be considered to be related to known tectonic structures and the random individual events that occur in the same tectonic province as the site but that cannot be related to tectonic structures. Where the occurrence of historic earthquakes can be correlated with tectonic structure, the ground motion at the site is determined assuming that the largest earthquake related to the tectonic structure is situated at the point on the structure closest to the site. Where the occurrence of the earthquake cannot be reasonably related to a tectonic structure, ground motion at the site is usually determined assuming that the largest historic earthquake in the tectonic province can occur near the site.

At the conclusion of the CP review, the staff considered an SSE of 0.20g at the bedrock-till interface to be an adequately conservative value for the Braidwood site. This was based on the assumed occurrence of a maximum Modified Mercalli intensity (MMI) VIII earthquake at the Braidwood site. Braidwood Station is located in the central stable region (CSR) tectonic province. Although the largest historical earthquake, in terms of intensity, which is not associated with tectonic structure is the 1937 Anna, Ohio, event (MMI VII-VIII), the staff's position was that the historical frequency of earthquakes in the site region, including three MMI VII events within 200 mi, is too high to consider an SSE of less than MMI VIII conservative. The staff also concluded that the maximum ground acceleration of 0.09g for the operating basis earthquake (OBE) was conservative and acceptable on the basis of the applicant's computed recurrence interval of 2,150 years for an earthquake of maximum MMI VI (CP-SER).

It is the staff's current position that the accelerations of 0.20g and 0.09g anchoring RG 1.60 spectra at the foundation level are adequate for the SSE and OBE, respectively. The staff's consultant, Lawrence Livermore National Laboratory (LLNL) is performing a probabilistic seismic hazard analysis for the Braidwood Station site. The probabilistic hazard analysis will aid the staff in assessing the conservatism of the SSE. The staff has not yet received the results of the LLNL study. It is not anticipated that the results will alter the staff's conclusion regarding the adequacy of the Braidwood Station site. This will be discussed in more detail in a supplement to the SER after the staff has received the LLNL report.

2.5.2.2 Tectonic Province

The Braidwood site lies within the CSR tectonic province described by Eardley (1962). The CSR is a region of relative consistency of surface geologic structural features characterized by a series of arches, basins, and domes formed during the Paleozoic era. King (1969) describes the area as "platform deposits on Precambrian foldbelts." The province is a rather extensive region that is, in general, characterized by a relatively low level of seismicity. However, a few areas within the province have experienced significant earthquakes and/or activity above this moderate level. Barstow et al. (NUREG/CR-1577) developed an earthquake frequency map of the Central and Eastern United States. Their work shows that the Braidwood site region has experienced up to three earthquakes per 11,680 km^2 in the period 1800 to 1977.

The staff has recognized that the surface geology of the CSR may not explain the fact that different areas of this large region exhibit different levels of seismicity. Earthquakes typically occur at depths (below ground surface) of 5 to 20 km in the Central United States; therefore, the relevant explanation of the geologic mechanism causing earthquakes is to be found in the geologic structural features at these depths rather than those at the surface. In the absence of any definite knowledge as to the causative geologic structure, levels of seismicity are an important means of assessing earthquake potential.

2.5.2.3 Maximum Earthquake

As discussed in Section 2.5.2.1 of this SER, to determine the vibratory ground motion under the tectonic province approach, the largest historical earthquakes in the site's tectonic province are considered. The largest historical earthquake, in terms of intensity, in the CSR tectonic province was the 1929 Attica, New York, event (maximum MMI VIII). This earthquake is associated with the Clarendon-Lindon structure (CP-SER Nine Mile Point Nuclear Station, Unit 2, June 1973; Erie Nuclear Plant, Units 1 and 2, NUREG-0423, July 1978). The largest historical earthquake, in terms of intensity, in the CSR tectonic province that has not been associated with tectonic structure is the 1937 Anna, Ohio, event (maximum MMI VII-VIII). As stated in the CP-SER, historically 3 earthquakes of maximum MMI VII, 6 of maximum MMI VI, 11 of maximum MMI V, and many smaller events have occurred within approximately 200 mi of the Braidwood site. The earthquake of May 26, 1909, which had an epicenter at 42.5 N, 89.0 W (U.S. Department of Commerce, 1982), probably produced the highest historical intensity (MMI VI) at the site. Generally, in the CSR tectonic province the controlling earthquake for nuclear power plant seismic design is an Anna, Ohio, type event (MMI VII-VIII). However, based on the seismicity level that was perceived to be relatively higher than other parts of the CSR tectonic province, the staff concluded, at the CP stage, that the likelihood that the site could experience MMI VII is too high for a controlling earthquake of MMI less than VIII to be considered conservative. Accordingly, the staff based the SSE for the Braidwood site on the postulated occurrence of a maximum MMI VIII near the site. The applicant, while accepting this position, maintains in Section 2.5.2.4 of the FSAR that the maximum earthquake that could be expected near the site should be MMI VII. The staff has not been made aware of any compelling information during the operating license review that would cause it to change its position as to the possible occurrence of an MMI VIII event near the site.

2.5.2.4 Safe Shutdown Earthquake

In the CP-SER the staff accepted an SSE of 0.20g to be an adequately conservative for the Braidwood site based on the postulated occurrence of a maximum MMI VIII earthquake near the site. While the seismological and geological evaluation of this controlling earthquake has not been altered since the CP review, the staff has in the interim adopted a Standard Review Plan (NUREG-0800) and Regulatory Guides that have the effect of changing the acceleration for a MMI VIII earthquake. Specifically, following the present SRP an MMI VIII earthquake is characterized by a peak acceleration of 0.25g, which is used as the high frequency anchor of an RG 1.60 spectrum. This higher reference acceleration is determined using the trend of the means relating peak acceleration to intensity shown by Trifunac and Brady (1975). The Standard Review Plan and Regulatory Guides represent one approach that the staff considers acceptable to establish conformance with NRC regulations. Another acceptable approach to establish the adequacy of the seismic design of nuclear power plants is the use of site-specific spectra (see Sequoyah SER (NUREG-0011, 1979), Watts Bar SER (NUREG-0847, 1982), and Fermi 2 SER (NUREG-0798, 1981)). To compute site-specific response spectra, it is necessary to characterize the earthquake size, the epicentral distance (distance between the surface location of the earthquake and the site), and the site conditions being modelled. There are relatively few recordings of strong ground motion at MMI VIII. This and the more dependable classification of strong motion records by magnitude has led the staff to use magnitude estimates in site-specific studies.

Nuttli and Hermann (1978) developed a relation between maximum MMI and magnitude for the Central United States. Using this relation results in an estimated magnitude of 5.75 for an MMI VIII. Nuttli and Brill (NUREG/CR-1577) estimate the magnitude of the May 26, 1909, northern Illinois earthquake (MMI VII) as 5.1. Estimates of the magnitude of the 1937 Anna, Ohio, earthquake (MMI VII-VIII) range from 5.0 to 5.3 (Nuttli and Hermann, 1978; NUREG/CR-1577).

Therefore, using the site-specific spectrum developed from magnitude 5.8 earthquakes provides a conservative estimate of the vibratory ground motion expected at the Braidwood site.

The staff has available for its use two site-specific spectra that are suitable for use in establishing the adequacy of the Braidwood seismic design for structures founded on rock. One of these was generated by the Tennessee Valley Authority for the justification of the seismic design of the Sequoyah, Watts Bar, and Bellefonte nuclear power plants (Tennessee Valley Authority, 1979) and the other was generated by LLNL for use in the NRC-sponsored seismic hazard analysis program (NUREG/CR-1582, Vol. 4).

Each of these spectra was generated from real accelerograms of earthquakes in the body wave magnitude range 5.8 ± 0.5 (5.3 to 6.3), recorded at rock sites, at epicentral distances of less than about 25 km. Using a magnitude range helps account for uncertainty in the characterization of the earthquake and also helps ensure an adequate amount of data. The distance range chosen, less than about 25 km, is the distance range to which maximum intensities are felt in the Central United States (Gupta and Nuttli, 1976). In addition, the differences in seismic wave attenuation between earthquakes east and west of the Rocky Mountains have their principal effect on ground motion at larger distances (Nuttli, 1981). It is the staff's position that the 84th percentile spectrum represents an appropriately conservative representation of the site-specific earthquake (NUREG-0011, NUREG-0847, NUREG-0798); and San Onofre Units 2 and 3 SER, 1981 (NUREG-0712)). While neither of the two site-specific spectra was established directly for the Braidwood site, they generally conform to the Braidwood site-specific spectrum criteria. Although the two site-specific spectra were developed from recordings at rock sites, it is the staff's position that they are appropriate for use at the Braidwood Station. This position is based on the following information provided by the applicant. The containment structure is founded on rock. The auxiliary/fuel-handling building complex is mostly founded on rock with only a small percentage of the foundation overlying soil. The major slabs of this structure are continuous diaphrams and connect the portions on soil to the portions on rock foundations. The soil is the Wedron Formation glacial till. The pond screen house rests entirely on Wedron Formation glacial till. The shear-wave velocity of the glacial till is reported to be 2,400 ft/sec and the shear-wave velocity of the underlying rock (Carbondale Formation) is reported to be 3,200 ft/sec. Because of the relatively low shear-wave velocity contrast between the till and the rock, no significant amplification of the rock vibratory motion through the till is expected and, therefore, the rock spectra are suitable for use at Braidwood. The staff has compared the Braidwood site SSE (RG 1.60 spectrum anchored at zero period by a peak acceleration of 0.20g) to both of these site-specific spectra and found it to be more conservative than both these site-specific spectra because it exceeds them at all frequencies.

The New Madrid earthquakes of 1811-1812 are the largest historical earthquakes in the United States east of the Rocky Mountains. Nuttli (1981) indicated that the New Madrid 1811-1812 type earthquake would have a body-wave magnitude of 7.2. The staff's position has been that the closest approach to the Braidwood site of a possible recurrence of a New Madrid-type earthquake is Vincennes, Indiana. This is about 300 km from the site. The staff has calculated the effect of a magnitude 7.2 earthquake at a distance of approximately 300 km and used the results in conjunction with the mean plus one standard deviation amplification factors from NUREG/CR-0098 to estimate a response spectrum. The Braidwood SSE response spectrum is greater than the estimated spectrum at all frequencies.

Therefore, it is the staff's position that the SSE with a high-frequency acceleration of 0.20g anchoring a RG 1.60 spectrum at the foundation level of the structures is adequate.

2.5.2.5 Operating Basis Earthquake

To justify an OBE of 0.09g, which is less than half the SSE, the applicant computed the recurrence interval for an earthquake of maximum MMI VI in the site region. The result obtained is 2,150 years. Using the trend of the means relating peak acceleration to an intensity as shown by Trifunac and Brady (1975) results in a peak acceleration of less than 0.07g for MMI VI. Therefore, the return period for a peak acceleration of 0.09g should be greater than 2,150 years. Other probabilistic estimates for this region yield return periods in the range of 200 to 1,000 years for vibratory ground motion of this size. This apparent conflict in return periods among these recurrence studies is most probably the result of the different methods and assumptions used. However, in light of the Appendix A to 10 CFR 100 definition of the OBE, these differences in estimated return period do not effect the staff's conclusion that the OBE of 0.09g is acceptable. This definition states that the OBE is "that earthquake which.... could reasonably be expected to affect the plant site during the operating life of the plant." The staff concludes that the OBE of 0.09g is an adequate estimate of the maximum earthquake motion likely to be experienced at the site during the operating life of the plant.

2.5.3 Surface Faulting

The applicant has shown through borehole data, geophysical studies and, since the CP-SER review, excavation mapping that there is no evidence of surface displacement or capable faults, within the meaning of Appendix A to 10 CFR 100, at or within 5 mi of the site. On the basis of the information provided, the staff concurs with this assessment.

2.5.4 Stability of Subsurface Materials and Foundations

2.5.4.1 Project Layout and Scope of Safety Evaluation

Braidwood Station is located in Reed Township, Will County, Illinois, approximately 22 mi south-southwest of Joliet, Illinois. The site is flat-terrain agricultural farmland, adjoining an abandoned strip mine, approximately 4 mi southwest of the Kankakee river. Figure 2.9 also shows the layout of the Braidwood Station and its cooling pond with reference to the strip mine. The hydrological description of the plant cooling water system, including cooling pond and essential service cooling pond, is in Section 2.4.1 of this SER. The natural ground surface within the Braidwood Station site ranges from el 580 to 610 ft mean sea level (MSL). Strip mining for coal has significantly altered the topography over large areas and has resulted in 100-ft-high vertical cuts in the mined area. Plant grade (i.e., finished ground surface next to the plant structures) is at el 600 ft.

Safety-related structures within the power block, the pond screen house, the essential service water discharge structure, the essential service cooling pond, and essential service water supply and discharge pipelines are classified as seismic Category I facilities that should be functional to safely shut down the plant in the event of a safe shutdown earthquake (SSE). The safety evaluation of the foundations of these is presented in the following sections.

2.5.4.2 Subsurface Conditions

2.5.4.2.1 Site Investigations

The investigations performed to determine the subsurface conditions at this site consisted of borings, test pits, and geophysical surveys.

Borings

The subsurface soil, rock, and ground water at the site were explored by drilling a total of 117 borings. The locations of these borings are shown in FSAR Figure 2.5-16. The depths of the borings ranged from a minimum of 35.5 ft to a maximum of 345.0 ft.

Undisturbed and disturbed samples of soil were obtained and the rock was cored. A graphical representation of the soils and rocks encountered in the borings, observed ground water table elevations, standard penetration test (SPT) data, and sampling and coring information are presented on the boring logs, which are in the FSAR as Figures 2.5-123 through 2.5-253. Results of water pressure tests in the bedrock are in FSAR Tables 2.5-20 through 2.5-23.

Test Pits

A total of 13 test pits were dug to perform in-place density tests and to obtain undisturbed bulk samples of coarse-grained soil for laboratory testing. The base and walls of the foundation excavations at the site were geologically mapped. These provided verification of the subsoil conditions used in the design.

Geophysical Surveys

The geophysical surveys conducted at the plant site consisted of a seismic refraction survey, surface shear-wave study, uphole velocity survey, downhill shear-wave study, ambient noise studies, and geophysical borehole logging. FSAR Section 2.5.4.4 gives the description and results of these surveys. Safety-related structures are founded on either glacial till and/or bedrock. The shear wave velocities for these foundation materials are 2,400 and 3,200 fps, respectively. FSAR Figure 2.5-63 shows the values of the compressional and shear wave velocities used in the design. Section 2.5.4.2.3 of this SER gives the stratigraphy at this site.

The field investigations performed to determine the subsurface conditions at the project site are in general accordance with the recommendations of RG 1.132. The staff finds that the field investigations performed at this site are reasonable and acceptable.

2.5.4.2.2 Laboratory Testing

The physical properties tests performed on the soil samples were in situ moisture and density determination (laboratory and field), Atterberg limits, particle size analysis, compaction, relative density, and permeability. The static strength tests performed on the soil samples were direct shear, unconfined compression (soil and rock), and triaxial compression and consolidation tests. The dynamic tests that were performed on the soil samples were cyclic triaxial compression tests and resonant column tests.

FSAR Section 2.5.4.2 presents the results of laboratory tests on soil and rock samples from the project site. The static and dynamic properties used in the soil-structure interaction analysis are in Table 2.5 of this SER.

The scope of the laboratory testing program is in accordance with the general recommendations of RG 1.138. The staff finds that the laboratory test results and the recommended design parameters are reasonable and acceptable.

2.5.4.2.3 Site Stratigraphy

Figure 2.10 of this SER shows the typical stratigraphy and description of soil and rock at this site. In descending order, the overburden soils at the project site consist of eolian, lacustrine, and glacial deposits. These three are designated as Parkland Formation sand, Equality Formation sand, and Wedron Formation glacial till. The soil thickness ranges from 26.0 to 62.0 ft with an average thickness of 38.0 ft. The soil is underlain by 2,500 to 3,000 ft of nearly horizontal bedded sedimentary rocks. The bedrock at this site is from the Pennsylvanian to the Precambrian System, in descending order. The Carbondale Formation and the Spoon Formation of the Pennsylvanian System form the upper 100 ft (±) of the bedrock and were penetrated in many borings. The Carbondale Formation consists of limestone, sandstone, siltstone, shale, and coal. FSAR Figure 2.5-19 shows a detailed geologic profile indicating the system, series, stage, group, formation, and stratigraphic classification of the bedrock at this site. The safety-related structures are founded on glacial till and/or bedrock.

The glacial till is very stiff, heavily overconsolidated, and is considered suitable to support safety-related structures. Some of the safety-related structures are founded on bedrock. The low rock quality designation (RQD) of the bedrock is attributed to the weathering of the rock, joints in the rock, and core samples parting along the bedding planes during core recovery. The degree of weathering and frequency of joints in the bedrock decreases with depth; this was substantiated by the results of water pressure tests in the bedrock. The coefficient of permeability of the bedrock (measured in field test), which is an indication of the joint openings in the rock, decreases with depth. The bedrock permeability is in the 10^{-5}cm/sec range, and with the till cover on top of the bedrock, excessive seepage of water from the cooling pond through the bedrock is not a potential problem. Moreover, no huge voids were encountered in any of the borings drilled at the project site. The unconfined

Table 2.5 Summary of static and dynamic properties of subsurface materials
SOURCE: FSAR Table 2.5-26

Property	Recompacted Sands	Wedron Till	Carbondale, Spoon	Fort Atkinson	Scales
Poisson's ratio	0.41	0.38	0.38	0.32	(0.32)*
Static modulus of elasticity, E (lb/ft^2)	0.2×10^6 to 1.0×10^6	0.9×10^6 to 5.0×10^6	0.1×10^8 to 0.5×10^8	3.5×10^8 to 7.5×10^8	1.5×10^8 to 3.5×10^8
Calculated dynamic modulus of elasticity (lb/ft^2) single-amplitude shear strain - 1.0%	5,600 $(\bar{\sigma}m)^{0.5}$**	0.3×10^8			
0.1%	36,600 $(\bar{\sigma}m)^{0.5}$	1.5×10^6			
0.01%	127,000 $(\bar{\sigma}m)^{0.5}$	2.0×10^6	0.8×10^8*** to 3.5×10^8	6.0×10^8*** to 11.0×10^8	2.0×10^8*** to 4.5×10^8
Static modulus of rigidity, G (lb/ft^2)	0.07×10^6 to 0.4×10^6	0.4×10^6 to 0.2×10^6	0.1×10^8 to 0.2×10^8	1.5×10^8 to 3.0×10^8	0.6×10^8 to 1.5×10^8
Damping factor (percent of critical damping) single-amplitude shear strain - 1.0%	26%	20%			
0.1%	17%	15%			
0.01%	6%	10%	3%**	2%**	2%**

*Values in parenthesis are estimated values.

**$\bar{\sigma}m$ is mean effective principal stress.

***These values represent the upper range of the deformation moduli and are valid for strain levels on the order of 10^{-4} to 10^{-5}%.

NOTE: Refer to FSAR Figure 2.5-19 and Figure 2.10 of this report for typical stratigraphy at the site.

Thickness, ft	Formation	Description
0-10 Avg 7 el 593.0	Parkland Formation (eolian)	Light brown to reddish-brown, very fine to fine, silty sand (SM). Avg D_R 50%.
	Equality Formation, 6ft	Yellowish-brown to buff, fine sand, some silt, slightly cemented (SP-SM). D_R 60-95%, Avg 80%.
14-31 Avg 18 el 575.0	(lacustrine) 12ft	Light gray, medium to fine sand, trace silt. Grades to coarse sand and occasional zones of gravel and rubble at the bottom (DP). D_R 60-95%, Avg 85%.
4.5-29.5 Avg 13	Wedron Formation (glacial till)	Gray to dark gray clayey silt to silty clay with interspersed sand and dolomitic gravel. Very stiff.
		Intermittent zones of grayish-brown interbedded well graded sand and gravel with cobbles and boulders. Very dense.
el 562.0		Brownish-gray to gray, very sandy silt with interspersed clay and residual soil - occasionally.
3-27 Avg 10 el 552.0	Carbondale Formation	Light gray, silty, fine to medium grained, thin bedded sandstone with occasional interbedded shale. Core recovery 8-100%, RQD 0-100%.
28-65 Avg 52 el 500.0	Carbondale Formation	Gray micaceous siltstone, sandy at top to finely micaceous silty shale at bottom. Core recovery 33-100%, RQD 5-100%.
1.5-4.7 Avg 3.3	Colchester Coal	Dull to bright black coal.
12-48 Avg 34	Spoon Formation	Interbedded clayey shales, silty shales with carbonaceous zones, and siltstones Core recovery 33-100%, RQD 5-100%.

NOTES: Avg = Average thickness; D_R = relative density; RQD = rock quality designation
Ground elevation is 600.0 ft. Design ground water table is 600.0 ft.
See FSAR Figure 2.5-19 for stratigraphy below the Spoon Formation; see Table 2.3 for the foundation elevation of Category I structures.

Figure 2.10 Typical stratigraphy at the Braidwood Station site compressive strength of the rock core samples ranged from 2,420 to 7,286 psi. After allowances are made for weathering and joints, the bedrock is also considered competent to support safety-related structures.

2.5.4.2.4 Ground Water Conditions

Ground water conditions are discussed in FSAR Section 2.4.13. A total of 15 observation wells were installed to monitor the ground water level. The ground water exists under free water table condition in the overburden soil. The ground water table varies from el 577.5 ft to el 595.0 ft at this site. The deeper bedrocks are not major aquifers. The safety-related structures are designed for a ground water level of el 600.0 ft, which is 1 ft below the plant floor grade (el 601.0 ft).

2.5.4.3 Backfill Materials

The backfill materials used at this site were lean concrete, granular material, and bash. Bash is a mixture of cement, flyash, sand, and water.

Lean concrete was used as a leveling course (mud mat) below the mat foundation, as a protective layer on the compacted structural fill or prepared base of the excavation, and as a structural fill. Concrete of 2,000-psi, 28-day strength was specified for these purposes.

Granular material recovered from the foundation excavation was used as a backfill material. The Parkland Formation sand and the Equality Formation sand, both occurring as overburden soil at the site, were used as backfill. They are medium to fine sand with 5 to 20% silt and were judged to be suitable fill material. The granular material was placed in horizontal lifts and compacted by vibratory rollers. The compaction criteria were a minimum relative density (ASTM D-2049, 1969) of 85% for fill placed beneath and immediately next to the structures and a minimum relative density of 80% for backfill placed in the remaining areas.

Bash was used to backfill the seismic Category I pipeline excavations. Section 2.5.4.4.4 of this SER presents details of placement of the bash.

2.5.4.4 Excavation and Backfill for Safety-Related Structures

2.5.4.4.1 Power Block Buildings

Excavation

The excavation for the power block foundation was approximately 750 ft by 450 ft at the plant grade level (el 600.0 ft) and, where necessary, extended down to bedrock (siltstone, el 523.0 ft). The reactor containment building, the auxiliary building, and the fuel-handling building are the seismic Category I structures in the power block. Conventional excavating equipment was used to excavate the overburden material and rippable rock. Blasting was required for excavation in competent bedrock. During construction, the ground water was controlled by a combination of a cement-bentonite slurry trench cut-off-wall around the excavation (FSAR Figure 2.5-74) and pumping out the seepage water collected in the sump pits at the bottom of the excavation.

The excavation slopes were 2 horizontal:1 vertical (2H:1V) in sands, 1H:1V in till, and 0.5H:1V in bedrock. The slopes and base of the excavation were stable during construction.

Backfill

The exposed bedrock and prepared till subgrade at the base of the excavation was covered by a 4- to 6-in.-thick mudmat to protect against possible deterioration from exposure to rain, sunshine, and freezing conditions. Lean concrete or granular backfill material was placed above the mudmat as structural fill up to the bottom of the foundation mat for all buildings of the power block complex. The elevation of the foundations of various buildings within the power block complex range from el 527.0 ft to el 579.0 ft. FSAR Table 2.5-29 and Table 2.6 of this SER give information on the foundations of the safety-related structures. The granular material was used as a backfill behind the outer walls of the power block buildings and also to fill the excavation up to the plant grade. The compaction criteria are discussed in Section 2.5.4.3 of this SER. Results of the in situ placement density tests indicate that the backfill has been placed in accordance with the compaction criteria. The average compressive strength of the lean concrete, as per quality control tests performed during construction, was 1,250 psi. The staff considers this strength adequate to fulfill its function.

2.5.4.4.2 Pond Screen House

Excavation

The excavation for the pond screen house foundation was approximately 200 by 200 ft at the top (el 600.0 ft) and extended down to the bearing stratum, till (el 565.0 ft). Conventional excavating equipment was used to excavate the overburden material. The construction dewatering system and excavation slopes were similar to those at the power block. The slopes and base of the excavation were stable during construction.

Backfill

The exposed surface of the prepared subgrade (till) was covered by a 4-in.-thick mudmat to protect against possible deterioration from ponding of water, construction activity, and exposure to weather conditions. Granular material was placed as both structural fill beneath the foundation and as backfill behind the outer walls of the pond screen house. The fill placement criteria are discussed in Section 2.5.4.3 of this SER. Results of the in situ placement density tests indicate that the backfill has been placed in accordance with the compaction criteria.

2.5.4.4.3 Essential Service Cooling Water Discharge Structure

Excavation

The excavation for the essential service cooling water (ESCW) discharge structure foundation was approximately 75 by 75 ft at the top (el 584.0 ft) and was excavated down to the foundation elevation (el 570.0 ft, or 5 ft below the top of till at this location). The excavation slopes were similar to those at the

Table 2.6 Foundation data for seismic Category I structures
Source: FSAR Table 2.5-29

Structure	Approx Plan Dimensions, ft	Foundation Elev, ft (MSL datum)	Bearing Stratum	Approx Bearing Pressure, KSF		Factor of Safety		Maximum Estimated Settlement, in.	
				Static	Dynamic	Static	Dynamic	Total	Differ-ential
Reactor containment (core)	160-ft diameter	565 (538)	Sandstone (Siltstone)	6-10	15	15	10	0.5 (0.25)	0.5 0.25
Auxiliary building	80 x 140	527 to	Siltstone	5-10	15	15	10	0.25	0.25
Fuel handling	90 x 90 90 x 120	579	Compacted fill over glacial till	4-5	8	5	2.5	0.75	0.75
Pond screen house	70 x 90	565	Glacial till	3	5	15	7.5	0.5	0.5
ESCW discharge structure	13 x 19	570	Glacial till	1-3	1-7	35	22	0.25	

power block. The ground water was controlled by pumping out the water collected in sump pits at the bottom of the excavation. The slopes and base of the excavation were stable during construction.

Backfill

Granular material recovered from the excavation was used as backfill. The backfill was compacted as described in Section 2.5.4.3 of this SER. Results of the in situ placement density tests indicate that the backfill has been placed in accordance with the compaction criteria.

2.5.4.4.4 Seismic Category I Pipelines

Excavation

The ESCW pipelines, consisting of both makeup water and discharge water pipelines, are seismic Category I. Both the makeup and discharge pipelines are in a common trench between the power block and the pond screen house. The ESCW discharge pipeline does not share its trench with any other pipeline from the lake screen house to the ESCW discharge structure. The trench for the ESCW pipelines was excavated down to the bearing stratum (till) at el 573.0 ft, using conventional excavating equipment. A soil-bentonite slurry trench cut-off-wall was installed around the pipeline trench to prevent excessive seepage into the excavation. The excavation was kept dry by draining any water that collected in the trench to a sump pit and pumping it out.

Backfill

The ESCW pipelines within the limits of the excavation of the power block complex, the pond screen house, and the ESCW discharge structure are supported by granular material. The granular material was placed in horizontal lifts and compacted using vibratory rollers or hand tampers to a minimum relative density (ASTM D-2049) of 80%. This compacted fill was placed up to the top of the pipeline excavation. In other areas (the ESCW pipelines between the limits of the power block excavation, the pond screen house excavation, and the ESCW discharge structure excavation), the pipes were supported on bash pads to facilitate welding. The entire excavation was then completely filled with bash to approximately 1 ft above the top of the pipe, thereby completely encasing the pipeline in bash. Above this level, granular material was placed and compacted to a minimum relative density of 80%. Results of in situ placement density tests indicate that the granular material has been placed in accordance with the compaction criteria. The average compressive strength of bash, as per quality control tests performed during construction, was 520 psi. The staff considers this strength adequate to fulfill its function.

2.5.4.5 Foundation Stability

2.5.4.5.1 Power Block Structures

All seismic Category I structures within the power block are supported on mat foundations bearing on either the bedrock (siltstone-sandstone) or compacted granular fill placed over till that, in turn, is underlain by the bedrock.

The ultimate bearing capacity of the foundation bedrock was evaluated on the basis of the foundation rock strength, core recovery, and RQD. The ultimate bearing capacity for the compacted granular fill and till were computed by the Terzaghi and Peck method (1967). The applicant has estimated that the factor of safety against bearing capacity failure under a postulated safe-shutdown-earthquake (SSE) event is more than 10 for structures bearing on bedrock and 2.5 for structures bearing on compacted granular fill. The staff finds the applicant's analysis and design for bearing capacity acceptable.

Because the structures are founded on either bedrock or granular material, most of the settlement is elastic and takes place during construction. The estimated settlements are minimal and will not cause any harmful effects to the structures. FSAR Table 2.5-29 and Table 2.6 of this SER present foundation data on size, elevation, bearing pressure, bearing strata, and estimated settlements for seismic Category I structures. The maximum total and differential settlements measured were 0.9 and 0.7 in., respectively. These values are close to the estimated settlements and are not detrimental to the safety of the structures. The staff agrees with the applicant's design for the foundation of the power block structures.

2.5.4.5.2 Pond Screen House

The pond screen house is supported on a mat foundation bearing on the glacial till. The ultimate bearing capacity of the till was determined using the Terzaghi and Peck method (1967). The applicant has estimated that the factor of safety against bearing capacity failure under a postulated SSE event is more than 7.

Because the structure is founded on heavily overconsolidated glacial till, most of the settlement takes place during construction. Table 2.6 presents the estimated settlements of this structure. The applicant has not previously monitored the settlement of this structure but has committed to monitor the settlement in the future. The staff requires this monitoring to be a part of the inservice inspection program of this plant. However, the applicant's estimate of the settlements are reasonable and the staff does not expect that settlement has been or will be detrimental to the function of this structure.

2.5.4.5.3 ESCW Discharge Structure

The ESCW discharge structure is founded on glacial till. The bearing pressure under the foundation is minimal (maximum 2.0 ksf), and the factor of safety against bearing capacity failure under the postulated SSE is more than 20. Settlement also is expected to be minimal (see Table 2.6). The staff concurs with the applicant's conclusions that the foundation of the ESCW discharge structure is safe against bearing failure and excessive settlement under the postulated SSE event.

2.5.4.5.4 Seismic Category I Pipelines

The seismic Category I ESCW pipelines are founded on bash over glacial till, and the pipe trenches have been backfilled with either compacted granular material or bash. In either case, there is a negligible increase in stresses in the soil beneath the pipes. Hence, the settlement of the pipeline is estimated to be minimal. The bedding and foundation of the pipeline is stable and is acceptable to the staff.

2.5.4.6 Lateral Earth Pressure

The subsurface rigid walls were designed to resist the static and dynamic lateral pressures from the surrounding earth and water. The total pressure on the walls was obtained by adding the incremental dynamic pressure to the static at-rest pressure. A coefficient of earth pressure at-rest of 0.88 for compacted granular fill was used in the analysis. Incremental dynamic earth pressure was calculated using the method of Mononobe and Okabe as modified by Seed and Whitman (1970). The procedure is in accordance with the state of the art and is acceptable to the staff.

2.5.4.7 Liquefaction Potential

Section 2.5.2 of this SER describes the design-basis earthquake (the safe shutdown earthquake), which has an effective peak acceleration level of 0.20 g at the bedrock-till interface. This is reported to result in a peak acceleration of approximately 0.26g at the ground surface. Only the fuel handling building is founded on compacted granular fill. All other seismic Category I structures are founded on either bedrock or till. Inadequately compacted granular fill below the water table is potentially susceptible to liquefaction. The granular material beneath and around the safety-related structures has been adequately compacted to a minimum relative density of 85%. The liquefaction potential of this material was evaluated using a procedure suggested by Seed and Idriss (1971). According to the analysis, both the compacted fill beneath the structures (relative density 85%) and the backfill around structures (relative density 80%) will not liquefy under the postulated SSE. The staff agrees with the applicant's conclusions that the fill beneath the structures and backfill is sufficiently resistant to liquefactions under SSE conditions.

2.5.4.8 Conclusions

On the basis of the applicant's design criteria, field investigations, field and laboratory tests, design analysis, construction reports, and applicant responses to questions, the staff has concluded that the foundations of the power block structures, pond screen house, ESCW discharge structure, and ESCW pipelines are designed to adequately support the structures and to permit a safe operation under both static and dynamic (SSE) conditions. The foundations also meet the requirements of Appendix A to 10 CFR 100, GDC 2, provisions of RG 1.132, 1.138, and SRP Section 2.5.4 (NUREG-0800) Rev. 3. The applicant has committed to monitor the settlement of the pond screen house. The staff requires the settlement monitoring to be part of the inservice inspection program.

2.5.5 Stability of Slopes

2.5.5.1 Scope of Safety Evaluation

Figure 2.9 shows the layout of the essential service cooling pond (ESCP) and the normal cooling pond, and its dikes. The ESCP is part of the seismic Category I cooling water system and is the ultimate heat sink for the plant. If the normal cooling pond drains as a result of dike failure, the ESCP should retain enough water for safe shutdown of the power plant. The slopes of the ESCP should be sufficiently stable under a design-basis SSE so that the design capacity of the ESCP is maintained. The evaluation of the stability of the ESCP slopes is presented in this section. The normal cooling pond dikes and the interior dikes are not part of the essential system needed for safe shutdown of the power plant, and they are not considered safety-related structures. There are no other natural or constructed slopes in the immediate vicinity of the power plant whose failures would adversely affect the safety of the plant.

2.5.5.2 ESCP Characteristics

The normal cooling pond is surrounded by a system of dikes and is excavated to el 590.0 ft. The ESCP is a 6-ft-deep pond, excavated to a bottom elevation of 584.0 ft, within the cooling pond. The cut slopes of the ESCP are 10H:1V, with the top of the slope being the bottom of the cooling pond (el 590.0 ft). The ESCP is approximately rectangular, with a surface area of approximately 99 acres and a capacity of 576 acre-ft. The ESCP cooling water intake structure is located at the northwest corner of the ESCP and the discharge facility for essential service water is at the southern end of the ESCP.

2.5.5.3 ESCP Subsurface Conditions

Sections 2.5.4.2.1 and 2.5.4.2.3 of this SER present details of the site investigations performed and the soil stratigraphy at the project site. The stratigraphy at the cooling pond consists of lacustrine sand overlying glacial till that, in turn, is underlain by sedimentary bedrock. FSAR Sections 2.5.4.2 and 2.5.6.5 present the results of the laboratory strength tests performed on these materials. Table 2.7 of this SER presents the soil parameters used in evaluating the stability of the ESCP slopes.

Table 2.7 Soil parameters for slope stability analysis
Source: FSAR Figure 2.5-97

Parameter	Sand	Glacial Till
Wet density (lb/ft^3)	125	140
Cohesion (lb/ft^2)	0	190
Angle of internal friction (degrees)	34	30

The applicant's definition of the soil stratigraphy and soil parameters for the ESCP slopes is reasonable and acceptable to the staff.

2.5.5.4 ESCP Stability Analysis

2.5.5.4.1 Static Stability

The stability of the excavated slopes of the ESCP was investigated for static conditions. FSAR Figure 2.5-97 presents the typical cross-section of the slope

(10H:1V) analyzed for stability, and Table 2.7 of this SER gives the soil parameters used in the stability analyses. The stability was investigated by the Bishop slip circle method of analysis, using the SLOPE computer program. The conditions for which the stability of the slope was computed and the corresponding minimum factors of safety are in Table 2.8 of this SER.

Table 2.8 Factors of safety from stability analysis (ESCP slope 10H:1V)

Loading Condition	Minimum Factor of Safety
End of construction: no water in pond	5.9
Normal operating conditions: ESCP water el 595.0 ft	7.0
Rapid drawdown, ESCP water drawdown from el 595.0 ft to el 590.0 ft	7.0

The design parameters are reasonable, and the method of analyses represents accepted practice. The staff agrees with the applicant's conclusions that the ESCP slopes are stable under static conditions.

2.5.5.4.2 Dynamic Stability

The dynamic stability of the ESCP slope was evaluated by a pseudostatic analysis (Bishop slip circle method) in which seismic coefficients of 0.2 and 0.26 were used. The effective acceleration of the design-basis SSE is 0.20g at the bedrock-till interface; this results in a calculated acceleration of 0.26g at plant grade (el 600.0 ft). Because the toe of the ESCP slope is at el 584.0 ft, the seismic coefficient of 0.26 is judged to be a sufficiently conservative coefficient for the ESCP slope. The factors of safety are in Table 2.9 of this SER. The staff has concluded that the slopes would be stable after an SSE.

The applicant has not evaluated the dynamic stability of the ESCP slope by the finite element method because (1) the slope is very flat and shallow (10H:1V and 6-ft-high slope), (2) the factor of safety for the static condition is very high (7.0), and (3) the factor of safety from the pseudostatic analysis (1.3) is generally considered to be acceptable. On the basis of the above, the applicant has concluded that the ESCP slope is stable under both static and dynamic conditions.

The staff agrees with the applicant's analysis and conclusion that the ESCP slopes will be stable under the effects of the SSE.

2.5.5.4.3 Liquefaction Potential

The liquefaction potential of the sand in the ESCP bottom and cooling pond bottom was evaluated by calculating a factor of safety defined as the ratio of the

Table 2.9 Factors of safety from pseudostatic analysis (ESCP Slope 10H:1V)

Loading Condition	Factor of Safety: Seismic Coefficient 0.20	Factor of Safety: Seismic Coefficient 0.26
Normal operating conditions: ESCP water el 595.0 ft	1.3	1.1
Rapid drawdown: ESCP water drawdown from el 595.0 ft to el 590.0 ft	1.3	1.1

shear stress required to cause liquefaction to the shear stress induced by the SSE. The shear stresses induced by the SSE were computed using the SHAKE computer code and the resulting stress distribution induced in 10 cycles was determined. The cyclic stress ratios required to cause initial liquefaction, ± 5% axial strain and ± 10% axial strain were calculated using data from laboratory cyclic strength tests on both reconstituted and intact specimens. These were modified by correction factors to account for (1) the difference in stress conditions between the field and laboratory and (2) effect of specimen reconstitution on the fabric of the intact sand. The factors of safety against initial liquefaction, ± 5% strain and ± 10% strain conditions were determined for two cases (1) level ground at el 590.0 ft to represent the cooling pond bottom and (2) level ground at el 584.0 ft to represent the ESCP bottom. For the case of level ground at el 590.0 ft, the lowest factors of safety were 1.17 for initial liquefaction, 1.59 for ± 5% strain condition, and 2.06 for ± 10% strain condition. For the case of level ground at el 584.0 ft the corresponding lowest factors of safety were 1.06, 1.45, and 1.86 respectively. FSAR Section 2.5.6.5.2 presents the applicant's analysis in detail. The applicant has concluded that the sand in the ESCP is not susceptible to liquefaction.

During the construction permit stage, the staff and its consultant, the U.S. Army Corps of Engineers, reviewed the applicant's analysis for liquefaction potential at the ESCP; the CP findings are reported in the CP-SER (NUREG-75/023). Although the staff agreed with the procedure, the staff has reservations about the magnitude of the correction factors developed by the applicant because these were based on extrapolation of limited test data. The NRC analyzed the same data using conservative assumptions such as a lower correction factor and cyclic stress ratios corresponding to a relative density lower than the average relative density determined for the ESCP sand. The staff finds the site to be marginally safe with respect to initial liquefaction. In the unlikely event that the site should experience an SSE, the sand in ESCP slope might experience excessive deformation. As a consequence of this, the slope could experience a flow-type failure, resulting in a configuration of 20H:1V. The applicant has demonstrated that even with a 20H:1V slope configuration, the ESCP will maintain its design capacity and the material from the slope will not be displaced or moved far enough to result in blockage of either the circulating water intake opening or the ESCP discharge structure opening. Hence, even in the event of a slope failure, the ESCP will fulfill its safety function.

The interior dike, located west and south of the ESCP, is not a seismic Category I structure (see Figure 2.9). The dike is 10 ft high, with slopes of 3H:1V, and the slopes are protected with riprap. The toe of the dike slope is approximately 80 ft from the top of the ESCP slopes. The applicant has demonstrated that, if this dike fails, the material from the dike may be displaced up to 30 ft from the toe of the slope and will still be approximately 50 ft from the top of the ESCP slope. The displaced material from the dike will not be a critical surcharge at the top of the ESCP slope and will not affect the stability of the ESCP slope. Hence, the failure of the interior dike will not have any effect on the ESCP, and the ESCP will still be able to fulfill its safety function. To ensure that these slopes are maintained, RG 1.127 describes acceptable inspection of water-control structures associated with nuclear power plants. As part of the inspection, the applicant will monitor the floor and slopes of the ESCP. The staff will review the details of the inspection program which will be proposed by the applicant as part of the plant Technical Specifications.

On the basis of the staff's evaluation of the applicant's analysis and a conservative liquefaction analysis by the staff, the staff concludes that the ESCP will continue to fulfill its safety function during an emergency shutdown following an SSE.

2.5.5.5 Seepage From the ESCP

As described above, the ESCP is a 6-ft-deep pond dug in the bottom of the cooling pond; it holds 576 acre-ft of water, with an approximate surface area of 99 acres. The ESCP is the ultimate heat sink (UHS) for this plant and is designed to provide water to cool essential service equipment for a 30-day period following a design-basis SSE.

The ESCP is cut into a medium to fine silty sand stratum, underlain by till that in turn, overlies bedrock. Field permeability tests performed in borings indicate that both till and bedrock are relatively impermeable (permeability, 10^{-5} cm/sec range) compared to the sand stratum. The average coefficient of permeability of the sand, as determined from laboratory tests, was 6.7×10^{-3} cm/sec. The water from the ESCP can seep through the sand stratum if a hydraulic gradient to cause the flow exists. It is postulated that in the event of an SSE, the cooling pond dike will be breached and the pond drained. The applicant has computed the seepage loss from the ESCP assuming that the water level in the ESCP is at el 590.0 ft and the water table outside the ESCP is at el 580.0 ft. The volume of seepage loss was estimated using the SEEPAGE computer code. The seepage for 30 days would result in an estimated drawdown of 1.5 ft in the ESCP. The staff performed a simplified analysis and verified the applicant's estimate of the volume of seepage. For the critical combination of temperature, humidity and other weather conditions, it is estimated that the loss of water as a result of evaporation during the plant cooling for 30 days will result in a 1-ft drawdown of the water level in the ESCP. This results in a total drawdown of 2.5 ft in the ESCP, where the ESCP is 6 ft deep. Hence, the capacity of the ESCP is at least twice the volume of water needed to cool the plant for 30 days.

This analysis was conservative because it did not consider the presence of the soil-bentonite seepage barrier installed for the full length of the perimeter dike around the cooling pond. Even if the perimeter dike fails at any critical section, the seepage barrier will be present at other locations and will retard seepage through or beneath the perimeter dike. After the cooling pond is drained as a result of dike failure, the pond bottom will still be saturated and it will take a long time before an effective differential head between the ESCP and the cooling pond bottom is created. Hence, the 10-ft differential head assumed to occur immediately after an SSE is very conservative. The staff has reviewed the applicant's seepage analysis and considers it conservative and acceptable.

On the basis of the above, the staff concludes that the ESCP will retain enough water to fulfill its safety function following an SSE and the design is in compliance with the geotechnical aspects of RG 1.27.

2.5.5.6 Conclusions

On the basis of the applicant's design criteria, field investigations, field and laboratory tests, and design analyses, the staff concludes that the geotechnical design of the essential services cooling pond is adequate and that it will fulfill its safety function following a design-basis SSE. The ESCP meets the requirements of Appendix A to 10 CFR 50 and geotechnical provisions of RG 1.27. The staff will review the geotechnical aspects of the inservice inspection program which will be prepared by the applicant as part of the plant Technical Specifications following RG 1.127.

2.5.6 Embankments and Dams

In the immediate vicinity of this site, there are no embankments or dams whose failure would effect the safety of this nuclear power plant.

APPENDIX 2A

NEARBY INDUSTRIAL, TRANSPORTATION, AND MILITARY FACILITIES (BRAIDWOOD)*

The applicant states that there are no known industrial establishments within the 1-1/8 mi low population zone. Industries within ten miles are mostly located in Coal City, 4-1/2 miles northwest and produce metal products, lubricants, clothing, furniture and aerosol products. The Peabody Coal Company is located in South Wilmington about five miles from the site. There are no airports within six miles. The nearest airport with a paved runway is located at Morris, Illinois, 17 miles from the plant. The nearest pipeline is a six-inch natural gas pipeline two miles west. These activities are sufficiently distant from the site so as to not pose a hazard to safe operation of the Braidwood plant.

The Joliet Arsenal is the nearest military facility, and is located eight miles northeast of the site. In the conduct of its operations, the arsenal ships and receives high explosive material on Illinois State Highway Routes 53 and 129 and on the Illinois Central Gulf Railroad which border the northwest portion of the Braidwood exclusion area. We have been advised by the Illinois Central Gulf Railroad Company that no other hazardous materials have been recently transported on the track in question.

We have evaluated the frequency and quantity of high explosive shipments on the railroad, which passes within 1700 feet of the nearest reactor containment structure. We were informed by the Military Traffic Management and Terminal Service that quantities of up to 1,100,000 pounds of high explosive may pass the site. The maximum shipment would consist of seven boxcars containing 132,000 pounds of flake TNT each, and 15 boxcars of 105 millimeter shells. In addition, smaller quantities of Composition "B" Explosive, which is a mixture of 40% TNT and 60% RDX, are routinely shipped to the arsenal. We requested the applicant to submit appropriate analyses to demonstrate that this traffic would not adversely affect any plant safety-related structure, nor result in such damage as to prevent a safe and orderly shutdown of the reactors.

The applicant submitted an analysis of the railroad explosion hazard. This analysis includes a reconstruction of U.S. explosives traffic over the past 60 years, a survey of actual shipments of TNT from the Joliet Arsenal during the first six months of 1974, an estimation of the explosive accident rate

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*From "Partial Safety Evaluation Report on Site Characteristics," by the Directorate of Licensing U.S. Atomic Energy Commission, related to the application of Commonwealth Edison Company for permits to construct Byron Station, Units 1 and 2, and Braidwood Station, Units 1 and 2 (Docket Nos. STN 50-454, STN 50-455, Docket Nos. STN 50-456, STN 50-457), Section 2.2, "<u>Nearby Industrial, Transportation, and Military Facilities</u>."

experienced by military munitions traffic, and a detailed discussion of the comparative insensitivity of flake TNT to inadvertent detonation. The applicant concludes that the probability of unacceptable damage (more than one boxcar of TNT exploding) to the Braidwood plant as a result of accidental explosions of railroad munitions traffic, as estimated by various methods, is between 8.3×10^{-10} and 4.5×10^{-8} per year during peacetime. There is no evidence of proposed increases in the rate at which Joliet Army Ammunition Plant produces munitions, although such increases could not be precluded in the event of war, increased arms exports, or the closure of other competing plants. Based upon maximum production rates during war years, exposure to the plant could be increased by as much as a factor of five during the affected period.

We concur with the applicant's reconstruction of past U.S. munitions traffic, and the rate of overall accidental munitions railway explosions estimated by the applicant. We consider the applicant's six-month survey to be representative of Joliet Arsenal traffic during the years of comparative peace, 1971-1974 and agree with the applicant's estimate of the exposure to the Braidwood site, under current traffic conditions, from munitions trains containing two or more boxcars of flake TNT.

We were unable to find any record of the accidental detonation of a flake TNT shipment while in transit by railroad since 1916, although Department of Transportation records report several railroad accidents in which boxcars of flake TNT were burned or otherwise damaged without explosion. In the opinion of our consultants at the Naval Ordnance Laboratory, it is extremely unlikely that flake TNT would detonate under conditions credibly expected in a railroad accident near the Braidwood site.

On the basis of available accident reports and our consultants' opinion, we conclude that the applicant's estimate that the accidental explosion rate of flake TNT is half that of total military munitions is conservative. Using the same basic probabilistic model as the applicant, but without credit for the ratio of TNT traffic to total munitions traffic (0.135) which we consider to be not valid, our results are a factor of seven higher than the applicant's.

We compute 3×10^{-7} per year as the probability of an accident resulting in explosion of two or more boxcars for the current level of operations at the arsenal. Our independent evaluation indicates that explosions of quantities of up to one boxcar load (132,000 pounds) yield peak reflected overpressures less than the applicant's stated tornado wind pressure of 3.3 psi at 1700 feet, if the cargo is assumed to be TNT, and thus are within the design bases established by the applicant for safety-related structures.

The above computed probability, however, does not include certain qualitative considerations which further reduce the probability of an accident on that section of track adjacent to the site. The railroad track which passes the Braidwood site is part of the Amtrak passenger train system, and is maintained at a quality higher than the average railroad track in the U.S. The track is level and there are no curves, road crossings, switching yards, or sidings near the plant. While no quantitative value can be assigned to these considerations, we believe that these factors significantly reduce the likelihood of a train accident near the proposed Braidwood plant.

We have informed the applicant that we will require suitable arrangements to provide for the inspection and maintenance procedures necessary to assure that the section of track near the plant is maintained as high quality track according to the track safety standards of the Federal Railroad Administration. The applicant has agreed to take whatever action is necessary for plant safety if maintenance of that section of track falls below its present level. On the basis of the above, we conclude that the probability of an accidental explosion of two or more boxcars adjacent to the Braidwood site is of the order of 10^{-7} per year.

In view of the low probability of the accidental explosion discussed above, we conclude that special design provisions to protect the Braidwood plant against the explosion of two or more boxcars are not required, and that the Braidwood plant can be operated with an acceptable degree of safety with regard to potential accidents resulting from activities at nearby industrial, military, and transportation facilities.

3 DESIGN CRITERIA FOR STRUCTURES, SYSTEMS, AND COMPONENTS

This section is the same as NUREG-0876 except for the following:

3.4 Water Level (Flood) Design

3.4.1 Flood Protection

The design of the facility for flood protection was reviewed in accordance with SRP Section 3.4.1 (NUREG-0800). An audit review of each of the areas listed in the "Areas of Review" portion of the SRP section was performed according to the guidelines provided in the "Review Procedures" portion of the SRP section. Conformance with the acceptance criteria formed the basis for the staff's evaluation of the design of the facility for flood protection with respect to applicable regulations of 10 CFR 50.

To ensure conformance with the requirements of GDC 2, with respect to protection against flooding, the staff reviewed the overall plant flood protection design including all systems and components whose failure resulting from flooding could prevent safe shutdown of the plant or result in the uncontrolled release of significant radioactivity. The applicant has provided protection from inundation and the static and dynamic effects for safety-related structures, systems, and components by the "Dry Site" method as defined in RG 1.102, "Flood Protection for Nuclear Power Plants," Position C.1, as described below.

All safety-related equipment is flood protected by virtue of its location above the probable maximum flood (PMF) elevation and by being housed within flood-protected structures. The PMF level has been determined to be 598.17 ft above mean sea level, in accordance with the guidelines of RG 1.59, "Design Basis Floods for Nuclear Power Plants," Positions C.1 and C.2. Refer to Section 2.4 of this SER for further discussion on the PMF level. Plant grade is at el 601 ft. Access openings to structures containing safety-related equipment are at or above plant grade. All piping penetrating the exterior walls of these structures below grade are provided with watertight penetration sleeves, and water stops are provided in horizontal and vertical construction joints in all exterior walls as protection against ground water entry.

Within plant structures, safety-related equipment is protected against flooding from failures in tanks, vessels, and fluid piping systems as identified in the guidelines of BTP ASB 3-1, "Protection Against Postulated Piping Failures in Fluid Systems Outside Containment," by equipment location and drainage as described under Section 9.3.3 of this SER.

On the basis of the staff's review of the design criteria and bases and safety classification of safety-related systems, structures, and components necessary for a safe plant shutdown during and following flood conditions, the staff concludes that the design of the facility for flood protection conforms to the requirements of GDC 2 with respect to protection against natural phenomena and conforms to the guidelines of RG 1.59 (Positions C.1 and C.2) and RG 1.102

(Position C.1) concerning flood protection pending resolution of the confirmatory issue described in Section 2.4.3.3 of this SER.

3.5 Missile Protection

3.5.2 Structures, Systems, and Components To Be Protected From Externally Generated Missiles

The design of the facility for providing protection from tornado generated missiles was reviewed in accordance with SRP Section 3.5.2 (NUREG-0800). An audit review of each of the areas listed in the "Areas of Review" portion of the SRP section was performed according to the guidelines provided in the "Review Procedures" portion of the SRP section. Conformance with the acceptance criteria formed the basis for the staff's evaluation of the design of the facility for providing protection from tornado generated missiles with respect to the applicable regulations of 10 CFR 50.

GDC 2 requires that all structures, systems, and components essential to safety be protected against natural phenomena; and GDC 4 requires that these same structures, systems, and components be protected from the effects of externally generated missiles. The natural phenomena generating external missiles of concern are tornadoes. The tornado missile spectrum is discussed in Section 3.5.1.4 of this SER. The applicant has identified all safety-related structures, systems, and components requiring protection from externally generated missiles. All safety-related structures (including the containment, auxiliary building, fuel-handling building, and main steam safety valve rooms) are designed to withstand postulated tornado-generated missiles without damage to safety-related equipment. With the exception of the diesel generator exhaust stacks, and main steam safety and relief valve exhaust stacks, all safety-related systems and components (including outside air intakes and exhausts in safety-related structures) and stored fuel are (1) located within tornado-missile-protected structures, or (2) are provided with tornado-missile barriers or other protection (such as burial underground), or (3) are oriented so that tornado missiles do not present a safety hazard. Protection for the diesel generator exhaust stacks and main steam safety and relief valve exhaust stacks is discussed further in this SER section. Thus, the plant design satisfies the guidelines of RG 1.117, Positions C.1, C.2, and C.3. Compliance with the guidelines of RG 1.115 is discussed in Section 3.5.1.3 of this SER.

The station diesel generator exhaust stacks are protected on one side by the turbine building outside wall but are exposed on the other side to tornado missiles. Tornado-generated missiles could collapse or crimp and block the diesel engine exhaust stacks and cause failure of the diesel generator to run. The applicant has committed to provide tornado missile protection for the diesel exhaust stack by installation of a tornado-missile-proof exhaust pressure relief system that will open to ensure proper diesel engine operation in the event the exhaust stack should become blocked. The staff concurs with the applicant's approach and considers this method of tornado missile protection acceptable.

The main steam safety and relief valve discharge exhaust stacks, which are exposed for a short distance above the valve house roof, are of sufficient wall thickness to prevent collapse or significant crimping of the stacks as a result of tornado missile impact.

In addition, although the fuel-handling building is designed to be tornado missile resistant, the rollup freight door (which is a large opening in the building) is not capable of resisting tornado missile impact. A tornado or tornado missile could destroy the door and may allow a relatively lighweight missile of large area, such as a steel panel or the door itself, to travel inside the fuel-handling building. However, the spent fuel pool is sufficiently far from the door that any resulting tornado missiles and debris could not enter the spent fuel pool and cause damage to the spent fuel assemblies or block coolant flow because of the low trajectory a missile would have to follow through the door and toward the fuel pool and because it must then turn 90° to enter the pool. Therefore, unacceptable radiological release is prevented; thus, the guidelines of RG 1.13, Position C.2, are satisfied.

The ultimate heat sink at Braidwood is a passive cooling pond and, therefore, physical tornado missile protection is not required. Thus, the guidelines of RG 1.27, Positions C.2 and C.3, are satisfied.

Based on the above, the staff concludes that applicant's list of safety-related structures, systems, and components to be protected from externally generated missiles and the provisions in the plant design providing this protection is in accordance with the requirements of GDC 4 with respect to missile effects and the guidelines of RG 1.13 (Position C.2), RG 1.27 (Positions C.2 and C.3), and RG 1.117 (Positions C.1, C.2, and C.3) concerning protection of spent fuel, the ultimate heat sink, and other safety-related plant features from tornado missiles and is, therefore, acceptable. The tornado missile protection design meets the acceptance criteria of SRP Section 3.5.2.

3.5.3 Barrier Design Procedures

The plant Category I structures, systems, and components are shielded from, or designed for, various postulated missiles. Missiles considered in the design of structures include tornado-generated missiles and various containment internal missiles, such as those associated with a loss-of-coolant accident (LOCA).

Information has been provided indicating that the procedures that were used in the design of the structure, shields, and barriers to resist the effect of missiles are adequate. The analysis of structures, shields, and barriers to determine the effects of missile impact was accomplished in two steps. In the first step, the potential damage that could be done by the missile in the immediate vicinity of impact was investigated. This was accomplished by estimating the depth of penetration of the missile into the impacted structure. Secondary missiles are prevented by fixing the target thickness well above that determined for penetration. In the second step of the analysis, the overall structural response of the target when impacted by a missile is determined using established methods of impact analysis. The equivalent loads of missile impact, whether the missile is environmentally generated or accidentally generated within the plant, are combined with other applicable loads as is discussed in Section 3.8 of this SER.

The applicant estimated the potential hazard to the Braidwood Station from missiles and pressure load generated in a postulated accidental explosion of one boxcar of TNT on the railroad located 1,550 ft from the station.

The spectrum of missiles and the pressure thus generated have been reviewed by the staff and are addressed in Section 2.2.2 of this SER. The applicant provided results at several locations whereby shears and moments resulting from the blast loads were compared with the corresponding shears and moments capacities. Out of the five results thus compared, all of the sections remain elastic in shear and one wall was inelastic for moment with ductility of 1.5, which is within the acceptable limits of 3.0, as specified in the RG 1.42, Rev. 1, for a localized area of a structure. The staff reviewed the methodology of analysis for dynamic pressure loading and found that it is acceptable.

At the audit meeting held October 20-23, 1981, at Sargent & Lundy offices, the staff asked that the applicant demonstrate that the Category I manholes are adequately protected against impact of tornado missiles.

By letter of March 22, 1982, the applicant informed the staff that Category I manhole covers have been designed in accordance with SRP Section 3.5.3 for tornado missiles. Ductile iron with 100 ksi strength has been provided for these manhole covers. The staff considers this to be acceptable.

The staff concludes that the barrier design is acceptable and meets the requirements of GDC 2 and 4 with respect to the capabilities of the structures, shields, and barriers to provide sufficient protection for equipment that must withstand the effects of natural phenomena (tornado missiles) and environmental effects, including the effects of missiles, pipe whipping, and discharging fluids. This conclusion is based on the following.

The procedures used to determine the effects and loadings on seismic Category I structures and missile shields and barriers induced by design-basis missiles selected for the plant are acceptable because these procedures provide a conservative basis for engineering design to ensure that the structures or barriers are adequately resistant to and will withstand the effects of such forces.

The use of these procedures provides reasonable assurance that if design-basis missiles strike seismic Category I structures or other missile shields and barriers, the structural integrity of the structures, shields, and barriers will not be impaired or degraded to an extent that will result in a loss of required protection. Seismic Category I systems and components protected by these structures are, therefore, adequately protected against the effects of missiles and will perform their intended safety function, if needed. Conformance with these procedures is an acceptable basis for satisfying, in part, the requirements of GDC 2 and 4.

3.7 Seismic Design

3.7.1 Seismic Design Parameters

The input seismic design response spectra for the operating basis earthquake (OBE) and safe shutdown earthquake (SSE) are defined at the ground surface. These spectra comply with RG 1.60. The design time history is obtained using the following two-step procedure:

(1) The north-south and vertical components of the 1940 El Centro earthquake records are modified so that the response spectra generated using these

synthetic records matched closely with the RG 1.60 response spectra for horizontal and vertical directions.

(2) The design time history obtained in Step (1) is applied at the ground surface of a one-dimensional shear layer system extending to the bedrock, and the time histories at the foundation and bedrock level are computed using the wave propagation theory.

In the process of developing the response spectra in Step (2), the response spectra at the foundation level have displayed a significant dip over a large range of frequencies. This reduction of motion at the foundation level is not acceptable to the staff. Therefore, the staff asked the applicant to perform the analysis based on the RG 1.60 free-field surface design response spectra applied at the foundation level and the response spectra resulting from synthetic time history should envelope the RG 1.60 design spectra at the foundation level.

As a result of a series of meetings between the applicant and the staff, an agreement was reached and the criteria for re-evaluation of the Braidwood plant were developed. The staff reviewed the applicant's submittal (FSAR Amendment 36, January 1982) of results of the reanalysis and concluded that it is acceptable. The reassessment was based on the design response spectra for the Marble Hill nuclear plant, which were developed in accordance with the provisions of RG 1.60. Because of the similarity between the Marble Hill and Braidwood plants, the response spectra developed for the Marble Hill plant could be used for re-evaluation of the Braidwood plant. Although there are some features unique to Marble Hill, the effect of these features is negligible for the purpose of seismic analysis. The reassessment was made using the average actual material strength and included the combined effect of the loss-of-coolant accident (LOCA) and SSE load combinations. Reassessment was made for OBE loads on a few randomly selected structural elements that were reassessed earlier for increased SSE loads. Marble Hill used a zero period acceleration of 0.08 OBE and Braidwood used 0.09 OBE; therefore, the Marble Hill loads were factored by 0.09/0.08 to determine the Braidwood OBE loads.

On the basis of the applicant's re-evaluation as outlined above, the staff concludes that the seismic design basis used in the Braidwood plant ensures that the integrity and functionality of safety-related structures is maintained.

The containment foundation is supported on rock. The auxiliary/fuel-handling building complex is founded partly on bedrock and partly on soil. To account for the soil beneath the foundation slab, the structure was modeled by introducing shear springs and associated mass representing the soil between the slab and bedrock. The soil-structure model was analyzed using the foundation spectra at the bedrock.

The pond screen house rests on 10 ft of hard glacial till. The applicant provided the staff with information, which indicates that the properties of the till and of the underlying rock are similar so that there will be no appreciable amplification of seismic motion between the rock and the top of the till in the critical frequency range of 1 to 20 Hz.

The essential service water discharge structure (ESWDS) is founded on approximately 22 ft of glacial till. Because of the simplicity of the structure (it is a simple block of concrete, 13 ft by 19 ft by 10 ft high embedding two pipes, 4 ft in diameter) relatively simple analysis was performed to compute the seismic forces acting on the structure. The soil was represented in the analysis model by elastic half-space springs using the appropriate soil properties, and the rigid structure was represented by a rigid mass. The wide band surface response spectra corresponding to RG 1.60 were used as the input motion.

The information provided by the applicant, pertinent to the soil condition and seismic environment, has been evaluated by the staff and is described in more detail in Sections 2.5.2 and 2.5.4 of this SER.

The staff concludes that the seismic design parameters used in the plant structure design are acceptable and meet the requirements of GDC 2 and Appendix A to 10 CFR 100. This conclusion is based on the following.

The applicant has met the relevant requirements of GDC 2 and Appendix A to 10 CFR 100 by appropriate consideration for the most severe earthquake recorded for the site with an appropriate margin and considerations for two levels of earthquakes (SSE and OBE). The applicant has met these requirements by the use of the methods and procedures indicated below.

The seismic design response spectra (OBE and SSE) applied in the design of seismic Category I structures, systems, and components comply with the recommendations of RG 1.60. The specific percentage of critical damping values used in the seismic analysis of Category I plant structures, systems, and components is in conformance with RG 1.61. The artificial synthetic time history used for the seismic design of Category I plant structures, systems, and components is adjusted in amplitude and frequency content to obtain response spectra that envelop the design response spectra specified for the site. Conformance with the recommendations of RGs 1.60 and 1.61 ensures that the seismic inputs to Category I structures, systems, and components are adequately defined for an earthquake whose zero period accelerations are 0.09 g for the OBE and 0.2 g for the SSE so as to form a conservative basis for the design of such structures, systems, and components to withstand seismic loadings.

3.7.2 Seismic Structural System and Subsystem Analysis

The scope of review of the Braidwood seismic system and subsystem analysis for the plant included the seismic analysis methods for all Category I structures, systems, and components. It included review of procedures for modeling, seismic soil-structure interaction, development of floor response spectra, inclusion of torsional effects, evaluation of Category I structure overturning, and determination of composite damping. The review included design criteria and procedures for evaluation of interaction of non-Category I structures and piping with Category I structures and piping and effects of parameter variations on floor response spectra. The review also included criteria and seismic analysis procedures for Category I buried piping outside containment.

The system and subsystem analyses were performed by the applicant on an elastic basis. Modal response spectrum and time history methods form the bases for the analyses of all major Category I structures, systems, and components. When the modal response spectrum method was used, governing response parameters were combined by the double-sum method. The square root of the sum of the squares of the maximum codirectional responses was used in accounting for three components of the earthquake motion for both the time history and response spectrum methods. Floor spectra inputs used for design and test verifications of structures, systems, and components were generated from the time history method, taking into account variation of parameters by peak widening. A vertical seismic system dynamic analysis is employed for all structures, systems, and components, where analyses show significant structural amplification in the vertical direction. Torsional effects and stability against overturning are considered.

The applicant reported that an average torsional moment of 8% of the maximum building dimension times the story shear resulted from the seismic analysis of plant complex structures. It is the staff's position that to account for accidental torsion, an additional eccentricity of 5% of the maximum building dimension at the level under consideration should be assumed. The applicant claimed that it is not appropriate to include arbitrary torsion in the complex (auxiliary, fuel-handling, and turbine buildings). The applicant justified his position as follows:

(1) It is understood that the staff position is intended for isolated structures.

(2) The mathematical model includes the entire power block--namely, turbine building, auxiliary building, and fuel-handling building--and because of the configuration of the plant, implementation of additional torsion would result in excessive shear forces on the walls.

(3) The mathematical model includes the permanent equipment, and the addition of 5% of torsional eccentricity is not necessary because mass distribution has been accounted for.

The staff considered these arguments and concluded that the position on accidental torsion should not be applied to Braidwood Station.

The applicant, at the audit meeting, indicated that the cable tray and support system are considered as electrical items and only qualified for the SSE. The staff requested that the applicant demonstrate that the cable tray supports can withstand the loads associated with the OBE. In resolution of this audit action item, the applicant submitted computer results of two representative cable tray supports using OBE response spectra. The results showed that the supports have been designed within design-basis allowables.

The staff also requested that the applicant provide it with the description of the method and summary of the analysis of the heating, ventilating, and air conditioning (HVAC) hangers. The applicant reported that as a result of reevaluation of the plant (see SER Section 3.7), 7 of the 347 systems examined may be expected to have inelastic behavior. Considering that the resulting ductility will be relatively low (less than 3), the staff considers that these systems are satisfactory.

The staff concludes that the plant design is acceptable and meets the requirements of GDC 2 and Appendix A to 10 CFR 100 with respect to the capability of the structures to withstand the effects of the earthquake so that their design reflects

(1) appropriate consideration for the most severe earthquake recorded for the site with an appropriate margin (GDC 2); consideration of the two levels of earthquakes (Appendix A, 10 CFR 100)

(2) appropriate combination of the effects of normal and accident conditions with the effect of the natural phenomena

(3) the importance of the safety functions to be performed (GDC 2); the use of a suitable dynamic analysis or a suitable qualification test to demonstrate that structures, systems, and components can withstand the seismic and other concurrent loads, except where it can be demonstrated that the use of an equivalent static load method provides adequate consideration (Appendix A, 10 CFR 100).

The applicant has met the requirements of Item (1) by use of the acceptable seismic design parameters, as per SRP Section 3.7.1. The combination of loads resulting from an earthquake with those resulting from normal and accident conditions in the design of Category I structures as specified in SRP Sections 3.8.1 through 3.8.5 will be in conformance with Item (2).

The staff concludes that the use of the seismic structural analysis procedures and criteria delineated above by the applicant provides an acceptable basis for the seismic design which is in conformance with the requirements of Item (3).

3.9 Mechanical Systems and Components

3.9.3 ASME Code Class 1, 2, and 3 Components, Component Supports, and Core Support Structures

3.9.3.2 Pump and Valve Operability Assurance

This section is the same as NUREG-0876 except that the applicant must identify all equipment and corresponding installation features for Braidwood Station, Units 1 and 2, that are not the same for Byron Unit 1. The applicant must submit complete qualification information for the site-specific equipment identified for Braidwood Station, Units 1 and 2. The applicant also must indicate whether the seismic input for all safety-related equipment for Braidwood Station, Units 1 and 2, is completely enveloped by the seismic input for the corresponding equipment in Byron Station, Unit 1. The status of the applicant's program will be reported in a future supplement to this SER.

3.10 Seismic and Dynamic Qualification of Mechanical and Electrical Equipment

This section is the same as NUREG-0876 except that the applicant must identify all equipment and corresponding installation features for Braidwood Station, Units 1 and 2, that are not the same for Byron Unit 1. The applicant must submit complete qualification information for the site-specific equipment identified for Braidwood Station, Units 1 and 2. The applicant also must indicate whether the seismic input for all safety-related equipment for Braidwood Station, Units 1 and 2, is completely enveloped by the seismic input for the corresponding equipment in Byron Station, Unit 1. The status of the applicant's program will be reported in a future supplement to this SER.

3.11 Environmental Qualification of Safety-Related Electrical Equipment

This section is the same as NUREG-0876 except that the applicant must identify all equipment for Braidwood Station, Units 1 and 2, that is not the same for Byron Unit 1, and/or is located in areas where environmental conditions are more severe than the demonstrated values for Byron Station Unit 1. The applicant must submit complete qualification information for the equipment identified or justification for interim operation before an operating license is issued for Braidwood Station, Units 1 and 2. The status of the applicant's program will be reported in a future supplement to this SER.

4 REACTOR

This section is the same as NUREG-0876.

5 REACTOR COOLANT SYSTEM

This section is the same as NUREG-0876 except for the following:

5.2 Integrity of Reactor Coolant Pressure Boundary

5.2.4 Reactor Coolant Pressure Boundary Inservice Inspection and Testing

5.2.4.1 Compliance with the SRP

This section is the same as NUREG-0876 with the exception of the reference to the January 5, 1982, public meeting. The staff will require that the applicant commit to use at Braidwood similar augmented procedures that exceed the minimum ASME Code requirements that were used at Byron Unit 1.

5.2.4.3 Evaluation of Compliance with 10 CFR 50.55a(g) for Braidwood Unit 1

This section is the same as NUREG-0876 with the exception of the reference to the January 5, 1982, public meeting and the preservice examinations, which have not begun at Braidwood.

The staff considers the review of the preservice inspection program (PSI) to be a confirmatory issue based on the staff review of the Byron PSI program, which was determined to be acceptable, and contingent upon the applicant's committing to

(1) docketing an acceptable PSI program

(2) submitting all relief requests with supporting technical justifications

(3) submitting conclusions regarding the ability to examine the cast stainless steel pipe elbows

The staff will complete its evaluation of the Braidwood Unit 1 PSI program in a supplement to the SER after the applicant provides an acceptable response.

The initial inservice inspection (ISI) program has not been submitted by the applicant. The staff will evaluate the program after the applicable ASME Code Edition and Addenda can be determined based on 10 CFR 50.55a(b), but before the first refueling outage when ISI commences.

5.3 Reactor Vessel

5.3.1 Reactor Vessel Materials and RCPB Materials

5.3.1.2 Evaluation of Compliance to 10 CFR 50, Appendix H

On the basis of its review of the applicant's submittal that described the extent of compliance of Braidwood with Appendix H, 10 CFR 50, the staff has determined that

(1) Braidwood 1 has met all the requirements of Appendix H, but has not indicated the schedule for removal of reactor vessel beltline surveillance capsules.

(2) Braidwood 2 has met all the requirements of Appendix H, but has not indicated the schedule and the materials in the reactor vessel beltline surveillance program.

The applicant indicated that the reactor vessel beltline surveillance program will comply with ASTM E-185 and Appendix H, 10 CFR 50. The applicant also indicated that he will provide the information identified above at a later date. The staff will review this report during its review of the applicant's Technical Specifications to confirm that the surveillance program complies with ASTM E-185 and Appendix H, 10 CFR 50.

5.3.3 Reactor Vessel Integrity

The completed review of reactor vessel materials for Byron is applicable to Braidwood. The Braidwood review will be completed once the staff has received the plant-specific data concerning reactor vessel materials for Braidwood Units 1 and 2.

5.3.4 Pressurized Thermal Shock

Pressurized thermal shock (PTS), as a consequence of certain postulated accident scenarios, is of concern primarily for vessels that have experienced significant degradation of material properties because of irradiation damage in the beltline region. The staff's Unresolved Safety Issue (USI) A-49 will eventually address this issue for all pressurized water reactor (PWR) facilities, but initially USI A-49 is concerned primarily with operating facilities.

As part of its review of USI A-49, the staff issued Commission Report SECY 82-465, "Pressurized Thermal Shock." In this report the staff concluded that the risk from PTS events for reactor vessels with RT_{NDT} values less than the proposed screening criteria (270°F for axial welds and 300°F for circumferential welds) is acceptable.

An increase in RT_{NDT} of a material is a measure of the amount of radiation damage to the material. The amount of nickel and copper in a material and its accumulated neutron fluence affect the amount of increase in a material's RT_{NDT}. The increase in a material's RT_{NDT} may be predicted by the "Guthrie Formula," which is identified in Appendix E to Commission Report SECY 82-465.

The predicted RT_{NDT}, using the Guthrie Formula, for the Braidwood reactor vessels is 111°F. This was calculated for the limiting Braidwood Units 1 and 2 beltline reactor vessel material--upper to lower shell circumferential weld WF-562, which had 0.04% copper, 0.67% nickel, and an initial RT_{NDT} of +40°F. The peak end-of-life fluence at the inside wall is predicted by the applicant to be 2.4×10^{19} n/cm^2 (E>1MeV).

The predicted end-of-life value includes the Guthrie Formula's two sigma value (upper 95% probability limit) of 48°F. Hence, the value at end-of-life is considered conservative.

The staff believes that PTS will probably not pose a threat to the Braidwood reactor vessels for 32 effective-full-power years because the predicted end-of-life RT_{NDT} (111°F) is substantially below the PTS screening criteria (300°F). However, the staff is continuing to study this issue as USI A-49 and, if necessary, may re-evaluate this conclusion within the next few years.

5.4 Component and Subsystem Design

5.4.3 Residual Heat Removal System

This section is the same as NUREG-0876, except it should be noted that if the Diablo Canyon natural circulation tests are not completed or do not provide satisfactory results, the applicant has committed to perform such tests at Byron Station before startup after the first refueling outage.

6 ENGINEERED SAFETY FEATURES

This section is the same as NUREG-0876 except for the following:

6.2 Containment Systems

6.2.7 Fracture Prevention of Containment Pressure Boundary

The completed review of the reactor containment pressure boundary components for Byron Station is applicable to Braidwood. The Braidwood review will be completed pending receipt of plant-specific certified material test data for the components identified in the Byron review.

6.6 Inservice Inspection of Class 2 and 3 Components

6.6.1 Compliance With the Standard Review Plan

This section is the same as NUREG-0876 with the exception of the reference to the January 5, 1982, public meeting. The staff will require that the applicant commit to use similar augmented procedures at Braidwood that exceed the minimum ASME Code requirements that were used at Byron Unit 1.

6.6.3 Evaluation of Compliance of Braidwood Unit 1 With 10 CFR 50.55a(g)

This section is the same as NUREG-0876 with the exception of the reference to the January 5, 1982, public meeting and the preservice examinations that have not begun at Braidwood.

The staff considers the review of the preservice inspection program (PSI) to be a confirmatory issue based on the staff review of the Byron PSI program, which was determined to be acceptable, and contingent upon the applicant committing to (1) docketing an acceptable PSI program and (2) submitting all relief requests with supporting technical justifications.

The staff will complete its evaluation of the Braidwood Unit 1 PSI program in a supplement to the SER after the applicant provides an acceptable response.

The initial inservice inspection (ISI) program has not been submitted by the applicant. The staff will evaluate the program after the applicable ASME Code Edition and Addenda can be determined based on 10 CFR 50.55a(b), but before the first refueling outage when ISI commences.

7 INSTRUMENTATION AND CONTROL

This section is the same as NUREG-0876, except in Section 7.1.3 it should be noted that the findings from the site visit to Byron Station Unit 1 and the results of the subsequent review of the physical arrangement and installation of the instrumentation and controls equipment are applicable to Braidwood Station, Units 1 and 2, because of the duplicate design configuration. Also, in Section 7.3.1.2.7 it should be noted that the Braidwood Station essential service water system takes suction from an essential service water cooling pond whereas Byron uses cooling tower basins.

8 ELECTRIC POWER SYSTEMS

This section is the same as NUREG-0876, except in Section 8.4.4 it should be noted that a site visit to Braidwood Station will be required to verify the implementation of the applicant's design criteria regarding physical identification, separation, and independence of redundant safety-related electrical systems.

9 AUXILIARY SYSTEMS

This section is the same as NUREG-0876 except for the following:

9.2 Water Systems

9.2.1 Station Service Water Systems

The essential and nonessential service water systems were reviewed in accordance with SRP Section 9.2.1 (NUREG-0800). An audit review of each of the areas listed in the "Areas of Review" portion of the SRP section was performed according to the guidelines provided in the "Review Procedures" portion of the SRP section. Conformance with the acceptance criteria formed the basis for the staff's evaluation of the essential service water system with respect to the applicable regulations of 10 CFR 50.

Both an essential and a nonessential service water system are provided for cooling various plant equipment. The nonessential (nonseismic Category I, Quality Group D) service water system serves only nonsafety-related components and performs no safety function. The staff's review verified that its failure would have no adverse effects on safety-related systems as it is separated from essential components. The system can be shared between Units 1 and 2.

The essential service water system supplies cooling water to safety-related equipment from the essential cooling pond (the ultimate heat sink discussed in Section 9.2.5 of this SER) and returns the water to the pond for dissipation of heat to the atmosphere. The essential service water system provides cooling for the component cooling heat exchangers, diesel generator coolers, containment fan coolers, auxiliary feedwater pump oil cooler, auxiliary building chillers, centrifugal charging pump room coolers, containment spray pump room coolers, safety injection pump room coolers, residual heat removal pump room coolers, auxiliary feedwater pump room cooler, and control room air conditioning units. The system operates in normal and emergency situations. The essential service water system operating pressure is lower than that for the component cooling water system and other potentially radioactive systems. Radiation monitors are provided in the system to detect potential inleakage of radioactivity.

The essential service water systems consist of two redundant, independent, full-capacity piping trains per unit, each of which serves redundant essential components and is capable of providing 100% of the required cooling function in all operating modes assuming a single failure. Each train contains one full-capacity pump that is powered from a separate emergency (Class 1E) power source. Both system trains are automatically started on receipt of a safeguards actuation signal. The trains are cross-connected with each unit and between units through redundant isolation valves powered from separate emergency (Class 1E) power supplies. Essential service water pump suction for both units is supplied through two separate buried headers from the pond screen house forebay on the cooling pond. These lines are fed by six supply lines (three for each unit) located in the basemat of the pond screen house. Warm essential service water is returned through two separate buried lines to the discharge structure on the opposite side of the essential cooling pond from the pond screen house. Thus, the requirements of GDC 5, "Sharing of Structures, Systems and Components," and GDC 44, "Cooling Water," are met.

The essential service water system is seismic Category I, Quality Group C. The system is housed in the seismic Category I, flood- and tornado-protected reactor building, auxiliary building, and discharge structure (refer to Sections 3.4.1 and 3.5.2 of this SER). In addition, a small portion of the essential service water supply piping protrudes above the pond screen house basemat floor. The pond screen house basemat is designed not to fail in a safe shutdown earthquake (SSE). Failure of nonseismic Category I structures and equipment within the pond screen house will not affect this portion of the piping by blocking it as it is separated from the nonseismic features within the pond screen house. The piping between the essential cooling pond and the auxiliary building is buried underground and within the turbine building basemat to provide tornado missile protection. The turbine building basemat is designed not to fail in an SSE. Therefore, the integrity of the essential service water system piping is maintained. Thus, the requirements of GDC 2, "Design Basis for Protection Against Natural Phenomena," and the guidelines of RG 1.29, Positions C.1 and C.2, are met.

During normal plant operation, one essential service water system pump is operating. Availability of the remaining pumps is ensured by periodic functional tests and inspections as delineated in plant Technical Specifications. The system components are located in accessible areas to permit inservice inspection as required. Thus, the requirements of GDC 45, "Inspection of Cooling Water System," and GDC 46, "Testing of Cooling Water System," are met.

On the basis of the above, the staff concludes that the essential service water system meets the requirements of GDC 2, 5, 44, 45, and 46 with repect to the system's protection against natural phenomena, shared systems, decay heat removal capability, inservice inspection, and functional testing, and the guidelines of RG 1.29 with respect to the system's seismic classification and is, therefore, acceptable. The essential service water system meets the acceptance criteria of SRP Section 9.2.1.

9.2.3 Demineralized Water Makeup System

The demineralized water makeup system was reviewed in accordance with SRP Section 9.2.3 (NUREG-0800). An audit review of each of the areas listed in the "Areas of Review" portion of the SRP section was performed according to the guidelines provided in the "Review Procedures" portion of the SRP section. Conformance with the acceptance criteria formed the basis for the staff's evaluation of the demineralized water makeup system with respect to the applicable regulations of 10 CFR 50.

The nonsafety-related (Quality Group D, nonseismic Category I) demineralized water makeup system provides treated and demineralized water to various plant systems and components that include: condensate makeup (to the condenser hotwell and condensate storage tank), auxiliary steam boiler makeup, primary and secondary process sampling makeup, chemical feed and handling makeup, waste disposal system, reactor coolant makeup, radwaste station makeup, boric acid processing, component cooling water makeup, chemical and volume control, and boron thermal regeneration. The Kankakee River or the Braidwood cooling pond provide the source of water to the demineralized water makeup system.

The system has no safety-related functions. Adequate isolation is provided at all demineralized water makeup connections to safety-related systems. Protection from flooding for safety-related equipment resulting from failure of the system is discussed in Section 9.3.3 of this SER. The system is capable of fulfilling the normal operating requirements of the facility for acceptable makeup water with the necessary component redundancy. At each point of discharge from the system, check valves prevent contamination of the makeup demineralizer system by backflow from the systems that it supplies. Alarmed instrumentation has been provided to prevent delivery of off-specification water to safety-related systems. Failure of the system does not affect the capability to safely shut down the plant as described above; thus, the requirements of GDC 2, "Design Basis for Protection Against Natural Phenomena," and 5, "Sharing of Structures, Systems, and Components," and the guidelines of RG 1.29, Position C.2, are met.

On the basis of its review, the staff concludes that the demineralized water makeup system meets the requirements of GDC 2 and 5 with respect to the need for protection against natural phenomena and shared systems as its failure does not affect safety system functions and meets the guidance of RG 1.29 concerning its seismic classification and is, therefore, acceptable. The demineralized water makeup system meets the acceptance criteria of SRP Section 9.2.3.

9.2.4 Potable and Sanitary Water System

The potable and sanitary water system was reviewed in accordance with SRP Section 9.2.4 (NUREG-0800). An audit review of each of the areas listed in the "Areas of Review" portion of the SRP section was performed according to the guidelines provided in the "Review Procedures" portion of the SRP section. Conformance with the acceptance criteria formed the basis for the staff's evaluation of the potable and sanitary water system with respect to the applicable regulations of 10 CFR 50.

The nonsafety-related (Quality Group D, nonseismic Category I) potable and sanitary water system provides clean water for drinking and sanitary purposes and include all components and piping from the potable supply connection from the Kankakee River or the Braidwood cooling pond to points of discharge.

There are no cross connections between the potable and sanitary water system and potentially radioactive systems, and, therefore, inadvertent contamination is prevented. Protection from flooding for safety-related equipment resulting from failure of the system is discussed in Section 9.3.3 of this SER.

On the basis of its review, the staff concludes that potable and sanitary water system meets the requirements of GDC 60 with respect to prevention of release of potentially radioactive water and is, therefore, acceptable. The potable and sanitary water system meets the acceptance criteria of SRP Section 9.2.4.

9.2.5 Ultimate Heat Sink

The ultimate heat sink (UHS) was reviewed in accordance with SRP Section 9.2.5 (NUREG-0800). An audit review of each of the areas listed in the "Areas of Review" portion of the SRP section was performed according to the guidelines provided in the "Review Procedures" portion of the SRP section. Conformance with the acceptance criteria formed the basis for the staff's evaluation of the UHS with respect to the applicable regulations of 10 CFR 50.

The UHS provides heat dissipation capability for the reactors and their essential auxiliaries through the essential service water system during normal operating and accident conditions. Heat dissipation is by evaporation to the atmosphere of warm water returned to the essential cooling pond via the essential service water system (refer to Section 9.2.1 of this SER).

The UHS is shared by both Braidwood Units 1 and 2 and consists of an excavated essential cooling pond integral with the main cooling pond, the essential portions of the pond screen house, and the essential service water discharge structure. The six essential service water supply lines are located in the basemat floor of the pond screen house (intake structure) on the cooling pond to utilize the entire depth of the essential cooling pond. Warm essential service water is returned from the plant by two lines to the essential service water discharge structure located on the opposite side of the cooling pond from the pond screen house, thereby providing maximum utilization of the essential cooling pond area for heat dissipation and preventing warm water recirculation to the supply lines. The above described design meets the criteria of RG 1.27, Position C.3, regarding single failures.

The UHS provides the capability to dissipate 100% of the heat load under accident conditions (including LOCA) in one unit and safe shutdown of the other assuming the highest historical ambient temperatures for a 30-day period without the need for makeup. To demonstrate this capability, the applicant has used BTP ASB 9-2, "Residual Decay Energy for Light-Water Reactors for Long-Term Cooling," to establish the heat input to the UHS resulting from fission product and heavy element decay. The applicant performed a heat transfer analysis assuming conservative worst site meteorology to verify the performance capability of the UHS. The analysis assumed the essential cooling pond level was at its lowest initial level, the point where communication with the main cooling pond is lost (the level following total loss of the main cooling pond retaining dikes). This analysis verified that the UHS is capable of providing sufficient cooling for normal shutdown of one unit and accident conditions in the other unit and maintaining the essential service water system temperature at the design maximum. For long-term UHS operation (beyond the 30-day design), temporary makeup provisions can be provided if the normal makeup from the circulating water makeup pumps at the river screen house is not available. The design described above ensures that adequate heat removal capability to maintain plant safety is provided by the UHS for its design modes of operation, including accidents coincident with a single active failure. Thus, the requirements of GDC 5 and 44, and the guidelines of RG 1.27, Positions C.1, C.3, and C.4, regarding the UHS ability to maintain proper system temperature under all design modes of operation are met. Because the UHS contains no active components, the requirements of GDC 45 and 46 are not applicable.

The excavated depression that forms the essential cooling pond within the main cooling pond is designed to maintain its integrity following an SSE. Failure of the nonseismic Category I main cooling pond retaining dikes does not result in loss of the minimum level assumed for the UHS design-basis heat removal capability discussed above. The essential service water discharge structure and essential portions of the pond screen house that house the essential service water supply lines are seismic Category I. Failure of nonseismic Category I portions of the pond screen house will not affect the essential service water supply piping (refer to Section 9.2.1 of this SER for further detail).

Because of the nature of the essential cooling pond as described above, tornado missile protection is not necessary. Further, the essential service water supply lines within the lake screen house and the essential service water discharge structure are protected from tornado missiles by the structures themselves.

The pond screen house and essential service water discharge structure are protected against the effects of the probable maximum flood (PMF) and corresponding wind wave activity as discussed in Section 2.4 of this SER. Thus, the requirements of GDC 2, and the guidelines of RG 1.27, Position C.2, regarding UHS protection against natural phenomena, and RG 1.29 are met.

On the basis of the above, the staff concludes that the UHS meets the requirements of GDC 2, 5, and 44 with respect to protection against natural phenomena, shared systems, and decay heat removal capability and meets the guidelines of RGs 1.27 and 1.29 with respect to design capability and seismic classification and is, therefore, acceptable. The UHS meets the acceptance criteria of SRP Section 9.2.5.

9.4 Heating, Ventilation, and Air Conditioning Systems

9.4.6 Pump House Ventilation System

This section of NUREG-0876 is not applicable to Braidwood Station.

9.5 Other Auxiliary Systems

9.5.1 Fire Protection Program

9.5.1.4 General Plant Guidelines

This section is the same as NUREG-0876 except it should be noted that the fire protection water supply system consists of two fire pumps separately connected to an underground 14-in. water main loop around the plant. The fire pumps take suction from the forebay of the pond screen house. The source of water is much larger than the 300,000 gal recommended by the guidelines; therefore, the staff finds this acceptable.

10 STEAM AND POWER CONVERSION SYSTEM

This section is the same as NUREG-0876 except for the following:

10.2 Turbine Generator

10.2.1 Turbine Disk Integrity

This issue is addressed in Section 3.5.1.3 of this SER.

10.4 Other Features of Steam and Power Conversion System

10.4.5 Circulating Water System

The circulating water system (CWS) was reviewed in accordance with SRP Section 10.4.5 (NUREG-0800). An audit review of each of the areas listed in the "Areas of Review" portion of the SRP section was performed according to the guidelines provided in the "Review Procedures" portion of the SRP section. Conformance with the acceptance criteria formed the basis for the staff's evaluation of the circulating water system with respect to the applicable regulations of 10 CFR 50.

The nonsafety-related (Quality Group D, nonseismic Category I) circulating system supplies cooling water to the main condenser of each unit. This water is circulated to the normal heat sink (main cooling pond) where heat is rejected to the environment. Circulating water from the cooling pond is directed back to the main condensers by three circulating water pumps for each unit, which are located in the pond screen house. Makeup to the cooling pond is provided by the three nonsafety-related circulating water makeup pumps, which are located in the nonsafety-related river screen house on the Kankakee River. The circulating water system is not required to maintain the reactor in a safe shutdown condition or mitigate the consequences of accidents.

The applicant has provided the results of an analysis of the effects of possible flooding of safety-related equipment as a result of a postulated failure in the circulating water system. The circulating water system has the potential for flooding the turbine building to a high elevation. This is due to the large volume contained in the main cooling pond and the elevation and location of a portion of the circulating water system within the turbine building. No safety-related equipment is located in the turbine building and no below-grade openings from the turbine building connect directly into the auxiliary building, which houses safety-related equipment. Once the water level reaches grade elevation, the water flows into the yard away from plant structures, except for small amounts that may seep under closed doors. The applicant indicates, however, that the main steam tunnel that connects directly to the turbine building will be flooded. Flooding of the auxiliary feedwater tunnel is prevented by watertight closures on the openings to the main steam tunnel. In examining the consequences of flooding in the main steam tunnel, the applicant indicated that the main steam isolation valves (MSIVs) will eventually be flooded and become inoperable (fail as is, in the open position). The applicant has determined that sufficient downstream isolation valves on the main steamlines and main steam branch lines are provided to prevent steam blowdown in excess of the capability of the auxiliary feedwater system should the operator fail to close the MSIVs before they are flooded. None of these valves are affected by the turbine building flooding. Thus, the requirements of GDC 4 are met with respect to ensuring a safe plant shutdown in the event of flooding resulting from a circulating water system failure.

On the basis of its review, the staff concludes that the circulating water system meets the requirements of GDC 4 with respect to protection against environmental effects (flooding) on safety-related equipment as a result of a failure (pipe breaks) in the system and is, therefore, acceptable. The circulating water system meets the acceptance criteria of SRP Section 10.4.5.

11 RADIOACTIVE WASTE MANAGEMENT

This section is the same as NUREG-0876.

12 RADIATION PROTECTION

This section is the same as NUREG-0876 except for the following:

12.5 Operational Radiation Protection Program

The staff has audited the organization, equipment, instrumentation, facilities, and procedures for radiation protection contained in the Braidwood Station FSAR against the criteria of SRP Section 12.5 (NUREG-0800). The plant's health physics program objectives are to provide reasonable assurance that the limits of 10 CFR 20 are not exceeded, to further reduce unavoidable exposures, and to ensure that individual and total person-rem occupational radiation doses are maintained as low as is reasonably achievable (ALARA). The staff review consisted of ensuring that the applicant had either committed to following the criteria of the regulatory guides and staff positions referenced in SRP Section 12.5 (NUREG-0800) or provided acceptable alternatives and selectively compared the applicant's FSAR against the acceptance criteria of the Standard Review Plan using the review procedures identified in NUREG-0800. This selective review found the plant acceptable in these areas. Details of the review follow.

12.5.1 Organization

The Station Health Physicist is the Radiation Protection Manager (RPM) at Braidwood Station and is responsible for implementing and enforcing the plant's health physics program. However, the ultimate responsibility of the health physics program lies with the Station Superintendent. The Station Health Physicist is a member of the station's ALARA Review Committee. The Radiation/Chemistry Supervisor, a staff Health Physicist, or a Radiation Chemistry Foreman will act as the backup to the RPM if the RPM is absent from the station.

The Braidwood Station Radiation Protection Organization has been evaluated in accordance with the criteria in NUREG-0731, "Guidelines for Utility Management Structure and Technical Resources," and RG 8.8 (Section C.1.b(2),(3)).

The paragraphs below present an evaluation of how the health physics organization for Braidwood Station compares with the various staff positions concerning plant organization and management criteria.

(1) The organization description for Braidwood shows that the Station Health Physicist (RPM) reports directly to the Radiation/Chemistry Supervisor and has direct access to the Station Superintendent in all radiation protection matters. In addition the RPM has access to other station supervisors through daily contact at morning meetings. This satisfies the criteria of RG 8.8 and is acceptable.

(2) The radiation protection section and chemistry section are combined as one section at Braidwood Station. Health physics appraisal findings from other Commonwealth Edison Company plants having similar health physics/chemistry structures had shown that weaknesses do exist in this type of

joint organization. To resolve these weaknesses and ensure proper operation of the radiation protection organization, the applicant has committed to improving his radiation/chemistry technician training program to ensure that radiation/chemistry technicans maintain adequate qualification in both chemistry and health physics.

In addition, the applicant is reorganizing the station Radiation-Chemistry Department to include

(a) direct supervision of the Radiation/Chemistry Foreman by the lead professionals in the areas of health physics and chemistry

(b) round-the-clock health physics supervision by the Health Physics Foreman to direct the activities of the radiation/chemistry technicians during each shift

(c) laboratory supervision by a dedicated foreman, on the day shift, Monday through Friday

(d) adequate staff to divest the professionals and foreman of clerical activities such as scheduling and record keeping

The staff finds the applicant's proposed organization of the Radiation/Chemistry Department acceptable.

(3) The staff has evaluated the qualifications of the Braidwood Station Health Physicist (RPM) against the requirements of RG 1.8, which requires the RPM to have (1) a bachelor's degree or the equivalent in a science or engineering subject and (2) 5 years of applied radiation protection experience, including 3 years' experience dealing with radiological problems similar to those encountered in a nuclear power station. The applicant's RPM currently has (1) a Bachelor of Science in Health Physics, (2) 2-1/4 years' experience at an operating station, and (3) 1 year's experience as the RPM at the Braidwood Station. In addition the applicant has committed to a training program for the Station Health Physicist to upgrade his qualifications to meet RG 1.8 requirements before the fuel loading date. The staff has reviewed the applicant's proposed training program for the RPM and finds it acceptable. Subject to the Station Health Physicist completing the proposed training program, the staff finds the proposed Braidwood RPM qualifications acceptable. This is a confirmatory item.

(4) The backup to the RPM during his absence from the station will be selected from the positions of Radiation/Chemistry Supervisor, staff Health Physicist, or Radiation/Chemistry Foreman. The applicant has committed to using the criteria of American National Standards Institute (ANSI) 3.1, December 1979 draft, in selecting the individual temporarily filling the RPM position. This satisfies the criteria of NUREG-0731 and is acceptable.

(5) The applicant has committed to having at least one Radiation/Chemistry technician on site at all times. This satisfies the criteria of NUREG-0731 and is acceptable.

The applicant has shown that the current health physics organization for Braidwood Station meets staff criteria as stated in NUREG-0731 and RG 8.8 for an acceptable radiation protection organization.

13 CONDUCT OF OPERATIONS

This section is the same as NUREG-0876 except for the following:

13.1 Organizational Structure

FSAR Amendments 36 (January 1982) and 39 (September 1982) state that the organizational arrangement of Commonwealth Edison Company is included in Topical Report CE-1A, Rev. 15, "Quality Assurance Program for Nuclear Generating Stations." During its review, the staff found, however, that the then current version of CE-1A, which was Rev. 23 (March 1983), included organizational arrangements that were different from those discussed in the currently effective Chapter 13 of the FSAR. This applies to both the management and technical support organization and to the plant operating organization.

Furthermore, the plant organization shown in FSAR Chapter 16, Section 6, "Administrative Controls," is different from that presented in FSAR Chapter 13 and from that presented in CE-1A, Rev. 23.

In a letter dated September 30, 1983, the applicant stated that the current version of CE-1A is Rev. 25, dated September 9, 1983. When the staff has completed its review of Rev. 25, it will report its findings in a supplement to this SER.

The staff has not completed its review of the qualification of the shift personnel with respect to previous commercial PWR operating experience. When this review is completed, the staff will report its findings in a supplement to this SER.

13.3 Emergency Planning

This review will be completed pending submittal of an onsite and offsite emergency plan by the applicant.

13.4 Review and Audit

13.4.1 Review

FSAR Amendment 36 (January 1982) states that the review and audit program will be conducted in accordance with Topical Report CE-1A, Rev. 15. During its review the staff found, however, that the then current version of CE-1A, which was Rev. 23 (March 1983), addressed the safety review function differently from that described in FSAR Chapter 16, Section 6, which is the only place in the FSAR where the review and audit functions are described.

In a letter dated September 30, 1983, the applicant stated that the current version of CE-1A is Rev. 25, dated September 9, 1983. When the staff has completed its review of Rev. 25, it will report its findings in a supplement to this SER.

13.4.2 Audit

The staff reviewed the description of the audit functions presented in Topical Report CE-1A, Rev. 23. As noted in Section 13.4.1 above, when the staff has completed its review of Rev. 25 of CE-1A, it will report its findings in a supplement to this SER.

FSAR Amendment 21 (July 1979) lists the audit areas in Technical Specifications, Section 16.6.1.G.1.b. These areas are acceptable, but the time intervals for the audits (at least once per 12 months, in each case) should be specified in Technical Specifications, Sections b.4, b.5 and b.6, and the position titles used in Technical Specifications, Section b.11, should be updated.

13.4.3 Independent Safety Engineering Group

In a letter dated October 5, 1981, the applicant committed to provide, as required by Item I.B.1.2 of NUREG-0737, an Independent Safety Engineering Group consisting of four dedicated full-time engineers located on site, reporting to the Supervisor, Safety Engineering Groups, Office of Nuclear Safety.

The functions of the onsite Safety Engineering Group - Braidwood Station will include the following:

(1) evaluation of all procedures important to the safe operation of the station for technical adequacy and clarity

(2) evaluation of plant operations from a safety perspective

(3) evaluation of the effectiveness of the quality assurance program

(4) evaluation of the operating experience of the station to provide recommendations on safety concerns, and of the operating experience of other plants of similar design for applicability to Braidwood Station

(5) overall assessment of the Braidwood Station staff performance regarding conformance to requirements relating to safety

(6) other matters relating to safe operation of Braidwood Station that independent review deems appropriate for consideration

(7) assessment of plant safety program

Personnel assigned to the Safety Engineering Group - Braidwood Station, shall meet the qualification requirements described in Section 4.7 of Draft American Nuclear Society/American National Standards Institute (ANS/ANSI) 3.1-1979.

Qualified experts in disciplines, which would not be fully utilized at one site, will be made available to the safety engineering groups of all Commonwealth Edison Company sites as needed.

13.4.4 Conclusions

On the basis of the staff evaluation of the Byron Independent Safety Engineering Group (ISEG) as presented in Sections 13.4.5 and 13.4.6 of NUREG-0876, the staff concludes that the applicant's proposal concerning ISEG satisfies the requirements of TMI Action Plan Item I.B.1.2 and is acceptable.

13.5 Plant Procedures

13.5.2 Operating and Maintenance Procedures

13.5.2.1 General

A review is being conducted of the applicant's plan for development and implementation of operating and maintenance procedures. The review is conducted to determine the adequacy of the applicant's program for ensuring that routine operating, offnormal, and emergency activities are performed in a safe manner. The following description and evaluation are based on information contained in the applicant's FSAR and the applicant's response to NRC Three Mile Island (TMI) Action Plan Items (NUREG-0660 and NUREG-0737, including Supplement 1 to NUREG-0737).

In determining the acceptability of the applicant's program, the criteria of SRP Section 13.5.2 (NUREG-0800) are used. The review consists of an evaluation of (1) the applicant's procedure classification system for procedures that are performed by licensed operators in the control room and the classification for other operating and maintenance procedures; (2) the applicant's plan for completion of operating and maintenance procedures during the initial plant testing phase to allow for correction of the procedures before fuel loading; (3) the applicant's program for compliance with the guidance contained in RG 1.33, Rev. 2, March 1978, regarding the minimum procedural requirements for safety-related operations; (4) the applicant's program for compliance with the guidance contained in ANSI N18.7-1976/ANS 3.2; and (5) the applicant's program for compliance with Generic Letter 82-33 (Supplement 1 to NUREG-0737) for the development of emergency operating procedures.

13.5.2.2 Operating and Maintenance Procedure Program

The applicant has committed in the FSAR to a program in which all activities are to be conducted in accordance with detailed written and approved procedures meeting the regulatory positions of RG 1.33, Rev. 2, February 1978, and ANSI N18.7-1976/ANS 3.2.

The applicant uses the following categories of procedures for those operations performed by licensed operators in the control room:

(1) system operating procedures
(2) general operating procedures
(3) abnormal operating procedures
(4) emergency operating procedures
(5) annunciator response procedures
(6) temporary procedures

Other procedures include the following areas:

(1) plant radiation protection
(2) emergency preparedness
(3) instrument calibration and test

(4) chemical/radiochemical control
(5) radioactive waste management
(6) maintenance
(7) materials control
(8) plant security
(9) surveillance

The staff review determined that the applicant's program for use of operating and maintenance procedures meets the relevant requirements of 10 CFR 50.34, and is consistent with the guidance provided in RG 1.33 and ANSI N18.7-1976/ANS 3.2. Therefore, the staff concludes that the applicant's program is acceptable.

13.5.2.3 Reanalysis of Transients and Accidents; Development of Emergency Operating Procedures

In letters of September 13 and 27, October 10 and 30, and November 9, 1979, the staff required licensees of operating plants, applicants for operating licenses, and licensees of plants under construction to perform analyses of transients and accidents, prepare emergency procedure guidelines, upgrade emergency procedures, and conduct operator retraining. Emergency operating procedures are required to be consistent with the actions necessary to cope with the transients and accidents analyzed. Analyses of transients and accidents were to be completed in early 1980 and implementation of procedures and retraining were to be completed 3 months after emergency procedure guidelines were established; however, some difficulty in completing these requirements has been experienced. Clarification of the scope of the task and appropriate schedule revisions were included in NUREG-0737, TMI Task Action Plan Item I.C.1.

The Westinghouse Owners' Group indicated in a meeting with the staff on June 18, 1981, that generic emergency operating procedures and supporting analysis needed to comply with Item I.C.1, as clarified in NUREG-0737, would be submitted in two parts. The first part, containing event-based optimal recovery guidelines, was submitted as an attachment to a letter dated November 30, 1981, from Mr. Robert W. Jurgensen (Westinghouse Owners Group) to Mr. D. G. Eisenhut (NRC). The second part, containing symptom-based critical safety functional restoration guidelines, was submitted by similar letters dated July 21, 1982, and January 4, 1983. The revised guidelines incorporate the short-term reanalysis of small-break loss-of-coolant accidents and inadequate core cooling that was performed for Task Action Plan Items I.C.1(1) and I.C.1(2), and previously approved by the staff. The staff's evaluation and approval of these guidelines is in Generic Letter 83-22 dated June 3, 1983. The Westinghouse Owners' Group is expected to submit a revision to the guidelines in December 1983.

In Generic Letter 82-33, the staff specified the requirements for programs to upgrade emergency operating procedures (EOPs). This implemented the staff's long-term plan for EOPs encompassing requirements of TMI Action Plan Items I.C.1, I.C.8, and I.C.9. This generic letter required applicants to submit for staff review and approval a procedures generation package (PGP) to include (1) plant-specific technical guidelines, (2) a plant-specific writer's guide, (3) a description of the program to validate the EOPs, and (4) a description of the training program for the EOPs.

The applicant's letter of April 14, 1983, from Mr. C. Reed (CECo) to Mr. H. R. Denton (NRC) responded to Generic Letter 82-33 for the Braidwood Station units.

The applicant's letter stated that the PGP would be submitted 12 months after NRC approval of the first revision to the Westinghouse generic technical guidelines. The PGP must be reviewed before the operating license is issued. This review will be addressed in a supplement to this SER.

13.6 Physical Security

The applicant has submitted security plans entitled "Braidwood Nuclear Power Station Physical Security Plan," "Braidwood Nuclear Power Station Safeguards Contingency Plan," and "Braidwood Nuclear Power Station Security Training and Qualification Plan," for protection against radiological sabotage. The training and qualification plan and contingency plan have been reviewed in accordance with SRP Section 13.6, "Physical Security" (NUREG-0800). The security plan is currently under staff review in accordance with the above.

As a result of the staff's evaluation, the training and qualification plan and contingency plan have been approved. Certain portions of the security plan have been identified as requiring additional information to satisfy the requirements of 10 CFR 73.55. The applicant is expected to make commitments acceptable to the staff, which, when formally incorporated into these planning documents, will bring them into compliance with the Commission's regulations contained in 10 CFR 50 and 73.

The identification of vital areas and measures used to control access to these areas, as described in the plan, may be subject to amendments in the future.

The applicant's security plans are being protected from unauthorized disclosure in accordance with 10 CFR 73.21.

14 INITIAL TEST PROGRAM

This section is the same as NUREG-0876.

15 ACCIDENT ANALYSES

This section is the same as NUREG-0876 except for the following:

15.4 Radiological Consequences of Accidents

This section is the same as NUREG-0876 except that the diffusion estimates (χ/Q) for Braidwood Station differ from those for Byron Station. The diffusion estimates for Braidwood are presented and discussed in Section 2.3.4 of this SER. Table 15.1 has been modified to include the offsite radiological consequences specified for Braidwood Station. The accident assumptions listed in Tables 15.2 through 15.7 have been modified to include the diffusion estimates for Braidwood Station.

15.4.1 Loss-of-Coolant Accident

15.4.1.2 Post-LOCA Leakage from ESF System Outside Containment

As part of the LOCA, the staff has also evaluated the consequences of leakage of containment sump water, which is circulated by the emergency core cooling system (ECCS) after that postulated accident. During the recirculation mode of operation, the sump water is circulated outside containment to the auxiliary building. If a leak should develop, such as a pump seal failure, a fraction of the iodine in the water could become airborne in the auxiliary building and exit to the atmosphere. For Braidwood, the ECCS area in the auxiliary building is served by an engineered safety features air filtration system (the auxiliary building exhaust system). Therefore, doses from passive failures were not considered (as specified in SRP Section 15.6.5, Appendix B).

In FSAR Table 15.6-5a, the applicant has identified a value of 3,760 cc/hour as the routine amount of leakage from ECCS equipment following an accident. Using the information in the Standard Review Plan, the staff evaluated the potential radiological consequences from this release pathway assuming a routine leakage rate of twice the applicant's value (7,520 cc/hour). The resultant radiological consequences were only 2.2 rems to the thyroid at the exclusion area boundary and 3.8 rems to the thyroid at the low population zone (LPZ). The staff also evaluated the potential radiological consequences from normal ECCS component leakage at a leak rate of 1 gpm. The resulting radiological consequences were 63 rems to the thyroid at the exclusion area boundary and 106 rems to the thyroid at the LPZ.

Table 15.1 Radiological consequences of design-basis accidents

Postulated accident	Exclusion area boundary, rems		Low population zone, rems	
	Thyroid	Whole body	Thyroid	Whole body
Loss of coolant:				
Containment leakage				
0-2 hr	141	4	-	-
0-8 hr			31	0.9
8-24 hr			12	0.2
24-96 hr			11	0.1
96-720 hr			13	0.1
Total containment leakage	141	4	67	1.3
ECCS component leakage	2	<0.01	4	<0.01
Total	143	4	71	1.3
Steamline break outside secondary containment:				
Long-term operation case (DEI-131 at 1 μCi/gm)	10	<1.0	2.6	<1.0
Short-term operation case (DEI-131 at 60 μCi/gm)	13	<1.0	2.5	<1.0
Control rod ejection:				
Containment leakage pathway	41	<1.0	47	<1.0
Secondary system release pathway	36	<1.0	0.4	<1.0
Fuel-handling accident in fuel-handling area	24	0.5	2.5	<0.1
Small line break	4.7	<0.1	0.5	<0.1
Steam generator tube rupture:				
Long-term operation case (DEI-131 at 1 μCi/gm)	9	<0.1	1.5	<0.1
Short-term operation case (DEI-131 at 60 μCi/gm)	50	<0.1	6.7	<0.1

Table 15.2 Assumptions used in the calculation of loss-of-coolant accident doses

Parameter and unit of measure	Quantity
Containment leakage	
Power level, MWt	3,565
Operating time, yr	3
Fraction of core inventory available for containment leakage, %	
Iodine	25
Noble gases	100
Initial iodine composition in containment, %	
Elemental	91
Organic	4
Particulate	5
Containment leak rate, %/day	
0-24 hr	0.1
After 24 hr	0.05
Containment volume, ft^3	
Sprayed volume	2.35×10^6
Unsprayed volume	4.1×10^5
Containment mixing rate from cooling fan operation, cfm	180,000
Containment spray system	
Maximum elemental iodine decontamination factor	100
Spray removal coefficients, hr^{-1}	
Elemental iodine	10
Particular iodine	0.45
Organic iodine	0
Relative concentration values, sec/m^3	
0-2 hr at the exclusion area boundary	5.6×10^{-4}
0-8 hr at the low population zone (LPZ) boundary	5.9×10^{-5}
8-24 hr at the LPZ boundary	4.4×10^{-5}
24-96 hr at the LPZ boundary	2.3×10^{-5}
96-720 hr at the LPZ boundary	9.4×10^{-6}
ECCS leakage outside containment	
Power, MWt	3,565
Sump volume, gal	484,000
Flash fraction	0.1
Leak rate, gph (twice the maximum operational leakage defined in FSAR Table 15.6-15a)	2.1
Leak duration, hr	720
Delay time, hr	0.50
Filter efficiency for iodine, %	
Elemental and particulate	90
Organic iodine	50

Table 15.3 Assumptions used to evaluate the radiological consequences following a postulated main steamline break accident outside containment

Power, MWt	3,565
Preaccident dose equivalent I-131 in primary coolant, μCi/gm	1.0*
Preaccident dose equivalent I-131 in primary coolant, μCi/gm	60.0**
Primary-to-secondary leak rate, as limited by Technical Specifications, gpm	1.0
All of the 1-gpm leak occurs in the affected steam generator	
All the iodine transported to the shell side of the steam generator by the leakage is lost to the environment without decay	
Iodine release rate from fuel increases by a factor of 500 as a result of the accident (Case 2)	
χ/Q values, sec/m^3	
0-2 hr at 485 m (exclusion area boundary)	5.6×10^{-4}
0-8 hr at 1,810 m (low population zone)	5.9×10^{-5}

*Long-term operation case
**Short-Term operation case

Table 15.4 Assumptions used for the calculations of the radiological consequences of a postulated steam generator tube rupture accident

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The rupture is a double-ended guillotine break, resulting in an average leakage of 66 lbs/sec from the reactor coolant system to the steam generator secondary side.

Concentration of iodine (as DE I-131) in primary coolant at start of accident (two different cases): 60 μCi/gm (Technical Specifications for maximum concentration allowed during a short time), and 1.0 μCi/gm (equilibrium Technical Specification limit).

Secondary coolant activity at start of accident: 0.1 μCi/gm DEI-131 Technical Specification limit.

Average ratio of iodine mass concentration in the steam to that in the secondary side water, for both the affected and unaffected (but leaking slighty) steam generators: 0.1.

Carry-over (in droplet form): 1%.

Isolation of affected steam generator at 40 min.

Primary-to-secondary leak rate of 500 gal per day to each of the three unaffected steam generators.

Iodine release rate from fuel increases by a factor of 500 over the equilibrium release rate.

Condenser use is lost at time of scram, about 11 min after the rupture.

Atmospheric dispersion factors (χ/Q)
0-2 hr at 485 m = 5.6×10^{-4} sec/m^3
0-8 hr at 1,810 m = 5.9×10^{-5} sec/m^3

Table 15.5 Assumptions used for estimating the radiological consequences following a postulated control rod ejection accident

Power = 3,565 MWt

Primary-to-secondary leak rate is 1.0 gpm as limited by Technical Specifications.

10% of the fuel rods experience cladding failure, releasing all their gap radioactivity (assumed to be 10% of the equilibrium core activity of iodines and noble gases). The released activity is mixed immediately with the primary coolant.

0.25% of the fuel rods experience fuel melting, and all the noble gases and 50% of the iodine in this fraction of fuel are released and mixed immediately with the primary coolant.

As a result of loss of offsite power and subsequent steam venting, 10% of the iodine transported to and mixed with the secondary coolant is lost during the course of the accident.

Primary and secondary system pressures equalize in about 3,300 sec, terminating the primary-to-secondary leak.

For the containment pathway calculation, 50% of the iodine released into the containment is plated out instantaneously.

Primary containment leak rate = 0.10% per day (containment leakage pathway).

The iodine concentration in the secondary coolant was assumed to be 0.1 μCi/gm DEI-131.

χ/Q values
- 0-2 hr at 485 m = 5.6×10^{-4} sec/m^3 (exclusion area boundary)
- 0-8 hr at 1,810 m = 5.9×10^{-5} sec/m^3 (low population zone)

Table 15.6 Assumptions used for estimating the radiological consequences following a postulated fuel handling accident

Parameter and unit of measure	Quantity
Power level, MWt	3,565
Number of fuel rods damaged	314
Total number of fuel rods in core	60,602
Radial peaking factor of damaged rod	1.65
Shutdown time, hour	100
Inventory released from damaged rods (iodines and noble gases), %	10
Pool decontamination factors	
Iodine	100
Noble gases	1
Iodine fractions released from pool, %	
Elemental	75
Organic	25
Iodine removal efficiencies for ABGTS (spent fuel pool area), %	
Elemental	90
Organic	70
χ/Q values, sec/m^3	
0-2 hr at 485 m (exclusion area boundary)	5.6×10^{-4}
0-8 hr at 1,810 m (low population zone)	5.9×10^{-5}

Table 15.7 Assumptions used in accidents involving small line breaks outside the containment

Parameter and unit of measure	Quantity
Coolant released, lb	17,000
Fraction of coolant released flashed to steam, %	39
Coolant contaminant concentration, µCi/gm	
Case 1, normal operating limit	1.0
Case 2, coincident "iodine spike," 500 times normal release rate, concentration varying with time	1.0-7.7

16 TECHNICAL SPECIFICATIONS

The Technical Specifications in a license define certain features, characteristics, and conditions governing operation of a facility that cannot be changed without prior approval of the staff. The finally approved Technical Specifications will be made a part of the Operating License. Included will be sections covering definitions, safety limits, limiting safety system settings, limiting conditions for operations, surveillance requirements, design features, and administrative controls.

Because Braidwood Station is a duplicate of Byron Station, the Technical Specifications for the duplicate portions of Braidwood will be based on the then-current (at the time the review takes place) Technical Specifications for Byron Station, unless there is significant new information that substantially affects the bases for the Byron Technical Specifications or other good cause.

The staff is working with the applicant to prepare a draft of the Technical Specifications for the Braidwood Station. On the basis of its review to date, the staff concludes that normal plant operation within the limits of the Technical Specifications will not result in offsite exposure in excess of the 10 CFR 20 limits. Furthermore, the limiting conditions for operation and surveillance requirements will ensure that necessary engineered safety features will be available in the event of malfunctions within the facility.

During its review of the Braidwood Station application, the staff identified certain issues that must be included in the Technical Specifications as a condition of staff acceptance. Most of these issues are already addressed in the "Standard Technical Specifications for Westinghouse Pressurized Water Reactors" (NUREG-0452, Rev. 4). Those issues that are not included in NUREG-0452 will be added to the Technical Specifications being prepared for the Braidwood Station. These issues are listed below and are discussed further in the sections of this SER as indicated.

Issue	Section
Ground water elevation	2.4.6
Safety-related water supply	2.4.8
Seismic instrumentation	3.7.4
Axial offset control band limitation	4.3.1
N-1 loop operation	4.4.3
Heat tracing the RWST	6.3.1
Ventilation system flowrates	6.5.1
Surveillance tests of reactor trip breakers	7.2.2.1
Surveillance tests of safety system instrumentation constant setpoints	7.2.2.4
Surveillance test of RTD bypass loop flow	7.2.2.7
Safety system trip set point methodology	7.3.2.4
Testing of slave relays with proving lamps	7.3.2.12
Auxiliary feedwater system	10.4.9

17 QUALITY ASSURANCE

This section is the same as NUREG-0876.

18 HUMAN FACTORS ENGINEERING

The status report of the human engineering evaluation of the Braidwood control room follows.

Position

All licensees and applicants for an operating license are required to conduct a detailed control room design review (DCRDR) in response to NRC Task Action Plan Item I.D.1. (NUREG-0660, May 1980; and NUREG-0737, November 1980, as supplemented by Generic Letter No. 82-33, December 17, 1982). The purpose of the DCRDR is to identify and correct human engineering discrepancies (HEDs) that might affect the operator's ability to prevent or cope with an accident. NUREG-0700, "Guidelines for Control Room Design Reviews" (September 1981), provides guidance for conducting the DCRDR. Operating license applicants whose first SER supplement (SSER) will be issued after June 1983 will be required to complete their DCRDR before licensing.

Discussion

The scheduled date for SSER No. 1 for Braidwood Station, Unit 1, is May 1984. Braidwood is being reviewed in conformance with the Commission's "Statement on Standardization of Nuclear Power Plants" (1973, 1978), under the duplicate plant concept for the Byron Station design. The applicant has committed to all control room modifications resulting from the Byron Preliminary Design Assessment (PDA) and is taking exception to the June 1983 limiting date for licensing based on a PDA. In accordance with Generic Letter 82-33, the applicant has submitted the DCRDR schedule and program plan stating the exception (letter from Cordell Reed (CECo), April 14, 1983). The staff agrees with the exception and the commitment to duplicate the Byron control room modifications, once reviewed and accepted by the staff, but requires the applicant to perform separate PDA reviews of (1) all nonstandard, site-specific panels; (2) items not resolved in the Byron PDA; and (3) environmental systems (HVAC, lighting, sound).

In a letter from E. D. Swartz of Commonwealth Edison Company dated September 30, 1983, site-specific panels 1PM01J, 1PM03J, 0PM01J, 0PM03J, and 1PM09J were identified as containing differences from those in the Byron Station control room.

Conclusion

The applicant shall implement all corrective measures listed in the May 9, 1983, letter from E. D. Swartz, once these measures have been reviewed and accepted by the staff. The staff requires that the applicant perform an evaluation of (1) all nonstandard, site-specific panels identified in the September 30, 1983, letter from E. D. Swartz; (2) all items not resolved in the Byron PDA; and (3) environmental systems (HVAC, lighting, sound) and that he submit his findings, proposed corrective actions, and schedule for implementing those actions. The report on these items shall be submitted for staff review and approval no later than 120 days before an operating license is issued. After licensing, the applicant shall conduct a DCRDR in accordance with a schedule to be approved by the NRC.

On the basis of the quality of those aspects of the Byron Station control room that were reviewed, the corrections proposed by the applicant and approved by the NRC, and the commitment by the applicant to review the three areas listed above, the staff believes that interim, suitable corrections of human engineering discrepancies can be implemented so as to provide an improved control room design until the applicant completes his DCRDR.

19 REVIEW BY ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

In its letter dated May 13, 1975, the Advisory Committee on Reactor Safeguards (Committee) indicated that certain matters would require further attention and resolution during construction of the Braidwood Station, Units 1 and 2. These items were addressed in Supplement 1 to the CP-SER dated August 1975.

Certain of these matters are addressed further in this report, as identified below. References are given to sections in this report related to the construction of the facility for further discussion.

(1) containment functional design (6.2.1)
(2) turbine missiles (3.5.1.3)
(3) emergency core cooling system performance evaluation (6.3.5)
(4) physical security (13.6)
(5) environmental and seismic qualification of Class 1E electrical equipment (3.10, 3.11)

The OL application for Braidwood Units 1 and 2 is being reviewed by the Committee. The staff will issue a supplement to this SER after the Committee's report on this application is available. The supplement will append a copy of the Committee's report and will address the significant comments made by the Committee. It also will describe steps taken by the staff to resolve any issues raised as a result of the Committee's review.

20 COMMON DEFENSE AND SECURITY

The applicant has stated that the activities to be conducted will be within the jurisdiction of the United States and that all the directors and principal officers of the applicant are citizens of the United States. Commonwealth Edison Company is not owned, controlled, or dominated by an alien, a foreign corporation, or a foreign government. The activities to be conducted do not involve any restricted data, but the applicant has agreed to safeguard any such data that might become involved in accordance with the requirements of 10 CFR 50. The applicant will rely on obtaining fuel as it is needed from sources of supply available for civilian purposes, so that no diversion of special nuclear material for military purposes is involved. For these reasons, and in the absence of any information to the contrary, the staff finds that the activities to be performed will not be inimical to the common defense and security.

21 FINANCIAL QUALIFICATIONS

On March 11, 1982, the Commission approved SECY-82-21, a final rule eliminating entirely the financial qualification review and findings for "electric utilities" applicants, providing that the financial qualifications of an electric utility applicant are not among the issues to be considered by atomic safety and licensing boards in construction permit or operating licensing proceedings. This aspect of the rule is effective immediately upon publication in the Federal Register (47 FR 13750) and applies to pending licensing proceedings and the issues or contentions raised therein.

Pursuant to 10 CFR 50.2(X) and 50.33(F), 47 FR 13750 (March 31, 1982), electric utility applicants will no longer be required to submit information on their financial qualifications and the staff shall not conduct any financial qualifications reviews of such applicants. "Electric Utility" includes investor-owned utilities, public utility districts, municipalities, rural electric cooperatives, and state or Federal agencies, and associations of these entities.

22 FINANCIAL PROTECTION AND INDEMNITY REQUIREMENTS

22.1 General

Pursuant to the financial protection and indemnification provisions of the Atomic Energy Act of 1954, as amended (Section 170 and related sections), the Commission has issued regulations in 10 CFR 140. These regulations set forth the Commission's requirements with regard to proof of financial protection by, and indemnification of, licenses for facilities such as power reactors under 10 CFR 50.

22.2 Preoperational Storage of Nuclear Fuel

The Commission's regulations in 10 CFR 140 require that each holder of a construction permit under 10 CFR 50, who also is the holder of a license under 10 CFR 70 that authorizes the ownership and possession for storage only of special nuclear material at the reactor construction site for future use as fuel in the reactor (after issuance of an operating license under 10 CFR 50), shall, during the interim storage period before licensed operation, have and maintain financial protection in the amount of $1,000,000 and execute an indemnity agreement with the Commission. Proof of financial protection is to be furnished before, and the indemnity agreement executed as of, the effective date of the 10 CFR 70 license. Payment of an annual indemnity fee is required.

The applicant has stated that he will furnish proof of financial protection in the amount of $1,000,000 in the form of a Nuclear Energy Liability Insurance Association Policy. Furthermore, the applicant has committed to execute an indemnity agreement with the Commission effective as of the date of the preoperational fuel-storage license. The applicant will pay the annual indemnity fee applicable to preoperational fuel storage.

22.3 Operating Licenses

Under the Commission's regulations (10 CFR 140), a license authorizing the operation of a reactor may not be issued until proof of financial protection in the amount required for such operation has been furnished and an indemnity agreement covering such operation (as distinguished from preoperational fuel storage only) has been executed. The amount of financial protection that must be maintained for Braidwood Station, Units 1 and 2 (which have a rated capacity in excess of 100,000 kWe), is the maximum amount available from private sources, which is currently $570 million.

Accordingly, licenses authorizing operation of Braidwood Station, Units 1 and 2, will not be issued until proof of financial protection in the requisite amount has been received and the requisite indemnity agreement has been executed.

The staff expects that, in accordance with the usual procedure, the nuclear liability insurance pools will provide, several days in advance of anticipated issuance of the operating license, evidence in writing, on behalf of the applicant, that prior coverage has been appropriately amended so that the policy limits have been increased to meet the requirements of the Commission's regulations for reactor operation. Similarly, an operating license will not be issued until an appropriate amendment to the present indemnity agreement has been executed. The applicant will be required to pay an annual fee for operating license indemnity, as provided in the NRC regulations.

On the basis of the above considerations, the staff concludes that the currently applicable requirements of 10 CFR 140 have been satisfied and that, before an operating license is issued, the applicant will be required to comply with the provisions of 10 CFR 140 applicable to operating licenses, including those as to proof of financial protection in the requisite amount and as to the execution of an appropriate indemnity agreement with the Commission.

23 CONCLUSIONS

On the basis of its evaluation of the application as set forth above, the staff has determined that, upon favorable resolution of the outstanding matters described herein, it will be able to conclude that

(1) The application for facility licenses filed by the applicant, dated June 27, 1978, complies with the requirements of the Atomic Energy Act of 1954, as amended (Act), and the Commission's regulations set forth in 10 CFR Chapter 1.

(2) Construction of Braidwood Station, Units 1 and 2, has proceeded, and there is reasonable assurance that both will be substantially completed in conformity with Construction Permits Nos. CPPR-132 and CPPR-133, the application as amended, the provisions of the Act, and the rules and regulations of the Commission.

(3) The facilities will operate in conformity with the application as amended, the provisions of the Act, and the rules and regulations of the Commission.

(4) There is reasonable assurance (a) that the activities authorized by the operating licenses can be conducted without endangering the health and safety of the public and (b) that such activities will be conducted in compliance with regulations of the Commission set forth in 10 CFR Chapter 1.

(5) The applicant is technically qualified to engage in the activities authorized by the licenses in accordance with the regulations of the Commission set forth in 10 CFR Chapter 1.

(6) The issuance of these licenses will not be inimical to the common defense and security or to the health and safety of the public.

Before operating licenses are issued to the applicant for operation of the Braidwood Station, Units 1 and 2, the units must be completed in conformity with the provisional Construction Permits, the application, the Act, and the rules and regulations of the Commission. Such completeness of construction as is required for safe operation at the authorized power levels must be verified by the NRC before the licenses are issued.

Furthermore, before operating licenses are issued, the applicant will be required to satisfy the applicable provisions of 10 CFR 140.

APPENDIX A

CHRONOLOGY OF NRC STAFF RADIOLOGICAL SAFETY
REVIEW OF BRAIDWOOD STATION, UNITS 1 AND 2

November 30, 1978	Letter from applicant transmitting Amendment 17 to the Final Safety Analysis Report (FSAR) consisting of an application for operating licenses for the Braidwood Station, Units 1 and 2.
November 30, 1978	Letter from applicant transmitting the Environmental Report, Volumes 1 and 2.
November 30, 1978	Letter from applicant transmitting oversized electric instrumentation and control drawings to Amendment 17 of the FSAR.
November 30, 1978	Application for operating licenses for the Braidwood Station, Units 1 and 2, docketed by NRC.
December 11, 1978	Letter from applicant transmitting the affidavit for distribution of Amendment 17 to the FSAR.
January 2, 1979	Letter from applicant transmitting revised pages to Topical Report CE-1A, Revision 6, Quality Assurance Program.
February 8, 1979	Letter from applicant transmitting Amendment 18 to the FSAR. This amendment consists of responses to NRC questions raised during the acceptance review.
February 13, 1979	Letter from applicant transmitting an affidavit stating that distribution of the Environmental Report--Operating License Stage--has been made.
February 15, 1979	Letter from applicant transmitting the 1978 Annual Financial Reports.
February 16, 1979	Letter from applicant transmitting an affidavit stating that Amendment 18 to the FSAR has been made.
February 22, 1979	Letter to applicant establishing a schedule for NRC review of the application for an operating license.
March 22, 1979	Letter from applicant transmitting antitrust information requested by NRC.

March 23, 1979	Letter from applicant transmitting the Safeguards Contingency Plan.
March 30, 1979	Letter from applicant transmitting Amendment 19 to the FSAR.
April 11, 1979	Letter from applicant transmitting an affidavit of service for Amendment 19 to the FSAR.
May 2, 1979	Letter from applicant transmitting Amendment 20 to the FSAR.
May 11, 1979	Letter from applicant transmitting an affidavit of service for Amendment 20 to the FSAR.
May 15, 1979	Letter to applicant requesting additional information in the areas of auxiliary systems, power systems, mechanical engineering, structural engineering, quality assurance and effluent treatment systems.
June 1, 1979	Letter to applicant requesting additional information.
June 21, 1979	Letter from applicant transmitting Revision 8 to the Quality Assurance Program.
June 27, 1979	Letter from applicant advising of decision to delay fuel load dates by 1 year.
July 9, 1979	Letter from applicant advising that ASME Code Case N-240 will be referenced in the FSAR.
July 20, 1979	Letter from applicant transmitting Amendment 21 to the FSAR.
July 30, 1979	Letter from applicant transmitting table showing that manning of technical positions by technical graduates is virtually complete.
July 31, 1979	Letter from applicant transmitting an affidavit of service for Amendment 21 to the FSAR.
August 1, 1979	Letter to applicant requesting a secondary water chemistry program to be incorporated in new license condition.
August 18, 1979	Letter from applicant transmitting the Security Personnel Training and Qualification Plan.
September 5, 1979	Letter to applicant requesting additional information in order to complete review of the operating license application. Transmits the third set of first-round questions for Mechanical Engineering and Quality Assurance Branches.

September 13, 1979	Letter to applicant discussing reorganization of design and operating requirements resulting from NRC review following TMI-2 accident.
September 14, 1979	Letter from applicant transmitting Amendment 22 to the FSAR.
September 17, 1979	Letter from applicant transmitting an affidavit of service for Amendment 22 to the FSAR.
September 21, 1979	Letter from applicant discussing reorganization of design, operations, and licensing activities.
September 21, 1979	Letter from applicant notifying NRC of internal reorganization of nuclear activities. All future correspondence should be addressed to D. L. Peoples.
September 27, 1979	Letter to applicant transmitting NRC position on status and applicability to pending operating license applications of results of TMI followup actions.
September 28, 1979	Letter from applicant transmitting the proposed secondary water chemistry programs.
October 10, 1979	Letter to applicant discussing schedule for upgraded emergency plans.
October 17, 1979	Letter to applicant discussing NRC plans to use generic analyses to develop early verification program to resolve anticipated transients without scram (ATWS) issue.
October 19, 1979	Letter from applicant advising that recent interim rate increase and additional rate increase by March 1980 will enable resumption of construction by June 1, 1980.
November 2, 1979	Letter from applicant transmitting Amendment 23 to the FSAR.
November 8, 1979	Letter to applicant requesting additional information for Reactor Systems and Quality Assurance Branches--fourth set of first-round questions.
November 9, 1979	Letter to applicant concerning discussions of TMI lessons-learned short-term requirements.
November 14, 1979	Letter from applicant transmitting an affidavit of service for Amendment 23 to the FSAR.
November 21, 1979	Letter to applicant transmitting information regarding upgraded emergency plans to plants under construction permit and operating license review.

November 26, 1979	Letter from applicant transmitting Revision 1 to the Physical Security Plan.
December 14, 1979	Letter from applicant responding to NRC letter concerning "TMI-2 Lessons Learned Task Force Status Report and Short-Term Recommendations."
December 19, 1979	Letter from applicant concerning title change of "Manager of Nuclear Operations" to "Vice President of Nuclear Operations."
December 20, 1979	Letter from applicant transmitting revised pages of QA Program for Nuclear Generating Stations, Revision 11.
December 21, 1979	Letter to applicant transmitting Revision 1 to branch technical position on radiological environmental monitoring.
December 21, 1979	Letter to applicant announcing regional workshops to discuss feasibility of proposed change to regulation on radiological emergency response plans for facilities.
December 26, 1979	Letter to applicant transmitting request for evacuation time estimates for areas near nuclear power plants.
January 15, 1980	Letter from applicant transmitting drawings for FSAR Section 1.2.
January 15, 1980	Letter from applicant transmitting a list of drawings for latest FSAR revision.
January 28, 1980	Letter from applicant transmitting responses to questions from the Structural Engineering Branch.
January 28, 1980	Letter to applicant authorizing the use of ASME Code Case N-240.
February 4, 1980	Letter to applicant requesting information on confirmatory piping analysis. Analysis will verify that piping system meets applicable ANSI/ASME Code stress criteria.
February 5, 1980	Letter to applicant transmitting additional guidance to NUREG-0588. Requests review of equipment qualification documentation to identify degree of compliance.
February 8, 1980	Letter from applicant transmitting Amendment 24 to the FSAR.
February 8, 1980	Letter from applicant transmitting structural and component drawings per NRC request.

February 14, 1980	Letter from applicant transmitting additional FSAR information on the method of determining heat rejection from PWR plant to ultimate heat sink.
February 15, 1980	Letter from applicant transmitting revised Page 5-2 to QA Program.
February 22, 1980	Letter from applicant transmitting an affidavit of service for Amendment 24 to the FSAR.
February 22, 1980	Letter from applicant transmitting the Annual Financial Report for 1979.
March 3, 1980	Letter to applicant requesting description of QA Program to prevent degradation of safety-related equipment, components, and structures during suspension of construction activities.
March 10, 1980	Letter to applicant requesting additional information on the auxiliary feedwater system per Bulletins and Orders Task Force review of TMI-2 accident.
March 11, 1980	Letter to applicant advising that submittal date for evacuation time estimates has been postponed pending GAO clearance under Federal Reports Act.
March 11, 1980	Letter from applicant transmitting a report entitled "Preliminary Evacuation Time Study of 10-Mile Radius Emergency Planning Zone."
March 21, 1980	Letter from applicant transmitting Amendment 25 to the FSAR.
March 25, 1980	Letter from applicant transmitting the QA Plan implemented in 1979.
March 28, 1983	Letter to applicant requesting additional information consisting of the fifth set of round-one questions regarding safeguards contingency plan.
April 1, 1980	Letter from applicant transmitting an affidavit of service for Amendment 25 to the FSAR.
April 2, 1980	Letter from applicant transmitting revised pages constituting Revision 13 to the QA Program.
April 21, 1980	Letter from applicant advising that first cores for Byron and Braidwood nuclear power plant units will employ Westinghouse manufactured 17x17 optimized fuel assemblies.
April 24, 1980	Letter from applicant transmitting draft of emergency plan addressing items in NUREG-0654.

April 24, 1980	Letter from applicant transmitting public version of draft emergency plan for generating stations.
April 25, 1980	Letter to applicant transmitting clarification of NRC requirements for emergency-response facilities.
April 28, 1980	Letter to applicant transmitting the sixth set of first-round questions covering the Auxiliary Systems Branch, Reactor Systems Branch, and Instrumentation and Control Branch reviews.
April 30, 1980	Letter from applicant transmitting information for Argonne National Laboratory (ANL) confirmatory piping analysis requested by NRC.
May 2, 1980	Letter from applicant transmitting Revision 2 to security plan.
May 6, 1980	Letter from applicant transmitting responses to ANL December 20, 1979, questions on FSAR Chapter 7.
May 15, 1980	Letter from applicant transmitting information for ANL to perform confirmatory piping analysis.
May 20, 1980	Letter to applicant transmitting the seventh set of first-round questions regarding training and requalification plan.
May 20, 1980	Letter to applicant advising BWR and PWR applicants of NRC decision to modify implementation plan presented in Section 4 of NUREG-0577, regarding the adequacy of applicable support structures.
May 22, 1980	Letter from applicant transmitting information on auxiliary feedwater system.
May 22, 1980	Letter from applicant requesting that E. R. Cross of Sargent & Lundy be added to the distribution list.
June 2, 1980	Letter to applicant transmitting the eighth set of first-round questions regarding safeguards contingency plan.
June 3, 1980	Letter from applicant transmitting drawing 583F499, which was inadvertently omitted from Item 7 of May 6, 1980, letter.
June 4, 1980	Letter from applicant transmitting public version of utility-revised draft emergency plan.
June 13, 1980	Letter to applicant advising of the reorganization of the Office of Nuclear Reactor Regulation.

June 16, 1980	Letter to applicant concerning independent structural analysis.
June 18, 1980	Letter from applicant responding to questions concerning the guard training and qualification plans.
June 20, 1980	Letter from applicant transmitting an update of scheduled commercial service dates for facilities.
June 23, 1980	Letter from applicant transmitting Revisions 11 and 12 of QA Program.
June 25, 1980	Letter from applicant transmitting safety-related electrical equipment qualification.
June 25, 1980	Letter to applicant (generic) concerning Commission Memorandum and Order on Union of Concerned Scientists.
June 26, 1980	Letter to applicant concerning Commission guidance for power reactor operating licenses.
June 27, 1980	Letter from applicant transmitting safeguards contingency plan amendment.
June 27, 1980	Letter from applicant transmitting Amendment 26 to the FSAR.
June 30, 1980	Letter from applicant transmitting affidavit of service for Amendment 26 to the FSAR.
June 30, 1980	Letter to applicant (generic) concerning region meetings for applicants, NRC staff, and architect/engineers.
July 2, 1980	Letter to applicant (generic) concerning evacuation times.
July 3, 1980	Letter from applicant transmitting additional FSAR drawings and information for auxiliary and containment building structures and components.
July 7, 1980	Letter from applicant transmitting drawings listed in FSAR Subsection 1.7.
July 8, 1980	Letter from applicant responding to NRC request for additional information on masonry wall design.
July 11, 1980	Letter from applicant transmitting Amendment 27 to the FSAR.
July 11, 1980	Letter to applicants of CP and OL requesting up-to-date completion schedules and fuel-load target dates so that NRC may establish licensing priorities.

July 16, 1980	Letter from applicant transmitting an affidavit of service for FSAR Amendment 27.
July 17, 1980	Letter from applicant notifying NRC of exception to Regulatory Guide (RG) 1.88 taken in Amendment 26 to the FSAR.
July 31, 1980	Letter from applicant advising of inability to submit security plan revisions until August 11, 1980.
July 31, 1980	Letter to applicant (generic) concerning interim criteria for shift staffing.
August 1, 1980	Letter to applicant transmitting a draft of NUREG-0696, "Functional Criteria for Emergency Response Facilities."
August 1, 1980	Letter from applicant transmitting errata changes to safeguards contingency plan.
August 1, 1980	Letter from applicant transmitting comments on NUREG-0577 regarding reactor coolant pumps, reactor vessels, pressurization and steam generators.
August 11, 1980	Letter from applicant transmitting a security plan revision on the guard training qualification plan.
August 13, 1980	Letter from applicant transmitting Revision 13 to the QA Program.
August 13, 1980	Letter from applicant advising that J. S. Abel has replaced D. L. Peoples as Utility Director of Nuclear Licensing.
August 14, 1980	Letter from applicant submitting construction completion schedules.
September 2, 1980	Letter to applicant notifying of acceptance of the Byron nuclear station safeguards contingency plans.
September 5, 1980	Letter to all CP and OL holders regarding preliminary clarification of TMI action plan requirements.
September 12, 1980	Letter to applicant concerning interim actions for plant operation pending final resolution of ATWS.
September 15, 1980	Letter from applicant transmitting preliminary topical outlines of proposed modules on thermal hydraulics and core damage mitigation to be included in revised reactor operator training program.
September 19, 1980	Letter to applicant transmitting errata sheets amending letter that provided preliminary clarification of TMI action plan requirements.

September 24, 1980	Letter from applicant transmitting drawing of block diagram on turbine trip protection.
September 24, 1980	Letter from applicant transmitting schematic drawings indicating circuits for each of the 12 motor-operated valves listed in the response to FSAR Question 040.12.
September 24, 1980	Letter from applicant transmitting a list of current qualification programs for the Electric Power Research Institute (EPRI).
October 1, 1980	Letter from applicant transmitting comments regarding preliminary clarification of TMI action plan requirements.
October 1, 1980	Letter to applicant (generic) concerning environmental qualification of safety-related equipment.
October 6, 1980	Letter to applicant (generic) concerning NUREG-0577.
October 31, 1980	Letter to applicant forwarding NUREG-0737 regarding TMI action plan requirements.
October 31, 1980	Letter from applicant transmitting Amendment 28 to FSAR.
October 31, 1980	Letter from applicant advising that information on environmental qualification tests to be conducted within next 2 years on equipment used or to be used in facilities will be provided by December 1, 1980.
November 3, 1980	Letter from applicant transmitting draft public information package regarding emergency preparedness for facilities.
November 5, 1980	Letter from applicant transmitting page 2-6 of Topical Report CE-1-A, "QA Program."
November 6, 1980	Letter from applicant transmitting an affidavit of service for Amendment 28 to the FSAR.
November 13, 1980	Letter to applicant (generic) concerning final regulations for emergency planning.
November 18, 1980	Letter from applicant advising of intentions to use ASME Code Case N-275.
November 26, 1980	Letter from applicant transmitting tabulation of planned environmental qualification testing programs.
November 26, 1980	Letter to applicant (generic) concerning environmental qualification of safety-related equipment.
December 8, 1980	Letter to applicant authorizing use of Code Case N-275.

December 9, 1980	Letter to applicant (generic) transmitting Revision 1 to NUREG-0654/FEMA-REP-1, November 1980.
December 22, 1980	Letter to applicant (generic) concerning control of heavy loads.
December 30, 1980	Letter to applicant advising that the revised guard training and qualification program submitted is acceptable.
December 31, 1980	Letter from applicant transmitting public version of Revision 2, Draft 1, of generating stations emergency plan and site-specific annexes.
January 2, 1981	Letter from applicant transmitting Revision 15 to CE-1A, QA Program topical report.
January 2, 1981	Letter to applicant transmitting the ninth set of round-one questions.
January 2, 1981	Letter to applicant transmitting the first set of round-two questions.
January 19, 1981	Letter to applicant (generic) concerning environmental qualification of safety-related electrical equipment.
January 19, 1981	Letter from applicant transmitting offsite dose calculation system description per meterological data requirements in NUREG-0737.
February 3, 1981	Letter from applicant advising that the utility intends to use ASME Code Case N-272, "Compiling Data Report Records," in completion of N-3 and N-5 data reports.
February 3, 1981	Letter from applicant requesting approval of ASME Code Case N-295 permitting use of existing construction materials certified per Code editions dated earlier than edition specified for construction.
February 3, 1981	Letter to applicant (generic) concerning control of heavy loads.
February 6, 1981	Letter from applicant transmitting Amendment 29 to the FSAR.
February 6, 1981	Letter from applicant transmitting 14 oversize drawings referenced in response to FSAR Questions 040.78 and 040.82.
February 19, 1981	Letter from applicant transmitting the Annual Financial Report for 1980.
February 20, 1981	Letter to applicant (generic) concerning NUREG-0619.

February 25, 1981	Letter to applicant (generic) concerning possible station blackout event.
February 25, 1981	Letter from applicant transmitting an affidavit of service for Amendment 29 to the FSAR.
February 26, 1981	Letter to applicant (generic) concerning periodic updating of FSAR.
March 4, 1981	Letter from applicant transmitting a corrected version of summary slide JW/OZ-12 used in the utility February 18, 1981, presentation in Bethesda, Maryland, on seismic reanalysis of facilities.
March 5, 1981	Letter to applicant authorizing use of ASME Code Case N-272 and Code Case N-295.
March 5, 1981	Letter to applicant (generic) concerning functional criteria for emergency response facilities, NUREG-0696.
March 10, 1981	Letter to applicant (generic) concerning environmental qualification of Class 1E electrical equipment, NUREG-0696.
March 13, 1981	Letter to applicant concerning OL hearing scheduling process and SER technical input schedule for Calendar Years 1981 and 1982.
March 16, 1981	Letter to applicant transmitting second set of second-round questions.
March 16, 1981	Letter from applicant transmitting Revision 16 to the QA Program.
March 23, 1981	Letter to applicant transmitting the tenth set of round-one questions.
March 26, 1981	Letter to applicant concerning human factors control room design review/site visit.
March 27, 1981	Letter from applicant transmitting a public version of the emergency plan implementing procedures for corporate command center and operating facility.
March 27, 1981	Letter from applicant advising that licensing schedule outlined in NRC status reports to Congressman Beville is unrealistic and will result in significant delays.
March 30, 1981	Letter from applicant transmitting public version of revised implementing procedures for emergency plan.

March 31, 1981	Letter from applicant transmitting outline of proposed seismic reassessment program regarding safety-related main plant structures, seismic review of system, equipment and components required for safe shutdown, and method for considering nonseismic interaction.
March 31, 1981	Letter from applicant transmitting Amendment 30 to the FSAR.
April 1, 1981	Letter from applicant transmitting an affidavit of service for Amendment 30 to the FSAR.
April 9, 1981	Letter to applicant concerning meeting review to discuss and resolve open issues related to analyses of mechanical systems and piping.
April 10, 1981	Letter from applicant transmitting confirmation of NUREG-0737, Item III.A.1.2., implementation dates and justification for delays.
April 15, 1981	Letter to applicant concerning instrumentation and control systems drawings.
April 21, 1981	Letter from applicant advising that ASME Code Case N-292 will be used regarding deposition of weld metal before preparing ends for welding.
April 22, 1981	Letter from applicant responding to NRC letter of April 3, 1981, requesting additional copies of revised pages to Topical Report CE-1-A, Revision 16, regarding QA Program.
April 30, 1981	Letter to applicant requesting information on four potential problem areas in the design and analysis of the instrumentation and control systems.
May 4, 1981	Letter to applicant (generic) concerning qualification of inspection, examination, and testing and audit personnel.
May 5, 1981	Letter to applicant concerning natural circulation cooldown.
May 5, 1981	Letter to applicant (generic) concerning engineering evaluation of the H. B. Robinson reactor coolant system leak on January 29, 1981.
May 7, 1981	Letter to applicant requesting additional information for Geotechnical Engineering Branch.
May 15, 1981	Letter from applicant transmitting interim actions to be taken regarding review of controls for handling heavy loads.

May 18, 1981	Letter to applicant concerning guard training and qualification plan.
May 19, 1981	Letter from applicant transmitting responses to NRC March 26, 1981, letter containing questions on the control room design review.
May 19, 1981	Letter from applicant transmitting handouts shown at the May 14, 1981, safety assessment systems meeting.
May 20, 1981	Letter to applicant concerning use of ASME Code Cases N-295 and N-292.
May 28, 1981	Letter from applicant transmitting revised environmental qualification testing schedules and locations for facilities.
June 3, 1981	Letter to applicant concerning fire protection program.
June 3, 1981	Letter to applicant concerning ASME Code Case N-292.
June 3, 1981	Letter from applicant transmitting Revision 4 to Initial Piping Stress Analysis.
June 3, 1981	Letter to applicant (generic) concerning qualification of safety-related electrical equipment.
June 4, 1981	Letter to applicant (generic) concerning Institute of Nuclear Power Operations (INPO).
June 10, 1981	Letter to applicant concerning natural circulation training and testing.
June 11, 1981	Letter to applicant concerning staff evaluation of Item I.C.1 for Westinghouse facilities.
June 11, 1981	Letter from applicant transmitting Amendment 31 to the FSAR.
June 12, 1981	Letter from applicant transmitting three oversized drawings referenced in FSAR Amendment 31.
June 13, 1981	Letter to applicant (generic) concerning upgraded emergency plans.
June 15, 1981	Letter from applicant transmitting 300 oversized drawings containing details of instrumentation and control systems design.
June 22, 1981	Letter from applicant transmitting a response to NUREG-0612.

June 22, 1981	Letter from applicant transmitting a response to NRC letter on emergency procedures and training for station blackout events.
June 22, 1981	Letter to applicant requesting additional information for the independent structural analyses.
June 25, 1981	Letter from applicant transmitting an affidavit of service for Amendment 31 to the FSAR.
June 25, 1981	Letter from applicant transmitting Revision 17 to the QA Program.
June 25, 1981	Letter from applicant advising that all information and analyses regarding auxiliary feedwater system will be provided by August 15, 1981.
June 26, 1981	Letter from applicant transmitting revised pages to the QA Program.
June 30, 1981	Letter from applicant transmitting minutes of NRC Mechanical Engineering Branch meetings in Bethesda, Maryland (May 11-13, 1981) to review portions of the SER.
July 6, 1981	Letter to applicant (generic) concerning INPO evaluation reports.
July 8, 1981	Letter to applicant requesting additional information on radiation protection program.
July 9, 1981	Letter to applicant (generic) concerning privacy and proprietary material in emergency plans.
July 10, 1981	Letter to applicant requesting additional information on operator licensing.
July 12, 1981	Letter from applicant transmitting Revision 18 to the QA Program.
July 13, 1981	Letter from applicant advising that L. DelGeorge has been named Director of Nuclear Licensing and replaces J. S. Abel as principal correspondent for NRC communications.
July 13, 1981	Letter to applicant (generic) requesting verification that prompt emergency notification system will be installed.
July 24, 1981	Letter from applicant transmitting additional information regarding structural design.

July 31, 1981	Letter from applicant advising that they intend to use ASME Code Case N-235 regarding calibration checks using automated ultrasonic examination system.
July 31, 1981	Letter from applicant transmitting July 1981 revision of generating station emergency plan telephone directory.
July 31, 1981	Letter to applicant (generic) concerning steam generator overfill.
August 3, 1981	Letter from applicant transmitting Revisions 17 and 18 to QA Program.
August 7, 1981	Letter from applicant transmitting Revision 19 to the QA Program.
August 7, 1981	Letter to applicant (generic) concerning simulator examinations.
August 11, 1981	Letter from applicant transmitting corrected Revisions 17 and 18 to the QA Program.
August 17, 1981	Letter from applicant responding to NRC request for commitment to meet Regulatory Guide 1.58, Revision 1.
August 18, 1981	Letter from applicant transmitting Amendment 32 to the FSAR.
August 28, 1981	Letter from applicant transmitting responses to Radiological Assessment Branch regarding FSAR and NUREG-0737 questions.
August 28, 1981	Letter from applicant transmitting an affidavit of service for Amendment 32 to the FSAR.
August 31, 1981	Letter from applicant concerning fire protection.
September 2, 1981	Letter to applicant requesting additional information.
September 3, 1981	Letter from applicant transmitting Overpressure Protection Report for Byron/Braidwood nuclear power stations.
September 3, 1981	Letter to applicant concerning structural engineering review.
September 3, 1981	Letter to applicant requesting the responses to all outstanding requests so the FSAR review can be completed.
September 14, 1981	Letter to applicant concerning additional information for seismic margin.
September 16, 1981	Letter from applicant transmitting "Seismic Soil-Structure Interaction Analysis of Nuclear Power Plants."

September 18, 1981	Letter from applicant transmitting responses to NRC questions regarding the FSAR.
September 22, 1981	Letter from applicant transmitting responses to NUREG-0612, "Control of Heavy Loads."
September 29, 1981	Letter to applicant concerning schedule requirements.
October 5, 1981	Letter from applicant transmitting advance portion of Appendix E to FSAR regarding TMI requirements.
October 5, 1981	Letter from applicant transmitting response to FSAR questions on operator training and requalification.
October 5, 1981	Letter from applicant transmitting responses to NRC questions on FSAR Chapter 11 text and figure changes.
October 6, 1981	Letter from applicant transmitting schedule for NRC examination including simulator examinations for remainder of Calendar Years 81 and 82.
October 9, 1981	Letter from applicant transmitting the revised corporate emergency plan implementing procedures.
October 14, 1981	Letter from applicant transmitting responses to FSAR questions.
October 16, 1981	Letter from applicant transmitting Amendment 33 to the FSAR.
October 21, 1981	Letter to applicant concerning Appendix R of 10 CFR 50, "Fire Protection Rule."
October 23, 1981	Letter from applicant transmitting revised Table 2.3 to summary report "Static Dynamic and Relaxation Testing of Expansion Anchors."
October 27, 1981	Letter from applicant transmitting responses to NRC questions regarding FSAR text changes including additions to Appendix L.
October 27, 1981	Letter from applicant transmitting additional information on reliability of auxiliary feedwater system.
October 28, 1981	Letter to applicant concerning use of ASME Code Case N-235.
October 30, 1981	Letter to applicant concerning the auxiliary feedwater system.
October 30, 1981	Letter from applicant transmitting revision to the FSAR regarding radiological impact and shielding design for the volume reduction system.

November 3, 1981	Representatives from NRC, Chemical Engineering (CE), Bio Technology, Sargent & Lundy, ARD Corp., and Lawrence Livermore Laboratory met in Bethesda, Maryland, to discuss the detailed control room design review, (summary issued November 9, 1981).
November 4, 1981	Letter from applicant transmitting an affidavit of service for Amendment 33 to the FSAR.
November 9, 1981	Letter from applicant concerning an update of EQEE testing schedules.
November 10, 1981	Letter from applicant concerning 10 CFR 50.55(e) Deficiency Report No. 81-04.
November 10, 1981	Letter from applicant concerning auxiliary feedwater reliability.
November 12 and 13, 1982	Meeting with NRC, CE, and Sargent & Lundy regarding Instrumentation and Control Systems Branch review of FSAR (summary issued 12/7/81).
November 13, 1981	Letter from applicant concerning action items resulting from the structural design audit.
November 17, 1981	Letter to applicant requesting additional information for Containment Systems Branch.
November 17, 1981	Meeting with NRC, CE, and Torrey Pines Technology regarding auxiliary feedwater system (AFWS) reliability review (summary issued 12/3/81).
November 18, 1981	Letter from applicant concerning unresolved safety issues.
November 19, 1981	Letter from applicant transmitting Amendment No. 34 to the FSAR.
November 23, 1981	Letter from applicant concerning structural design audit.
November 25, 1981	Letter from applicant concerning responses to FSAR questions.
November 30, 1981	Letter to applicant concerning request for additional information concerning Materials Engineering Branch (MTEB) review.
November 30, 1981	Letter to applicant (generic) concerning NRC policy on low-level radwaste reduction.
December 2, 1981	Letter from applicant concerning schedule for responding to FSAR questions.

December 4, 1981	Letter to applicant concerning Reactor System Branch review.
December 4, 1981	Letter from applicant transmitting Amendment No. 35 to the FSAR.
December 7, 1981	Letter from applicant concerning control room design review.
December 9, 1981	Letter from applicant concerning loop blowdown force computation transmitting Westinghouse letter requesting withholding pursuant to 10 CFR 2.790.
December 9, 1981	Representatives from NRC and CE meet in Bethesda, Maryland, to discuss issues raised by the Instrumentation and Control Systems Branch (summary issued December 18, 1981).
December 15, 1981	Letter from applicant concerning auxiliary feedwater system reliability.
December 16, 1981	Letter from applicant transmitting an affidavit that service was provided for Amendment No. 34 to the FSAR.
December 16, 1981	Letter from applicant transmitting an affidavit that service was provided for Amendment No. 35 to the FSAR.
December 22, 1981	Letter to applicant (generic) regarding licensing operator written exams.
December 23, 1981	Letter from applicant concerning reponses to FSAR questions.
December 24, 1981	Letter from applicant concerning control of heavy loads.
December 28, 1981	Letter from applicant transmitting responses to FSAR questions.
December 29, 1981	Letter from applicant transmitting responses to FSAR questions.
December 30, 1981	Letter from applicant transmitting additional information requested by the Effluent Treatment Systems Branch.
December 31, 1981	Letter from applicant transmitting advance FSAR information.
January 2, 1982	Letter from applicant transmitting advance FSAR information.
January 4, 1982	Letter from applicant transmitting additional informa- requested by the Effluent Treatment Systems Branch.

January 5, 1982	Letter from applicant transmitting advance FSAR information.
January 6, 1982	Letter from applicant transmitting advance FSAR information.
January 9, 1982	Letter from applicant transmitting advance FSAR information.
January 12, 1982	Letter from applicant transmitting advance FSAR information.
January 12, 1982	Letter to applicant (generic) regarding license application review, scheduling, manpower, budgeting.
January 13, 1982	Letter from applicant transmitting advance FSAR information.
January 14, 1982	Letter from applicant transmitting advance FSAR information.
January 15, 1982	Letter from applicant transmitting advance FSAR information.
January 19, 1982	Letter from applicant transmitting advance FSAR information.
January 19, 1982	Letter to applicant requesting additional information on preservice inspection program.
January 20, 1982	Letter from applicant transmitting advance FSAR information.
January 21, 1982	Letter from applicant transmitting advance FSAR information.
January 22, 1982	Letter from applicant transmitting advance FSAR information.
January 26, 1982	Letter from applicant transmitting advance FSAR information.
January 26, 1982	Letter from applicant concerning Instrumentation and Control Systems Branch review meeting notes.
February 1, 1982	Letter from applicant concerning preservice inspection program.
February 3, 1982	Letter from applicant concerning advance FSAR information.
February 4, 1982	Letter from applicant transmitting Amendment No. 36 to the FSAR.

February 5, 1982	Letter from applicant concerning control of heavy loads.
February 8, 1982	Generic letter 82-02 to applicant concerning policy on factors causing fatigue of operating personnel at nuclear plants.
February 10, 1982	Letter to Westinghouse with carbon copy to Bryon/ Braidwood concerning withholding information transmitted on loop blowdown force computation.
March 9, 1982	Letter to applicant (generic) regarding INPO/program.
March 11, 1982	Letter to applicant concerning evacuation time estimates.
March 19, 1982	Letter to applicant (generic) concerning post-TMI requirements.
April 2, 1982	Letter to applicant concerning NUREG-0737, Item I.D.1.
April 2, 1982	Letter to applicant regarding Human Factors Engineering Branch control room review.
April 15, 1982	Generic Letter 82-08 to applicant transmitting NUREG-0909 relating to the Ginna tube rupture.
April 20, 1982	Generic Letter 82-09 to applicant concerning qualification of safety-related electrical equipment.
April 26, 1982	Letter from applicant concerning auxiliary building flooding.
April 26, 1982	Letter from applicant concerning reactor vessel forces and moments.
April 26, 1982	Letter from applicant concerning containment stresses.
April 27, 1982	Letter from applicant concerning steam generator tube inservice inspection.
May 4, 1982	Letter from applicant concerning turbine missile evaluation.
May 5, 1982	Letter from applicant concerning boration of reactor coolant system.
May 5, 1982	Letter from applicant concerning improved thermal design procedures.
May 6, 1982	Representatives from NRC, CE, and Sargent & Lundy met in Bethesda, Maryland to discuss applicant's program for seismic qualification of equipment (summary issued May 27, 1982).

May 12, 1982	Letter from applicant concerning testing of P-4 Interlock.
May 14, 1982	Letter from applicant concerning auxiliary feedwater systems reliability.
May 26, 1982	Letter from applicant concerning reactor vessel head temperature.
May 26, 1982	Letter from applicant concerning status of research programs.
June 7, 1982	Letter to applicant withholding from public disclosure the improved thermal design procedures transmitted by Westinghouse (CAW-82-15).
June 7, 1982	Letter from applicant concerning detection of inadequate core cooling.
June 7, 1982	Letter from applicant concerning locked rotor and shaft break transients.
June 10, 1982	Letter to applicant concerning loose-parts monitoring program.
June 15, 1982	Generic Letter 82-12 to applicant transmitting revised pages to NUREG-0737.
June 15, 1982	Letter from applicant concerning control room human factors review.
June 16, 1982	Letter from applicant transmitting Amendment No. 38 to the FSAR.
June 16, 1982	Letter from applicant concerning offsite dose calculation system.
June 16, 1982	Letter to applicant concerning SER scope and schedule.
June 16, 1982	Letter to applicant transmitting final duplicate design approval with topics considered outside scope of design approval listed.
June 17, 1982	Generic Letter 82-13 to applicant concerning NRC January 6, 1982, meeting with utilities concerning changes to operator examinations.
June 17, 1982	Letter from applicant concerning seismic qualification of equipment.
June 22, 1982	Letter to applicant concerning additional information on the volume reduction system.

June 24, 1982	Letter from applicant transmitting an affidavit of service for Amendment No. 38 to the FSAR.
June 28, 1982	Letter from applicant concerning advance FSAR information.
July 2, 1982	Letter from applicant concerning Rad/Chem Department Organization.
July 15, 1982	Letter to applicant concerning Auxiliary Systems Branch questions on safe shutdown report.
July 20, 1982	Letter from applicant concerning locked rotor and shaft break transients.
July 20, 1982	Letter from applicant concerning pump and valve operability assurance.
July 30, 1982	Letter to applicant requesting additional information on the water hammer prevention.
August 2, 1982	Letter to applicant transmitting a draft technical evaluation report based on responses to Section 2.1 of NRC generic letter, "Control of Heavy Loads," dated December 22, 1980.
August 4, 1982	Letter from applicant concerning fossil collection.
August 6, 1982	Letter from applicant concerning volume reduction system.
August 6, 1982	Letter from applicant concerning containment pressure analyses.
August 9, 1982	Letter from applicant concerning pipe support anchor plates.
August 9, 1982	Generic Letter 82-14 to applicant concerning 10 CFR Chapter 1 requirements.
August 10, 1982	Letter from applicant concerning high and moderate energy pipe break analyses.
August 10, 1982	Letter from applicant concerning main steam line break.
August 11, 1982	Letter from applicant concerning residual heat removal system.
August 12, 1982	Letter from applicant concering inservice inspection of snubbers.
August 13, 1982	Letter from applicant concerning detection of inadequate core cooling.

August 16, 1982	Letter from applicant concerning control room human factors review.
August 17, 1982	Letter from applicant concerning containment isolation.
August 18, 1982	Letter from applicant concerning turbine missile study.
August 25, 1982	Letter from applicant concerning vessel material surveillance capsules.
August 26, 1982	Letter from applicant concerning containment stresses.
August 26, 1982	Letter from applicant concerning postaccident sampling.
September 3, 1982	Letter from applicant concerning containment sump and atmosphere temperature monitoring.
September 9, 1982	Letter from applicant concerning water hammer prevention.
September 9, 1982	Letter from applicant concerning charging pump miniflow lines.
September 22, 1982	Letter from applicant concerning locked rotor and shaft break transients.
September 22, 1982	Letter from applicant concerning water hammer prevention.
September 23, 1982	Letter from applicant transmitting an affidavit of service for Amendment No. 39 to the FSAR.
September 30, 1982	Letter from applicant transmitting an application for construction permit extension. The applicant requests extension of the latest completion dates for CPPR-133 to April 30, 1987, and April 30, 1988, respectively.
October 1, 1982	Generic Letter 82-17 to applicant concerning the inconsistency between requirements of 10 CFR 50.54(t) and Standard Technical Specifications for performing audits of emergency preparedness programs.
October 4, 1982	Letter to applicant transmitting question Q321.43.
October 5, 1982	Letter to applicant transmitting a list of deviations from the Chemical Engineering Branch Technical Position 9.5.1 regarding fire protection.
October 5, 1982	Letter from applicant concerning pipe support anchor plates.
October 6, 1982	Generic Letter 82-21 to applicant concerning technical specification for fire protection.

October 7, 1982	Letter to applicant requesting additional information regarding venting of high points in reactor coolant system (RCS) method of depressurization to cold shutdown and depressurization following steam generator tube rupture events.
October 12, 1982	Generic Letter 82-18 to applicant concerning reactor operator and senior reactor operator requalification exams.
October 14, 1982	Letter from applicant concerning Rad/Chem Department Organization.
October 19, 1982	Letter from applicant concerning turbine missile study.
October 25, 1982	Letter from applicant concerning control of heavy loads.
October 26, 1982	Letter from applicant concerning pressurizer safety and relief valves.
October 26, 1982	Letter from applicant concerning postaccident sampling systems.
October 26, 1982	Generic Letter 82-20 to applicant concerning NUREG-0906 to be used until Regulatory Guide 1.70 is revised.
October 27, 1982	Letter to applicant requesting additional information to complete review of TMI Item II.F.2 regarding detection of inadequate core cooling.
October 27, 1982	Letter from applicant concerning main steamline break subcompartment analyses.
October 30, 1982	Generic Letter 82-23 to applicant concerning inconsistency between requirements of 10 CFR 73.40(g).
November 3, 1982	Letter from applicant concerning volume reduction system.
November 4, 1982	Letter from applicant concerning inservice testing of pumps and valves.
November 10, 1982	Letter from applicant concerning containment isolation.
November 15, 1982	Letter to applicant transmitting an Order extending the latest construction completion dates for Units 1 and 2 to April 30, 1987 and April 30, 1988, respectively.
November 29, 1982	Letter from applicant transmitting proprietary and non-proprietary Westinghouse information on the Turbine Missile Study and requesting previous information be withdrawn from the public domain.

December 17, 1982	Generic Letter 82-33 to applicant concerning Supplement 1 to NUREG-0737, "Requirements for Emergency Response Capability."
December 17, 1982	Letter from applicant transmitting an affidavit that service has been made for Amendment No. 40 to the FSAR.
December 22, 1982	Generic Letter 82-38 to applicant concerning recent developments for licensing examinations.
December 22, 1982	Generic Letter 82-39 to applicant concerning problems with submittals of 10 CFR 73.21 safeguards information for licensing review.
December 28, 1982	Generic Letter 82-30 to applicant concerning 10 CFR 50 production and utilization facilities.
December 29, 1982	Letter to Westinghouse withholding from public disclosure the Turbine Missile Study for Byron and Braidwood Stations submitted by Commonwealth Edison Company.
January 6, 1983	Letter from applicant concerning NUREG-0737, Item III.A.1.2, "Upgrade Emergency Support Facilities," status of implementation.
January 11, 1983	Letter to applicant requesting additional information Auxiliary Systems Branch, Meteorology and Effluent Treatment Branch, Structural Engineering Branch, Geosciences Branch, Hydrologic and Geotechnical Engineering Branch.
January 11, 1983	Generic Leter 83-01 to applicant concerning operator licensing examination site visit.
January 14, 1983	Letter to applicant requesting additional information for FSAR information submitted through Amendment 40.
January 14, 1983	Letter from applicant concerning minimum containment pressure analysis.
January 20, 1983	Letter from applicant concerning proposed Technical Specifications.
January 31, 1983	Generic Letter 83-06 to applicant concerning certificates and revised format for reactor operator and senior reactor operator licenses.
February 1, 1983	Generic Letter 83-04 to applicant concerning Supplement 1 to NUREG-0737.
February 1, 1983	Letter to applicant requesting additional information concerning Section 2.5 of FSAR through Amendment 40.

February 1, 1983	Letter to applicant transmitting comments on utility response to NRC letter of May 7, 1981, regarding geotechnical engineering.
February 1, 1983	Letter to applicant requesting additional information--Geosciences Branch, Geology Section.
February 2, 1983	Letter from applicant concerning notification of CECo Personnel Change (correspondence should now be addressed to Mr. Dennis L. Farrar).
February 8, 1983	Generic Letter 83-09 to applicant concerning CE Owners Group emergency procedures guideline program.
February 8, 1983	Letter from applicant concerning instrumentation for the detection of inadequate core cooling.
February 9, 1983	Letter from applicant concerning steam generator tube vibration.
February 14, 1983	Letter from applicant concerning additional FSAR information.
February 15, 1983	Letter from applicant concerning water hammer prevention.
February 16, 1983	Letter from applicant concerning reactor trip breaker test appeal meeting.
February 16, 1983	Generic Letter 83-07 to applicant concerning the Nuclear Waste Policy Act of 1982.
February 16, 1983	Letter to applicant concerning reactor trip breaker testing.
February 23, 1983	Letter from applicant transmitting FSAR Amendment 41.
February 24, 1983	Generic Letter 83-12 to applicant concerning NRC Form 398, "Personal Qualifications Statement - Licensee."
March 1, 1983	Letter from applicant concerning inservice inspection of snubbers.
March 2, 1983	Letter from applicant concerning ASME Code Case N-340.
March 2, 1983	Generic Letter 83-13 to applicant concerning clarification of surveillance requirements for HEPA filters and charcoal adsorbor units in STS on ESF cleanup systems.
March 3, 1983	Letter from applicant concerning Environmental Report--operating license stage.
March 7, 1983	Generic Letter 83-14 to applicant concerning definition of "Key Maintenance Personnel."

March 8, 1983	Letter from applicant concerning additional FSAR information.
March 8, 1983	Letter from applicant concerning additional FSAR information--Radiation/Chemistry and Station Health Physicist.
March 15, 1983	Letter to applicant requesting additional information--Qualification Program.
March 16, 1983	Letter from applicant concerning additional FSAR information.
March 18, 1983	Letter from applicant transmitting an affidavit of service for Amendment No. 41 to the FSAR.
March 23, 1983	Generic Letter 83-15 to applicant concerning implementation of Regulatory Guide 1.150.
March 23, 1983	Letter from applicant transmitting additional FSAR information.
March 30, 1983	Letter to applicant requesting additional information--Mechanical Engineering Branch.
March 30, 1983	Letter to applicant requesting additional information concerning pipe whip restraints.
April 8, 1983	Generic Letter 83-17 to applicant concerning integrity of requalification exams.
April 11, 1983	Letter to applicant concerning NRC's position on source range neutron flux monitoring.
April 11, 1983	Letter to applicant concerning positions regarding remaining open items concerning Appendix R criteria for postfire safe shutdown.
April 14, 1983	Letter from applicant responding to NUREG-0737, Supplement 1--Generic Letter No. 82-33.
April 14, 1983	Letter from applicant concerning additional information on Environmental Qualification Program.
April 18, 1983	Letter from applicant concerning additional FSAR information.
April 21, 1983	Letter from applicant concerning additional FSAR information.
April 22, 1983	Letter from applicant transmitting Revision 6 to the security plan for Braidwood.

April 26, 1983	Letter from applicant transmitting supplemental information concerning the control of heavy loads when new fuel is being stored in the spent fuel pit during construction phase of Byron/Braidwood Stations.
May 3, 1983	Letter to applicant concerning use of ASME Code Case N-340.
May 3, 4 & 5, 1983	Representatives from NRC, CEC and interested members of the public visit various points of interest at the Braidwood site.
May 4, 1983	Letter from applicant concerning additional information on crushable material.
May 6, 1983	Letter from applicant concerning additional FSAR information.
May 9, 1983	Letter from applicant concerning control room preliminary design assessment.
May 9, 1983	Generic Letter 83-20 to applicant concerning integrated scheduling for implementation of plant modes.
May 11, 1983	Generic Letter 83-21 to applicant concerning clarification of access to control procedures for law enforcement visits.
May 20, 1983	Letter to applicant concerning identification of safety-related mechanical equipment located in harsh environmental areas.
May 20, 1983	Letter to applicant concerning mechanical equipment environmental qualification program for Bryon/Braidwood.
May 23, 1983	Letter from applicant concerning additional FSAR information required by Regulatory Guide 1.8-1977.
May 23, 1983	Letter from applicant concerning additional FSAR information (responses to Questions 241.3 and 330.1 and revised response to Questions 241.5 and 241.7).
May 24, 1983	Letter to applicant requesting additional information for Site Analysis Branch.
May 24, 1983	Letter to applicant concerning environmental review of the Braidwood Station.
May 24, 1983	Letter to applicant concerning pipe line locations and frequency of munitions shipments.
May 31, 1983	Letter from applicant transmitting Amendment 42 to the FSAR.

June 14, 1983	Letter from applicant transmitting an affidavit for distribution of Amendment 42 to the FSAR.
June 16, 1983	Letter from applicant concerning power operational relief valves.
June 17, 1983	Letter from applicant concerning fire protection.
June 17, 1983	Letter from applicant concerning safety parameter display system.
June 17, 1983	Letter from applicant concerning additional FSAR information responding to Question 423.40 concerning suction conditions for the ESW pumps.
June 17, 1983	Letter from applicant concerning control of heavy loads.
June 20, 1983	Letter from applicant concerning spent fuel pool liner.
June 21, 1983	Letter from applicant concerning nonaccessible area filters and fuel-handling building filters.
June 21, 1983	Letter from applicant advising of reorganization of the State of Illinois Department of Nuclear Safety and requests the State official's title and address be changed.
June 21, 1983	Letter to applicant concerning moisture separation of heaters in filter systems serving nonaccessible area and fuel-handling building for SER review.
July 5, 1983	Generic Letter 83-26 to applicant concerning clarification of surveillance requirements for diesel fuel impurity.
July 7, 1983	Letter from applicant concerning containment leak rate testing.
July 15, 1983	Letter from applicant concerning additional FSAR information (responses to Questions 330.3 and 330.4, 241.3, 241.8 and 362.1).
July 15, 1983	Letter from applicant concerning Environmental Report--operating license stage.
July 15, 1983	Letter to applicant concerning draft SER for Braidwood (NRC transmits 2 xerox copies).
July 18, 1983	Letter from applicant concerning counterflow steam generator owners review group evaluation of Westinghouse proposed modifications to Model D4, D5, and E steam generators.

July 21, 1983	Letter to applicant concerning pipe whip restraint design for Byron/Braidwood--Status Report.
July 22, 1983	Letter to applicant concerning environmental review of Braidwood.
July 26, 1983	Letter from applicant transmitting an affidavit advising that Amendment 2 to the Environmental Report has been distributed.
August 3, 1983	Letter to applicant requesting additional information on the station physical security plan.
July 7, 1983	Letter from applicant transmitting an application for withholding proprietary information from public disclosure (information concerns Westinghouse Model D4, D5, and E steam generators).
July 27, 1983	Letter from applicant concerning topical report on benchmark of PWR nuclear design methods.
August 1, 1983	Letter from applicant concerning counterflow steam generator owners review group evaluation of Westinghouse proposed modifications to Model D4, D5, and E steam generators.
August 11, 1983	Letter to applicant concerning FSAR changes.
August 24-26, 1983	Representatives from NRC, CEC, members of the public, petitioners, and intervenors meet for open meeting at the site on August 24 and then NRC staff and CEC meet on August 25 and 26 in Joliet, Illinois (Braidwood plant site), to assess status of construction and completion schedules.
August 31, 1983	Letter from applicant transmitting Amendment 3 to the Braidwood Environmental Report--operating license stage.
September 2, 1983	Letter from applicant concerning schedules for submittals in response to NRC Generic Letter 83-28.
September 7, 1983	Letter to applicant concerning environmental review of Braidwood Station.
September 7, 1983	Letter from applicant concerning physical security plan.
September 8, 1983	Letter from applicant concerning pipe whip restraint energy absorbing material (EAM) test program.
September 13, 1983	Letter from applicant transmitting an affidavit for service of Amendment 3 to the Braidwood Environmental Report--operating license stage.

September 14, 1983	Letter to applicant requesting additional information regarding noise impact assessment for Braidwood Station.
September 16, 1983	Letter from applicant concerning charging pump deadheading.
September 19, 1983	Letter from applicant transmitting Revision 7 of the Braidwood security plan.
September 20, 1983	Letter to applicant requesting additional information on fire protection.
September 23, 1983	Letter from applicant concerning additional FSAR information.
September 23, 1983	Letter from applicant transmitting Amendment 43 to the FSAR.
September 28, 1983	Letter from applicant transmitting an affidavit for service of Amendment 43 to the FSAR.
September 30, 1983	Letter from applicant responding to unresolved issues contained in draft SER, NUREG-1002.
October 6, 1983	Letter from applicant transmitting an advance copy of the majority of Amendment 4 to the Braidwood Station Environmental Report.
October 12, 1983	Letter to applicant requesting additional information concerning the environmental review of Braidwood Station.
October 14, 1983	Letter from applicant transmitting Amendment 4 to the Environmental Report--operating license stage.

APPENDIX B

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APPENDIX C

NUCLEAR REGULATORY COMMISSION UNRESOLVED SAFETY ISSUES

This Appendix is the same as NUREG-0876 except for the following status changes:

C.5 Discussion of Tasks as They Relate to Braidwood Units 1 and 2

A-11 Reactor Vessel Materials Toughness

This issue has been resolved by issuance of NUREG-0744, "Resolution of the Task A-11, Reactor Vessel Materials Toughness Safety Issue," Volumes I and II, Revision 1.

A-12 Fracture Toughness of PWR Steam Generator and Reactor Coolant Pump Supports

This issue has been resolved by issuance of NUREG-0577, "Potential for Low Fracture Toughness and Lamellar Tearing in PWR Steam Generator and Reactor Coolant Pump Supports," Revision 1. However, it should be noted that the requirements for resolving this issue are applied only to new construction permit (CP) and preliminary design approval (PDA) plants. Therefore, this issue is not applicable to Braidwood Units 1 and 2.

A-48 Hydrogen Control Measures and Effects of Hydrogen Burns on Safety Equipment

This issue is limited to plants with pressure suppression containments, i.e., an ice condenser for PWR plants and Mark I, II, and III containments for BWR plants. The containment for Braidwood is a large dry containment. Therefore, this issue is not applicable to Braidwood Units 1 and 2.

APPENDIX D

EMERGENCY PREPAREDNESS EVALUATION REPORT

This review will be completed pending submittal of an onsite and offsite emergency plan by the applicant.

APPENDIX E

REPORT ON THE SEISMOLOGICAL ASPECTS OF THE BRAIDWOOD STATION, UNITS 1 AND 2, BY LAWRENCE LIVERMORE NATIONAL LABORATORY

The Lawrence Livermore National Laboratory has been requested by the staff to perform a probabilistic hazard analysis on the seismological aspects of Braidwood Station. This report will be included as Appendix E in a future supplement to this SER.

APPENDIX F

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This SER is a product of the NRC staff and its consultants. The NRC staff members listed below were principal contributors to this report. A list of consultants follows the list of staff members.

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APPENDIX G

FINAL DUPLICATE DESIGN APPROVAL (FDDA) FOR THE BYRON STATION DUPLICATE DESIGN

COMMONWEALTH EDISON COMPANY
DOCKET NOS. 50-454, 455, 456, 457
BYRON STATION DUPLICATE DESIGN
FINAL DUPLICATE DESIGN APPROVAL (FDDA)

(1) The Commonwealth Edison Company has submitted to the Nuclear Regulatory Commission's (NRC) staff for its review a proposed design for major portions of a nuclear power reactor and associated balance of plant which may be duplicated or replicated at different sites by one or more utility applicants. The design is decribed in the Byron/Braidwood Station Final Safety Analysis Report (FSAR) along with 37 amendments thereto.

(2) The Byron/Braidwood FSAR contains standardized design information in accordance with 10 CFR Part 50, Appendix N, for licenses to operate power reactors of duplicate design at multiple sites. The duplicate design encompasses the nuclear steam supply systems, balance of plant systems as well as associated auxiliary systems. The Byron reference design is designed to operate at a core thermal power level of 3425 megawatts.

(3) The Bryon reference design has been reviewed by the NRC staff and by the Advisory Committee on Reactor Safeguards (ACRS) for specific application to the Byron site in Ogle County, Illinois. The results of the NRC staff evaluation of the Byron references design are presented in the Safety Evaluation Report (SER) (NUREG-0876) dated February 1982. The ACRS comments are set forth in its letter of March 9, 1982 (Appendix G of NUREG-0876, Supplement 1).

(4) Based on its review, and the findings set forth in Section 23 of the SER, the NRC staff has concluded that subject to the conditions set forth herein, the information provided in the Byron/Braidwood FSAR with respect to the major portions of the design encompassed by the Byron/Braidwood FSAR as they apply to the Byron Station, complies with the requirements of 10 CFR Part 50, Appendix N and the NRC's "Statement on Standardization of Nuclear Power Plants," dated August 31, 1978, (43FR38954) and is acceptable for incorporation by reference in applications for operating licenses. In accordance with the above and subject to the conditions set forth herein the approved Byron standardized design shall be utilized and relied upon by the staff in its review of duplicated or replicated plants which incorporates by reference the approved design, unless there exists significant new information which substantially affects the determination set forth in this Final Duplicate Design Approval or other good cause.

(5) Duplicate or Replicate plant applications which incorporate the standard Byron design may be reviewed to evaluate the compatibility of the design with site-related characteristics; changes to the duplicate plant design; the status of matters identified for the duplicate plant design in the SER and Supplements thereto, or matters subsequently identified by the Advisory Committee on Reactor Safeguards, or during public hearings on

applications referencing the duplicate plant design. Differences between major contractors and the manner in which any duplicate plant conforms to the Commission's regulations which have become effective since the issuance of the FDDA may also be reviewed by staff for duplicate or replicate plants which incorporate the Byron design.

(6) This Final Duplicate Design Approval is applicable to those systems and design features of the design described and evaluated in Sections 1 through 18 of the Byron SER. Those systems listed in Appendix A to this FDDA are not within the scope of the duplicate design and are not included in this Final Duplicate Design Approval.

(7) This Final Duplicate Design Approval and all constructions permit and operating license applications incorporating it by reference, are subject to all applicable provisions of the Atomic Energy Act of 1954, as amended, and the rules and regulations and Orders of the Commission now or hereafter in effect.

(8) This Final Duplicate Design Approval does not constitute a commitment to issue a permit or license or in any way affect the authority of the

Commission, Atomic Safety and Licensing Appeal Board, Atomic Safety Licensing Boards and other presiding officers in any proceeding under Subpart G of 10 CFR Part 2.

Appendix G-A

Systems/Topics Not Included in FDDA
(in SER Sections 2 through 18)

I. Site-related matters, including:

1. Site investigation program including geography and demography, geology-seismology, foundation engineering, hydrology, and meteorology (Section 2.1 - 2.5);

2. Site-related design criteria, such as wind and tornado loadings (Section 3.3), flood level (Section 3.4), missile protection (Section 3.5), and seismic design (Section 3.7);

3. Radiological consequences of accidents (Sections 6.4 and 15.4);

4. Radioactive releases - liquid (Section 11.2) and gaseous effluents (Section 11.3).

II. Changes from the Byron Station design, including:

1. Offsite power systems (Section 8.2);

2. Water systems - ultimate heat sink (Section 9.2);

3. Pumphouse ventilation system (Section 9.4.6), pumphouse diesel generator fuel oil system (Section 9.5);

4. Circulating water system (Section 10.4.5).

III. Utility-oriented safety-related matters, including:

1. ALARA policy (Sections 12.1.1 and 12.1.3);

2. Radiation protection organization and procedures (Sections 12.5.1 and 12.5.3);

3. Organizational structure (Section 13.1);

4. Operator training (Section 13.2);

5. Emergency plan (Section 13.3)

6. Operational review (Section 13.4);

7. Plant procedures (Section 13.5);

8. Industrial security (Section 13.6);

9. Quality assurance (Section 17).

IV. Other items, including

1. RCPB materials and reactor vessel materials (Sections 5.2 and 5.3);
2. Inservice inspection program (Section 6.6);
3. Fire protection program (Section 9.5.1);
4. Secondary water chemistry program (Section 10.3.3);
5. Plant shielding (TMI Item II.B.2) (Section 12.3);
6. Initial test program (Section 14);
7. Human factors review (Section 18).

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16. ABSTRACT *(200 words or less)*

This report provides the results of the NRC staff review of Commonwealth Edison Company's application for licenses to operate the Braidwood Station, Units 1 and 2. The facility is located in northeastern Illinois within Reed Township, Will County, Illinois. Subject to favorable resolution of the items discussed in the Safety Evaluation Report, the staff concludes that the facility can be operated by the applicant without endangering the health and safety of the public.

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