Safety Evaluation Report

NUREG-75/023
Supplement 2

U. S. Nuclear
Regulatory Commission

related to construction of

Byron Station, Units 1 and 2

Braidwood Station, Units 1 and 2

Office of Nuclear
Reactor Regulation

Docket No. STN 50-454
STN 50-455
STN 50-456
STN 50-457

Commonwealth Edison Company

October 1975

Supplement No. 2

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NUREG-75/023 Supplement 2
October 17, 1975

SUPPLEMENT NO. 2
TO THE
SAFETY EVALUATION REPORT
BY THE
OFFICE OF NUCLEAR REACTOR REGULATION
IN THE MATTER OF
COMMONWEALTH EDISON COMPANY

BYRON STATION, UNITS 1 AND 2
AND
BRAIDWOOD STATION, UNITS 1 AND 2

DOCKET NOS. STN 50-454, STN 50-455, STN 50-456, and STN 50-457

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APPENDICES

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1.0 INTRODUCTION

The Nuclear Regulatory Commission's (NRC or Commission) Safety Evaluation Report in the matter of the application by Commonwealth Edison Company (applicant) to construct and operate the Byron Station, Units 1 and 2, and the Braidwood Station, Units 1 and 2, was issued on April 4, 1975. Supplement No. 1 to the Safety Evaluation Report, which updated the information presented in the Safety Evaluation Report, was issued on August 1, 1975.

In Supplement No. 1 to the Safety Evaluation Report we stated that the resolution of several matters that were under review by the NRC staff would be reported in a future supplement to the Safety Evaluation Report. These matters were identified as (1) the seismic system analysis, (2) the performance evaluation of the emergency core cooling system to satisfy the requirements of 10 CFR Part 50.46, and (3) the evaluation of radioactive waste management systems to meet the dose objectives of Appendix I to 10 CFR Part 50.

The purpose of this Supplement is to update our review by providing (1) our evaluation of the outstanding matters identified in Supplement No. 1, (2) our evaluation of additional information submitted by the applicant since the issuance of Supplement No. 1, and (3) our evaluation of new safety issues that have been identified since the issuance of Supplement No. 1.

There are two new safety issues discussed in this Supplement: (1) the small geological faults discovered at the Byron Station site during routine geological investigations of the excavations at the site, and (2) the possible underestimation of loadings on the reactor pressure vessel support system. These two new issues are discussed in Sections 2.5.3 and 5.2.1, respectively, of this Supplement.

With this Supplement we conclude that all of the outstanding matters have been satisfactorily resolved. We conclude that the Byron Station, Units 1 and 2, and the Braidwood Station, Units 1 and 2, can be constructed and operated as proposed without endangering the health and safety of the public.

Each of the following sections of this Supplement is numbered the same as the section in the Safety Evaluation Report and Supplement No. 1 that is being updated, and is supplementary to and not in lieu of the discussion in the Safety Evaluation Report and Supplement No. 1.

Appendix A to this Supplement is a continuation of the chronology of our principal actions related to the processing of the application. The Report of the Advisory Committee on Reactor Safeguards is attached as Appendix B. That report was inadvertently omitted from Supplement No. 1 to the Safety Evaluation Report.

2.0 SITE CHARACTERISTICS

2.5 Geology and Seismology

2.5.3 Site Geology (Byron)

On July 30, 1975 the applicant notified us of the discovery of four possible faults at three different locations in the excavations at the Byron St tion site. These offsets were discovered during routine geological investigations conducted at the excavations. An NRC staff geologist accompanied by a geologist from the U.S. Geological Survey and geologists from the Illinois Geological Survey, visited the site on August 6-7, 1975 to examine the displacement features that were discovered by the applicant's consulting geologists.

Five small vertical displacements and several joints were observed. These vertical displacements were classified as small or minor faults. The faults showed vertical displacements ranging from one to six inches.

Upon discovering these faults, the applicant developed a fault specific geotechnical investigation for the purpose of describing more fully the small faults, including their regional and site specific characteristics, and to confirm that the faults were non-capable as that term is defined in Appendix A to 10 CFR Part 100. A report, "Fault Specific Geotechnical Investigation-Byron Station," was filed with the Commission by applicant's letter dated August 29, 1975. The report was supplemented by additional information in a letter from the applicant dated September 16, 1975. The additional information was developed at the request of our staff geologist during a site visit on September 9-10, 1975.

The faults identified as Faults Nos. 10 and 34 were selected by the applicant for detailed analyses for the following reasons: (1) the faults are two exposures of the same fault; (2) these faults occurred on opposite sides of the excavation and can be traced continuously across the floor of the exposed excavation until they meet; (3) these faults had the maximum recorded vertical displacement; (4) the faults were overlaid by unfaulted soils in the immediate proximity of the excavation; and (5) these faults were considered to have the identical history of development as the other faults with lesser displacements and poorer expression.

In the report "Fault Specific Geotechnical Investigation - Byron Station," the applicant states that the data indicate that the faults are non-capable faults and that there is no relationship between the reported historical seismicity of northern Illinois or southern Wisconsin and faulting at the Byron Station site. The applicant has based the determination of the relative age of the faults on an interpretation of

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the regional geological history, an interpretation of the age of the unfaulted residual soil which overlies Faults Nos. 10 and 34, and by the age of the clay filling in Faults Nos. 10 and 34 as determined by correlation of the clay mineralogy and the source of the clay filling.

At a public hearing before the Atomic Safety and Licensing Board in Bethesda, Maryland on August 26, 1975 the applicant, the NRC staff, and the Illinois Geological Survey presented testimony related to the faults at the Byron Station site. This testimony was presented in support of findings by the NRC staff and the applicant that a prior determination of site suitability is still valid in the light of this most recent knowledge of the faults and that the site meets the requirements of Appendix A to 10 CFR Part 100.

Studies of the general regional tectonics of the site area indicate that the faulting most likely occurred 65 million years before the present time. Further geological studies of the site areas, accomplished by the applicant and now being reviewed by the NRC staff, indicate that it can be demonstrated that the last fault movement occurred prior to 700,000 years before the present time.

Based on our review of these studies and our inspections of the site, we conclude that the latest fault movement occurred prior to 70,000 years before the present time and, therefore, the faults discovered at the site are not capable faults as defined in Appendix A to 10 CFR Part 100.

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3.0 DESIGN CRITERIA FOR STRUCTURES, COMPONENTS, EQUIPMENT AND SYSTEMS

3.7 Seismic Design

3.7.2 Seismic System Analysis

We stated in Supplement No. 1 to the Safety Evaluation Report that we were reviewing the additional information related to the seismic system analysis that the applicant had provided in a letter to us dated July 22, 1975. The applicant subsequently incorporated this additional information into the Preliminary Safety Analysis Report by Amendment 14 on July 30, 1975.

We completed our review of that additional information and determined that further clarification was necessary regarding (1) the use of critical damping values in the seismic system analysis, (2) the method of determining the seismic response of interconnected components due to differential seismic movement, and (3) the method of combining all significant modal responses to obtain the combined response in each direction.

We informed the applicants of the need to clarify these matters and suggested that an acceptable method of providing such clarification would be the criteria that we found to be acceptable for the Standardized Nuclear Unit Power Plant System (Docket Nos. STN 50-482, 50-483, 50-484, 50-485, and 50-486). In a letter to us dated September 16, 1975, the applicant stated that the proposed seismic system analysis would be amended to include these criteria.

On the basis of our review of the seismic system analysis, and the commitment made by the applicant to amend the seismic system analysis according to the letter of September 16, 1975, we conclude that the seismic system analysis for the nuclear steam supply system portion of the plant, which is within the scope of RESAR-3, Consolidated Version, is acceptable.

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5.0 REACTOR COOLANT SYSTEM

5.2 Reactor Coolant Pressure Boundary

5.2.1 Component Design

On May 7, 1975 we were informed by a licensee of a pressurized water reactor, Virginia Electric and Power Company, that an asymmetric loading resulting from a postulated pipe rupture at a particular location in the reactor coolant system had not been taken into account in the original design of the reactor pressure vessel support system for the North Anna Units 1 and 2 (Docket Nos. 50-338 and 339). This loading results from the forces induced on the internals within the reactor vessel caused by differential pressure conditions within the vessel immediately following a postulated loss-of-coolant accident. In addition, the asymmetric loading from transient differential pressures that would exist around the exterior of the reactor vessel from the same postulated pipe rupture was not included in the original design analysis. However, the symmetric loadings from such a postulated pipe rupture were included in the original analysis of the reactor pressure vessel supports.

It is our opinion that these factors related to the design of the reactor pressure vessel supports are generic in nature and may apply to the Byron and Braidwood Stations. Accordingly we are taking steps to review this problem on a generic basis to determine the extent of the problem.

We have informed the applicant of the nature of this problem and have requested the applicant to verify that the design procedures for the reactor pressure vessel support system will properly include the asymmetric forces described above in the final design of the supports. In a letter dated September 30, 1975 the applicant provided verification that the final design will include the asymmetric forces.

Based on our review of this generic problem to date, we have determined that the methodology necessary to model the complete reactor coolant system in sufficient detail to determine analytically the magnitudes and phase relationships of the vessel support system loads from the transient pressure differentials has been developed extensively by Westinghouse, and that the calculational techniques have been refined so that it is practical to evaluate the actual dynamic system response to all the known transient loads. Furthermore, Westinghouse has informed us that structural analyses based on the loads developed by the worst case loading (which is a rupture of the reactor coolant pipe at the cold leg nozzle) demonstrate that Westinghouse reactor coolant support systems now being designed can sustain these loads and remain within the conservative design basis stress limits comparable to those stress limits specified in Appendix F of Section III of the ASME Boiler and Pressure Vessel Code.

On the basis of our review of this problem to date, we conclude that the applicant can properly account for these forces during the final design of the reactor vessel support system.

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6.0 ENGINEERED SAFETY FEATURES

6.2 Containment Systems

6.2.1 Containment Functional Design

In Supplement No. 1 to the Safety Evaluation Report we stated that we require the applicant to design the shielding canisters which are intended to be placed in the inspection openings of the reactor vessel cavity so that they will not interfere with the venting of the reactor cavity and will not become potential damaging missiles or interfere with any other safety-related equipment or functions.

We have evaluated the information presented in Amendment 14 to the Preliminary Safety Analysis Report and determined that further information was necessary in order for us to verify that our requirements as stated above would be met. Accordingly, we requested the applicant to provide additional information in our letter of September 5, 1975, to confirm that these requirements would be incorporated into the design criteria.

The applicant has provided additional information related to the design of the reactor cavity in response to our request and has confirmed that our requirements will be included in the proposed design criteria for the reactor cavity.

On the basis of our review of the additional information, including the confirmation that our requirements regarding the design of the cavity vents and shielding canisters will be met, we reaffirm our conclusion that the proposed design criteria are acceptable.

6.3 Emergency Core Cooling Systems (ECCS)

6.3.3 Performance Evaluation

In Supplement No. 1 to the Safety Evaluation Report we stated that we were reviewing the performance evaluation of the emergency core cooling system and that we would report the results of our review in a future supplement to the Safety Evaluation Report.

We have completed our review of the performance evaluation submitted by the applicant in Amendment 12 to the Preliminary Safety Analysis Report and the additional information submitted by letter dated July 22, 1975. The analyses submitted were based on the Westinghouse emergency core cooling system evaluation model which was previously

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reviewed and determined to be an acceptable model by the NRC staff for a certain class of pressurized water reactors. The Byron and Braidwood Stations are within this class of pressurized water reactors.

The applicant submitted large break and small break analyses for the postulated loss-of-coolant accidents in Amendment 12 to the Preliminary Safety Analysis Report for the Byron and Braidwood Stations. The large break analyses were limited to a spectrum of three double-ended guillotine breaks with a Moody multiplier of 1.0, 0.6 and 0.4. These analyses were specific for the Byron and Braidwood Stations. To supplement these analyses, the applicant referenced Westinghouse Topical Report WCAP-8565-P, "Westinghouse ECCS - Four Loop Plant (17x17) Sensitivity Studies," July 1975, which is a generic report that presents analyses of other break sizes, types and locations, and demonstrates that the guillotine breaks are the worst cases for this plant type.

The analyses identified the worst break as the double-ended cold leg guillotine break with a Moody multiplier of 0.6. The calculated peak clad temperature was 2178 degrees Fahrenheit which is within the acceptable limit of 2200 degrees Fahrenheit as specified in Section 50.46(b) of 10 CFR Part 50. In addition the calculated maximum local metal-water reaction of seven percent and the total core wide metal-water reaction of less than 0.3 percent are well below the allowable limits of 17 percent and one percent, respectively. The analyses were based on a total peaking factor of 2.32, 102 percent of the rated power level of 3411 MWt and a peak linear power density of 12.6 kilowatts per foot. The results were based on a containment pressure transient which was calculated using an acceptably conservative set of containment parameters which is representative of the containment design for the Byron and Braidwood Stations.

The small break analyses included a spectrum of three break sizes. These analyses were specifically related to the Byron and Braidwood Stations design and referenced Westinghouse Topical Report WCAP - 8340, "Westinghouse Emergency Core Cooling System - Plant Sensitivity Studies," July 1974, which is a generic Westinghouse report containing additional break analyses. The four-inch diameter pipe break was identified as the limiting small break with a calculated peak clad temperature of 1673 degrees Fahrenheit. These analyses clearly indicate that the small break is not the limiting case.

Appendix K to 10 CFR Part 50 requires that the combination of emergency core cooling subsystems to be assumed operative shall be available after the most damaging single failure of equipment has occurred. The worse single failure which would minimize core cooling and provide the maximum containment cooling was identified as the loss of a low-head safety injection pump.

Our review of the piping and instrumentation diagrams for the Byron and Braidwood Stations indicates that spurious actuation of specific motor-operated valves was not considered in the selection of the worst single failure. In this regard we have

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identified the following motor-operated valves, which in case of a single failure could result in consequences that were not considered in the applicant's performance evaluation: (1) MOV 8806 at the suction to the high pressure safety injection pumps; (2) MOV 8835 at the high pressure safety injection pump discharge to the reactor coolant system cold leg; (3) MOV 8813 at the high pressure safety injection pump recirculation to the refueling water storage tank; (4) MOV 8808 A, B, C and D which are the accumulator isolation valves; (5) MOV 8809 A and B at the residual heat removal pump discharge to the reactor coolant system cold legs; (6) MOV 8840 at the residual heat removal pump discharge to the reactor coolant system hot legs; and (7) MOV 8802 A and B at the high pressure safety injection system pumps discharge to the reactor coolant system hot legs.

With regard to these motor-operated valves listed above, we require the applicant to provide appropriate design criteria to protect the reactor system in the event of a postulated loss-of-coolant accident and a simultaneous failure of one of the above mentioned valves. An acceptable criterion is specified in the document "Branch Technical Position EICSB 18, Application of the Single Failure Criterion to Manually-Controlled Electrically-Operated Valves." This document is attached to this Supplement as Appendix C. In a letter dated September 23, 1975 the applicant stated that the criteria recommended in the document referenced herein would be incorporated into the design for the valves identified above.

With regard to long term cooling and the prevention of boron concentration buildup in the nuclear core following a postulated loss-of-coolant accident, the applicant has proposed to provide for changing over from cold leg recirculation to hot leg recirculation at 24 hours following the postulated accident. We have reviewed the emergency core cooling system for the Byron and Braidwood Stations and determined that the proposed system design will permit this procedure.

With regard to the containment pressure analysis, we concluded in Supplement No. 1 to the Safety Evaluation Report that the containment pressure for the performance evaluation is conservative and that the containment pressure has been calculated in accordance with Appendix K to 10 CFR Part 50.

6.3.5 Conclusion

Based on our review of the performance evaluation of the emergency core cooling system we conclude that (1) the postulated loss-of-coolant accident analyses that were performed are in conformance with the requirements of Appendix K to 10 CFR Part 50, (2) the performance evaluation conforms to the peak clad temperature, maximum oxidation and hydrogen generation criteria specified in Section 50.46 of 10 CFR Part 50, (3) the emergency core cooling system performance will be adequate despite any postulated failure of a single component and (4) adequate systems are available to provide long term core cooling. On these bases, we conclude that the proposed design of the emergency core cooling system is acceptable.

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11.0 RADIOACTIVE WASTE MANAGEMENT

We have evaluated the radioactive waste management systems proposed for Byron Station, Units 1 and 2, and the Braidwood Station, Units 1 and 2, to reduce the quantities of radioactive materials released to the environment in liquid and gaseous effluents in accordance with 10 CFR Part 50.34a. These systems have been previously described in Sections 11.2 and 11.3 of the Safety Evaluation Report dated April 4, 1975 and in Section 3.5 of the Final Environmental Statements for each of the stations, dated July 1974. Based on more recent information applicable to the Byron/Braidwood Stations and changes in our calculational model, we have revised the liquid and gaseous source terms given in the Final Environmental Statements. These changes occurred subsequent to issuing the Final Environmental Statements for the Byron/Braidwood Stations. The revised source terms were calculated using the model and methodology described in Draft Regulatory Guide 1.BB, "Calculation of Releases of Radioactive Materials in Liquid and Gaseous Effluents from Pressurized Water Reactors (PWRs)," September 9, 1975.

In Supplement No. 1 to the Safety Evaluation Report issued August 1, 1975 we indicated that we had not completed our review of these systems to meet the requirements of Appendix I to 10 CFR Part 50 issued May 5, 1975. On September 4, 1975 (40 FR 40816) the Commission amended Appendix I to 10 CFR Part 50 to provide persons who have filed applications for construction permits for light-water-cooled nuclear power reactors which were docketed on or after January 2, 1971 and prior to June 4, 1976 the option of dispensing with the cost-benefit analysis required by Paragraph II.D of Appendix I. This option permits an applicant to design its radioactive waste management systems to satisfy the Guides on Design Objectives for Light-Water-Cooled Nuclear Power Reactors proposed in the Concluding Statement of Position of the Regulatory Staff in Docket RM-50-2, dated February 20, 1974. As indicated in the Statement of Consideration included with this amendment the Commission noted that it is unlikely that further reductions to radioactive material releases would be warranted on a cost-benefit basis for light-water-cooled nuclear power reactors having radwaste systems and equipment determined to be acceptable under the proposed staff design objectives set forth in RM-50-2.

In a letter to the Commission dated September 30, 1975, Commonwealth Edison Company chose to comply with the September 4, 1975 amendment to Appendix I rather than submit a cost-benefit analysis as required by Paragraph II.D.

Based on our reassessment of the liquid radioactive waste management systems we estimate that the quantity of radioactive materials released in liquid effluent, excluding tritium and dissolved noble gases, will be less than 5 Ci/year/reactor and that the total calculated quantity of radioactive materials released in liquid effluents from each station will not result in an annual dose or dose commitment to the total body or

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to any organ of an individual in an unrestricted area from all pathways of exposure will not exceed 5 mrem. Based on our reassessment of the gaseous radioactive waste management systems, we estimate that the total quantity of radioactive materials released in gaseous effluents from each station will not result in a calculated annual gamma air dose in excess of 10 mrads or a beta air dose in excess of 20 mrads at any location near ground level, at or beyond the site boundary, which could be occupied by individuals. We estimate that the annual total quantity of iodine-131 released in gaseous effluents will not exceed 1 Ci/yr/reactor and that the calculated annual total quantity of radioiodine and radioactive particulates released in gaseous effluents from each station will not result in an annual dose or dose commitment to any organ of an individual in an unrestricted area from all pathways of exposure in excess of 15 mrem.

Our evaluation of the proposed liquid and gaseous radioactive waste management systems for Byron Station, Units 1 and 2, and the Braidwood Station, Units 1 and 2, shows these systems to be capable of meeting the criteria given in Appendix I to 10 CFR Part 50 for keeping releases of radioactive materials to the environment "as low as is reasonably achievable," and therefore we find the proposed systems to be acceptable.

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21.0 CONCLUSIONS

Our conclusion that the issuance of permits for the construction of the facilities will not be inimical to the common defense and security or to the health and safety of the public, as stated in our Safety Evaluation Report, was conditioned on the favorable resolution of the remaining outstanding issues indicated in Supplement No. 1 to the Safety Evaluation Report. We have discussed each of these outstanding issues in this Supplement and have indicated that the outstanding issues have been satisfactorily completed. We have also indicated that the two new safety issues that were identified since the issuance of Supplement No. 1 have also been satisfactorily completed.

Accordingly, we reiterate our conclusions as stated in the Safety Evaluation Report and conclude that the issuance of permits for construction of the facilities will not be inimical to the common defense and security or to the health and safety of the public.

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APPENDIX A

CONTINUATION OF CHRONOLOGY OF RADIOLOGICAL REVIEW OF BYRON STATION, UNITS 1 AND 2 AND BRAIDWOOD STATION, UNITS 1 AND 2

July 17, 1975	Letter from Department of the Army concerning stability of slopes at Byron site
July 22, 1975	Letter from applicant containing additional information relating to ECCS-FAC
July 30, 1975	Letter from applicant transmitting RESAR-3, Consolidated Version, including Amendments 1 through 6 as the reference documents for this application
July 30, 1975	Preliminary notification of the discovery of possible fault area at the Byron construction site
July 30, 1975	Submittal of Amendment No. 14, consisting of revised information concerning reactor protection system, effects of displacements of slopes at Braidwood essential cooling pond, seismic analysis, outages of power grid, and miscellaneous corrections
August 1, 1975	Issuance of Supplement No. 1 to the Safety Evaluation Report
August 6-7, 1975	Meeting with the applicant and site visit to the Byron Station site by NRC staff geologist, USGS geologist, and geologists from the State of Illinois Geological Survey
August 7, 1975	Letter from applicant requesting authorization to undertake additional construction activities at the Braidwood Station
August 11, 1975	Letter from applicant regarding the faults at the Byron Station site and the proposed geotechnical investigation program
August 19, 1975	Letter to applicant regarding the results of our review of WCAP-7705
August 20, 1975	Letter to the applicant transmitting staff positions regarding the geotechnical investigation of the faults at the Byron Station site

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August 29, 1975	Letter from the Department of the Army concerning the densities of sands in the foundation for the river screen house at the Byron site
August 29, 1975	Letter from applicant transmitting the report "Fault Specific Geotechnical Investigation - Byron Station"
September 2, 1975	Letter to the applicant requesting clarification of the seismic system design criteria
September 5, 1975	Letter to applicant related to the reactor cavity vent design criteria
September 16, 1975	Letter from applicant transmitting information concerning seismic analysis
September 16, 1975	Letter from applicant transmitting additional information on the Fault Specific Geotechnical Investigation - Byron Station
September 23, 1975	Letter from applicant regarding design criteria for the reactor cavity vent
September 23, 1975	Letter from applicant regarding design to meet the single failure criterion for ECCS
September 30, 1975	Letter from applicant regarding reactor pressure vessel support designs
October 2, 1975	Letter from applicant regarding inspection of excavations at the Byron Station site

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ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

May 13, 1975

Honorable William A. Anders
Chairman
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Subject: REPORT ON THE BYRON STATION UNITS 1 AND 2 AND BRAIDWOOD STATION UNITS 1 AND 2

Dear Mr. Anders:

At its 181st meeting, May 8-10, 1975, the Advisory Committee on Reactor Safeguards completed its review of the application of the Commonwealth Edison Company for authorization to construct Byron Station Units 1 and 2 and Braidwood Station Units 1 and 2. These plants were previously considered at Subcommittee meetings at Des Plaines, Illinois, on January 23, 1975, and April 24, 1975. Members of the Committee visited the sites on January 22, 1975. The Committee reviewed site-related aspects at its 178th meeting, February 6-8, 1975. This review is the first in which the Committee simultaneously has considered similar reactor designs at two widely separated sites. During its review, the Committee had the benefit of discussions with representatives and consultants of the Commonwealth Edison Company, the Westinghouse Electric Corporation, and the NRC Staff. The Committee also had the benefit of the documents listed.

The Byron Station is located in Ogle County, Illinois, about 17 miles southwest of Rockford, Illinois, the nearest population center (1970 population 147,370). The minimum exclusion radius is 1510 feet; the low population zone is three miles in radius.

The Braidwood Station is located in Will County, Illinois, about 20 miles south-southwest of Joliet, Illinois, the nearest population center (1970 population about 80,000). The minimum exclusion radius is 1500 feet; the low population zone is 1.13 miles in radius.

The safe shutdown earthquake for both Byron and Braidwood Stations is 0.20g horizontal acceleration at the bedrock-till interface. The operating basis earthquake is 0.09g.

High explosives, primarily flake TNT, are regularly shipped from the Joliet Arsenal on the Illinois Central Gulf Railroad, which passes within 1700 feet of the nearest Braidwood reactor containment structure. The Committee has reviewed the Applicant's analysis of the probability of an accidental explosion adjacent to the site and agrees that the probability and consequences of such an explosion are acceptably low.

The ultimate heat sinks differ at Byron and at Braidwood. The Byron Station will utilize two mechanical draft cooling towers with a makeup system for each tower. The Braidwood Station essential cooling is provided by a cooling pond that is an integral part of a larger man-made lake which supplies cooling during normal operation. All critical items are Seismic Category I.

Byron and Braidwood each will utilize two Westinghouse four-loop pressurized water nuclear steam supply systems having design power levels of 3411 MW(t), essentially identical to the Catawba and Vogtle Stations, previously reported on by the Committee in its letters of November 13, 1973, and April 16, 1974, respectively.

In the unlikely event of a guillotine break of a cold leg pipe, a sufficiently large bypass flow area is required to prevent over-pressurization of the containment subcompartments. The region surrounding the cold leg pipe is to be filled with a boron-containing material to absorb neutrons. The viability of the bypass flow concept depends on administrative control to assure that the refueling deck cover plates are unbolted or removed during operation. There is also an incomplete assessment of the characteristics of the blocks of neutron absorber with regard to their acting as missiles or fragmenting with the possibility of plugging of sump filters and spray nozzles. This problem should be resolved to the satisfaction of the NRC Staff.

The ACRS considered the problem of turbine missiles in its report of April 18, 1973, where recommendations were made concerning overspeed systems, optimum turbine orientation, and projectile penetration. The Committee requests that the NRC Staff continue to review the combination of overspeed systems and low angle missile barriers to determine if changes would improve the acceptability of Byron and Braidwood Stations, recognizing that design of these plants, which utilize a non-optimum turbine orientation, was well advanced prior to 1973. For future plants, the ACRS reiterates its recommendation that a peninsular arrangement, optimized to be non-interactive with critical components in both single and multi-station plants, is preferred.

The NRC Staff has determined that the ECCS performance evaluation for the Byron and Braidwood Stations meets the Interim Acceptance Criteria of June, 1971. In addition, the Applicant's ECCS performance evaluation, using an approved Westinghouse model, to show compliance with the Final Acceptance Criteria of 10 CFR 50.46 and Appendix K, must be reviewed and approved by the NRC Staff. The Committee wishes to be kept informed.

The Committee recommended in its report of September 10, 1973, on acceptance criteria for ECCS, that significantly improved ECCS capability should be provided for reactors for which construction permits were filed after January 7, 1972. The Byron and Braidwood Stations are in this category. These units will use 17x17 fuel assemblies. Although calculated peak clad temperatures in the event of a hypothetical LOCA are less for 17x17 assemblies than for a 15x15 array, the Committee believes that the Applicant should continue studies that are responsive to the Committee's September 10, 1973 report. If studies establish that significant further ECCS improvements can be achieved, consideration should be given to incorporating them into this plant.

The Committee recommends that further attention be given by the Applicant and the NRC Staff to those provisions of Regulatory Guide 1.17, which address design features to prevent or mitigate the consequences of acts of sabotage.

A problem considered to be generic by the ACRS is the environmental and seismic qualification of Class IE electrical equipment. An important aspect is that of defining what represents an acceptable aging procedure for multi-component electrical systems. This issue should be resolved by the Applicant and the NRC Staff. The Committee wishes to be kept informed.

Generic problems relating to large water reactors have been identified by the NRC Staff and the ACRS and discussed in the Committee's report dated March 12, 1975. These problems should be dealt with appropriately by the NRC Staff and the Applicant.

The ACRS believes that the above items can be resolved during construction and that, if due consideration is given to these items, the Byron Station Units 1 and 2 and Braidwood Station Units 1 and 2 can be constructed with reasonable assurance that they can be operated without undue risk to the health and safety of the public.

Sincerely,

WKerr

William Kerr
Chairman

References:

1. Commonwealth Edison Company, Byron/Braidwood Stations, Preliminary Safety Analysis Reports (PSAR) Volumes 1 - 7.

2. Amendments 1 - 9 to the PSAR.

3. Office of Reactor Regulation (USNRC), "Safety Evaluation of the Byron Station, Units 1 and 2, and the Braidwood Station, Units 1 and 2," April 14, 1975.

4. Naval Surface Weapons Center, letter discussing hazards of flake TNT, November 6, 1974.

5. Department of the Army (Corps of Engineers), letter on soils and foundation aspects of the Byron Station, October 3, 1974.

6. Commonwealth Edison Company letters:
   a) on analysis of consequences of anticipated transients without scram, September 26, 1974;

   b) on quality of railroad track adjacent to the Braidwood Station, September 18, 1974;

   c) on the description of activities to be undertaken at the Byron Station under a limited work authorization (LWA), August 15, 1974;

   d) on the description of activities to be undertaken at the Braidwood Station under an LWA, August 1, 1974;

   e) on the quality assurance program for the Byron Station grouting work, February 6, 1974.

APPENDIX C

BRANCH TECHNICAL POSITION EICSB 18

APPLICATION OF THE SINGLE FAILURE CRITERION TO MANUALLY-CONTROLLED ELECTRICALLY-OPERATED VALVES

A. BACKGROUND

Where a single failure in an electrical system can result in loss of capability to perform a safety function, the effect on plant safety must be evaluated. This is necessary regardless of whether the loss of safety function is caused by a component failing to perform a requisite mechanical motion, or by a component performing an undesirable mechanical motion.

This position establishes the acceptability of disconnecting power to electrical components of a fluid system as one means of designing against a single failure that might cause an undesirable component action. These provisions are based on the assumption that the component is then equivalent to a similar component that is not designed for electrical operation, e.g., a valve that can be opened or closed only by direct manual operation of the valve. They are also based on the assumption that no single failure can both restore power to the electrical system and cause mechanical motion of the components served by the electrical system. The validity of these assumptions should be verified when applying this position.

B. BRANCH TECHNICAL POSITION

1. Failures in both the "fail to function" sense and the "undesirable function" sense of components in electrical systems of valves and other fluid system components should be considered in designing against a single failure, even though the valve or other fluid system component may not be called upon to function in a given safety operational sequence.

2. Where it is determined that failure of an electrical system component can cause undesired mechanical motion of a valve or other fluid system component and this motion results in loss of the system safety function, it is acceptable, in lieu of design changes that also may be acceptable, to disconnect power to the electric systems of the valve or other fluid system component. The plant technical specifications should include a list of all electrically-operated valves, and the required positions of these valves, to which the requirement for removal of electric power is applied in order to satisfy the single failure criterion.

3. Electrically-operated valves that are classified as "active" valves, i.e., are required to open or close in various safety system operational sequences, but are manually-controlled, should be operated from the main control room. Such valves may not be included among those valves from which power is removed in order to meet the single failure criterion unless: (a) electrical power can be restored to the valves from the main control room, (b) valve operation is not necessary for at least ten minutes following occurrence of the event requiring such operation, and (c) it is demonstrated

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that there is reasonable assurance that all necessary operator actions will be performed within the time shown to be adequate by the analysis. The plant technical specifications should include a list of the required positions of manually-controlled, electrically-operated valves and should identify those valves to which the requirement for removal of electric power is applied in order to satisfy the single failure criterion.

4. When the single failure criterion is satisfied by removal of electrical power from valves described in (2) and (3), above, these valves should have redundant position indication in the main control room and the position indication system should, itself, meet the single failure criterion.

5. The phrase "electrically-operated valves" includes both valves operated directly by an electrical device (e.g., a motor-operated valve or a solenoid-operated valve) and those valves operated indirectly by an electrical device (e.g., an air-operated valve whose air supply is controlled by an electrical solenoid valve).

C. REFERENCES

1. Memorandum to R. C. DeYoung and V. A. Moore from V. Stello, October 1, 1973.

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