ATTACHMENT B

PROPOSED CHANGES TO APPENDIX A
TECHNICAL SPECIFICATIONS FOR
FACILITY OPERATING LICENSES
NPF-37, NPF-66, NPF-72 AND NPF-77

Revised Page

6-6

6.3 UNIT STAFF QUALIFICATIONS

6.3.1 Each member of the unit staff shall meet or exceed the minimum qualifications of ANSI N18.1-1971, except for the Health Physics Supervisor or Lead Health Physicist, who shall meet or exceed the qualifications of Regulatory Guide 1.8, September 1975, for a Radiation Protection Manager. ~~The licensed Operators and Senior Operators shall also meet or exceed the minimum qualifications of the supplemental requirements specified in Sections A and C of Enclosure 1 of the March 28, 1980 NRC letter to all licensees.~~

6.4 TRAINING

6.4.1 A retraining and replacement training program for the unit staff shall be maintained under the direction of the Production Training Department and shall meet or exceed the requirements and recommendations of Section 5 of ANSI/ANS 3.1-1978 ~~and Appendix A of 10 CFR Part 55 and the supplemental requirements specified in Sections A and C of Enclosure 1 of the March 28, 1980 NRC letter to all licensees~~, and shall include familiarization with relevant industry operational experience from the program managed by Quality Programs and Assessment.

6.5 REVIEW INVESTIGATION AND AUDIT

The Review and Investigative Function and the Audit Function of activities affecting quality during facility operations shall be constituted and have the responsibilities and authorities outlined below.

OFFSITE

6.5.1 The Superintendent of the Offsite Review and Investigative Function shall be appointed by the Manager of Quality Assurance/Nuclear Safety (QA/NS) responsible for nuclear activities. The corporate audit function shall be the responsibility of the Manager of QA/NS and shall be independent of operations.

The Manager of QA/NS reports directly to the Chief Executive Officer and has the responsibility to set Corporate Policy for both the areas of Quality Assurance and Nuclear Safety. Policy is promulgated through a central policy committee directed by the Manager of QA/NS. The Manager of QA/NS has the responsibility for the performance of periodic audits of each nuclear station and corporate department to determine that QA/NS policy is being carried out.

a. Offsite Review and Investigative Function

The Superintendent of the Offsite Review and Investigative Function shall: (1) provide directions for the review and investigative function and appoint a senior participant to provide appropriate direction, (2) select each participant for this function, (3) select a complement of more than one participant who collectively possess background and qualifications in the subject matter under review to provide comprehensive interdisciplinary review coverage under this function, (4) independently review and approve the findings and recommendations developed by personnel performing the review

6.2.4 SHIFT TECHNICAL ADVISOR (Continued)

To assure capability for performance of all STA functions:

(1) The shift foreman (SRO) shall participate in the SCRE shift relief turnover.

(2) During the shift, the shift engineer and the shift foreman (SRO) shall be made aware of any significant changes in plant status in a timely manner by the SCRE.

(3) During the shift, the shift engineer and the shift foreman (SRO) shall remain abreast of the current plant status. The shift foreman (SRO) shall return to the control room two or three times per shift, where practicable, to confer with the SCRE regarding plant status. Where not practicable to return to the control room, the shift foreman (SRO) shall periodically check with the SCRE for a plant status update. The shift foreman (SRO) shall not abandon duties original to reactor operation, unless specifically ordered by the shift engineer.

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ATTACHMENT C

SUMMARY OF THE EVALUATION OF SIGNIFICANT HAZARDS CONSIDERATIONS

Commonwealth Edison Company (CECo) has evaluated this proposed amendment and determined that it involves no significant hazards considerations. According to Title 10, Code of Federal Regulations, Part 50, Section 92, Paragraph (c) [10 CFR 50.92(c)], a proposed amendment to an operating license involves no significant hazards considerations 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 basis for this determination of no significant hazards considerations is presented below.

The proposed amendment would delete the reference to the March 28, 1980 NRC letter to all licensees in Specifications 6.3.1 and 6.4.1. Additionally, a reference to Appendix A of 10CFR Part 55 is being removed. This appendix was previously incorporated into the body of 10CFR Part 55.

The purpose of the March 28, 1980 NRC letter to all licensees was to set forth revised criteria to be used by the NRC staff in evaluating reactor operator training and licensing under the regulations in effect at the time of the letter. These revised criteria were established based on the Commission review of the Three Mile Island Unit 2 accident which occurred on March 28, 1979. The letter stated that the Commission review in the area of operator training and qualification would continue and could be expected to result in additional criteria being established. The letter also stated that final requirements would be established through rulemaking proceedings.

The continued Commission review in the area of operator training and qualification resulted in revisions to 10 CFR 55 and the issuance of NUREG-1021, Operator Licensing Examiner Standards, which provides guidance regarding the implementation of 10 CFR 55 requirements. These requirements supercede those delineated in the March 28, 1980 NRC letter to all licensees.

Since the initial license examinations at both Byron and Braidwood Stations, operator training and qualification has been evaluated against the requirements of NUREG-1021.

Additionally, a reference to Appendix A of 10CFR Part 55 is being removed. This appendix was previously incorporated into the body of 10CFR Part 55. These changes are considered to be administrative in nature.

The Chapter 15 Analyses are unaffected by this request. The proposed change will have no effect on the sequence of events leading to the initiation of a transient. The chapter 15 Analyses assume a range of mechanical failures (i.e. pipe breaks, loss of heat sink, etc.) as the initiating event for the purpose of demonstrating the continued protection of the public. These failures are not assumed to be triggered by personnel actions. As such, the specific training requirements for the plant staff are not related to the probability of the occurrence of a previously analyzed accident.

The consequences of any previously analyzed accidents are not increased. The licensed Operators and Senior Operators at Byron and Braidwood Stations are trained in accordance with a Systems Approach to Training based program that has been accredited by the Institute for Nuclear Power Operations and evaluated against the requirements of NUREG-1021. Successful completion of this training provides a high level of confidence that operator actions assumed by the accident analyses will occur when required to mitigate the consequences of the accidents. Additionally, specific, short term actions assumed by the analyses are procedurally directed.

There is no possibilty of an accident or malfunction of a type different from those described in the SAR occurring as a result of this proposed change. This change does not alter the installed configuration of any plant equipment. The plant equipment is not operated in a new or different manner, nor are there any equipment or system interactions introduced by these revisions. As such, no new or different failures are introduced, and hence, accidents of a new or different type will not be created by the change to the administrative requirements governing the qualification of licensed operators.

There will be no reduction in the margin of safety as a result of the proposed change. The proposed change to the Technical Specifications are administrative in nature and have no effect on plant equipment or the setpoints at which the equipment would be actuated to mitigate the consequences of an accident.

ATTACHMENT D

ENVIRONMENTAL ASSESSMENT STATEMENT

Commonwealth Edison has evaluated the proposed amendment against the criteria for and identification of licensing and regulatory actions requiring environmental assessment in accordance with 10 CFR 51.21. It has been determined that the proposed changes meet the criteria for a categorical exclusion as provided for under 10 CFR 51.22(c)(9). This determination is based on the following:

1. These changes are being proposed as an amendment to a license for a reactor pursuant to 10 CFR 50 which changes a requirement with respect to the qualification and training of licensed Operators and Senior Operators,

2. the amendment involves no significant hazards consideration,

3. there is no significant increase in the amounts of any effluents that may be released offsite, and

4. there is no significant increase in individual or cumulative occupational radiation exposure.