ATTACHMENT 3

PROPOSED TECHNICAL SPECIFICATIONS

Technical Specification 5.0

"DESIGN FEATURES"

5.0 DESIGN FEATURES

5.1 SITE

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5.1.A [INTENTIONALLY BLANK]

Low Population Zone

5.1.B The low population zone shall be as shown in Figure 5.1.B-1.

Radioactive Gaseous Effluents

5.1.C Information regarding radioactive gaseous effluents shall be located in the OFFSITE DOSE CALCULATION MANUAL.

Radioactive Liquid Effluents

5.1.D Information regarding radioactive liquid effluents shall be located in the OFFSITE DOSE CALCULATION MANUAL.

FIGURE 5.1.A-1

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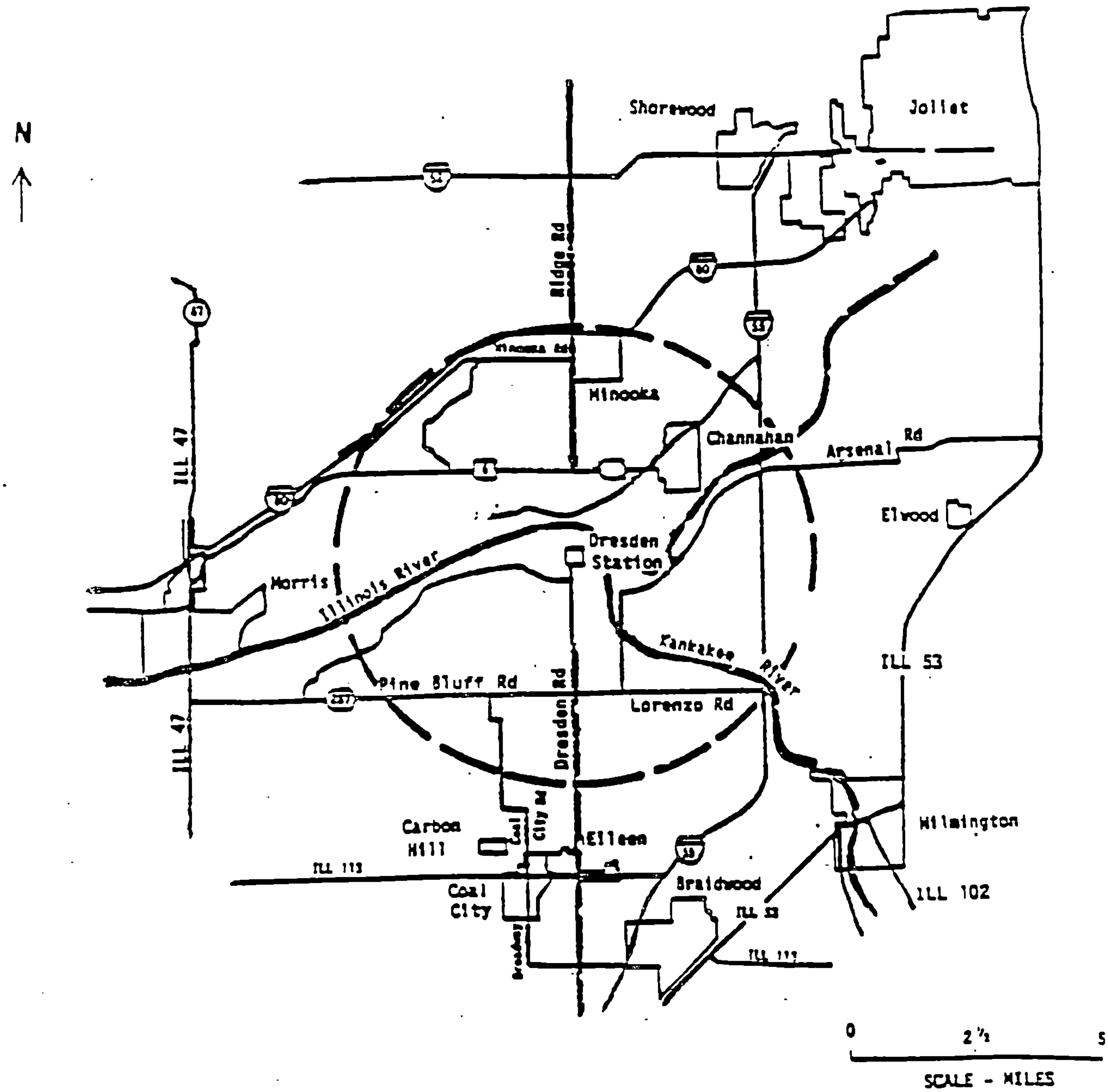
FIGURE 5.1.B-1

LOW POPULATION ZONE



5.0 DESIGN FEATURES

5.2 CONTAINMENT

Configuration

5.2.A The primary containment is a steel lined concrete structure consisting of a drywell and suppression chamber. The drywell is a steel structure composed of a spherical lower portion, a cylindrical middle portion, and a hemispherical top head. The drywell is attached to the suppression chamber through a series of downcomer vents. The drywell has a minimum free air volume of 158,236 cubic feet. The suppression chamber has an air region of 116,300 to 112,800 cubic feet and a water region of 116,300 to 119,800 cubic feet.

Design Temperature and Pressure

5.2.B The primary containment is designed and shall be maintained for:

1. Maximum internal pressure: 62 psig.
2. Maximum internal temperature: drywell 281°F.
suppression pool 281°F.
3. Maximum external pressure: drywell 2 psig.
suppression pool 1 psig.

Secondary Containment

5.2.C The secondary containment consists of the Reactor Building and a portion of the main steam tunnel and has a minimum free volume of 5,760,000 cubic feet.

5.0 DESIGN FEATURES

5.3 REACTOR CORE

Fuel Assemblies

5.3.A The reactor core shall contain 724 fuel assemblies. Each assembly consists of a matrix of Zircaloy clad fuel rods with an initial composition of natural or slightly enriched uranium dioxide as fuel material and water rods. Limited substitutions of zirconium alloy or stainless steel filler rods for fuel rods, in accordance with NRC-approved applications of fuel rod configurations, may be used. Fuel assemblies shall be limited to those fuel designs that have been analyzed with applicable NRC staff-approved codes and methods, and shown by tests or analyses to comply with all fuel safety design bases. A limited number of lead test assemblies that have not completed representative testing may be placed in non-limiting core regions.

Control Rod Assemblies

5.3.B The reactor core shall contain 177 cruciform shaped control rod assemblies. The control material shall be boron carbide powder (B_4C) and/or hafnium metal. The control rod assembly shall have a nominal axial absorber length of 143 inches.

5.4 REACTOR COOLANT SYSTEM

Design Pressure and Temperature

5.4.A The reactor coolant system is designed and shall be maintained:

1. In accordance with the code requirements specified in Section 4.1 of the FSAR, with allowance for normal degradation pursuant to the applicable Surveillance Requirements,

2. For a pressure and temperature of:

   a. 1175 psig at 565°F on the suction side of the recirculation pump.

   b. 1450 psig at 575°F from the recirculation pump discharge to the outlet side of the discharge shutoff valve.

   c. 1325 psig at 580°F from the discharge shutoff valve to the jet pumps.

Volume

5.4.B The total water and steam volume of the reactor vessel and recirculation system is approximately 14,626 cubic feet at 68°F.

5.0 DESIGN FEATURES

5.5 [INTENTIONALLY BLANK]

5.5.A [INTENTIONALLY BLANK]

5.0 DESIGN FEATURES

5.6 FUEL STORAGE

Criticality

5.6.A The spent fuel storage racks are designed and shall be maintained with:

1. A k_{eff} equivalent to $\leq$0.95 when flooded with unborated water, including all calculational uncertainties and biases as described in Section 9.1 of the FSAR.
2. A nominal 6.30 inch center-to-center distance between fuel assemblies placed in the storage racks.

Drainage

5.6.B The spent fuel storage pool is designed and shall be maintained to prevent inadvertent draining of the pool below elevation 589' 2.5".

Capacity

5.6.C The spent fuel storage pool is designed and shall be maintained with a storage capacity limited to no more than 3537 fuel assemblies.

5.0 DESIGN FEATURES

5.1 SITE

[INTENTIONALLY BLANK]

5.1.A [INTENTIONALLY BLANK]

Low Population Zone

5.1.B The low population zone shall be as shown in Figure 5.1.B-1.

Radioactive Gaseous Effluents

5.1.C Information regarding radioactive gaseous effluents shall be located in the OFFSITE DOSE CALCULATION MANUAL.

Radioactive Liquid Effluents

5.1.D Information regarding radioactive liquid effluents shall be located in the OFFSITE DOSE CALCULATION MANUAL.

FIGURE 5.1.A-1

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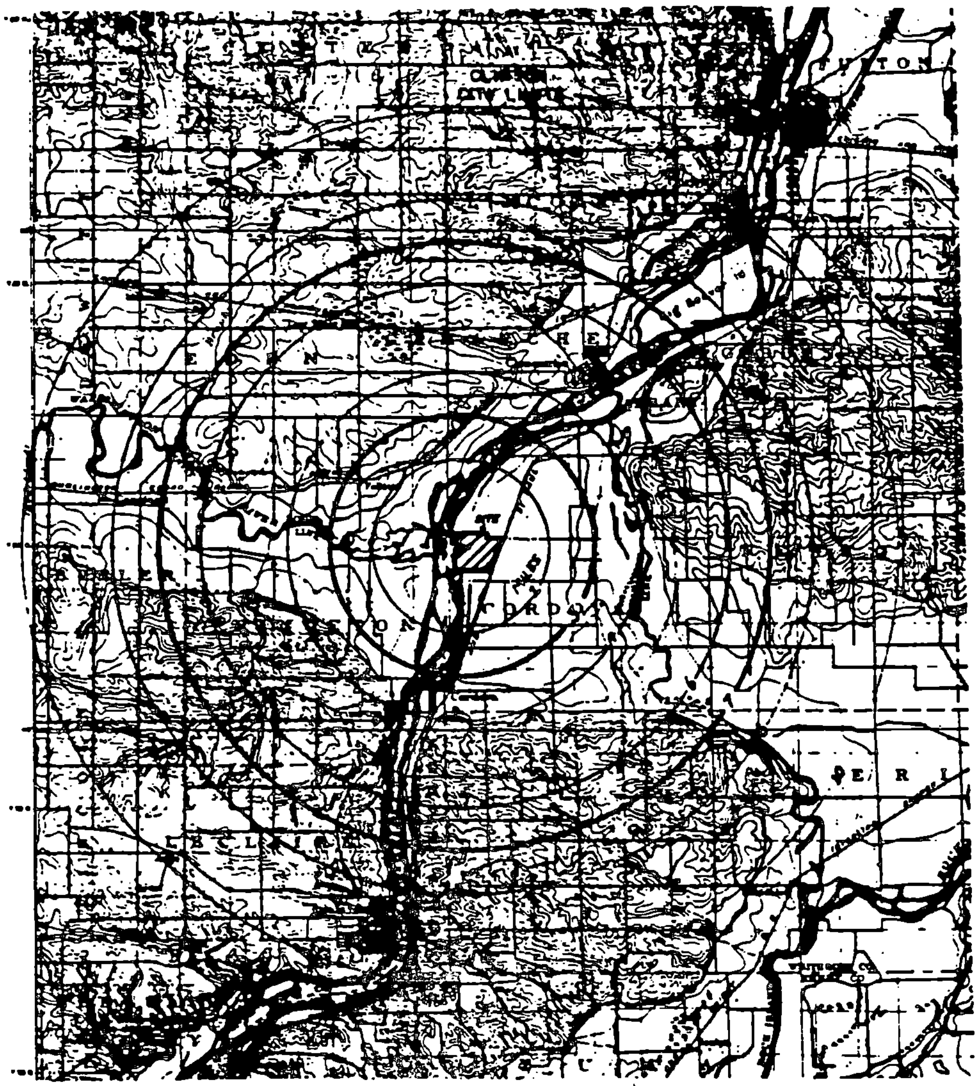
FIGURE 5.1.B-1

LOW POPULATION ZONE



POPULATION CENTER DISTANCE, 7 MILE RADIUS
LOW POPULATION ZONE, 3 MILE RADIUS

5.0 DESIGN FEATURES

5.2 CONTAINMENT

Configuration

5.2.A The primary containment is a steel lined concrete structure consisting of a drywell and suppression chamber. The drywell is a steel structure composed of a spherical lower portion, a cylindrical middle portion, and a hemispherical top head. The drywell is attached to the suppression chamber through a series of downcomer vents. The drywell has a minimum free air volume of 158,236 cubic feet. The suppression chamber has an air region of 120,800 to 117,300 cubic feet and a water region of 111,500 to 115,000 cubic feet.

Design Temperature and Pressure

5.2.B The primary containment is designed and shall be maintained for:

1. Maximum internal pressure: 56 psig.
2. Maximum internal temperature: drywell 281°F.
suppression pool 281°F.
3. Maximum external pressure: drywell 2 psig.
suppression pool 1 psig.

Secondary Containment

5.2.C The secondary containment consists of the Reactor Building and a portion of the main steam tunnel and has a minimum free volume of 5,760,000 cubic feet.

5.0 DESIGN FEATURES

5.3 REACTOR CORE

Fuel Assemblies

5.3.A The reactor core shall contain 724 fuel assemblies. Each assembly consists of a matrix of Zircaloy clad fuel rods with an initial composition of natural or slightly enriched uranium dioxide as fuel material and water rods. Limited substitutions of zirconium alloy, in accordance with NRC-approved applications of fuel rod configurations, may be used. Fuel assemblies shall be limited to those fuel designs that have been analyzed with applicable NRC staff-approved codes and methods, and shown by tests or analyses to comply with all fuel safety design bases. A limited number of lead test assemblies that have not completed representative testing may be placed in non-limiting core regions.

Control Rod Assemblies

5.3.B The reactor core shall contain 177 cruciform shaped control rod assemblies. The control material shall be boron carbide powder (B_4C) and/or hafnium metal. The control rod assembly shall have a nominal axial absorber length of 143 inches.

5.4 REACTOR COOLANT SYSTEM

Design Pressure and Temperature

5.4.A The reactor coolant system is designed and shall be maintained:

1. In accordance with the code requirements specified in Section 5 of the FSAR, with allowance for normal degradation pursuant to the applicable Surveillance Requirements,

2. For a pressure and temperature of:

   a. 1175 psig at 565°F on the suction side of the recirculation pump.

   b. 1450 psig at 575°F from the recirculation pump discharge to the outlet side of the discharge shutoff valve.

   c. 1325 psig at 580°F from the discharge shutoff valve to the jet pumps.

Volume

5.4.B The total water and steam volume of the reactor vessel and recirculation system is approximately 15,679 cubic feet at 68°F.

5.0 DESIGN FEATURES

5.5 [INTENTIONALLY BLANK]

5.5.A [INTENTIONALLY BLANK]

5.6 FUEL STORAGE

Criticality

5.6.A The spent fuel storage racks are designed and shall be maintained with:

1. A k_{eff} equivalent to $\leq$0.95 when flooded with unborated water, including all calculational uncertainties and biases as described in Section 9.1 of the FSAR.

2. A nominal 6.22 inch center-to-center distance between fuel assemblies placed in the storage racks.

Drainage

5.6.B The spent fuel storage pool is designed and shall be maintained to prevent inadvertent draining of the pool below elevation 666' 8.5".

Capacity

5.6.C The spent fuel storage pool is designed and shall be maintained with a storage capacity limited to no more than 3657(Unit 1)/3897(Unit 2) fuel assemblies.

ATTACHMENT 4

EXISTING TECHNICAL SPECIFICATIONS

Technical Specification 5.0

"<u>DESIGN FEATURES</u>"

ATTACHMENT 4

DELETION OF CURRENT TECHNICAL SPECIFICATIONS

This technical specification amendment will replace the current sections 5.0, Design Features, for the Dresden Unit 2 and Unit 3 Technical Specifications. The specifications are replaced in its entirety with revised pages that combine the Unit 2 and Unit 3 specifications.

Delete the following pages:

DPR - 19	DPR - 25
5-1	5-1
5-2	5-2

ATTACHMENT 4

DELETION OF CURRENT TECHNICAL SPECIFICATIONS

This technical specification amendment will replace the current sections 5.0, Design Features, for the Quad Cities Unit 1 and Unit 2 Technical Specifications. The specifications are replaced in its entirety with revised pages that combine the Unit 1 and Unit 2 specifications.

Delete the following pages:

DPR - 29	DPR - 30
5.0-1	5.0-1

ATTACHMENT 5

DRESDEN 2/3 DIFFERENCES

Technical Specification 5.0

"DESIGN FEATURES"

ATTACHMENT 5

COMPARISON OF DRESDEN UNIT 2 AND UNIT 3 TECHNICAL SPECIFICATIONS FOR THE IDENTIFICATION OF TECHNICAL DIFFERENCES

SECTION 5.0
"DESIGN FEATURES"

Commonwealth Edison has conducted a comparison review of the Dresden Unit 2 and Unit 3 Technical Specifications to identify any technical differences in support of combining the Technical Specifications into one document. The intent of the review was not to identify any differences in presentation style (e.g. table formats, use of capital letters, etc.), punctuation or spelling errors, but rather to identify areas which the Technical Specifications are technically or administratively different.

The review of Section 5.0 "Design Features" did not reveal any technical differences.

ATTACHMENT 5

COMPARISON OF QUAD CITIES UNIT 1 AND UNIT 2 TECHNICAL SPECIFICATIONS FOR THE IDENTIFICATION OF TECHNICAL DIFFERENCES

SECTION 5.0
"DESIGN FEATURES"

Commonwealth Edison has conducted a comparison review of the Quad Cities Unit 1 and Unit 2 Technical Specifications to identify any technical differences in support of combining the Technical Specifications into one document. The intent of the review was not to identify any differences in presentation style (e.g. table formats, use of capital letters, etc.), punctuation or spelling errors, but rather to identify areas which the Technical Specifications are technically or administratively different.

The review of Section 5.0 "Design Features" did not reveal any technical differences.

ATTACHMENT 6

SIGNIFICANT HAZARDS CONSIDERATIONS AND ENVIRONMENTAL ASSESSMENT EVALUATION

Technical Specification 5.0

"DESIGN FEATURES"

ATTACHMENT 6

EVALUATION OF SIGNIFICANT HAZARDS CONSIDERATION

Commonwealth Edison has evaluated this proposed amendment and determined that it involves no significant hazards consideration. According to 10 CFR 50.92(c), a proposed amendment to an operating license involves no significant hazards consideration if operation of the facility, in accordance with the proposed amendment, would not:

1) Involve a significant increase in the probability or consequences of an accident previously evaluated; or

2) Create the possibility of a new or different kind of accident from any accident previously evaluated; or

3) Involve a significant reduction in a margin of safety.

The proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated because:

> The proposed changes to Technical Specification Section 5.0 will delete those portions of the present requirements which are not used in the safety analysis of an accident previously evaluated. Those portions which do reflect parameters which have been determined necessary and sufficient to assure the assumptions and techniques used in the safety analysis remain valid. The added requirements follow STS, 10CFR 20, and GL 89-01 guidelines that are in use at many operating BWRs with similar design and operating configurations as Dresden and Quad Cities Stations. Therefore, the proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

Create the possibility of a new or different kind of accident from any previously evaluated because:

> The proposed changes for Dresden and Quad Cities Technical Specification Section 5.0 are based on present provisions and STS, 10 CFR 20, and gl 89-01 guidelines or later operating BWR plants' NRC accepted changes. These proposed changes have been reviewed for acceptability at the Dresden and Quad Cities Nuclear Stations considering similarity of system or component design versus the STS or later operating BWRs. No new modes of operation are introduced by the proposed changes, considering the acceptable operational modes in present specifications, the STS, or later operating BWRs. The proposed changes do not modify existing setpoints or design assumptions for system or component operation. The proposed changes add new requirements that are not in the present technical specifications, adopt requirements that have been used successfully at other operating BWRs with designs similar to Dresden and Quad Cities, or omit STS provisions that are

not applicable or no longer required. The proposed changes maintain at least the present level of requirements which still provide a proven acceptable level of operability. Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any previously evaluated.

Involve a significant reduction in the margin of safety because:

The proposed changes to Technical Specification Section 5.0 implement present requirements, the intent of present requirements, or provisions that have been found acceptable for use on other operating BWRs with system designs similar to that at Dresden and Quad Cities. The proposed changes are intended to include the design features which have been determined to be necessary and sufficient to maintain acceptable levels of safe operation. The proposed changes have been evaluated and found to be acceptable for use at Dresden and Quad Cities based on system design, safety analysis requirements and operational performance. Since the proposed changes are based on NRC accepted provisions at other operating plants that are applicable at Dresden and Quad Cities and maintain necessary levels of system, component or parameter operability, the proposed changes do not involve a significant reduction in the margin of safety.

Conclusion

Guidance has been provided in 51 FR 7744 for the application of standards to license change requests for determination of the existence of significant hazards considerations. This document provides examples of amendments which are not likely considered to involve significant hazards considerations.

This proposed amendment does not involve a significant relaxation of the criteria used to establish safety limits, a significant relaxation of the bases for the limiting safety system settings or a significant relaxation of the bases for the limiting conditions for operations. The proposed amendments most closely fit the example of changes that constitute an additional limitation, restriction, or control not presently included in the Technical Specifications, e.g., a more stringent surveillance requirement (e.(ii) of 51 FR 7751) and/or a purely administrative change (e.(i) of 51 FR 7751). Therefore, based on the guidance established in 10 CFR 50.92(c), the proposed changes do not constitute a significant hazards consideration.

ATTACHMENT 6

ENVIRONMENTAL ASSESSMENT STATEMENT APPLICABILITY REVIEW

Commonwealth Edison has evaluated the proposed amendment against the criteria for the identification of licensing and regulatory actions requiring environmental assessment in accordance with 10 CFR 51.20. It has been determined that the proposed changes meet the criteria for a categorical exclusion as provided under 10 CFR 51.22 (c)(9). This conclusion has been determined because the changes requested do not pose significant hazards consideration or do not involve a significant increase in the amounts, and no significant changes in the types, of any effluent that may be released offsite. Additionally, this request does not involve a significant increase in individual or cumulative occupational radiation exposure. Therefore, the Environmental Assessment Statement is not applicable for these changes.