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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

JUN 1 1979

Docket Nos.: STN 50-454/455
STN 50-456/457

Mr. Cordell Reed
Assistant Vice President
Commonwealth Edison Company
P. O. Box 767
Chicago, Illinois 60690

Dear Mr. Reed:

SUBJECT: FIRST ROUND QUESTIONS ON THE BYRON AND BRAIDWOOD OL APPLICATION

In our review of your application for operating licenses for the Byron Station, Units 1 and 2, and the Braidwood Station, Units 1 and 2, we have identified a need for additional information which we require to complete our review. The specific requests contained in the enclosure to this letter are the second set of our round one questions and cover those areas of our review performed by the following: (1) Accident Analysis Branch and (2) Quality Assurance (Conduct of Operations).

In order to maintain our present schedule as stated in our letter of February 22, 1979, we need a completely adequate response to all questions in the enclosure by July 21, 1979.

Please contact us if you desire any discussion or clarification of the enclosed requests.

Sincerely,

Steven A. Varga, Chief
Light Water Reactors Branch No. 4
Division of Project Management

Enclosure:
Request for Additional
Information

cc: See next page

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311.0 SECTION A, ACCIDENT ANALYSIS BRANCH

311.14 (Table 2.1-10) For the urban centers listed in Table 2.1-10 for Braidwood, provide the estimated population for the year 2020 and the basis for such estimates.

311.15 (2.2.2) For the Byron site, please clarify the intentions of Mr. Yost with respect to his restricted use airport. In particular, will the FAA designation of this facility be changed and will it be removed from future editions of the Chicago FAA Sectional Aeronautical Chart?

311.16 (3.5.1.4) Some of the missile velocities presented in Table 3.5-4 appear to be lower than the Revision 0 spectrum of Standard Review Plan Section (SRP) 3.5.1.4 (Revision 1). As noted in SRP 3.5.1.4, mixing of velocities between spectra is not considered acceptable. We require demonstration that the plant is protected against one of the missile spectra of SRP 3.5.1.4 Rev. 1.

311.17 (3.5.1.4) Provide a table which lists all systems or system components for which tornado missile protection is provided in accordance with Regulation Guide 1.117. Include in the table the location of the system or component and the means and degree of protection (e.g. roof and wall thicknesses, concrete strength, etc.).

311.18 (6.4) Section 6.4.4 and Table 6.4-1 indicate that your calculated occupational doses to control room operators after a DBA are within the levels required by Criterion 19 of the General Design Criteria. Clarify the doses listed in the referenced table to indicate the applicable plant (Byron or Braidwood) and the type of dose (i.e., beta, gamma, whole body, thyroid). In addition, outline the analysis that was performed, identifying all assumptions for each plant used in the analysis including, as appropriate:

a. Assumed credit for engineered safety features such as containment cleanup systems and the control room makeup air filters.

b. Assumed rate of unfiltered air inleakage after the DBA, including such leak paths as control room doors, ducts, penetrations, outside air isolation dampers and contaminated air from rooms adjacent to those served by the control room HVAC.

c. Assumed atmospheric dispersion (X/Q) factors for the control room air intake vents, the data source (e.g., meteorological records, literary references) for these X/Q values, and other assumptions made in reaching the X/Q values used in your analysis (release height, distance and direction to receptor-control room air intake vents, building wake factor, projected containment area, wind direction changes, control room occupancy factor). These data need to be supplied, as appropriate, for both the Byron and Braidwood stations.

d. The volume of the control room envelope.

For your reference in this dose analysis, see the following:

a. U.S. NRC Standard Review Plan Section 6.4, "Habitability Systems," (NUREG-75/087, Section 6.4).

b. Murphy, K.G. and K.H. Campe, "Nuclear Power Plant Control Room Ventilation System Design for Meeting General Design Criterion 19," Proceedings of the Thirteenth AEC Air Cleaning Conference, August 1974.

c. U.S. NRC Standard Review Plan Section 15.6.5, Appendix A,

"Radiological Consequences of a Design Basis Loss-of-Coolant Accident: Containment Leakage Contribution," and Appendix B, ". . . Leakage from Engineered Safety Features Components outside Containment."

311.19 (6.5.2) It is not clear from your description of the spray system what type of spray nozzle you propose to use in the containment. Clarify your description by including the model number and nozzle manufacturer for the containment spray nozzles in your proposed design.

311.20 (6.5.2) Your description of the spray ring header placement is not sufficient to permit a determination of the fraction of the containment which will not be sprayed directly. Provide the volumes for the following regions of the containment: region which is sprayed directly; region which is not sprayed and with no communication with the sprayed region; region which is not sprayed and in poor communication with the sprayed region, and the region which is not sprayed and in good communication with the sprayed region. Provide all the assumptions used to calculate these volumes including appropriate figures and analyses.

311.21 (6.5.1) (9.4.5) Provide additional information regarding the negative pressure that will be established in the fuel handling building following a postulated fuel handling accident. Your response should include the specific negative pressure required, the basis for this pressure, the time required to establish the design negative pressure, the areas or buildings that are maintained at the negative pres-

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sure, the necessary supply and exhaust flow rates, the methods of detecting and controlling changes in the negative pressure, and the fuel handling building isolation procedures. Conformance to the criteria of SRP 15.7.4 should be demonstrated.

311.22 (6.5.1) (9.4.5) (3.2.1) Appendix A A1.29

The fuel handling building exhaust system is designed to Safety Class I requirements. To meet the guidelines of Regulatory Guide 1.29, this system should be designed to Seismic Category I requirements. It is not apparent from the discussion in FSAR Section 3.2.1 if your Safety Class I requirements meet the staff requirements as given in Regulatory Guide 1.29 and Appendix A to 10 CFR Part 100. Provide the necessary clarification.

311.23 (9.4.5) (6.5.1)

According to the FSAR, the Fuel Handling Building Exhaust System (FHBES) operates continuously during all normal plant operating conditions, discharging the exhaust from the plant via the charcoal bypass line. Additional information and clarification is required with respect to the following:

1) Two dampers are installed in series in the bypass line, one labeled normally open, the other normally closed (Fig. 9.4-5, sheet 12). Explain or correct this apparent functional inconsistency.
2) As stated in the FSAR, the bypass line is closed automatically on a high radiation signal in the exhaust duct. However, on FSAR Fig. 9.4-5, sheet 12, the damper actuators are manual hand switches (HS). Explain this inconsistency.

3) The high radiation signal for isolation of the charcoal bypass line and opening of the dampers in the charcoal booster fan lines is generated by a single radiation monitor in the exhaust duct of the FHBES (Fig. 9.4-5, sheet 3). Since the exhaust system is an ESF system, this arrangement does not appear to satisfy the requirement for the delivery of the isolation and switching signal in the event of a single failure. Clarify this inconsistency.

4) Since the FHBES does not include a means for humidity control of the exhaust prior to entering the charcoal filters, provide a discussion of the operation of the FHBES for the case of high radioactivity and high humidity in the exhaust flow upstream of the exhaust filter plenums (Figure 9.4-5, sheet 12). Provide a discussion of the potential problems which could arise from operating the FHBES under these conditions.

5) In FSAR Section 6.5.1.2.3, the FHBES is referred to as the Auxiliary Building Exhaust System. Correct this inconsistency.

311.24 (15.1.5) Table 15.1-3 contains an entry for the long term steam release from a defective steam generator and quotes a value of 1,000 pounds for the 0-8 hour period. However, the initial steam release in the defective steam generator over a 0-2 hour period is given as 163,000 pounds. Provide some discussion which clearly defines the differences between these two values for the affected steam generator over the period 0-8 hours.

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311.25 (15.3.3) Provide either in FSAR Section 15.3.3.2 or in Table 15.3-3 the amount of fuel that would be expected to experience DNB and therefore would be assumed to experience clad perforation and/or fuel melt for the Reactor Coolant Pump Shaft Seizure accident.

311.26 (15.4.8) There is an apparent discrepancy between the steam release values and other important parameters of Table 15.4-4 and those provided in the text of FSAR Section 15.4.8.3. For example, the text states that for the case of loss-of-offsite power, 83,000 pounds of steam will be released by a steam dump through the relief values for a period of only 350 seconds.

Table 15.4-4, on the other hand, states that this same steam dump would release 113,000 pounds of steam over a period of 500 seconds. In light of this information, revise FSAR Section 15.4.8.3 and Table 15.4-4 as appropriate such that the assumptions given in the text and listed in the table are consistent with each other.

Also Table 15.4-4 indicates that even though the condenser is available in the realistic case, no steam dump to the condenser would occur. Clarify why no steam dump to the condenser occurs in the realistic analysis for the consequences of this accident.

311.27 (15.5.2) Since Acceptance Review Question 311.10 incorrectly identified the assumed location of the letdown line break, a response to staff question 311.10 is not required. However, perform an analysis of the radiological consequences of a CVCS letdown line rupture outside containment and downsteam of the outboard isolation

valve. The analysis should be performed using the most restrictive single failure. Include in your analysis the time required to generate the isolation signal, isolation valve closure time, single failure of one isolation valve, mass of reactor coolant released prior to isolation and increased iodine reactor coolant concentrations because of the transient conditions (iodine spike). Provide all the assumptions and parameters used in the radiological consequence analysis.

311.28 (15.6.5) (9.4) Describe in more detail the design and operation of the proposed containment minipurge system. Include in your discussion the projected number of hours of operation per year and a radiological consequence analysis of a LOCA while the minipurge is in operation.

311.29 (15.6.5) Table 15.6-15 is incomplete. Provide all possible sources of ECCS recirculation leakage and the leakage rates. If these are referenced from the FSAR, provide sufficient references for these values.

311.30 (15.6.5) Provide an analysis adiological consequences from expected ECCS leakage for the first 24 hours after a LOCA and from the assumed failure described in SRP Section 15.6.5 Appendix B for the course of the accident. Also provide a description of the operation of all systems which will be used to mitigate any such consequences. If filters are proposed for iodine dose reduction, provide sufficient information for the staff to determine that the proposed filters meet the recommendations of Regulatory Guide 1.52 and to determine that the ECCS pump rooms or any rooms where radioactive leakage might occur can be maintained at a negative pressure with respect to the ambient atmosphere to insure that any leakage will be processed prior to release to the environment.

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311.31 (15.6.5) In Table 15.6-9 the applicant uses iodine spray removal coefficient of 29.9 hr^{-1} in their conservative analyses. The branch position states that a maximum iodine spray removal coefficient of 10 hr^{-1} can be used when 50% instantaneous plateout of radioactive iodine is assumed. Provide a revised LOCA radiological consequence analysis using the spray removal coefficient of 10 hr^{-1}.

311.32 (13.3) In an appendix to the Generating Station Emergency Plan (GSEP) specific to the Byron and the Braidwood sites, provide the information cited at the following places in Appendix A to Regulatory Guide 1.101, (Revision 1, March 1977): 3., 4.1.3, 4.1.4, 4.1.5, 4.2, 5.1, 5.2.1, 5.3.2, 5.4 (Local agreements) 6.4, 6.4.1, 6.4.2, 6.4.3.2, 6.5, 7., 7.1, 7.3, 7.4 (re: reassembly area for site evacuees, including construction workers, and offsite emergency operation centers), 7.5, 7.6, 8.1.2, 8.2 (last sentence), 8.3 and 10.

311.33 (13.3) With regard to section 5.4 of Regulatory Guide 1.101: in lieu of submitting State and local agencies' radiological response plans as evidence of reasonable assurance that appropriate and timely response measures can and will be taken in behalf of the population-at-risk in the plant environs, you may address the applicable elements of the list below for each State and local agency having a response role in support of the Byron and the Braidwood Emergency Plan. If you choose to submit any State or local plans in lieu of the above, ensure that the plans are reviewed for completeness with respect to the applicable elements listed below. If necessary, request the State or local agencies to include such information in their plans, or you may supplement their plans with the necessary information in Section 13.3 of the FSAR. Note that in the absence of a State or local agency plan, the written agreement with that agency should reflect their concurrence with your docketed description of the applicable elements from the following list as related to that agency's role in support of the emergency response plans developed for the Byron and the Braidwood plant.

1. The identity of the agency.

2. A description of the authority and responsibility for emergency response functions.

3. A description of the concept of operations including the operational interrelationships of all organizations having emergency response roles.

4. The designation and location of the Emergency Operations Center for the direction and/or coordination of emergency support activities.

5. The established relationship and interface with State and/or local government emergency response plans.

6. The provisions established with the Department of Energy Reg nal Coordinating Office for radiological assistance under the RAP and IRAP programs.

7. A description of the communication plan for emergencies including titles and alternates for both ends of the communication links, and primary and backup means of communication. Where consistent with the agency function, include the following:

   a. Provision for 24-hour/day manning of communication link.

   b. Provision for administrative control methods for ensuring the effective coordination and control of the emergency support activities.

   c. Provision for communications arrangements with contiguous local governments where applicable.

   d. Provision for communications arrangements with Federal emergency response organizations.

   e. Provision for communications with the nuclear facility, State and/or local emergency operations centers, and field assessment teams.

8. A description of the communications methods for issuing emergency instructions to the public in the potentially affected environs of the nuclear facility.

9. A description of the methods and equipment to be employed in determining the magnitude and locations of any radiological hazards following liquid or gaseous radioactivity releases.

10. The designation of protective action guides and/or other criteria to be used for implementing specific protective actions and the information needs (e.g., dose rates, projected dose levels, contamination levels, airborne or waterborne activity levels) for implementing such actions.

11. A description of the methods for protecting the public from consumption of contaminated foodstuffs.

12. A description of the evacuation plans for the Low Population Zone (LPZ) including survey maps for the facility environs showing evacuation routes as well as relocation and shelter areas. The plans may extend to areas beyond the LPZ and should include the following:

    a. Population and their distribution around the nuclear facility.

    b. Means for notification of the potentially affected population.

    c. Disabilities, institutional confinement, or other factors which may impair mobility of parts of the population.

    d. Means of effecting relocation.

    e. Potential egress routes and their traffic capacities.

    f. Potential impediments to use of egress routes.

13. The provisions for maintaining dose records of all potentially exposed emergency workers involved in response activities.

14. The provisions for emergency drills and exercises to test and evaluate the response role of the agency, including provisions for critique by qualified observers.

15. A description of the training program established for those individuals having an emergency response assignment.

16. The provisions for periodic review and updating of the emergency response plans of the agency.

311.34 (13,3) We note that section 13.3 references the GSEP filed February 18, 1975. We have on file a document dated December 1977 (Revision 0) with page changes dated January 1978 (Revision 1). Please identify the appropriate document to be used for the Byron and Braidwood plants.

422.0 QUALITY ASSURANCE BRANCH - CONDUCT OF OPERATIONS

422.3 (13.1.1.3) Describe the number of persons reporting to the Project Engineers, PWR section of the SNED, in terms of total numbers by discipline (Electrical Engineers, Mechanical Engineers, Nuclear Engineers) of those assigned specifically to the Byron/Braidwood Station project and those assigned to other projects.

422.4 (13.1.1.3) Describe the number of persons reporting to each of the supervisors (tech staff supervisor, supervisor of radiation protection..., supervisor for nuclear waste processing...) reporting to the Assistant Superintendent - Rad/Chem Systems for Production System Analysis. Include the proportion of time they will be available to work on the Byron/Braidwood Stations project. In addition, provide the resumes of the principal supervisory personnel in the areas of health physics and rad-chemistry.

422.5 (13.1.1.3) Describe the number of professional persons on the staff of the Nuclear Fuel Services Department. In addition, provide the resume of the person filling the position of Director, Nuclear Fuel Services.

422.6 (13.1.2.1) The organization chart provided in subsection 16.6.1 does not indicate the number of persons assigned to common or duplicate positions. Additionally, these numbers should not be included in the proposed Technical Specifications. Therefore, provide in a new plant staff figure, or some other manner, the number of persons assigned to the positions of Operating Engineers, Shift Engineers, Shift Foreman, Nuclear Station Operator, Equipment Operator, Equipment Attendant, Instrument Maintenance Foreman, Mechanical Maintenance Foreman, Electrical Maintenance Foreman, Fuel Handling Foreman, Rad/Chem Foreman and Engineers, Group Leader, Rad/Chem Technicians, Control System Technicians and Mechanics for both single-unit and two-unit operation.

422.7 (13.1.2.2) Expand this section to describe the responsibilities and authority of your Shift Foreman, Startup Foreman, Nuclear Station Operator, Equipment Operator, Equipment Attendant, Station Manager, Rad Foreman and Engineers, and Group Leaders. In addition, expand your station organization chart to show these organizational units.

422.8 (13.1.2.2) Expand your plant staff organizational chart to show the functional units responsible for nuclear engineering and plant quality control. For these functional units, show any further functional units reporting to those foremen and the number of persons to be assigned to these units for both single-unit and two-unit operation.

422.9 (13.1.2.3) Clarify Table 13.1-2 to show the type of NRC license to be held by those persons filling the positions for which you state an NRC license will be required.

422.10 (13.1.2.2) Please clarify the specific line of succession of authority and responsibility for station operation when the Station Superintendent is unavailable by describing the line of succession by position title.

422.11(RSP) (13.1.2.3) Your proposed shift crew complement for two-unit operation is not acceptable. It is the staff's position that, in regard to licensed operators, that your shift crew composition for two-unit operation should include at least two senior licensed operators and three licensed operators.

422.12 (13.1.3.1) Expand Table 13.1-2 to show the qualification requirements for the positions of Assistant Superintendent, Administrative Assistant, Master Instrument Mechanic, Instrument Mechanical Foreman, Electrical Foreman, Mechanical Foreman, Operating Engineers, Equipment Operators, Equipment Attendents, Rad Foreman and Engineers, Quality Control Engineer, and Training Supervisor. This may be done by reference to a specific ANSI N18.1 position title or by describing your proposed qualification requirements for the position.

422.13(RSP) (13.1.3.1) Your qualification requirements for the position of Rad/Chem Supervisor is unacceptable. It is the staff's position, in regard to the health physics qualifications, that the qualification requirements for this position should meet those described in Revision 1 to Regulatory Guide 1.8 for the position of Radiation Protection Manager.

422.14(RSP) (13.1.3.1) The qualification requirements for the position of Radiation-Chemistry Technicians are not satisfactory. It is our position that the qualifications for this position should meet that described in Section 4.5.2 of ANSI N18.1-1971.

422.15 (9.5.1) Identify the upper level offsite management position(s) which has overall responsibility for the formulation, implementation, and assessment of the effectiveness of the station fire protection program.

422.16 (9.5.1) While the station superintendent is generally responsible for all activities at the facility, describe any further delegation of these responsibilities for the fire protection program such as training, maintenance of fire protection systems, testing of fire protection equipment, fire safety inspections, fire fighting procedures, and fire drills.

422.17 (9.5.1) Describe the composition of your shift fire brigade in terms of numbers and job titles.

422.18 (9.5.1) Describe the authority of your shift fire brigade leader relative to that of your shift engineers.